Dominion Energy Nuclear Connecticut, Inc
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DominionEnergy.com

June 25, 2025

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

Serial No.: 25-165
NRA/NDM: R0
Docket No.: 50-336
License No.: DPR-65

DOMINION ENERGY NUCLEAR CONNECTICUT, INC
MILLSTONE POWER STATION UNIT 2
UPDATE TO THE FINAL SAFETY ANALYSIS REPORT – REVISION 43

Pursuant to 10 CFR 50.71(e), Dominion Energy Nuclear Connecticut (DENC) submits Revision 43 to the Updated Final Safety Analysis Report (UFSAR) for Millstone Power Station Unit 2 (MPS2).

The complete MPS2 UFSAR, Revision 43, is provided in Enclosure 1 and contains Security-Related Information that DENC requests be withheld from public disclosure in accordance with 10 CFR 2.390. A public version of the MPS2 UFSAR, Revision 43, is provided in Enclosure 2.

A list and brief description of changes since the previous annual update is provided in the Revision Summary located in the Frontmatter of each enclosure. Third-party documents included by reference in the MPS2 UFSAR may not be text searchable.

A review in accordance with 54.37(b) has been completed, which identified no new systems, structures or components subject to aging management or time-limited aging analyses for MPS2.

If you have any questions or require additional information regarding this submittal, please contact Nick Maynard at (804) 273-3910.

Sincerely,

James E. Holloway
Vice President – Nuclear Engineering & Fleet Support

ALLEN FULMER
NOTARY PUBLIC
REG. #00286846
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES JUNE 30, 2028

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by James E. Holloway, who is Vice President – Nuclear Engineering & Fleet Support of Dominion Energy Nuclear Connecticut. He has affirmed before me that he is duly authorized to execute and file the foregoing document on behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 25 day of June, 2025.

My Commission Expires: June 30, 2028.

Notary Public

ENCLOSURE 1 TO THIS LETTER CONTAINS SECURITY-RELATED INFORMATION WHICH IS TO BE WITHHELD FROM PUBLIC DISCLOSURE UNDER 10 CFR 2.390. UPON REMOVAL OF ENCLOSURE 1, THIS LETTER IS DECONTROLLED.

Enclosures:

1. Millstone Power Station Unit 2, Updated Final Safety Analysis Report, Revision 43 **[Security-Related Information – Withhold Under 10 CFR 2.390]**
2. Millstone Power Station Unit 2, Updated Final Safety Analysis Report, Revision 43 [Redacted]

Commitments made in this letter: None.

cc: U. S. Nuclear Regulatory Commission
Region I
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NRC Senior Resident Inspector (w/enclosures)
Millstone Power Station

Serial No.: 25-165
Docket No.: 50-336

ENCLOSURE 2

MILLSTONE POWER STATION UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT

REVISION 43

[REDACTED]

Dominion Energy Nuclear Connecticut, Inc
Millstone Power Station Unit 2



MILLSTONE POWER STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

Revision 43—06/30/25

Revision Summary

Revision 43—06/30/25

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
43	01/30/25	MP2-UCR-2024-006	14.6.5, 14.6.5.2, 14.6.5.2.7, 14.6.5.4, 14.6.6, Tables 14.1.5.2-6, 14.6.5.4-1 through 7, 14.6.5.2-7, Figures 14.0.7-1, 5, &6, 14.6.5.4-1 through 19	Millstone Unit 2 Cycle 30 Updates to Millstone Unit 2 Final Safety Analysis Report EVAL-ENG-RSE-M2C30, Revision 0
43	01/30/25	MP2-UCR-2024-007	5.2.6.2.3, Table 5.2-12	Containment Penetrations 15 and 16 Historical Material Changes MP2-11-01093, MP2-12-01140
42.01	09/30/24	MP2-UCR-2024-002	Table 11.1-1	FSAR Table 11.1-1 for Radwaste Component Information contains Historical Errors and Incomplete Info CR1258711
42.01	09/30/24	MP2-UCR-2024-004	Table 8.3-3	Editorial Correction to MP2 FSAR Table 8.3-3 CA12378363

Revision 42—06/27/24

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
42	06/27/24	MP2-UCR-2023-001	9.8.2.1.1.2	[Update to Section 9.8.2.1.2-7 describing the ISFSI, updating to include EOS spent fuel storage system ETE-NAF-2022-0113]
42	06/27/24	MP2-UCR-2023-006	6-i, 6.2.2.1, 6.2.5	[Markups for MPS U2 GL 2004-02 Closure LA8491301]

Revision 42—06/27/24

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
41.01	01/31/24	MP2-UCR-2017-016	7.10.2, 9.10.2.1, 9.13.2.1	Update Unit 2 FSAR due to Retiring the Fire Water Storage Tank Auxiliary Steam Heating System. MPG-17-00116
41.01	01/31/24	MP2-UCR-2022-006	2.5.4.22, 2.5.4.2.3, Figure 2.5-18	FSAR Change for EC MP2-22-01087 to Replace Turbine Building Stoplog Barrier with Flood Door MP2-22-01087
41.01	01/31/24	MP2-UCR-2022-007	14.8.2.1.4	2-MS-4B Piping Reconfiguration & Bypass Line Removal EC MP2-22-01094
41.01	01/31/24	MP2-UCR-2023-005	14.1.5.2.5, 14.1.5.2.5.2, 14.1.5.2.6.1.5, 14.1.5.2.6.2.5, 14.1.5.2.7, 14.1.6, Table 14.1.5.2-6	Implementation of New CHF Correlation for Analysis of the Post-Scram Main Steam Line Break Accident EVAL-ENG-RSE-M2C29, Revision 0, Addendum A

Revision 41—06/29/23

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
41	06/29/23	MP2-UCR-2022-004	1.2.5, 1.2.12.d, 5.E, 11.1.3.1, 14.6.5.2.5.3, 14.6.5.2.5.4, Figures 8.1-1D and 14.6.5.2-15	Correct identified MPS 2 FSAR errors CR1194532, MP2-UCR-2015-020, MP2-UCR-2016-013, MP2-UCR-2016-016, MP2-UCR-2018-010, MP2-UCR-2018-015
41	06/292/23	MP2-UCR-2019-010	7.5.6.2.2.2.3, 7.5.6.2.2.2.4	[MP2 PIOPS Upgrade MP2-19-01091]

Revision 41—06/29/23

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
41	06/29/23	MP2-UCR-2023-002	3,5.2, 3.5.5, 14.0.7, 14.0.13, 14.1.5, 14.1.6, 14.3.3, 14.4.3, 14.4.8, 14.4.10, Table 14.1.5.2-6, Figure 1.0.7-(1,5,6)	Millstone Unit 2 Cycle 29 Updates to Millstone Unit 2 Final Safety Analysis Report EVAL-ENG-RSE-M2C29, Revision 0
41	06/29/23	MP2-UCR-2020-005	7.2.3.2.2	New Trip Units Applicable to All RPS Channels MP2-15-01097 Rev 7
41	06/29/23	MP2-UCR-2022-001	8.5.22, 8.5.4.1	MP2 Battery Charger Replacement (201B/DC2) MP2-21-01158
40.01	09/29/22	MP2-UCR-2022-003	2.5.4.2.1.b.1.B	Statement on PMW as Backup to AFW Discrepancy ETE-MP-2021-1131, Revision 1 CR1198357

Revision 40—06/30/22

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
39.03	03/31/22	MP2-UCR-2021-007	2.5.4.2.1.A.12, 2.5.4.2.1.B.2.c, 2.5.4.2.1.B.3.c	Service Water Pump Motor Restoration Flood Water Level CR1171836
39.03	03/31/22	MP2-UCR-2022-002	Table 14.6.3-4, Table 14.6.3-5	Corrects transcription errors identified by the NRC CR1191354
39.02	01/27/22	MP2-UCR-2016-017	Tables 6.2-1, 6.4-1, 9.4-1, 9.5-2, and 11.1-1	Change to allow ANSI class ratings other than 600 or 800 to be used in various systems. CA 3033065

Revision 40—06/30/22

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
39.02	01/27/22	MP2-UCR-2018-013	7.5.7	MP2 Loose Parts Monitoring System Replacement MP2-17-00158-000
39.02	01/27/22	MP2-UCR-2020-007	Figure 8.1-1D	345kV Switch 15G-10T-4 Replacement MP2-19-01182
39.02	01/27/22	MP2-UCR-2020-009	8.5.22, 8.5.4.1	MP2 Battery Charger Replacement (201A/DC1) MP2-19-01174
39.02	01/27/22	MP2-UCR-2021-005	3.3.4, Table 14.1.5.2-6, Figures 14.0.7-1, 14.0.7-5, 14.0.7-6	Millstone Unit 2 Cycle 28 Updates to Millstone Unit 2 Final Safety Analysis Report EVAL-ENG-RSE-M2C28, Revision 0
39.02	01/27/22	MP2-UCR-2021-006	Table 5.2-11	MP2 10CFR50 Appendix J Scope Reduction Shutdown Cooling Penetration 10 ETE-MP-2021-1088
39.01	09/30/21	MP2-UCR-2018-017	7.8	FSAR change to update Communications System for replacement of Trunked Radio System. MPG-15-01007, CR499765
39.01	09/30/21	MP2-UCR-2021-002	Table 7.5-3	Update MP2 FSAR Accident Monitoring Instrument Table 7.5-3 ETE-MP-2020-1068

Revision 39—06/30/21

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
39	03/31/21	MP2-UCR-2020-011	7.11, 12.10	Addition of 50.69 Program P3 Technical Specification Amendment 337 Risk-Informed Categorization and Treatment of Structure, Systems, and Components of Nuclear Power Reactors
39	03/31/21	MP2-UCR-2021-001	14.6.1.4, 14.6.1.15, 14.6.1.6, 14.6.6, 14.6.7, Tables 14.6-1 and 14.6.1-2	Removal of Inadvertent Opening of a Power Operated Relief Valve in Mode 2 from the Millstone Unit 2 FSAR ETE-NAF-2020-0118
39	03/31/21	MP2-UCR-2021-003	1.10, 12.8	Incorporating Quality Assurace Program Description by Reference CA8102163
38.02	03/31/21	MP2-UCR-2016-011	8.1.2.2.c	MP2 FSAR Update To Document The Addition Of The Open Phase Detection Relay Schemes On The High Voltage Side Of The Generator Step-up And Rsst Transformers. MP2-15-01026, MP2-17-00156
38.02	03/31/21	MP2-UCR-2016-013	1.2.12.d, 11.1.3.1, 11.1.3.2, 11.1.5.1, Table 11.1-1, Figures 1.2-9, 9.12-1, 10.3-1, 11.1-4, 11.1-3, 11.2-5	Update the affected sections due to the removal of demineralizers T141A, T141B, T141C, and the installation of the Energy Solutions ALPS/AIM and SEDS systems. MP2-14-01169, MP2-15-01044, DC MP2-15-01045
38.02	03/31/21	MP2-UCR-2016-014	Table 11.1-1	Update Liquid and Aerated Waste Valve information to allow other stainless steel material grads and other ANSI ratings. MP2-14-01169

Revision 39—06/30/21

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
38.02	03/31/21	MP2-UCR-2018-015	Appendix 5.E.VI	Add NUPIPE II computer code to UFSAR Appendix 5E.VI MP2-17-00178
38.02	03/31/21	MP2-UCR-2019-001	14.1.5.3, 14.6.3.6.2, Table 14.1.5.3-1 , 14.1.5.3-2, 14.6.3-4 through 14.6.3-6	Updated Main Steam Line Break and Steam Generator Tube Rupture Dose Consequences. ETE-NAF-2019-0010, CR 1094790, CR 1103271
38.02	03/31/21	MP2-UCR-2019-007	9.8.2.2.3.5	Permanently Bypass RFM Aux. Hoist Load Cell Interlock, M2H12. MP2-19-01136
38.02	03/31/21	MP2-UCR-2020-001	Figure 8.1-1D	345kV Switch 15G-18T-8 Replacement. MP3-18-01094
38.02	03/31/21	MP2-UCR-2020-008	5.2.8.4.1, 5.2.8.4.2	Revision of "MP2 Tendon Surveillance Program Description". LBDCR 19-MP2-009
38.02	03/31/21	MP2-UCR-2020-010	7.8.3	Replacing ENRS (ARCOS) with DEENS (Everbridge) MP-20-04-00 - EP-AA-101-Attachment 2, 50.54(q), for change from Emergency Notification & Response System (ENRS) to Dominion Energy Emergency Notification System (DEENS)
38.01	09/30/20	MP2-UCR-2015-010	8.6.2.2, 5.6.4.1	Reflects replacement of inverters (INV 2, INV4, INV6). MP2-14-01158
38.01	09/30/20	MP2-UCR-2015-020	Figure 8.1-1D	Reflects revised discussion regarding piping weld examination method(s). LBDCR 14-MP2-019

Revision 39—06/30/21

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
38.01	09/30/20	MP2-UCR-2016-005	7.2.3.2.2	Delete the word "reed" in the first paragraph in MP2 FSAR section 7.2.3.2.2. MP2-15-01097
38.01	09/30/20	MP2-UCR-2020-004	6.7.2	Update Unit 2 FSAR to reflect changes for License Amendment No. 338. License Amendment No. 338

Revision 38—06/30/20

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
38	06/30/20	MP2-UCR-2019-030	8.3	Increasing the TRM required Diesel Fuel Oil Storage for T-148. ETE-MP-2020-1002
38	06/30/20	MP2-UCR-2020-002	3.3.1.1.2, 3.3.4, 3.5.2.2.1, 3.5.5, Figure 14.0.7-1, Figure 14.0.7-5, Figure 14.0.7-6, Table 14.1.5.2-6	Millstone Unit 2 Cycle 27 Updates to Millstone Unit 2 Final Safety Analysis Report. EVAL-ENG-RSE-M2C27
38	06/30/20	MP2-UCR-2020-003	5.2.6.1.1, 8.7.1.2, 8.7.2.2	Update the MP2 FSAR to reflect IEEE Standard 317-1976 as applicable to replacement medium voltage electrical containment penetration modules. MP2-19-01025
37.03	03/31/20	MP2-UCR-2019-006	9.2.4.1	Interm Retirement of MP2 Boric Acid Heat Tracing Circuits. ETE-MP-2019-1081

Revision 38—06/30/20 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
37.03	03/31/20	MP2-UCR-2018-010	1.2.5, 1.4.3.1, 4.3.2, Table 4.3-2, Table 10.2-1	Updating The FSAR to Accurately Reflect The New Unit 2 Main Steam Moisture Content. MP2-19-01066
37.03	03/31/20	MP2-UCR-2019-011	8.5.1.2	Changing of the Unit 2 Vital Battery Discharge Testing and Battery Charger Load Testing frequencies. ETE-MP-2019-1055
37.02	12/30/19	MP2-UCR-2017-017	Figure 14.0.7-5	Departure from Nucleate Boiling Limiting Condition for Operation Verification Results. ETE-NAF-2017-0169

Revision 38—06/30/20 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
37.02	12/3019	MP2-UCR-2019-005	4.3.8.1.1, 4.3.8.3.1, 7.5.1.2.1.1.a, 7.5.1.2.1.1.c, 7.5.1.2.2, 7.5.5.1, 7.5.5.3, 7.5.5.4, 7.5.6.2.2.2.6, 7.5.6.3.2.1.8, 7.5.6.3.2.2.7, 7.6.2.2.2	Trend Recorder Description. MP2-17-00101, DCR M2-96064, M2-05003
37.01	09/30/19	MP2-UCR-2017-010	2.5.4.2.1	AOP 2560 - Actions Due to Intake Water Level Elevation Approaching 16’-6”

Revision 38—06/30/20 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
37.01	09/3019	MP2-UCR-2019-002	7.3.4.2.3	SAR section 7.3.4.2.3 not consistent with Specification 7604-M-480, Specification For Engineered Safety Actuation Systems for The Millstone Point Company Millstone Nuclear Power Station Unit 2 Waterford Connecticut. STI-M2-2018-001

Revision 37—06/27/19

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
36.2	12/31/18	MP2-UCR-2018-001	8.6.2.2	Unit 2 inverter replacement. MP2-14-01130
36.2	12/31/18	MP2-UCR-2018-005	7.4.7.2.2	Millstone Unit 2 Main Control Room Recorders Replacement on panel C-05. MP2-17-00171
36.2	12/31/18	MP2-UCR-2018-011	8.1.2.2	Generator Backup Audio-Tone Equipment Replacement. MP2-18-00102
36.2	12/31/18	MP2-UCR-2018-012	Table 11.1-1	Correction of Volume of Waste Gas Surge Tank in FSAR Table 11.1.-1. CR1078780, CR1091106
36.2	12/31/18	MP2-UCR-2018-014	14.0.7.2, 14.0.7-5, Figure 14.0.7-6, Table 14.1.5.2-6	Millstone Unit 2 Cycle 26 Updates to Millstone Unit 2 Final Safety Analysis Report. EVAL-ENG-RESE-M2C26

Revision 37—06/27/19 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
36.2	12/31/18	MP2-UCR-2018-016	Table 14.6.3-5	Specification of the Basis for the Steam Generator Partition Factor for Particulates. ETE-NAF-2018-0099
36.1	09/27/18	MP2-UCR-2015-017	8.4.2.1, 8.5.1.1, 8.5.2.1, 8.5.2.2, 8.5.3.1, 8.5.3.2	Reflects removal of 125 VDC lighting loads from Vital Station Batteries & installation of LED lighting. MP2-14-01185
36.1	09/27/18	MP2-UCR-2018-007	14.1.5.2.5.1.4, 14.1.5.2.5.1.6, 14.1.5.2.6.2.1, 14.1.5.2.6.2.3, Table 14.1.5.2-5, Table 14.1.5.2-6, Table 14.1.5.2-8, Figure 14.1.5.2.-9, Figure 14.1.5.2.-10, Figure 14.1.5.2.-11, Figure 14.1.5.2.-12, Figure 14.1.5.2.-13, Figure 14.1.5.2.-14, Figure 14.1.5.2.-15, Figure 14.1.5.2.-16	MPS2 FSAR Update for Post-Scram Main Steam Line Break with Decreased MTC and Revised AFW Flow Distribution. ETE-NAF-2018-0019
36.1	09/27/18	MP2-UCR-2018-009	14.6.5.2, Table 14.6.5.2-7	MPS3 FSAR Update to the Small Break Loss of Coolant Accident (SBLOCA) Results Due to Error Correction. ETE-NAF-2018-0060

Revision 36—06/28/18

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
36	06/28/18	MP2-UCR-2017-005	10.4.5.5	Update FSAR Section 10.4.5.5 to read as follows: "The plant can carry 55 percent load with one-half of the feedwater heaters out of service and isolated from the system, with only one condensate pump in operation and with only one SGFP in service". LA3022214
36	06/28/18	MP2-UCR-2018-002	Table 1.3-1, Table 11.1-1	Correction of Volume of Waste Gas Test Tank in FSAR table 11.1-1 CR1078780 and CR1080672
36	06/28/18	MP2-UCR-2018-006	Table 4.6-2	Update to MPS2 Pressurizer RTNDT values determined through Drop Weight Testing. AREVA CR 2017-2164
35.3	03/29/18	MP2-UCR-2017-017	Table 14.0.7-5, Figure 14.0.7-5	Departure from Nucleate Boiling Limiting Condition for Operation Verification Results. ETE: ETE-NAF-2017-0169 CA3060853
35.3	03/29/18	MP2-UCR-2018-003	Table 9.11-2	Change Pressure Values on Table 9.11-2 for Instrument Air Receiver Tanks T-51E and T-51F. DC: MP2-17-00175
35.2	12/28/17	MP2-UCR-2017-013	9.2.3.3, 11.1.3.1	Modify U3 FSAR to correct item related to previous degasifier related FSAR change. FSARCR: 98-MP2-167
35.2	12/28/17	MP2-UCR-2017-014	9.2.2.1	Modify U2 FSAR to correct item related to VCT level controls related to a previous FSAR change. FSARCR: 97-MP2-66

Revision 36—06/28/18 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
35.1	09/28/17	MP2-UCR-2016-001	1.A, 8.1.2.2, 8.1.2.3	Eversource 15G Breaker Failure Relay Replacement. ETE-MP-2015-1186
35.1	09/28/17	MP2-UCR-2016-018	8.1.1.2, 8.1.2.1, 8.1.2.2, Fig 8.1-1B, Fig 8.1-1C, Fig 8.1-1D	Eversource Energy proposes to reconfigure the 345 kV Millstone - Haddam - Besesk 348 line. ETE-MP-2016-1135
35.1	09/28/17	MP2-UCR-2016-019	7.6.5.4.1	App R fire panel door lock removal. CR1045068
35.1	09/28/17	MP2-UCR-2016-020	7.5.5.3	Plant Process Computer Data Diode Installation. MP2-16-01035
35.1	09/28/17	MP2-UCR-2017-001	Figure 6.5-1, Table 6.5-2	Replacement of “C” CAR Fan M2F14C. MP2-17-01004

Revision 36—06/28/18 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
35.1	09/28/17	MP2-UCR-2017-002	1.4.2, 3.1, 3.2.3, 3.3.1, 3.3.1.1, 3.3.1.1.1, 3.3.1.1.2, 3.3.1.3.1, 3.3.1.3.2, 3.3.4 Refs, Table 3.3-1, Figure 3.3-1A, Figure 3.3-1B, 3.4.2, 3.4.3.2.3, 3.5.1.1, 3.5.2.1.2, 3.5.2.1.3, 3.5.3.3.1, 3.5.3.3.4, Table 3.5-1, 9.5.2.1, 9.5.3.1, 9.5.4.1, 14.0.2, 14.0.7.1, 14.0.10, Table 14.0.2-2, Table 14.0.7-4, Table 14.0.7-5, Figure 14.0.7-1, Figure 14.0.7-2, Figure 14.0.7-5, Figure 14.0.7-6, 14.1.5.1.6.1, 14.1.5.1.6.1.3, 14.1.5.1.7, 14.1.5.2.5.1.4, 14.1.5.2.5.1.6, 14.1.5.2.6.1.1, 14.1.5.2.6.1.2, 14.1.5.2.6.1.3, Table 14.1.5.1-5, Table 14.1.5.1-6, able 14.1.5.1-7, Figure 14.1.5.1-1, Figure 14.1.5.1-2	This SAR Change Request updates the Millstone Unit 2 FSAR for the Cycle 25 core design and supporting safety analyses. ETE-NAF-2017-0001, EVAL-ENG-RSE-M2C25

Revision 36—06/28/18 (Continued)

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
35.1	09/28/17	MP2-UCR-2017-002 (cont)	Figure 14.1.5.1-3, Figure 14.1.5.1-4, Figure 14.1.5.1-5, Figure 14.1.5.1-6, Table 14.1.5.2-4, Table 14.1.5.2-5, Table 14.1.5.2-6, Table 14.1.5.2-7, Figure 14.1.5.2-2, Figure 14.1.5.2-3, Figure 14.1.5.2-4, Figure 14.1.5.2-5, Figure 14.1.5.2-6, Figure 14.1.5.2-7, Figure 14.1.5.2-8, Figure 14.1.5.2-9, Table 14.2.4-3, Figure 14.2.4-1, Figure 14.2.4-2, Figure 14.2.4-3, Figure 14.2.4-4, Figure 14.2.4-5, 14.4.6.5, 14.4.6.6, 14.4.10 Refs, Table 14.4.6-2, Table 14.4.6-3, Table 14.4.6-4, 14.6.5, 14.6.5.1 through 14.6.5.8, 14.6.5.4 through 14.6.5.4.8 (new)	This SAR Change Request updates the Millstone Unit 2 FSAR for the Cycle 25 core design and supporting safety analyses. ETE-NAF-2017-0001, EVAL-ENG-RSE-M2C25

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Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
35.1	09/28/17	MP2-UCR-2017-002 (cont)	14.6.6 Refs, Table 14.6.5.1-3, Table 14.6.5.1-4, Table 14.6.5.1-5, Table 14.6.5.1-6, Table 14.6.5.1-7, Table 14.6.5.1-8, Figures 14.6.5.1-1 through Figure 14.6.5.1-17, Tables 14.6.5.4-1 through Table 14.6.5.4-7, Figures 14.6.5.4-1 through Figures 14.6.5.4-19	This SAR Change Request updates the Millstone Unit 2 FSAR for the Cycle 25 core design and supporting safety analyses. ETE-NAF-2017-0001, EVAL-ENG-RSE-M2C25
35.1	09/28/17	MP2-UCR-2017-006	4.3.8.5.11	Update to reflect that during normal operation, strainers for the B and C RCPs are operated with their strainer baskets removed and the A and D RCPs do not have strainers. MP2-17-01033
35.1	09/28/17	MP2-UCR-2017-007	7.3.1.2.8	Engineered Safety Feature Actuation UFSAR Response Time Change. ETE-MP-2015-1030
35.1	09/28/17	MP2-UCR-2017-008	4.3.2, 4.3.5	MP2 Plant Process Computer Replacement additional FSAR changes missed during the initial project. MP2-13-01067
35.1	09/28/17	MP2-UCR-2017-009	8.3.3.1	Implementation of Revised Combustible Gas (Hydrogen) Control Requirements in Containment additional FSAR changes missed during the initial project. M2-04009

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35.1	09/28/17	MP2-UCR-2017-011	9.10.2.1	Editorial correction to MP2 FSAR. Change last 2 sentences of 4th paragraph of 9.10.2.1 on page 9.10-2 "Pump Running and trouble signals for pump P-82 are available in the Unit 2" (Should be Unit 3) Control Room. DM3-01-0064-00
35.1	09/28/17	MP2-UCR-2017-012	14.8.2.1.2	Modify FSAR sections 10.4.5.5 and 14.8.2.1.2 to correct editorial errors. LA3022214, CR1047811

Revision 35—06/29/17

Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
35	06/29/17	Administrative	As identified in the 2017 NRC Submittal List of Changed Pages and submitted Summary of Change.	Major Revision issued for the purpose of 2017 NRC Submittal. Updated footer on all pages to reflect Revision 35.
35	06/29/17	MP2-UCR-2017-004	10.3.4.2, Table 10.3-1	LA3022214 - Modify FSAR Section 10.3.4.2 Tests and Inspection to read "See Section 5.2.7.4.1…" in lieu of 5.2.8.4.2.
35	06/29/17	MP2-UCR-2015-025	14.0.11, 14.6.1.1, 14.6.1.6	Confirmatory Order EA-13-188, AREVA analysis ANP-3406, Rev. 0 - Remove credit for charging from the long-term portion of the inadvertent opening of the PORV (IOPORV) event.

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Revision	Revision Release Date	Change Activity	Document Elements Affected (Sections, Tables, Figures)	Summary Description of Changes Made under the provisions of 10 CFR 50.59 except where indicated in brackets.
34.3	03/30/17	MP2-UCR-2016-008	9.5.1.1, 9.5.2.1, 9.5.3.1	The NRC requested additional information be provided in the FSAR to describe the procedural controls in place to maintain the Millstone 2 Spent Fuel Pool temperature below 150°F. LAR 15-571
34.2	12/22/16	MP2-UCR-2016-016	14.6.5.2, 14.6.5.2.5, 14.6.5.2.5.1, 14.6.5.2.5.2, 14.6.5.2.5.3, 14.6.5.2.5.4, 14.6.5.2.5.5, 14.6.5.2.5.8, 14.6.5.2.5.9, 14.6.5.2.6, 14.6.5.2.7, 14.6.5.2.8, Figures 14.6.5.2-1, 14.6.5.2-2, 14.6.5.2-3, 14.6.5.2-4, 14.6.5.2-5, 14.6.5.2-6, 14.6.5.2-7, 14.6.5.2-8, 14.6.5.2-9, 14.6.5.2-10, 14.6.5.2-11, 14.6.5.2-12, 14.6.5.2-13, 14.6.5.2-14, 14.6.5.2-15, 14.6.5.2-16, 14.6.5.2-17, 14.6.5.2-18, 14.6.5.2-19, 14.6.5.2-20, 14.6.5.2-21, 14.6.5.2-22, Tables 14.6.5.2-3, 14.6.5.2-4, 14.6.5.2-5, 14.6.5.2-6, 14.6.5.2-7	This SAR change request updates the Small Break Loss of Coolant Accident (SBLOCA) analysis presented in FSAR Section 14.6.5.2.
34.2	12/22/16	MP2-UCR-2016-012	15.2.1.21	MP3 HVK Chiller Service Water Thermal Performance Testing Elimination.
34.2	12/22/16	MP2-UCR-2016-010	15.2.1.2	Cancellation of Millstone 2 Boraflex Monitoring Program.

Revision 35—06/29/17 (Continued)

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34.2	12/22/16	MP2-UCR-2016-009	11.E.1	RM-8123 and RM-8262 are described incorrectly in the FSAR. This change removes the reference to a gamma geiger muller detector chamber type and replace it with the correct beta scintillator type detector.
34.2	12/22/16	MP2-UCR-2016-003	9.3.2.3	Update Subsection 9.3.2.3 to reflect changes made by DCN DM2-00-0166-07 that eliminated the packing leak-off lines on M22-SI-306.
34.2	12/22/16	MP2-UCR-2015-007	1.2.10.4, 1.A, 9.8.2.1.1, 9.8.2.1.2, 9.8.2.2.2, Figure 9.8-7, Figure 9.8-8A, Figure 9.8-8B, Figure 9.8-8C, 14.7.4.1	Reflects changes associated with LBDCR 12-MP2-008.
34.2	12/22/16	MP2-UCR-2013-017	Table 6.7-1, Table 9.9-11	Reflects correction of technical value in the respective tables.
34.1	9/29/16	MP2-UCR-2016-007	Chapter 1 section 1.A; Chapter 8 List of Figures; Chapter 8 Section 8.1.1.4, 8.1.2.1; 8.1.2.2; 8.1.2.3 and Figure 8.1-1A; 8.1-1B	Unit 2 SLOD Removal (LBDCR 15-MP2-006)
34.1	9/29/16	MP2-UCR-2016-006	Section 14.8.2.2.4	Millstone 2 NRC License Amendment 326, “Millstone Power Station Unit 2 - Issuance of Amendment to Revise Technical Specifications for Containment Leak Rate Testing (CAC No. MF5838),” March 31, 2016 (ADAMS Accession Number ML16068A312).
34.1	9/29/16	MP2-UCR-2016-004	Table 11.2-1	Correct typo identified by CR1017686.

LIST OF ABBREVIATIONS

Abbreviation	
ADH	Aerated Drain Header
AEAS	Auxiliary Exhaust Actuation Signal
AES	Auxiliary Exhaust System
ATWS	Anticipated Transient Without Scram
BOL	Beginning of Life
BWR	Boiling Water Reactor
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEDS	Control Element Drive System
CFR	Code of Federal Regulations
CIAS	Containment Isolation Actuation Signal
CRFS	Control Room Filtration System
CSAS	Containment Spray Actuation Signal
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
CVH	Containment Vent Header
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DRL	Division of Reactor Licensing
EB	Enclosure Building
EBFR	Enclosure Building Filtration Region
EBFS	Enclosure Building Filtration System

List of Abbreviations (Continued)

Abbreviation	
ECCS	Emergency Core Cooling System
EOL	End of Life
ESCS	Engineered Safety Features Control System
F	Degrees Fahrenheit
FPS	Fire Protection System
FT	Fuel Transfer
FW	Feed Water
HMP	High Mechanical Performance
HPSI	High Pressure Safety Injection
HTP	High Thermal Performance
IA	Instrument Air
ID	Inside Diameter
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MPC	Maximum Permissible Concentration
MSL	Mean Sea Level
MSS	Main Steam System
MTU	Metric Tons Uranium
MWD	Megawatt Days
NDT	Non-Destructive Testing
NDTT	Nil Ductility Transition Temperature
NS	Nitrogen Supply
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake

List of Abbreviations (Continued)

Abbreviation	
OD	Outside Diameter
PA	Purge Air
PDH	Primary Drain Header
PDT	Primary Drain Tank
PPM	Parts Per Million
PVMP	Preoperational Vibration Monitoring Program
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RBCCWS	Reactor Building Closed Cooling Water System
RCS	Reactor Coolant System
RWS	Radioactive Waste System
RWST	Refueling Water Storage Tank
SA	Station Air
SGBS	Steam Generator Blowdown System
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SRAS	Sump Recirculation Actuation Signal
SRST	Spent Resin Storage Tank
SS	Sampling System
TBCCWS	Turbine Building Closed Cooling Water System
WGH	Waste Gas Header
WREC	Westinghouse Rector Evaluation Center
WTR	Westinghouse Test Reactor

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a. Incorporated by reference this engineering controlled drawing is, coincidentally, an MPS-2 FSAR figure. Refer to the controlled plant drawing for the latest revision.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 1: Introduction And Summary

CHAPTER 1—INTRODUCTION AND SUMMARY

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CHAPTER 1 – INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) was initially submitted in support of the application of The Connecticut Light and Power Company (CL&P), The Hartford Electric Light Company (HELCO), Western Massachusetts Electric Company (WMECO), and Northeast Nuclear Energy Company (NNECO), for a license to operate the second nuclear powered generating unit at the site of the Millstone Power Station. Since the initial licensing of the unit, unless otherwise indicated, the FSAR has been updated a number of times to reflect current design and analysis information. On the basis of the information presented in the FSAR and referenced material at the time of application for operating license, the applicants concluded that Millstone Unit 2 is designed and constructed and will be operated without undue risk to the health and safety of the public.

Construction of Millstone Unit 2 was authorized by the United States Atomic Energy Commission (AEC) when it issued Provisional Construction Permit CPPR-76 on December 11, 1970. Commercial operation of Millstone Unit 2 commenced in December 1975 at a gross electrical output of 865 megawatts.

Millstone Unit 2 is located Millstone Point in the Town of Waterford, Connecticut. It is located immediately to the north of the first unit (Millstone Unit 1) and south of the third unit (Millstone Unit 3). Commercial operation of Millstone Unit 1 was authorized by the AEC by issuing Provisional Operating License DPR-21 on October 7, 1970. Commercial operation of Millstone Unit 1 commenced in December, 1970. Commercial operation of Millstone Unit 3 was authorized by the United States Nuclear Regulatory Commission (NRC) (formerly the AEC) by issuing the Low Power License on November 25, 1985, and the Full Power License on January 31, 1986. Commercial operation of Millstone Unit 3 commenced in April 1986. A licensing history for the Millstone Unit 2 plant is presented in Table 1.1-1.

Millstone Unit 2 utilizes a pressurized water nuclear steam supply system (NSSS). The unit is similar, in this respect, to the former Yankee Atomic Electric Company generating plant in Rowe, Massachusetts, (NRC Docket Number 50-29), the former Haddam Neck Plant operated by the Connecticut Yankee Atomic Power Company on the Connecticut River at Haddam, Connecticut (NRC Docket Number 50-213), and the Maine Yankee Atomic Power Company plant at Wiscasset, Maine (NRC Docket Number 50-309). The NSSS for Millstone Unit 2 is supplied by Combustion Engineering, Inc. (CE) which also supplied the steam supply system for the Maine Yankee plant. The Millstone Unit 2 NSSS is similar to the systems supplied by CE for the initial two units of the Baltimore Gas and Electric Calvert Cliffs Nuclear Power Plant (NRC Docket Numbers. 50-317 and 50-318).

Millstone Unit 2 has been designed to operate safely under all normal operating conditions and anticipated transients. Although the unit produces small amounts of radioactive waste, the offsite disposal of these wastes is rigidly controlled and maintained below established limits.

In 2001, Millstone Units 1, 2 and 3 operating licenses were transferred from Northeast Nuclear Energy Company to Dominion Nuclear Connecticut, Inc. (DNC).

DNC is an indirect wholly-owned subsidiary of Dominion Energy, which is in turn owned by Dominion Resources, Inc. (DRI). Virginia Power, which is the licensed owner and operator of the North Anna and Surry nuclear stations, is also a subsidiary of DRI.

The transmission and distribution assets on the site will continue to be owned by Connecticut Light and Power (CL&P) and will be operated under an Interconnection Agreement between CL&P and DNC.

The FSAR will retain references to Northeast Utilities and Northeast Nuclear Energy Company documents/activities when they are used in a historic context and are required to support the plant licensing bases.

Upon license transfer, all records and design documents necessary for operation, maintenance, and decommissioning were transferred to DNC. Some of these drawings are included (or referenced) in this FSAR. These drawings often have title blocks (or drawing numbers) which list Northeast Nuclear Energy Company (et. al) or Northeast Utilities Service Company (et. al). In general, no changes to these title blocks will be made at this time. Based on this general note, these drawings shall be read as if the title blocks list Dominion Nuclear Connecticut, Inc.

Millstone Unit 2 has been designed to operate reliably without accident. Nevertheless, to ensure that no reasonably credible accident could result in dangerous releases of radioactive material, the unit incorporates a number of features designed to minimize the effects of such an accident. The adequacy of these safety features under the conditions of various postulated accidents is discussed in Chapter 14.

The initial license to operate Millstone Unit 2 was at a full power core thermal output of 2560 megawatts. This corresponded to a NSSS thermal rating, which includes core power and other reactor coolant heat sources such as reactor coolant pumps and pressurizer heaters, of 2570 MWt.

Millstone Unit 2 is currently licensed for a steady state reactor core power level of 2700 MWt, corresponding to a NSSS rating of 2715 MWt. All Chapter 14 analyses have been evaluated on the basis of these current values.

Since the construction permit was issued, and during the design and construction of the unit, there have been no major deviations from the information supplied in the Preliminary Safety Analysis Report (PSAR). However, changes in various specific design features have been found desirable and these are covered in the appropriate sections of this report. A summary of the more significant design changes incorporated in the plant since the issuance of the PSAR up to the time of application for an operating license is provided in Section 1.7.

TABLE 1.1-1 LICENSING HISTORY

EVENT	DATE
Construction Permit Issued	December 11, 1970
Final Safety Analysis Report Filed	August 15, 1972
Full Term Operating Licensing Issued	September 26, 1975
Full Power License	September 26, 1975
Initial Criticality	October 17, 1975
100% Power	March 20, 1976
Commercial Operation	December 26, 1975
“Stretch Power”	June 25, 1979
Operating License Extension Requested	December 22, 1986
Operating License Extension Issued	January 12, 1988
Full Term Operating License Expires	December 11, 2010
Operating License Expires	July 31, 2035

1.2 SUMMARY DESCRIPTION

1.2.1 GENERAL

A summary description of Millstone Unit 2 of the Millstone Nuclear Power Station is provided in this section. The description includes the following:

a. Site

b. Arrangement

c. Reactor

d. Reactor coolant system

e. Containment system

f. Engineered safety features systems

g. Protection, control and instrumentation system

h. Electrical systems

i. Auxiliary systems

j. Steam and power conversion system

k. Radioactive waste processing system

l. Interrelation with Millstone Units 1 and 3

m. Summary of Codes and Standards

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

The containment houses the NSSS, consisting of the reactor, steam generators, reactor coolant pumps, pressurizer, and some of the reactor auxiliaries. The containment is equipped with a polar crane.

The enclosure building completely envelopes the containment and provides a filtration region between the containment and the environment.

The turbine building houses the turbine generator, condenser, feedwater heaters, condensate and feedwater pumps, turbine auxiliaries and certain of the switchgear assemblies.

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

1.2.4 REACTOR

The reactor is a pressurized light water cooled and moderated type fueled by slightly enriched uranium dioxide. The uranium dioxide is in the form of pellets and is contained in pressurized Zircaloy-4 tubes fitted with welded end caps. These rods are arranged into fuel assemblies each consisting of 176 fuel rods arranged on a 14 rod square matrix. Space is left in the fuel rod array to allow for the installation of five guide tubes. These guide tubes provide for the smooth motion of control element assembly fingers. The assembly is fitted with end fittings and spacer grids to maintain fuel rod alignment and to provide structural support. The end fittings are also drilled with flow holes to provide for the flow of cooling water past the fuel tubes.

The reactor is controlled by a combination of chemical shim and solid absorber. The solid absorber is boron carbide pellets or stainless steel contained in tubular Inconel elements. Some earlier elements had used stainless steel as the absorber material. Five absorber elements are connected together by a spider yoke in a square matrix with a center element. The five elements constitute a control element assembly (CEA). The 73 CEAs are connected, either singly or dually, through extension shafts, to 61 magnetic jack type control element drive mechanisms (CEDMs) which are mounted on nozzles on the reactor vessel head. Each CEA is aligned with and can be inserted into the guide tubes of fuel assemblies. The dual CEAs are utilized for shutdown rods. The single CEAs are divided into regulating groups. The eight part length control rods of Cycle One were replaced by dummy flow plugs. Two of the flow plugs were replaced by reactor vessel level indication system detectors, then in Cycle Twelve, the last six remaining flow plugs were removed. The resulting increase in core bypass flow has been accounted for in the safety analysis.

The replacement head has a total of 78 nozzle penetrations. 67 of these nozzles are suitable for supporting control element drive mechanisms (61 are in use, while the other 6 nozzles are capped with nozzle adapters). Two nozzles are used for heated junction thermocouples, which enable monitoring reactor vessel between the top of the vessel dome and the area directly above the fuel bundles. Eight nozzles are used for nuclear instrumentation and one nozzle is used for the reactor vessel head vent. The location, size and the number of nozzles on the replacement reactor vessel closure head are maintained in the same configuration as before (prior to cycle 16).

Chemical shim control is provided by boric acid dissolved in the coolant water. The concentration of boric acid is maintained and controlled as required by the chemical and volume control system.

The reactor core rests on the core support plate assembly which is supported by the core support barrel. The core support barrel is a right circular cylinder supported from a machined ledge on the inside surface of the vessel flange forging. The support plate assembly transmits the entire weight of the core to the core support barrel through a structure made of beams and vertical columns. Surrounding the core is a shroud which serves to limit the coolant which bypasses the core. An upper guide structure, consisting of upper support structure, control element assembly shrouds, a fuel alignment plate and a spacer ring, serves to support and align the upper ends of the fuel assemblies, prevents lifting of the fuel assemblies in the event of a loss-of-coolant accident (LOCA) and maintains spacing of the CEAs. Chapter 3 contains more detailed information on the reactor.

1.2.5 REACTOR COOLANT SYSTEM

The reactor coolant system consists of two closed heat transfer loops in parallel with the reactor vessel. Each loop contains one steam generator and two pumps to circulate coolant. An electrically heated pressurizer is connected to one loop hot leg. The coolant system is designed to operate at a thermal power level of 2715 MWt to produce steam at a nominal pressure of 880 psia.

The reactor vessel, loop piping, pressurizer and steam generator plenums are fabricated of low alloy steel, clad internally with austenitic stainless steel. The pressurizer surge line and coolant pumps are fabricated from stainless steel and the steam generator tubes are fabricated from Inconel.

Overpressure protection is provided by power-operated relief valves and spring-loaded safety valves connected to the pressurizer. Safety and relief valve discharge is released under water in the quench tank where the steam discharge is condensed.

The two steam generators are vertical shell and U-tube steam generators each of which produces 5.9×10^6 lb/hr of steam. Steam is generated in the shell side of the steam generator and flows upward through moisture separators. Steam outlet moisture content is less than 0.2 percent.

The reactor coolant is circulated by four electric motor-driven, single-suction, centrifugal pumps. Each pump motor is equipped with a non reverse mechanism to prevent reverse rotation of any pump that is not being used during operation with less than four pumps energized. Chapter 4 contains more detailed information on the reactor coolant system.

1.2.6 CONTAINMENT SYSTEM

A double containment system is used for Unit 2. The containment system consists of a prestressed concrete cylindrical structure referred to as the containment, which is completely enclosed by the enclosure building (EB). The enclosure building filtration region (EBFR) includes the region between the containment and the enclosure building, the penetration rooms and engineered safety feature equipment rooms. In the unlikely event of a LOCA the EBFR is maintained at a slightly negative pressure by the enclosure building filtration system (EBFS). Air in the EBFR would be processed through charcoal filters and released through the 375 foot Millstone stack during a LOCA.

The containment uses a prestressed post-tensioned concrete design. The containment is a vertical right cylindrical structure with a dome and a flat base. The interior is lined with carbon steel plate to further ensure leak tightness.

Inside the containment, the reactor and other NSSS components are shielded with concrete. Access to portions of the containment during power operation is permissible.

The containment, in conjunction with the engineered safety features, is designed to withstand the highest internal pressure and coincident temperature resulting from the main steam line break accident (Section 14.8.2). The structural design conditions are for an internal pressure of 54 psig and a coincident equilibrium temperature of 289°F.

The enclosure building is a limited leakage steel framed structure partially supported off the containment and auxiliary building with uninsulated metal siding and an insulated metal roof deck.

1.2.7 ENGINEERED SAFETY FEATURES SYSTEMS

The engineered safety features systems (ESFS) provide protection for the public and plant personnel against the incidental release of radioactive products from the reactor system, particularly as a result of postulated LOCA. These safety features localize, control, mitigate and

terminate such accidents to hold exposure levels below the applicable limits of 10 CFR Part 50.67.

The engineered safety features consist of the following systems:

a. Safety injection

b. Containment spray

c. Containment air recirculation and cooling

d. Enclosure building filtration

e. Hydrogen control

f. Auxiliary feedwater automatic initiation system

Each of these systems is divided into two redundant independent subsystems which in turn are powered by the associated redundant independent emergency electrical subsystem (see Section 1.2.9). The first three are cooled by the associated redundant independent reactor building closed cooling water headers (see Section 1.2.10.3).

Following a postulated LOCA, borated water is injected into the reactor coolant system by either high and/or low pressure safety injection pumps and safety injection tanks. This provides cooling to limit core damage and fission product release, and assures an adequate shutdown margin. The safety injection system also provides continuous long term post-accident cooling of the core by recirculating borated water from the containment sump through shutdown cooling heat exchangers and back to the reactor core (see Section 6.2).

Four safety injection tanks are provided, each connected to one of the four reactor inlet lines. The volume of each tank is 2019 cubic feet. Each tank contains about 1100 cubic feet of borated water at refueling concentration and is pressurized with nitrogen at 200 psig. In the event of a LOCA, the borated water is forced into the reactor coolant system by the expansion of the nitrogen. The water from three tanks adequately cools the entire core. Borated water is injected into the same nozzles by two low pressure and three high pressure injection pumps taking suction from the refueling water storage tank (RWST). For maximum reliability, the design capacity from the combined operation one high pressure and one low pressure pump provides adequate injection flow for any LOCA; in the event of a design basis accident (DBA), at least one high pressure and one low pressure pump will receive power from the emergency power sources if preferred power is lost and one of the emergency diesel generators is assumed to fail. When the refueling water storage tank supply is nearly depleted, the high pressure pump suctions automatically transfer to the containment sump and the low pressure pumps are shut down. One high pressure pump has sufficient capacity to cool the core adequately at the start of recirculation. During recirculation, heat in the recirculating water is removed through the shutdown cooling heat exchangers via either the low pressure injection pumps or containment spray pumps.

The safety injection pumps are located outside the containment to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during plant operation.

The safety injection system is designed in accordance with AEC General Design Criteria 35, 36, and 37 in Appendix A to 10CFR50 and General Criteria as described in Section 6.1. An analysis of the performance of the safety injection system (emergency core cooling system) following a postulated LOCA is given in Section 14.6.

Two independent, full capacity systems are provided to remove heat from the containment atmosphere by containment sprays and/or air recirculation and cooling after the postulated LOCA.

a. The containment spray system supplies borated water to cool the containment atmosphere. The spray system is sized to provide adequate cooling with two containment spray pumps. The pumps take suction from the refueling water storage tank. When this supply is nearly depleted, the pump suction is transferred automatically to the containment sump (see Section 6.4).

b. The containment air recirculation and cooling system is designed to cool the containment atmosphere. The cooling coils and fans are sized to provide adequate containment cooling with three of the four units in service (see Section 6.5).

c. A combination of one containment spray pump aligned with the shutdown cooling heat exchanger and two containment air recirculation units provides adequate cooling of the containment. Each spray pump and two associated containment air recirculation units are cooled by one of two associated redundant reactor building cooling water and service water subsystems. They are powered by the associated emergency electrical subsystem.

The enclosure building filtration system would collect and filter all potential containment leakage and minimize environmental radioactivity levels resulting from the discharge of all sources of containment leakage into the enclosure building filtration region in the unlikely event of a LOCA. The enclosure building filtration system would also collect and filter any radioactive releases in the unlikely event of a fuel handling accident inside the containment or spent fuel pool areas (see Section 6.7).

The hydrogen control system is provided to mix and monitor the concentration of hydrogen gas within the containment. This system consists of the post-accident recirculation system for mixing the containment environment and the hydrogen monitoring system for continuous monitoring of the post-accident containment atmosphere. The hydrogen purge system and hydrogen recombiners which are not credited in accident analyses are provided for reducing containment hydrogen concentrations.

The auxiliary feedwater automatic initiation system, (AFAIS), is provided to ensure delivery of sufficient feedwater to the steam generators in event of the loss of main feedwater. This system automatically actuates two motor driven auxiliary feedwater pumps (see Section 10.4.5.3), and opens the two auxiliary feedwater flow control valves via the automatic initiation control circuitry (see Section 7.3.2.2.h). The AFAIS is actuated upon completion of a 2-out-of-4 logic matrix initiated by a low steam generator level. Upon receipt of an actuation signal both pumps are started and the flow control valves to both steam generators are opened (see Section 7.3).

1.2.8 PROTECTION, CONTROL AND MONITORING INSTRUMENTATION

Various instrumentation systems provide protection, control, and monitoring functions for the safe and efficient operation of Millstone Unit 2.

Protection instrumentation systems function to shut down the reactor and activate safety systems if continuously monitored key plant process parameters exceed predetermined limits. Specific protection instrumentation systems include the Reactor Protective System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The RPS functions to shut down or trip the reactor if any two of four safety channels generate coincident trip signals. An RPS trip removes power from the reactor control rods, allowing them to drop into the reactor, and shut it down. The ESFAS functions to actuate the engineered safety features systems described in FSAR Section 1.2.7. The exception to this is the containment purge valve isolation where one of four containment air radiation detectors can generate a trip signal. Actuation of the ESFS occurs if any two of four safety channels generate coincident trip signals.

Control instrumentation systems function to maintain plant parameters within operational limits during both steady state and normal operating transients. Major control systems include the Control Element Drive System (CEDS), the Reactor Regulating System (RRS), Pressurizer Level Regulating System (PLRS), Reactor Coolant Pressure Regulating System (RCPRS), Feed Water Regulating System (FWRS), and Turbine Generator Control System (TGCS).

Indications are provided to monitor normal and abnormal plant operation. Indicators are located within the control room and throughout the plant. The indicators are used to monitor the status and operation of the protective and control systems, and the status of other support systems. Major indication systems include the Control Element Assembly (CEA) Position Indication, Nuclear Instrumentation (NI), In-Core Instrumentation (ICI), Radioactivity Monitoring System (RMS), Integrated Computer System (ICS), Control Room Annunciation, and Post Accident Monitoring Instrumentation (PAMI).

Details of the above and other protective, control, and monitoring instrumentation systems are provided in Chapter 7.

1.2.9 ELECTRICAL SYSTEMS

The Millstone Nuclear Power Station consists of Millstone Unit 1 which is no longer generating power, Millstone Unit 2 with a 1011-MVA, 0.90 power factor generator, and Millstone Unit 3 with a 1354.7-MVA, 0.925 power factor generator (see Chapter 8).

The Millstone Unit 2 generator output is fed through a step up transformer bank to the 345 kV switchyard. The switchyard is connected to the high voltage transmission system through four 345 kV transmission lines. The switchyard, in addition to carrying the electrical output of the station, also provides a means of supplying power to the units from external sources. Startup power and reserve auxiliary power for Millstone Unit 2 are taken from the 345 kV switchyard through the reserve station service transformer. Normal station service power is taken from the generator main leads through the normal station service transformer. A second source of off site power for the engineered safety features is provided from normal station service transformer 15G-3SA or reserve station service transformer 15G-23SA, both associated with Millstone Unit 3 via a 4160V crosstie connection. Two diesel generators provide the on site emergency power for Millstone Unit 2. The 4160V crosstie from Unit 3 can also be configured (by operator action) to supply power directly from the Unit 3 Alternate AC (SBO) diesel generator to provide an alternate AC source for Unit 2 Appendix R and Station Blackout requirements.

Auxiliary power for Millstone Unit 2 is provided at 6900, 4160, 480, and 120/208 volts. Direct current 125 volt systems are also available for emergency power, engineered safety feature control, and essential nuclear instrumentation, control and relaying.

The preferred and on site emergency sources of electrical power are each adequate to permit prompt shutdown and maintain safe conditions under all credible circumstances. The on site emergency power source consists of two separate and redundant diesel generators. Each diesel is capable of carrying all required auxiliary loads following postulated LOCA without exceeding its continuous rating.

Each of the two separate and redundant station batteries is capable of carrying essential 125 volt DC and 120 volt AC inverter loads associated with a postulated LOCA.

The redundant channel wiring associated with these emergency electrical sources is physically separated.

1.2.10 AUXILIARY SYSTEMS

1.2.10.1 Chemical and Volume Control System

The chemistry of the reactor coolant is controlled by purification of a regulated letdown stream of reactor coolant. Water removed from the reactor coolant system is cooled in the regenerative heat exchanger. The fluid pressure is then reduced and flow is regulated by the letdown control valves. Temperature is reduced further in the letdown heat exchanger. From there, the flow passes through a filter and a purification ion exchanger to remove corrosion and fission products. A small fraction of the flow is diverted prior to entering the ion exchanger. This stream of coolant flows through a process radiation monitor. Upon leaving the ion exchanger, the coolant flows through a strainer and another filter and is then sprayed into the volume control tank.

Coolant is returned to the reactor coolant system by the charging pumps, through the regenerative heat exchanger. Prior to entering the charging pumps, the coolant boron concentration is adjusted

to meet the reactor reactivity requirements. In addition, provision is made to inject chemical additives to the suction of the charging pumps for coolant chemistry control.

The volume control system automatically controls the rate at which coolant must be removed from the reactor coolant system to maintain the pressurizer level within the prescribed control band, thereby compensating for changes in volume due to coolant temperature changes. Using the volume control tank as a surge tank decreases the quantity of liquid and gaseous wastes which would otherwise be generated.

Reactor coolant system makeup water is taken from the primary water storage tank and the two concentrated boric acid storage tanks. The boric acid solution is maintained at a temperature which prevents crystallization. The makeup water is pumped through the regenerative heat exchanger into the reactor coolant loop by the charging pumps.

Boron concentration in the reactor coolant system can be reduced by diverting the letdown flow away from the volume control tank to the radioactive waste processing system. Demineralized water is then used for makeup.

When the boron concentration in the reactor coolant system is low, the feed and bleed procedure previously described would generate excessive volumes of waste to be processed. Therefore, the chemical and volume control system is equipped with a deborating ion exchanger which reduces boron concentration late in cycle life. A complete description is given in Section 9.2.

1.2.10.2 Shutdown Cooling System

The shutdown cooling system (see Section 9.3) is used to reduce the reactor coolant temperature, at a controlled rate, from 300°F to a refueling temperature of approximately 130°F. It also maintains the proper reactor coolant temperature during refueling. Once entry conditions are met, the shutdown cooling system can provide long term cooling capability in the event of a LOCA after the reactor coolant system has refilled (see Section 14.6.5.3).

The shutdown cooling system utilizes the low pressure safety injection pumps to circulate the reactor coolant through two shutdown cooling heat exchangers. It is returned to the reactor coolant system through the low pressure safety injection header.

The reactor building closed cooling water system (RBCCW) supplies cooling water for the shutdown heat exchangers.

1.2.10.3 Reactor Building Closed Cooling Water System

The RBCCW system consists of two separate independent headers, each of which includes a RBCCW pump, a service water (seawater)-cooled RBCCW heat exchanger, interconnecting piping, valves and controls. A third RBCCW pump and a third RBCCW heat exchanger are provided as installed spares. The corrosion inhibited, demineralized water in this closed system is circulated through the RBCCW heat exchanger where it is cooled to 85°F by seawater which has a maximum design inlet temperature of 80°F (see Section 9.4).

The RBCCW system removes heat from the containment atmosphere, engineered safety feature components and various auxiliary system/components handling the reactor coolant. Items cooled by the RBCCW system include:

Containment air recirculation and cooling unit
Reactor vessel support concrete cooling coils
Containment spray pump seal coolers
High and low pressure safety injection pump seal coolers
Shutdown cooling heat exchangers
Engineered safety feature room air recirculation coils
Reactor coolant pump thermal barrier and oil coolers
Primary drain and quench tanks heat exchanger
CEDM coolers
Letdown heat exchanger
Degasifier effluent cooler
Degasifier vent condenser
Sample coolers
Spent fuel pool heat exchangers
Waste gas compressor aftercoolers
Steam generator blowdown quench heat exchanger

Each of the independent headers supply cooling water to components in the associated redundant safety related sub-systems (see Section 1.2.7). The RBCCW heat exchangers, connected to each independent RBCCW headers, are cooled by the associated independent service water header (see Section 1.2.10.6). Components in each independent RBCCW header, the associated safety related subsystems, and the associated service water header are powered from the associated redundant independent emergency electrical power subsystem (see Section 1.2.9).

Remote manually operated valves allow the spare RBCCW pump and/or heat exchanger to be operated with either of the two independent headers. The RBCCW surge tank absorbs the volumetric changes caused by temperature changes of the water within the RBCCW headers.

A chemical addition system is provided for the RBCCW system to maintain the corrosion inhibitor concentration as required.

During normal plant operation and normal shutdown, both of the independent RBCCW headers are in service.

Following a postulated LOCA, each of the RBCCW headers, in conjunction with the associated service water header and electrical subsystem, would provide the necessary cooling capacity to the associated engineered safety feature subsystems.

1.2.10.4 Fuel Handling and Storage

The fuel handling systems provide for the safe handling of fuel assemblies and control element assemblies and for the required assembly, disassembly, and storage of the reactor vessel head and

internals. These systems include a refueling machine located inside the containment above the refueling pool, the fuel transfer carriage, the upending machines, the fuel transfer tube, a fuel handling machine over the spent fuel pool, a new fuel elevator in the spent fuel pool, a spent fuel cask crane, a new fuel inspection machine in the fuel handling area of the auxiliary building, and various devices used for handling the reactor vessel head and internals (see Section 9.8).

New fuel is stored dry in vertical racks within a storage vault near the spent fuel pool in the auxiliary building. Storage space is provided for approximately one-third of a core.

The vault is designed to avoid criticality by spacing fuel assemblies at 20.5 inches, center to center. The spent fuel pool, located in the auxiliary building, is constructed of reinforced concrete lined with stainless steel. The spent fuel storage racks are separated into four regions, designated Regions 1, 2, 3, and 4. Section 9.8.2.1 contains a detailed description of spent fuel storage design and components.

Cooling and purification equipment is provided for the spent fuel pool water (see Section 9.5). This equipment can also be used to clean up the refueling water during and after its use in the refueling pool. Backup cooling methods are also available.

1.2.10.5 Sampling System

The sampling system consists of Sampling Stations 1 and 2, the Post Accident Sampling System (PASS), the Corrosion Monitoring Sample Station, and the Waste Gas Sample Sink. These provide the means for determining physical, chemical and radioactive conditions of process fluids, waste gas and containment air. The system is supplemented by independent sampling of nonradioactive fluids in numerous locations within the unit, including sampling of the chlorinated water. (See Section 9.6.)

1.2.10.6 Cooling Water Systems

The exhaust steam from the main turbine and steam generator feedwater pump turbines is condensed in the condenser, which is cooled, in turn, by circulating water flowing through the condenser tubes, (see Section 9.7.1).

Four circulating water pumps, with 548,800 gpm total capacity, take suction from and discharge to Long Island Sound. The circulating water system is designed to maintain condenser back pressure at 2 inches Hg absolute with a 60.8°F inlet circulating water temperature.

The service water system (see Section 9.7.2) provides cooling water to the RBCCW, TBCCW, diesel engine cooling water, chilled water system heat exchangers, vital switchgear room cooling coils and the circulating water pump bearings. Three vertical, centrifugal, half capacity service water pumps have a design flow of 12,000 gpm, each with a total dynamic head of 100 feet of water. These pumps take suction from and discharge to Long Island Sound.

The service water system consists of two redundant, independent cross-connected supply headers with isolation valves to all heat exchangers and two discharge headers for the RBCCW heat

exchangers. Two discharge headers exist for the emergency diesel generator cooling water; once underground these headers combine prior to entering the discharge canal. Service Water discharge from the TBCCW, chilled water system and vital switchgear room cooling heat exchangers combine into a common header prior to entering the discharge canal. Each of the supply headers is supplied by one of the service water pumps. During normal operation and shutdown and following a postulated LOCA, the two pumps connected to the two redundant supply headers are in service. However, only one service water pump and header is required to provide cooling of the RBCCW and diesel following a LOCA or for unit shutdown. Remote manually operated valves allow the third service water pump to be connected to either of the redundant headers.

The intake structure consists of four independent bays. The intake structure is equipped with a chlorination system, consisting of two 1800 gallon sodium hypochlorite storage tanks and two injection systems with one supplying sodium hypochlorite to the service water system and the other to the circulating water intake.

1.2.10.7 Ventilation Systems

Normally the containment environment is cooled by the containment air recirculation and cooling system. Following a postulated LOCA, these units reduce the temperature and pressure of the containment atmosphere to a safe level (see Sections 1.2.7, 6.5 and 9.9.1). The containment auxiliary circulation fans maintain uniform containment environmental temperature by mixing the air. Normally, the environment for the control element drive mechanisms is maintain by the CEDM fan-coil units. A forced outside air purge system is provided to maintain a suitable environment within the containment whenever access is desired. The exhaust of this containment air purge system is monitored to assure that radioactive effluents are maintained within acceptable limits.

The auxiliary building is served by separate ventilation systems in the fuel handling area, the radioactive waste area and for the nonradioactive waste area. Each area is provided with a heating and ventilating supply unit and separate exhaust fans. Exhausts from the potentially contaminated areas are filtered through high efficiency particulate air (HEPA) filters, monitored, and discharged through the Unit 2 stack. Exhaust from clean areas is discharged directly to the atmosphere (see Section 9.9.6).

Handling of irradiated fuel or moving a cask over the spent fuel pool does not require fuel handling area integrity or ventilation but it may be desirable to use the main exhaust or EBF systems, if available, as the exhaust discharge paths. If boundary integrity is set then these discharge paths provide a monitored radiological release pathway. If boundary integrity is not assured then suitable radiological monitoring is recommended per the Millstone Effluent Control Program.

The ventilation systems (main exhaust and EBFS) are normally available to provide for a filtered and monitored release pathway for effluents from the fuel handling area. If ventilation is not available, releases from the fuel handling area are monitored per the Millstone Effluent Control Program to ensure appropriate radiological effluent controls are in place.

Two full capacity and redundant air conditioning systems are provided for the control room. In the event of an accident, a bypass through either of the two full capacity and redundant control room filtration systems, which contain charcoal filters, is provided to protect control room operating personnel from exposure to high radiation levels.

The turbine building is equipped with supply and exhaust fans for year round ventilation.

The access control area is air conditioned for year-round comfort. All other areas are provided with ventilation for cooling during summer and unit heaters for heating during the winter.

1.2.10.8 Fire Protection System

The fire protection systems' (see Section 9.10) function to protect personnel, structures, and equipment from fire and smoke. The fire protection systems have been designed in accordance with the applicable National Fire Protection Association (NFPA) Codes and Standards, regulatory requirements, industry standards, and approved procedures. The design of the various fire protection systems has been reviewed by American Nuclear Insurers (ANI).

The fire detection and protection systems are designed such that a fire will be detected, contained, and/or extinguished. This is accomplished through the use of noncombustible construction, equipment separation, fire walls, stops and seals, fire detection systems, and automatic and manual water suppression systems. As a minimum, portable extinguishers, hose stations, and fire hydrants are available for all areas to control or extinguish a fire.

1.2.10.9 Compressed Air Systems

The instrument air system consists of one 640 scfm and two 237 scfm (each) instrument air compressors, receivers, dryers, and after-filters to provide a reliable supply of clean, oil free dry air for the unit pneumatic instrumentation and valves. Station air for normal unit maintenance is provided by a separate 630 scfm station air compressor. Operating pressures for both systems range between 80 to 120 psig depending on how the compressors are aligned and how the systems are interconnected.

The station air is used as a backup to the instrument air with tie-in points at the receiver inlets and inside the containment. The compressed air systems for Units 3 and 2 are interconnected by piping and manually operated valves.

Descriptions of the compressed air systems are given in Section 9.11.

1.2.11 STEAM AND POWER CONVERSION SYSTEM

The turbine generator for Unit 2 is furnished by General Electric Company. It is an 1800 rpm tandem compound, four flow exhaust, indoor unit designed for saturated steam conditions.

Under nominal steam conditions of 870 psia and 528°F at the turbine stop valve inlets and with turbines exhausting against a condenser pressure of 2 inches Hg absolute, the gross electrical

output is 935 MWe. Turbine output corresponds to a NSSS thermal power level of approximately 2715 MWt.

The condensate and feedwater system consists of three condensate pumps, one steam packing exhauster, two steam jet air ejectors, two external drain coolers, two trains each having five stages of low pressure feedwater heaters, two turbine-driven steam generator feedwater pumps, two high pressure feedwater heaters as well as the associated piping, valves and instrumentation.

Normally, the steam generator feedwater pump turbines are driven by extraction steam. At low loads, main steam is used to drive the steam generator feedwater pump turbines.

A complete description of the steam and power conversion system is given in Section 10.

1.2.12 RADIOACTIVE WASTE PROCESSING SYSTEM

The radioactive waste processing system provides controlled handling and disposal of liquid, gaseous and solid waste from Unit 2 (see Section 11.1). Gaseous and liquid wastes discharged to the environment are controlled to comply with the limits given in the Technical Specifications and established to meet the requirements of 10 CFR Part 20 Sections 1301 and 1302 and Appendix B and the "as low as reasonably achievable (ALARA)" requirement of 10 CFR Part 50, Appendix I. The radioactive waste processing system consists of the following parts.

a. Clean Liquid Waste Processing System

   The clean liquid waste processing system collects and processes reactor coolant wastes from the chemical and volume control system, primary drain tank and the closed drains system. The system is comprised of pumps, filters, degasifier, demineralizers, receiver tanks, monitor tanks and the necessary instrumentation, piping, controls and accessories.

   The processed clean liquid wastes are collected in monitor tanks, sampled, and monitored prior to discharge to the circulating water system after ensuring that the predetermined limits for release are not exceeded.

b. Aerated Liquid Waste Processing System

   Aerated liquid wastes, consisting of radioactive liquid wastes exposed to the atmosphere, are collected in drain tanks and processed through filters, and demineralizers. The processed wastes are collected in a monitor tank, sampled, and monitored prior to discharge to the circulating water system after ensuring that the predetermined limits for release are not exceeded.

c. Gaseous Waste Processing System

   Radioactive waste gases are collected through the waste gas header into the waste gas surge tank. These gases are drawn from the surge tank by one of two

compressors and are pumped into a waste gas decay tank for storage to allow radioactive decay. After decay, the tank contents are sampled and monitored prior to discharge and released through a particulate filter, at a predetermined controlled rate, into the Millstone stack. The discharge is monitored prior to its entering the stack and while in the stack, thus ensuring that the predetermined limits for release are not exceeded. The six waste gas decay tanks which are provided allow a minimum of 60 days storage capacity prior to release.

d. Solid Waste Processing System

Radioactive solid wastes are collected and placed in suitable containers for off site disposal. Spent demineralizer resins are held for radioactive decay prior to being dewatered and placed in a shielded cask for removal. Contaminated filter elements are placed in shielded drums for subsequent storage and off site disposal.

Low activity compactible solid wastes such as contaminated rags, paper, etc., are compacted at the Millstone Radwaste Reduction Facility prior to being shipped for disposal. Noncompactible solid wastes may be shipped to an off site processor for volume reduction prior to disposal.

1.2.13 INTERRELATION WITH MILLSTONE UNITS 1 AND 3

A number of the facilities of the Millstone Nuclear Power Station are common to Millstone Units 1, 2, and 3. The safe shutdown of any unit will not be impaired by the failure of the facilities and systems which are shared. A list of these facilities and systems follows:

a. Facilities

Radiochemistry laboratory
Radioactive and clean change facilities, including showers, lockers, clothing storage, and toilets
Radiation Protection offices
Instrument repair room
Warehouse machine shop
Millstone stack (for Unit 2 waste gas), main condenser air ejector and enclosure building filtration system discharge
General offices
First aid station
Lunch room
Visitors gallery
345 kV switch yard
Millstone Unit 3 normal station service transformer 15G-3SA
Millstone Unit 3 reserve station service transformer 15G-23SA
Millstone Unit 3 SBO diesel generator system
Makeup water treatment (Millstone Units 2 and 3 only)
Bulk storage chemical ton (Millstone Units 2 and 3 only)

Millstone Unit 2 Control Room (for monitoring and controlling Millstone Unit 1 systems)

b. Systems

Low pressure nitrogen storage
Fire protection (water supply and fire detection)
Auxiliary steam
Makeup water treatment
Building heating
Sanitary sewers
Plant water
Communications
Station Air (A system cross-tie between Unit 3 service air and Unit 2 station air headers is provided)

Operating and maintenance personnel are employed in all three units as described in Section 12.1.

Both units have a double containment system with rectangular outer envelopes.

The 40 CFR 190 off site radiation dose limits will not be exceeded by simultaneous operation of Millstone Units 1, 2, and 3.

The Millstone Station Physical Security Plan has been implemented in accordance with 10 CFR 73.55 “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Industrial Sabotage” to prohibit unauthorized access to vital areas.

This plan includes measures to deter or prevent malicious actions that could result in the release of radioactive materials into the environment through sabotage. Section 12.7 contains a description of the Security Plan.

1.2.14 SUMMARY OF CODES AND STANDARDS

To ensure the integrity and operability of pressure-containing components important to safety, established codes and standards are used in the design, fabrication and testing. Table 1.2-1 lists these codes and standards for components relied upon to prevent or mitigate the consequences of incidents and malfunctions originating within the reactor coolant pressure boundary, to permit shutdown of the reactor, and to maintain the reactor in a safe shutdown condition.

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS [1]

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A, 1968 Edition, Addenda through Summer 1969	ASME Boiler and Pressure Vessel Code, Section III, Class C (1968 Edition including Addenda through Summer 1969)	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or Equivalent
	Reactor Vessel [2]	Safety Injection Tanks [4]	Reactor Building Closed Cooling Water Heat Exchangers [3]	Service Water Strainers [3]
	Pressurizer [2]		Reactor Building Closed Cooling Water Surge Tank	Vital Chilled Water System Condensers/ Evaporators
	Steam Generators [3]	Shutdown Heat Exchangers [2] Concentrated Boric Acid Storage Tanks [2]		
		Refueling Water Storage Tank [4]		
0-15 psig Storage Tanks	—	API-620 with the NDT Examination Requirements in Table NST-1, Class 2	API-620 with the NDT Examination Requirements in Table NST-1, Class 3	API-620 or Equivalent Condenser Storage Tank
Atmospheric Storage Tanks	—	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B96.1 With the NDT Examination Requirements in Table NST-1, Class 2	Applicable Storage Tank Codes Such as API-650 AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 3	API-650, AWWAD100 or ANSI B 96.1 or Equivalent Diesel Oil Supply Tanks

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS [1] (CONTINUED)

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Pumps and Valves	1. ASME Standard Code for Pumps and Valves for Nuclear Power, Class 1, March 1970 Draft	Draft ASME Code for Pumps and Valves, Class II, November 1968. See Footnote [5].	Draft ASME Code for Pumps and Valves Class III	Valves - ANSI B 31.1.0 or Equivalent Pumps - Draft ASME Code for Pumps and Valves Class III or Equivalent
	2. ASME Section III, Paragraph N153 in Summer 1969 Addenda			
	3. ASME Section III, Appendix IX Reactor Coolant Pumps and Valves	High Pressure Safety Injection Pumps and Valves	Vital Chilled Water Pump	Vital Chilled Water Valves
		Low Pressure Safety Injection Pumps and Valves ASME Section III 1971 Edition, 1971 Winter Addenda Reactor Coolant System Branch Connection Valves beyond Second Isolation Valves ASME Standard Code for Pumps and Valves, Class 2, March 1970 draft All Containment Penetration Isolation Valves ASME Section III, 1971; Draft ASME Pump and Valve Code, 1980, 1983		Service Water Pumps and Valves Standards of the Hydraulic Institute, ANSI G16.5 Class 1 RBCCW Pumps and Valves Standards of the Hydraulic Institute, ANSI B16.1, ANSI B31.1 Auxiliary Feedwater Pumps ASME Code for Pumps and Valves for Nuclear Power, Class II NEMA Standard SM20-1958 Hydraulic Institute
		Chemical and Volume Control System-Concentrated Boric Acid Service-Pumps Acid Service-Pumps and Valves Draft ASME Code for Pump and Valves, Class II, November 1968		
		Containment Spray Pumps and Valves		
Pressurizer Safety Valves	1. ASME Section III, Class A, 1968 Edition, Addenda through summer of 1970. Code Case 1344-1.			
Pressurizer Power Operated Relief Valves	ASME Section III Class 1, 1977 Edition through winter 1979 Addenda			

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS (1) (CONTINUED)

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Piping	1. ANSI B 31.7 Class I, 1969 Edition	ANSI B 31.7, Class II 1969 Edition	ANSI B 31.7, Class III 1969 Edition	ANSI B 31.1.0 or Equivalent
	2. ASME Section III, Paragraph N153 in Summer 1969 Addenda 3. Code Case 70 to B31.7			
	Primary Coolant Piping and Surge Line	High Pressure Safety Injection Piping		
		Low Pressure Safety Injection Piping		
	4. Other Reactor Coolant Pressure Pressure Boundary Class I Piping–ASME Section III Code - 1971 Edition, Class I.	Reactor Coolant System Branch Piping beyond Second Isolation Valves Chemical and Volume Control System Concentrated Boric Acid Service Piping ANSI B31.1.0 modified (inside Containment) Containment Spray Piping		Service Water Piping RBCCW Piping
		All Containment Piping Penetrations		
		1. ANSI B-31.1, Piping Code, ANSI B31.7 Nuclear Piping Code, Class I or II as a minimum, 1969 Edition.		
		3. ASME Section III, Class 1 or 2, 1971 Edition		

1 This table summarizes the Codes and Standards used for major pressure retaining components. Not all components are listed. Later codes and standards may be employed for plant modifications if permitted by applicable design and regulatory requirements in effect at the time of the modification.

2 The reactor vessel head and the replacement pressurizer are constructed in accordance with ASME Boiler & Pressure Vessel Code, Section III, Subsection NB 1998 Edition, through 2000 Addenda.

3 Including ASME Code Case N-416.

4 1971 ASME Boiler and Pressure Code, Section III, Class 3.

5 All pressure-retaining cast parts shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified in the applicable class in the code.

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–1 SITE LAYOUT



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–2 PLOT PLAN



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–3 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT OPERATING FLOOR ELEVATION 54 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–4 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT MEZZANINE FLOOR ELEVATION 31 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–5 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT GROUND FLOOR ELEVATION 14 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–6 GENERAL ARRANGEMENT CONTAINMENT PLAN AT FLOOR ELEVATION 14 FEET 6 INCHES AND ELEVATION 36 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–7 GENERAL ARRANGEMENT AUXILIARY BUILDING PLAN AT ELEVATION 36 FEET 6 INCHES AND ELEVATION 38 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–8 GENERAL ARRANGEMENT AUXILIARY BUILDING SECTIONS "G-G" AND "H-H"



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–9 GENERAL ARRANGEMENT AUXILIARY BUILDING GROUND FLOOR ELEVATION 14 FEET 6 INCHES AND CABLE VAULT ELEVATION 25 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–10 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)5 FEET 0 INCHES AND ELEVATION (-)3 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–11 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)25 FEET 6 INCHES AND ELEVATION (-)22 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–12 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)45 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–13 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING SECTION "A-A"



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–14 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING SECTION "B-B"



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–15 GENERAL ARRANGEMENT TURBINE BUILDING SECTIONS "C-C" AND "E-E"



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–16 GENERAL ARRANGEMENT TURBINE BUILDING SECTIONS "D-D" AND "F-F"



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–17 GENERAL ARRANGEMENT INTAKE STRUCTURE AUXILIARY STEAM BOILER ROOM PLAN AND SECTION



1.3 COMPARISON WITH OTHER PLANTS

Table 1.3-1 presents a summary of the characteristics of the Millstone Unit 2 Nuclear Power Plant at the time of application for operating license. The table includes similar data for Calvert Cliffs Units 1 and 2, Maine Yankee Unit Number 1, Turkey Point Units Numbers. 3 and 4 and Palisades Unit Number 1. Bechtel Corporation and Combustion Engineering (CE), Inc. are identified as contractors in Section 1.6. The Palisades plant is included in the table because its coolant system is similar to that of Millstone Unit 2, because both Bechtel Corporation and CE, Inc. are Palisades contractors and because it is an example of a CE, Inc. nuclear steam supply system which is operating. Calvert Cliffs and Maine Yankee were selected because their cores are similar to that of Millstone Unit 2 and the most contemporaneous plants for which operating licenses have been issued with which CE is associated. Turkey Point is included because it is another comparable plant with which Bechtel Corporation is associated.

TABLE 1.3-1 COMPARISON WITH OTHER PLANTS

HYDRAULIC and THERMAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Total Core Heat Output, MWt	3.5	2,560	2,200	2,200	2,560	2,440
Total Core Heat Output, Btu/hr	3.5	$8,737 \times 10^6$	$7,479 \times 10^6$	$7,509 \times 10^6$	$8,740 \times 10^6$	$8,328 \times 10^6$
Heat Generated in Fuel, %	3.5	97.5	97.4	97.5	97.5	97.5
Maximum Overpower, %	3.5	12	12	12	12	12
System Pressure, Nominal, psia	3.5	2,250	2,250	2,100	2,250	2,250
System Pressure, Minimum Steady State, psia	3.5	2,200	2,200	2,050	2,200	2,200
Hot Channel Factors, Overall Heat Flux, F_q	3.5	3.00	3.23	3.8	3.00	2.89
Hot Channel Factors, Enthalpy Rise, $F_{\Delta H}$	3.5	1.65	1.77	2.51	1.65	1.62
DNB Ratio at Nominal Conditions	3.5	2.30	1.81	2.00	2.18	2.45
Coolant Flow: Total Flow Rate, lb/hr	3.5	134×10^6	101.5×10^6	125×10^6	122×10^6	122×10^6
Coolant Flow: Effective Flow Rate for Head Transfer, lb/hr	3.5	130×10^6	97.0×10^6	121.25×10^6	117.5×10^6	117.5×10^6
Coolant Flow: Effective Flow Area for Heat Transfer, ft^2	3.5	53.5	41.8	58.7	53.5	53.5
Coolant Flow: Average Velocity along Fuel Rods, ft/sec	3.5	16	14.3	12.7	13.6	13.9
Coolant Flow: Average Mass Velocity, lb/hr-ft^2	3.5	2.4×10^6	2.32×10^6	2.07×10^6	2.20×10^6	2.29×10^6
Coolant Temperatures, °F: Nominal Inlet	3.5	542	546.2	545	543.4	538.9
Coolant Temperatures, °F: Maximum Inlet due to Instrumentation Error and Deadband, °F	3.5	544	550.2	548	548	546
Coolant Temperatures, °F: Average Rise in Vessel, °F	3.5	45	55.9	46	52	51.1
Coolant Temperatures, °F: Average Rise in Core, °F	3.5	46	58.3	47	54	53.1
Coolant Temperatures, °F: Average in Core, °F	3.5	565	575.4	568.5	570.4	565.4
Coolant Temperatures, °F: Average in Vessel	3.5	564	574.2	568	569.5	564.4
Coolant Temperatures, °F: Nominal Outlet of Hot Channel	3.5	640	642	642.8	643	636

TABLE 1.3-1 Comparison with Other Plants (Continued)

HYDRAULIC and THERMAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Average Film Coefficient, Btu/hr-ft^2-F	3.5	5270	5400	4860	5240	5300
Average Film Temperature Difference, °F	3.5	34.5	31.8	30	33.5	33
Heat Transfer at 100% Power: Active Heat Transfer Surface Area, ft^2	3.5	48,400	42,460	51,400	48,416	47,000
Heat Transfer at 100% Power: Average Heat Flux, Btu/hr-ft^2	3.5	176,600	171,600	142,400	176,000	170,200
Heat Transfer at 100% Power: Maximum Heat Flux, Btu/hr-ft^2	3.5	527,800	554,200	541,200	527,900	502,300
Heat Transfer at 100% Power: Average Thermal Output, kw/ft	3.5	5.94	5.5	4.63	5.94	5.74
Heat Transfer at 100% Power: Maximum Thermal Output, kw/ft	3.5	16.6	17.6 (2)	17.6 (2)	17.8	16.9
Maximum Clad Surface Temperature at Nominal Pressure, °F	3.5	657	657	648	657	657
Fuel Center Temperature, °F: Maximum at 100% Power	3.5	3,780	4,030	4,040	3,780	3,640
Fuel Center Temperature, °F: Maximum at Over Power	3.5	4,070	4,300	4,350	4,070	3,940
Thermal Output, kw/ft at Maximum Over Power	3.5	19.6	20.0	19.7 (2)	20.0	19.0

CORE MECHANICAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Fuel Assemblies: Design	3.3	CEA	RCC	Cruciform	CEA	CEA
Fuel Assemblies: Rod Pitch, inches	3.3	0.58	0.563	0.550	0.58	0.580
Fuel Assemblies: Cross-Section Dimensions, inches	3.3	7.98 x 7.98	8.426 x 8.426	8.1135 x 8.1135	7.98 x 7.98	7.98 x 7.98
Fuel Assemblies: Fuel Weight (as UO_2), pounds	3.3	207,035	176,200	210,524	207,269	203,934
Fuel Assemblies: Total Weight, pounds	3.3	282,500	226,200	295,800	282,570	279,235

TABLE 1.3-1 Comparison with Other Plants (Continued)

CORE MECHANICAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Fuel Assemblies: Number of Grids per Assembly	3.3	8	7	8	8	8
Fuel Rods: Number	3.3	36,896	32,028	43,168	36,896	36,352
Fuel Rods: Outside Diameter, inches	3.3	0.44	0.422	0.4135	0.44	0.440
Fuel Rods: Diametral Gap, inches	3.3	0.0085	0.0065	0.0065	0.0085	0.0085
Fuel Rods: Clad Thickness, inches	3.3	0.026	0.0243	0.022	0.026	0.026
Fuel Rods: Clad Material	3.3	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Fuel Pellets: Material	3.3	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered
Fuel Pellets: Diameter, inches	3.3	0.3795	0.367	0.359	0.3795	0.3795
Fuel Pellets: Length, inches	3.3	0.650	0.600	0.600	0.650	0.650
Control Assemblies: Neutron Absorber	3.3	B_4C / S.S. Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%) Cruciform	B_4C / S.S. / Cd-In-Ag	B_4C / S.S. / Cd-In-Ag
Control Assemblies: Cladding Material	3.3	NiCrFe Alloy (Inconel 625)	304 SS-Cold Worked Welded to 13.5 inch span	304 SS Tubes, E.B.	NiCrFe Alloy	NiCrFe Alloy
Control Assemblies: Clad Thickness	3.3	0.040	0.109	0.016	0.040	0.040
Control Assemblies: Number of Assembly, full / part length	3.3	73	53	41 / 4 Cruciform Rods	77 / 8	77 / 8
Control Assemblies: Number of Rods per Assembly	3.3	5	20	117 Tubes per Rod	5	5
Core Structure: Core Barrel ID / OD, inches	3.3.2.2	148 / 151.5	133.875 / 137.875	149.75 / 152.5	148 / 149.75	148 / 149.75
Core Structure: Thermal Shield ID / OD, inches	3.3.2.5	156.75 / 162.75	142.625 / 148.0	None	None	156 / 162

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Structural Characteristics: Core Diameter, inches (Equivalent)	3.3.1	136	119.5	136.71	136.0	136.0
Structural Characteristics: Core Height, inches (Active Fuel)	3.3.1	136.7	144	132	136.7	136.7
H_2O/U, Unit Cell (Cold)	3.4.1	3.50	4.18	3.50	3.44	3.44
Number of Fuel Assemblies	3.3	217	157	204	217	217
UO_2 Rods per Assembly, Unshimmed / Shimmed		-	204	212 / 208	-	-
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch A	3.3	176	-	-	176	176
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch B	3.3	164	-	-	164	160
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch C	3.3	(176 / 164 / 164)	-	-	(176 / 164 / 164)	(176 / 164 / 160)
Performance Characteristics Loading Technique	3.4.1	3 Batch Mixed Central Zone	3 Regions Non-uniform	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Fuel Discharge Burnup, Mwd/MTU: Average First Cycle	3.4.1	12,770	13,000	10,180	13,775	13,795
Fuel Discharge Burnup, Mwd/MTU: First Core Average	3.2.1	22,000	14,500	17,600	22,550	30,000
Feed Enrichment (weight percent): Region 1	3.4.1	1.93	1.85	1.65	2.05	2.01
Feed Enrichment (weight percent): Region 2	3.4.1	2.33	2.55	2.08 / 2.54	2.45	2.40
Feed Enrichment (weight percent): Region 3	3.4.1	2.82	3.10	2.54 / 3.20	2.99	2.95
Feed Enrichment (weight percent): Equilibrium		-	-	2.54 / 3.20	-	-
Control Characteristics Effective Multiplication (beginning of life): Cold, No Power, Clean	3.4.1	1.170	1.180	1.212	1.194	1.170
Control Characteristics Effective Multiplication (beginning of life): Hot, No Power, Clean	3.4.1	1.129	1.38	1.175	1.152	1.129
Control Characteristics Effective Multiplication (beginning of life): Hot, Full Power, Xe Equilibrium	3.4.1	1.078	1.077	1.111	1.094	1.075
Control Assemblies: Material	3.3	B_4C / S.S. Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%)	B_4C / S.S.-Cd-In-Ag	B_4C / S.S.-Cd-In-Ag
Control Assemblies: Number of Control Assemblies	3.4.1	73	53	45 Cruciform	85	85
Number of Absorber Rods per RCC (or CEA) Assembly	3.3	5	20	117 Tubes Welded to Form 13.5 inches span	5	5
Total Rod Worth (Hot), % Δρ	3.4.1	11.0	7	8.6	≥ 9.6	≥ 9.9
Boron Concentrations – To shut reactor down with no rods inserted, clean, ppm: Cold / Hot, ppm	3.4.1	945 / 935	1,250 / 1,210	1,180 / 1,210	1,120 / 1,095	945 / 935
Boron Concentrations – To shut reactor down with no rods inserted, clean, ppm: To control at power with no rods inserted, clean / equilibrium xenon, ppm	3.4.1	820 / 590	1,000 / 670	1,070 / 830	960 / 725	820 / 590
Kinetic Characteristics, Range Over Life: Moderator Temperature Coefficient (3) Δρ/°F	3.4.1	-0.4 x 10^{-4} to -2.1 x 10^{-4}	+.3 x 10^{-4} to -1.96 x 10^{-4} -3.5	-0.08 x 10^{-4} to -2.25 x 10^{-4}	-.20 x 10^{-4} to -1.96 x 10^{-4}	-0.40 x 10^{-4} to -2.20 x 10^{-4}

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Kinetic Characteristics, Range Over Life: Moderator Pressure Coefficient (3) $\Delta\rho$/psi	3.4.1	-0.65×10^{-6} to $+2.39 \times 10^{-6}$	-0.3×10^{-6} to $+3.4 \times 10^{-6}$	$+0.10 \times 10^{-6}$ to $+1.7 \times 10^{-6}$	$+0.65 \times 10^{-6}$ to $+2.39 \times 10^{-6}$	$+0.65 \times 10^{-6}$ to $+2.39 \times 10^{-6}$
Kinetic Characteristics, Range Over Life: Moderator Void Coefficient (3) $\Delta\rho$/% Void	3.4.1	-0.41×10^{-3} to -1.43×10^{-3}	$+0.5 \times 10^{-3}$ to -2.5×10^{-3}	-0.06×10^{-3} to -1.0×10^{-3}	-0.41×10^{-3} to -1.43×10^{-3}	-0.41×10^{-3} to -1.43×10^{-3}
Kinetic Characteristics, Range Over Life: Doppler Coefficient (4) $\Delta\rho$/°F	3.4.1	-1.45×10^{-5} to -1.07×10^{-5}	-1.0×10^{-5} to -1.6×10^{-5}	-1.56×10^{-5} to -1.08×10^{-5}	-1.46×10^{-5} to -1.06×10^{-5}	-1.45×10^{-5} to -1.07×10^{-5}

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Reactor Vessel	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Steam Generator: Tube Side	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Steam Generator: Shell Side	4.2.2	ASME III Class A	ASME III Class C	ASME III Class A	ASME III Class A	ASME III Class A
Pressurizer	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Pressurizer Relief (or Quench) Tank	4.2.2	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C
Pressurizer Safety Valves	4.2.2	ASME III	ASME III	ASME III	ASME III	ASME III
Reactor Coolant Piping	4.2.2	ANSI B 31.7	USAS B 31.1	USAS B 31.1	USAS B 31.7	USAS 31.1

PRINCIPAL DESIGN PARAMETERS OF THE COOLING SYSTEM

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Operating Pressure, psig	4.2.1	2235	2235	2085	2235	2235
Reactor Inlet Temperature, °F	4.2.1	539.7	546.2	545	544.5	540
Reactor Outlet Temperature, °F	4.2.1	595.1	602.1	591.1	599.4	592.8
Number of Loops	4.1	2	3	2	2	3
Design Pressure, psig	4.3.4	2,485	2,485	2,485	2,485	2,485

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE COOLING SYSTEM

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Design Temperature, °F	4.3.4	650	650	650	650	650
Hydrostatic Test Pressure (cold), psig	4.2.1	3,110	3,110	3,110	3,110	3,110
Total Coolant Volume, cubic feet	4.2.1	11,101	9,088	10,809	11,101	11,026

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Material	4.3.1, 4.5.6	SA-533, Grade B Class I, low alloy steel plates and SA-508-64, Class 2 forgings, internally clad with Type 304 (5) austenitic SS	SA-302, Grade B, low alloy steel, internally clad with Type 304 austenitic SS	SA-302, Grade B, low alloy steel, internally clad with Type 304 austenitic SS	SA-533, Grade B, Class I, steel, internally clad Type 304 austenitic SS	SA-533, Grade B, forgings-A-508-64 Class 2, cladding weld deposited 304 SS equivalent
Design Pressure, psig	4.3.1	2,485	2,485	2,485	2,485	2,485
Design Temperature, °F	4.3.1	650	650	650	650	650
Operating Pressure, psig	4.2.1	2,235	2,235	2,085	2,235	2,235
Inside Diameter of Shell, inches	4.3.1	172	155.5	172	172	172
Outside Diameter across Nozzles, inches	4.3.1	253	236	254	253	266-5/8
Overall Height of Vessel and Enclosure Head, feet-inches to top of CRD Nozzle	4.3.1	41 feet 11.75 inches	41 feet 6 inches	40 feet 1-13/16 inches	41 feet 11.75 inches	42 feet 1-3/8 inches
Minimum Clad Thickness, inches	4.3.1	1/8	5/32	3/16	1/8	1/8

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Number of Units	4.3.2	2	3	2	2	3
Type	4.3.2	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-tube with integral moisture separator
Tube Material	4.3.2	Ni-Cr-Fe Alloy	Ni-Cr-Fe-Alloy	Ni-Cr-Fe Alloy	Ni-Cr-Fe Alloy	Ni-Cr-Fe Alloy
Shell Material	4.3.2	SA-533 Gr. B Class 1 and SA-516 gr 70	SA-302	Carbon Steel	SA-533 Gr. B Class 1 and SA-516 gr 70	SA-533 Gr. B Class 1 and SA-516 gr 70
Tube Side Design Pressure, psig	4.3.2	2,485	2,485	2,485	2,485	2,485
Tube Side Design Temperature, °F	4.3.2	650	650	650	650	650
Tube Side Design Flow, lb/hr	4.3.2	61×10^6	33.93×10^6	62.5×10^6	61×10^6	40.67×10^6
Shell Side Design Pressure, psig	4.3.2	1,000	1,085	985	985	985
Shell Side Design Temperature, °F	4.3.2	550	556	550	550	550
Operating Pressure, Tube Side, Nominal, psig	4.3.2	2,235	2,235	2,085	2,235	2,235
Operating Pressure, Shell Side, Maximum, psig	4.3.2	885	1,020	885	885	885
Maximum Moisture at Outlet at Full Load, %	4.3.2	0.2	0.25	0.2	0.2	0.2
Hydrostatic Test Pressure, Tube Side (cold), psig	4.3.2	3,110	3,107	3,110	3,110	3,110
Steam Pressure, psig, at full power	4.3.2	800	730	755	835	800
Steam Temperature, °F, at full power	4.3.2	520.3	510	513.8	525.2	520.3

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMP

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Number of Units	4.3.3	4	3	4	4	3
Type	4.3.3	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design Pressure, psig	4.3.3	2,485	2,485	2,485	2,485	2,485
Design Temperature, °F	4.3.3	650	650	650	650	650
Operating Pressure, nominal psig	4.3.3	2,235	2,235	2,085	2,235	2,235
Suction Temperature, °F	4.3.3	540	546.5	545	543.4	538.9
Design Capacity, gpm	4.3.3	81,200	89,500	83,000	81,200	108,000
Design Head, feet	4.3.3	243	260	260	300	290
Hydrostatic Test Pressure, (cold), psig	4.3.3	3,110	3,107	3,110	3,110	3,110
Motor Type	4.3.3	AC Induction	AC Induction	AC Induction	AC Induction	AC Induction
	4.3.3	Single Speed	Single Speed	Single Speed	Single Speed	Single Speed
Motor Rating, hp	4.3.3	6,500	6,000	6,250	7,200	9,000

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Material	4.3.4	SA516 - GR 70 with minimum 1/8 304L SS clad	Austenitic SS	SA212B clad with SS	SA516 - gr 70 with nominal 7/32 SS clad	SA516 - gr 70 with SS clad
Hot Leg - ID, inches	4.3.4	42	29	42	42	33.5
Cold Leg - ID, inches	4.3.4	30	27.5	30	30	33.5
Between Pump and Steam Generator - ID, inches	4.3.4	30	31	30	30	33.5

TABLE 1.3-1 Comparison with Other Plants (Continued)

CONTAINMENT SYSTEM PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Type	5.2.1	Double containment with steel lined prestressed post tensioned concrete cylinder, curved dome roof completely enclosed by Enclosure Building	Steel lined prestressed post tensioned concrete cylinder, shallow dome roof	Steel lined prestressed post tensioned concrete cylinder, curved dome roof	Steel lined prestressed post tensioned concrete cylinder, curved dome roof	Steel lined reinforced concrete flat bottom and hemispherical dome
Containment Parameters: Inside Diameter, feet	5.2.1	130	116	116	130	135
Containment Parameters: Height, feet.	5.2.1	175	169	190.5	181-2/3	169.5
Containment Parameters: Free Volume, ft^3	5.2.1	1,920,000 (5)	1,550,000	1,640,000	2,000,000	1,855,000
Containment Parameters: Reference Incident Pressure, psig	5.2.1	54	59	55	50	55
Containment Parameters: Concrete Thickness, feet						
Containment Parameters: Vertical Wall	5.2.1	3.75	3.75	3	3.75	4.5
Containment Parameters: Dome	5.2.1	3.25	3.25	2.5	3.25	2.5
Containment Leak Prevention and Mitigation Systems	6.7.2.1	Completely enclosed containment has leaktight penetrations and continuous steel liner. Enclosure Building Filtration region at small negative pressure during LCI. Automatic isolation where required. The exhaust from filtration region passed through charcoal filters to 375 feet Millstone stack following incident.	Leak tight penetration and continuous steel liner, automatic isolation where required	Leak tight penetration and continuous steel liner, automatic isolation where required	Leak tight penetration and continuous steel liner, automatic isolation where required. The exhaust from penetration rooms to vent.	Leak tight penetration and continuous steel liner, automatic isolation where required
Gaseous Effluent Purge	11.1.2.1.3	Discharge through Unit 2 stack	Through particulate filter & monitors part of main exhaust system	Discharge through stack	Discharge through vent	Discharge through stack

TABLE 1.3-1 Comparison with Other Plants (Continued)

ENGINEERED SAFEGUARDS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Safety Injection System: Number of High Head Pumps	6.3.2.1	3	4 (shared)	3	3	3 (charging)
Safety Injection System: Number of Low Head Pumps	6.3.2.1	2	2	2	2	2
Safety Injection System: Safety Injection Tank, number	6.3.2.1	4	3	4	4	3
Containment Fan Coolers: Number of Units	6.5.1.2	4	3	4	4	6
Containment Fan Coolers: Air Flow capacity, each at emergency condition, cfm	6.5.2.2	34,800	65,000	30,000	55,000	N/A
Post-Incident Filters Inside Containment: Number of Units		None	None	None	None	None
Post-Incident Filters Inside Containment: Type		None	None	None	None	None
Containment Spray Number of Pumps	6.4.2.1	2	2	-	2	3
Emergency Power Diesel Generator Units	8.3.1.1	2	2 total for both units	2	3 total for both units	2
Enclosure Building Filtration System Number of Units	6.7.2.1	2	-	-	-	0

RADIOACTIVE WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Design Failed Fuel, %	11.1.1.1	1	1	1	1	1
Gaseous Waste Processing System	11.1.2.1					
Annual Volume of Gases Discharge, ft^3	11.1.2.1	14,344	(6)	4,539	66,240	(6)
Annual Activity Discharge, Curies	11.1.2.1	556	14,758	(6)	6)	(6)
Decay Storage Time for Gases, Days	11.1.2.1	60 (Minimum)	45	30 (Minimum)	60	(6)
Compressors: Number		2	2 (7)	2	2 (7)	2

TABLE 1.3-1 Comparison with Other Plants (Continued)

RADIOACTIVE WASTE PROCESSING SYSTEMS

<*Parameter*>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Compressors: Capacity, each	11.1.2.2	25 SCFM	40 CFM	2.35 SCFM	4 to 7 SCFM	(6)
Decay Tanks: Number		6	6 (7)	3	3 (7)	3
Decay Tanks: Capacity, (each), ft^3		Specified 582 cuft As Built 627 cuft	525	100	610	200

LIQUID WASTE PROCESSING SYSTEMS

<*Parameter*>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Clean Liquid Waste (Reactor Coolant Wastes)	11.1.2.1					
Design Volume Wastes per Year	11.1.2.1	14 Reactor Coolant System Volumes (840,000 Gallons)		(6)	14 Reactor Coolant System	(6)
Expected Volume of Waste Discharge Per Year, Gallons	11.1.2.1	404,234	(Design Incorporates Recycle of Waste to R.C. System Clean Liquid Waste System Not Compared)	724,300	805,542	(6)
Annual Expected Activity Discharged, curies	11.1.2.1	286 (includes H^3)		(6)	(6)	(6)
Percentage of 10 CFR Part 20	11.1.4.1	0.6%		(6)	(6)	(6)
Degasifier: Number	11.1.2.2	1	1	2	2	
Degasifier: Type	11.1.2.2	Packed Column Utilizing Internal Generated Stripping Steam	Vacuum	Packed Tower	Flashing	
Degasifier: Design Flow Rate, gpm	11.1.2.2	132	160	120	100	
Degasifier: Decontamination Factors	11.1.2.2	1,000 (Kr & Xe)	40	10	(6)	
Storage Tanks: Number	11.1.2.2	4		4	4	2
Storage Tanks: Total Capacity	11.1.2.2	3 Reactor Coolant System (180,000 Gallons)		200,000 Gallons	6 Reactor Coolant System Volumes (7)	250,000 Gallons
Storage Tanks: Vent Discharge	11.1.2.2	To Gaseous Waste System for storage and decay		To Exhaust Plenum	Plant Vent	To ventilation System and stack
Demineralizers: Number	11.1.2.2	3		3	4	2

TABLE 1.3-1 Comparison with Other Plants (Continued)

LIQUID WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Demineralizers: Type	11.1.2.2	Mixed Bed Non Regenerative		Mixed bed	Mixed Bed Non Regenerative	Cesium Removal
Demineralizers: Decontamination	11.1.2.2	1,000		10	100	(6)
Demineralizers: Factors	11.1.2.2			(0 for Y, Mo, H^3)		
Evaporator (Boron Recovery): Number	11.1.2.2	1		N/A	2	1
Evaporator (Boron Recovery): Type	11.1.2.2	Vacuum, Submerged U-Tube			Horizontal Spray Film	Forced Calculating, Single Effect
Evaporator (Boron Recovery): Capacity, GPM Distillate	11.1.2.2	25			20	30
Evaporator (Boron Recovery): Decontamination	11.1.2.2				10^5 (Nonvolatiles)	(6)
Evaporator (Boron Recovery): Factors	11.1.2.2	1,000 (Nonvolatiles), 50 (Halogens), 100 (Dissolved Gases)			10^4 (Gases)	

Aerated Liquid Waste Processing System (Miscellaneous Wastes)

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Design Volume of Waste per year	11.1.2.1	3,639,400 (Gallons)	(6)	(6)	(6)	(6)
Expected Volume of Waste Discharged per year, Gallons	11.1.2.1	313,000	508,620	(6)	1,330,320	(6)
Annual Expected Activity Discharged, Curies	11.1.2.1	1.11 (includes H^3)	0.077	(6)	(6)	(6)
Percentage of 10 CFR Part 20	11.1.4.1	Less than 0.1%				
Storage Tanks: Number	11.1.2.2	1	2	1	2	2
Storage Tanks: Total Capacity	11.1.2.2	5,000 Gallons	2,000 Gallons	5,500 Gallons	8,000 Gallons	24,800 Gallons
Demineralizers: Number	11.1.2.2	1	(6)	N/A	1	N/A
Demineralizers: Type	11.1.2.2	Mixed Bed Non Regenerative	(6)		Mixed Bed Non Regenerative	
Demineralizers: Decontamination Factors	11.1.2.2	500	(6)		100	

TABLE 1.3-1 Comparison with Other Plants (Continued)

Aerated Liquid Waste Processing System (Miscellaneous Wastes)

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Evaporator:	11.1.2.2	N/A		N/A	N/A	
Evaporator: Number	11.1.2.2		1 (7)			1
Evaporator: Type	11.1.2.2		(6)			(6)
Evaporator: Capacity, Distillate GPM	11.1.2.2		(6)			(6)
Evaporator: Decontamination Factors	11.1.2.2		10^6		(6)	(6)

Solid Waste Processing System

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Evaporator Concentrates	11.1.2.1	Solidified in Concrete in 55 Gallon drums	Solidified in Concrete in 55 Gallon drums	N/A	Solidified in Concrete in 55 Gallon drums	55 Gallon drums
Spent Resins Shipping & Volumes	11.1.2.1	Shipping cask after dewatering, 225 ft^3	Dewatered 55 Gallon Drums	(6)	Solidified in Concrete in 55 Gallon Drums	Shipping cask
Contaminated Filter Cartridges & Volumes	11.1.2.1	55 Gallon drums	55 Gallon drums	55 Gallon drums	Solidified in Concrete in 55 Gallon Drums	Cask or 55 Gallon drums
Annual Activity Shipped, curies	11.1.2.1	4,250	(6)	(6)	(6)	(6)

1 The values listed for these plants were taken from public documentation.
2 Based on total heat output of the core rather than heat generated in the fuel alone.
3 Values shown are for beginning of life full power / end of cycle full power.
4 Values shown are for beginning of life zero power/beginning of life cycle full power.
5 Measured value from pre-operational volume verification test and used for integrated leak rate testing. Includes volume of vented pressurizer, safety injection tanks, and other tanks.
6 Not Specifically Available in Public Documents.
7 Shared by Two (2) Units.

1.4 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR DESIGN

The principal architectural and engineering features used in the design of Unit 2 of the Millstone Nuclear Power Station are summarized in the following material.

1.4.1 PLANT DESIGN

Principal structures and equipment which may serve either to prevent accidents or to mitigate their consequences have been designed, fabricated and erected in accordance with applicable codes so as to withstand the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phenomena which could be reasonably assumed to occur at the site during the lifetime of this plant. Systems and components designed for Seismic Category I requirements are listed in Table 1.4-1. It should be noted that the terms ‘Category’ and ‘Class’ are used interchangeably throughout the MP2 FSAR in defining seismic design classifications of Structures, Systems and Components. Unit 2 was designed so that the safety of one unit will not be impaired in the unlikely event of an accident in the other unit. Principal structures and equipment were sized for the maximum expected nuclear steam supply system (NSSS) and turbine outputs.

Redundancy is provided in the reactor and safety systems so that the single failure of any active component of either system cannot prevent the action necessary to avoid an unsafe condition. The unit is designed to facilitate inspection and testing of systems and components whose reliabilities are important to the protection of the public and plant personnel.

Provisions have been made to protect against the hazards of such events as fires or explosions.

Systems and components which are significant from the standpoint of nuclear safety are designed, fabricated and erected to quality standards commensurate with the safety function to be performed. Appendix 1.A of this FSAR addresses the implementation of Atomic Energy Commission (AEC) General Design Criteria for Nuclear Power Plants, 10 CFR Part 50, Appendix A. Section 12.8 describes the Quality Assurance Program.

1.4.2 REACTOR

The following criteria (see Chapter 3) apply to the reactor:

a. The reactor is of the pressurized water-type, designed to provide heat to steam generators which, in turn, provide steam to drive a turbine generator. The initial full power core thermal output was 2560 megawatts (the NSSS rating was 2570 megawatts) prior to its uprating to the current 2700 megawatts thermal power level (NSSS rating of 2715 megawatts).

b. The reactor is refueled with slightly enriched uranium dioxide contained in zirconium alloy tubes.

c. Minimum departure from nucleate boiling ratio during normal operation and anticipated transients will not be below that value which could lead to fuel rod failure or damage. The maximum fuel centerline temperature evaluated at the design overpower condition will be below that value which could lead to fuel rod failure. The melting point of the UO_2 will not be reached during routine operation and anticipated transients.

d. Fuel rod clad is designed to maintain cladding integrity throughout fuel life. Fission gas release within the rods and other factors affecting design life will be considered for the maximum expected exposures.

e. The reactor and control systems are designed so that any xenon transients can be adequately damped.

f. The reactor is designed to accommodate the anticipated transients safely and without fuel damage.

g. The reactor coolant system (RCS) is designed and constructed to maintain its integrity throughout the expected plant life. Appropriate means of test and inspection are provided.

h. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by deformation or rupture, to the pressure vessel or impair operation of the engineered safety features (ESF).

i. Control element assemblies (CEA) are capable of holding the core subcritical at hot zero power conditions following a trip, and providing a safety margin even with the most reactive CEA stuck in the full, withdrawn position.

j. The chemical and volume control system (CVCS) can add boric acid to the reactor coolant at a sufficient rate to maintain an adequate shutdown margin when the RCS is cooling down following a reactor trip. This is accomplished at a maximum design rate. This system is independent of the CEA system.

k. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

1.4.3 REACTOR COOLANT AND AUXILIARY SYSTEMS

1.4.3.1 Reactor Coolant System

The design bases in this section are those used for the integrated design of the RCS or those which apply to all of the system components. The design bases unique to each component are discussed in Section 4.3.

The RCS is designed for the normal operation of transferring 2715 MWt (9.26 x 10 Btu/hr) from the reactor core (2700 MWt) and reactor coolant pumps (15 MWt) to the steam generators. In the steam generator, this heat is transferred to the secondary system forming 5.9 x 10^6 lb/hr of 880 psia saturated steam per generator with a 2.0 percent maximum moisture content.

The RCS is designed to accommodate the normal design transients listed. These transients include conservative estimates of the operational requirements of the systems and are used to make the required component fatigue analyses.

a. 500 heatup and cooldown cycles at a maximum heating and cooling rate of 100°F/hr. The pressurizer is designed for a maximum cooldown rate of 200°F/hr.

b. Pressurizer spray piping is limited to 160 plant heatup and cooldown cycles. Primary manway studs of the replaced steam generators are limited to 200 heatup and cooldown cycles.

c. 15,000 power change cycles in the range between 15 and 100 percent of full load with a ramp load change of 5 percent of full load per minute increasing or decreasing. This will occur without reactor trip.

d. Primary manway studs for the replaced steam generators are limited to 1,000 cycles with a ramp load change of 5% per minute decreasing and 30% per hour increasing (plant loading/unloading).

e. 2,000 step power changes of 10 percent, both increasing and decreasing between 15 and 100 percent of full load. Primary manway studs for the replaced steam generator are limited to 1,500 step power changes.

f. 10 cycles of hydrostatic testing at 3,110 psig and a temperature at least 60°F above the nil ductility transition temperature (NDTT) of the component having the highest NDTT.

g. 200 cycles of leak testing at 2,485 psig and a temperature at least 60°F greater than the NDTT of the component with the highest NDDT.

h. Primary manway studs for the replaced steam generators are limited to 80 cycles of leak testing at 2,485 psig.

i. 10^6 cycles of operating pressure variations of ±100 psi from the normal 2,235 psig operating pressure and ±6°F at operating temperature and pressure.

j. 400 reactor trips when at 100 percent power. Primary manway studs for the replaced steam generator are limited to 200 reactor trips when at 100% power.

In addition to these normal design transients, the following abnormal transients are also considered to arrive at a satisfactory usage factor as defined in Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code:

a. 40 cycles of loss of turbine load from 100 percent power.

b. 40 cycles of loss of reactor coolant flow when at 100 percent.

c. 5 cycles of loss of main steam system pressure.

Components of the RCS are designed and will be operated so that no deleterious pressure or thermal stress will be imposed on the structural materials. The necessary consideration has been given to the ductile characteristics of the materials at low temperature.

1.4.3.2 Chemical and Volume Control System

The major functions of the CVCS (see Section 9.2) are to:

a. Maintain the required volume of water in the RCS.

b. Maintain the chemistry and purity of the reactor coolant.

c. Maintain the desired boric acid concentration in the reactor coolant.

d. Provide a controlled path to the waste processing system.

The system is designed to accept the discharge when the reactor coolant is heated at the design rate of 100°F/hr and to provide the required makeup when the reactor coolant is cooled at the design rate of 100°F/hr. Discharge is automatically diverted to the waste processing system when the volume control tank is at its highest permissible level. The system will also supply makeup or accept discharge due to power decreases or increases. The design transients are ±10 percent of full power step changes and ramp changes of ±5 percent of full power per minute between 15 to 100 percent power. On power increases, the letdown flow is automatically diverted to the waste processing system when the volume control tank reaches the highest permissible level. On power decreases, sufficient coolant is in the volume control tank to allow a full to zero power decrease without additional makeup, in the event of a makeup system failure or override.

For an assumed 1 percent failed fuel condition, the activity in the reactor coolant does not exceed 411 μCi/cc at 77°F. The system is also designed to maintain the reactor coolant chemistry within the limits specified in Section 4.4.3.

The rate of boron addition is sufficient to counteract the maximum reactivity increase due to cooldown and xenon decay. Any one of the three charging pumps is capable of injecting the required boron (as boric acid). The maximum rate at which the reactor coolant boron concentration can be reduced must be substantially less than the equivalent maximum rate of reactivity insertion by the CEA.

Prior to refueling, the system is capable of increasing the reactor coolant boron concentration from zero to 1720 ppm by feed and bleed when the reactor coolant is at hot standby operating temperature.

Provisions to facilitate the plant hydrostatic testing and to leak test the RCS are included.

1.4.3.3 Shutdown Cooling System

The shutdown cooling system (see Section 9.3) is designed to cool the RCS from approximately 300° to 130°F in 24 hours, assuming that the component cooling water inlet temperature is at its maximum design value of 95°F. The design RCS cooldown rate is 100°F/hr. A temperature of 130°F or less can be achieved 27.5 hours after reactor shutdown, assuming an infinitely exposed core. The maximum allowable pressure for the RCS during shutdown cooling is approximately 285 psig.

1.4.4 CONTAINMENT SYSTEM

The containment (see Sections 5.2 and 14.8), including the associated access openings and penetrations, is designed to contain pressures and temperatures resulting from a postulated main steamline break (MSLB) in which:

a. A range of power level, break sizes, and single failures are considered.

b. Cases with the loss of offsite power and with AC power available are analyzed to determine which scenario maximizes the energy removal into containment.

c. Safety injection is not assumed since it would tend to reduce the energy released into containment.

d. The containment air recirculation cooling system and the containment spray system are credited to mitigate the containment pressure and temperature consequences.

Containment response to a loss-of-coolant (LOCA) accident was also analyzed. It was found that the peak containment pressure and temperature of the MSLB accident bound the LOCA.

The containment is designed to assure integrity against postulated missiles from equipment failures and against postulated missiles from external sources.

Means are provided for pressure and leak rate testing of the containment system. This includes provisions for leak rate testing of individual piping and electrical penetrations that rely on gestated seals, sealing compounds, expansion bellows, and the interior of the containment.

The enclosure building (see Section 5.3) is designed to withstand a wind loading of 115 mph, with gusts of 140 mph, snow load of 60 psf and seismic loads. The Enclosure Building is designed so that is structural framing will withstand tornado loads, but the siding will be blown away (see Section 5.3.3).

1.4.5 ENGINEERED SAFETY FEATURES SYSTEMS

The design incorporates redundant independent full capacity engineered safety features systems (ESFS). These, in conjunction with the containment, ensure that the release of fission products, following any postulated occurrence, at least the minimum ESF required to terminate that occurrence are operable. The following are required as minimum safety features:

One high pressure safety injection (HPSI) train

One low pressure safety injection (LPSI) train

Four safety injection tanks (water quantity of three is required to reach the core)

One containment spray and two containment air recirculation and cooling subsystems, or equivalent (Section 6.4)

One hydrogen control subsystem

One enclosure building filtration train

One auxiliary feedwater trains

Each of these subsystems is independent of its redundant counterpart with the exception of the safety injection subsystems. The HPSI and LPSI subsystems (Section 6.3) are independent up to the common pipe connections to the four reactor coolant cold legs. Remote manually operated valves provide appropriate cross-connections between redundant subsystems for backup and to allow maintenance. Redundant components are physically separated.

The ESFS are designed to perform their functions for all break sizes in the RCS piping up to and including the double-ended rupture of the largest reactor coolant pipe. The safety injection system limits fuel and cladding damage to an amount which will not interfere with adequate emergency core cooling and holds metal-water reactions to minimal amounts. Two full capacity systems, based on different principles remove heat from the containment to maintain containment integrity, the containment spray system (Section 6.4) and the containment air recirculation and cooling system (Section 6.5). The enclosure building filtration system (EBFS) (Section 6.7) maintains the enclosure building filtration region (EBFR) at a slightly negative pressure and filters the exhaust from this space. The containment postaccident hydrogen control system (Section 6.6) mixes and

monitors the accumulation of hydrogen gases within the containment. Purge and recombiners are not credited for any mitigating function.

1.4.6 PROTECTION, CONTROL AND INSTRUMENTATION SYSTEM

A reactor protective system (RPS) (see Section 7.2) is provided which initiates reactor trip if the reactor approaches an unsafe condition.

Interlocks and automatic protective systems are provided along with administrative controls to ensure safe operation of the plant.

Sufficient redundancy is installed to permit periodic testing of the RPS so that failure or removal from service of any one protective system component or portion of the system will not preclude reactor trip or other safety action when required.

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 Number 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 Section 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 1301 and 1302 and Appendix B of 10 CFR Part 20 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.

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.

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS

System	Components
Safety Injection System	HPSI pumps and motors
	LPSI pumps and motors
	Safety Injection Tanks
	Refueling Water Storage Tank
	Piping and supports
	Valves and valve operators
Containment Spray System	Containment spray pumps and motors
	Shutdown cooling heat exchangers
	Refueling water storage tank
	Piping and supports
	Valves and valve operators
	Containment sump screen
Containment Air Recirculation and Cooling System	Fans and motors
	Cooling Coils
	Housing
Enclosure Building Filtration System and Emergency Spent Fuel Pool Cleanup	Fans and motors
	Filters and housing
	Electric heaters
	Piping, ductwork and supports
	Dampers and damper operators
Hydrogen Control System	Hydrogen recombiners
	PIR fans and motors
	Piping and supports
	Hydrogen purge valves and valve operators
	Hydrogen monitoring system

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Control Room Air Conditioning System (including the control room filtration system)	Fans and motors
	Direct expansion and condenser coils
	Housings
	Compressor
	CRFS Filters
	Ductwork and supports
	Dampers and damper operators
	Refrigeration piping and supports
	Refrigerant valves and valve operators
	Temperature control system
	Control Panels
Engineered Safety Feature Room Air Recirculation System	Fans and motors
	Cooling coils
	Ductwork and supports
	Dampers and damper operators
Diesel Generator Ventilation System	Fans and motors
	Ductwork and supports
	Dampers
Vital Switchgear Ventilation System	Fans and Motors
	Cooling Coils
	Chillers and control panels
	Pumps and motors
	Piping; valves and supports
	Ductwork and supports
	Dampers and Damper Operators
Containment Isolation System	Piping and sleeves
	Valves and valve operators

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Electrical Power Supply System	Station batteries, racks and chargers
	125 VDC Switchgear
	DC/AC Inverters
	Vital AC and DC distribution panels
	4160 Volt Emergency Switchgear
	480 Volt Emergency Load Centers
	480 Volt Emergency Motor Control Centers
Electrical Distribution System	Vital tray system and supports
	Vital underground duct banks
	Penetration assemblies
Reactor Coolant System	Reactor vessel and internals
	Control element assemblies and drives
	Pressurizer
	Reactor coolant pumps and motors
	Reactor coolant piping
	Pressurizer surge line and supports
	Pressurizer safety and relief valves
	Steam generators
	Vent, sampling and drain piping, supports and valves up to and including second isolation valve
	Quench tank *
	Pressurizer safety and relief valves
	piping and supports to quench tank *
	Reactor coolant pump supports

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Chemical and Volume Control System	Boric acid storage tanks
	Boric acid pumps and drivers
	Boric acid piping supports and valves
	Charging pumps and drivers
	Charging line piping, supports, valves and pulsation dampeners
	Letdown line piping, supports and valves up to and including second isolation valve
	Regenerative Heat exchanger
	Letdown heat exchanger *
	Letdown line piping, supports, and valves downstream of reactor coolant system isolation valves *
	Letdown filters *
	Ion exchangers *
	Volume control tank *
Spent Fuel Pool Cooling System	Piping, supports and valves between spent fuel pool and shutdown heat exchangers
	Spent fuel pool cooling pumps
	Spent fuel pool heat exchangers
	Spent fuel pool cooling pump drivers *
	Piping, supports, and valves associated with normal spent fuel cooling (up to and including pipe support beyond isolation valve on branch lines) *
Gaseous Waste Processing System	Waste gas decay tanks *
	Waste gas compressors *
	Waste gas filter *
	High pressure (150 psig) service piping, supports, and valves *

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Fuel and Reactor Component Handling Equipment	Containment polar crane
	Spent fuel cask crane
	Spent fuel platform crane *
	Refueling machine *
	Fuel transfer machine *
	Fuel tilting mechanisms *
	Fuel transfer tube and isolation valve
	New and spent fuel storage racks
	New fuel elevator *
	Spent fuel inspection machine *
RBCCW System	RBCCW Pumps and Motors
	RBCCW Heat Exchangers
	RBCCW Surge Tank
	Piping and Supports
	Expansion Joints
	Valves and Valve Operators
Service Water System	Pumps and Drivers
	Piping and Supports
	Valves and Valve Operators
	Service Water Strainers
Emergency Diesel Generators Diesel Oil System	Air Intake and Exhaust Piping
	Control Panels
	Diesel Oil Supply Tanks
	Piping, Valves and Supports
Lube Oil System	Pumps and motors
	Coolers
	Piping and supports
	Heaters
	Piping and supports

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Jacket Water Cooling System	Pumps and motors
	Coolers
	Piping and supports
	Heaters
	Jacket water expansion tank
	Valves and valve operators
*Designated seismic Class II components but designed for Class I earthquake basis.	
Air Cooling System	Pumps
	Coolers
	Piping and supports
	Valve and valve operators
Starting Air System	AC and DC Motor Driven Compressors
	Starting Air tanks
	Piping and supports upstream of check valves
Auxiliary Feedwater System	Auxiliary. feedwater pumps and drivers
	Condensate storage tank
	Piping and supports
	Valves and valve operators
Main Steam System (Upstream of isolation valves	Main steam safety relief valves
	Atmospheric dump valves
	Main Steam isolation valves
	Piping and supports
	Valves and valve operators
Engineered Safety Actuation System, Status Panel	
Reactor Protection System	
Seismic Measurement Instrumentation	
Main Control Boards	
Main Steam Isolation Panel	

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Hot Shutdown Control Boards	
Boric Acid Heat Tracing Panels	
Radiation Monitoring System	

* Designated seismic Class II components but designed for Class I earthquake basis.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

1.5.1 GENERAL

The design of Millstone Unit 2 is based upon concepts which have been successfully applied in the design of other pressurized water reactor power plants. However, certain programs of theoretical analysis or experimentation (constituting “research and development” as defined in the Atomic Energy Act, as amended, and in Nuclear Regulatory Commission (NRC) Regulations) have been undertaken to aid in plant design and to verify the performance characteristics of the plant components and systems. This section describes the results and status of these analytical and test programs, including experimental production and testing of models, devices, equipment and materials at time of application for an operating license.

Combustion Engineering (CE), Inc., which conducted these programs, had taken into consideration information derived from research and development activities of the NRC and other organizations in the nuclear industry.

All CE research and development programs required to justify the design to Millstone Unit 2 were completed and all test results were factored into design of the plant.

1.5.2 FUEL ASSEMBLY FLOW MIXING TESTS

In 1966, a series of single-phase tests on coolant turbulent mixing was run on a prototype fuel assembly which was geometrically similar to the Palisades assembly. The model enabled determination of flow resistance and vertical subchannel flow rates using pressure instrumentation and the average level of eddy flow using dye-injection and sampling equipment. The tests yielded the value of the inverse Peclet number characteristic of eddy flow (0.00366). During the course of the tests the value was shown to be insensitive to coolant temperature and to vertical coolant mass velocity. The design value of the inverse Peclet Number was established as 0.0035 on the basis of the experimental results.

As part of a CE sponsored research and development program, a new series of single-phase dye injection mixing tests were conducted in 1968. The tests were performed on a model of a portion of control element assembly (CEA) type fuel assembly which was sufficiently instrumented to enable measurement, via a data reduction computer program, of the individual lateral flows across the boundaries of 12 subchannels of the model. Although these tests were not intended for that purpose, some of the test results could be used to determine the average level of turbulent mixing in the reference design assembly. The inverse Peclet Number calculated from the average of 56 individual turbulent missing flows (two for each subchannel boundary) obtained from the applicable data was 0.0034. With respect to general turbulent mixing, therefore, the more recent study on the CEA verifies the constancy of the inverse Peclet number for moderately different fuel assembly geometries and confirms the design value of that characteristic.

1.5.3 CONTROL ELEMENT ASSEMBLY DROP TESTS

A series of tests was completed on both single and dual CEAs in a cold water, low pressure facility to satisfy the following objectives:

a. Determine the mechanical and functional feasibility of the CEA type control rod concept.

b. Experimentally determine the relationship between CEA drop time and CEA drop weight, annular clearance between CEA fingers and guide tubes, and coolant flow rate within the guide tube.

c. Experimentally determine the relationship between flow rate and pressure drop within the guide tube as a function of CEA axial position and of finger-to-guide-tube clearance.

d. Determine the effects on drop time of adding a flow restriction or of plugging the flow holes in the lower portion of a guide tube (as might occur under accident conditions).

e. Determine the effects of misalignment within the CEA guide tube system on drop time.

The results of these tests were used as the basis for selecting the final CEA and guide tube geometrics. The tests have demonstrated that the five-finger CEA concept is mechanically and functionally feasible and that the CEA design has met the criteria established for drop time under the most adverse conditions. The testing has also verified that the analytical model used for predicting the drop times gives uniformly conservative results.

The effects on drop time of all possible combinations of frictional restraining forces in the control element drive mechanism (CEDM), angular and radial misalignment of the CEDM, bowing of the guide tubes, and misalignments of the CEA should have been experimentally investigated and defined. The conditions tested simulated all the effects of tolerance buildup, dynamic loadings, and thermal effects. The tests demonstrated that misalignments and distortions in excess of those expected from tolerance buildup or any other anticipated cause would still result in acceptable drop times.

1.5.4 CONTROL ELEMENT DRIVE ASSEMBLY PERFORMANCE TESTS

An accelerated life test of a magnetic jack coupled to a CEA was completed. This test consisted of continuous operation of the mechanism for a total accumulated travel of 32,500 feet at conditions similar to those it will encounter when installed on the operating reactor. The mechanism was operated at a speed of 40 inches per minute. without malfunction or adjustments. In addition, 200 full height drops were completed with all drop times less than 2.5 seconds for 90 percent insertion. Subsequent testing at various conditions was conducted to determine maintenance cycles.

Tests have shown that the magnetic jack type mechanism will operate in the anticipated containment environment after a Design Basis Accident. Among various other tests documented in Reference 1.5-2, a magnetic jack type CEDM, similar to that installed at Unit 2 was verified to be capable of withstanding a complete loss of air cooling for a 4 hour period with the plant at normal operating temperature and pressure (600°F and 2250 psi) without damage to the CEDM and holding the CEA. In addition, the coils stacks were later subjected to a steam environment for 15 minutes without affecting their electrical capabilities.

The design of the CEDM is such that loss of CEDM cooling will not prevent the CEDM from releasing the CEA. The ability of the CEDM to release the rods is not dependent on the cooling flow provided by the CEDM cooling system. Cooling function is only to ensure reliability of the CEDM coil stack.

1.5.5 FUEL ASSEMBLY FLOW TESTS

Velocity and static pressure measurements were made in an oversized model of a fuel assembly to determine the flow distributions present. Effects of the distributions on thermal behavior and margin are to be evaluated, where necessary, with the use of a CE version of the COBRA thermal and hydraulic code (Reference 1.5-1). Subjects investigated include the following:

a. Assembly inlet flow distribution as affected by the core support plate and bottom header plate flow hole geometry: Flow distribution was measured and results indicate that uniform nominal value is achieved within 10 percent of core height. The normal inlet flow distribution arising from the geometric configuration of the core support plate and lower end fitting of the fuel assembly was shown to have an effect on thermal margin which was small enough so that no allowance had to be made in the context of CE current conservative thermal-hydraulic calculational techniques.

b. Assembly inlet flow distribution as affected by a blocked core support plate flow hole: Flow distribution was measured and indicated that flow was recovered to at least 50 percent of the uniform nominal value at an elevation corresponding to 10 percent of core height. Analysis of several of the flow maldistributions arising from the unlikely blockage of a flow hole in the core support plate or from the blockage of one to nine subchannels indicated that flow recovery is rapid enough downstream of the obstruction so that the complete blockage of a core support-plate flow hole or of a single subchannel during 120 percent of full power operation would not result in a W-3 departure from boiling ratio (DNBR) of less than 1.0. The experimental data also indicated that the upstream influence of a subchannel blockage diminished very rapidly in that direction.

c. Flow distribution within the assembly as affected by complete blockage of one to nine subchannels: The flow distributions were measured and indicated very little upstream effect on such blockage, followed by recovery to normal subchannel flow conditions within 10 to 15 percent of core height, depending upon the number of subchannels blocked.

d. Flow distribution below the top header plate, as affected by the header plate and alignment plate flow hole geometry and by the presence of the CEA shroud: Measurements of the flow distribution near the top of the active core demonstrated that there was a negligible effect of the fuel assembly end fitting, alignment plate, and CEA shroud on that distribution.

1.5.6 REACTOR VESSEL FLOW TESTS

Tests were conducted with one-fifth scale models of CE reactors to determine hydraulic performance. The first tests were performed for the Palisades plant which has a reactor coolant system (RCS) similar to that of Millstone Unit 2. The tests investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps operating and various part-loop configurations. Air was used as the test medium. CE has also conducted tests on a one-fourth scale model of the Fort Calhoun reactor using air as the test medium.

Similar one-fifth scale model tests have been performed for Maine Yankee, which has a core similar to that of Millstone Unit 2. These tests were conducted in a cold water loop. All components for the model were geometrically similar to those in the reactor except for the core where 217 cylindrical core tubes were substituted for the fuel bundles. The core tubes contained orifices to provide the proper axial flow resistance.

Flow characteristics for Millstone Unit 2 were determined by taking into consideration similarities between Millstone Unit 2 and other CE reactors in conjunction with the experimental data from the flow model programs.

1.5.7 IN-CORE INSTRUMENTATION TESTS

Tests on in-core thermocouples and flux detectors were performed to ensure that the instrumentation will perform as expected at the temperatures to be encountered and that it does not vibrate excessively and cause excessive wear or fretting. Cold flow testing has been completed on a similar detector cable; no adverse vibrations or wear effects were encountered. Hot flow testing is also complete. After 2,000 hours at 590°F and 2,100 psig in a test loop, no breach of mechanical integrity was observed.

Mechanical tests of the insertion and removal equipment and instrumentation were performed on thimbles of the same approximate configuration as those used on Millstone Unit 2. The top entry in-core instrumentation design provides a means of eliminating the need of handling instrument assemblies separately, thus, minimizing downtime and personnel exposure. A full-scale mockup was built to accommodate three in-core instrumentation thimble assemblies. Major components and subassemblies of the mockup included:

a. An in-core instrumentation test assembly, including the upper guide structure support plate, three thimble guide sleeves, fuel alignment plate, three fuel bundle guide tubes, and the core support plate.

b. A thimble assembly consisting of the instrument plate, three in-core instrumentation thimbles and the lifting sling.

c. An upper guide tube with the guide tube attached to the thimble extension in and the detector cable partially inserted in the guide tube.

Insertion and withdrawal tests were performed to determine the frictional forces of a multi-tube instrument thimble assembly during insertion and withdrawal from a set of fuel bundles. This test simulated the operation that will be performed during the refueling of the reactor. To determine whether jamming of the thimbles would occur during this operation, bending loads were applied to the thimble assembly by tilting the instrument plate in 0.5 degree increments up to a total of five degrees from horizontal. Guide tubes were filled with water. The assembly was raised and lowered approximately five times for each tilt setting. Results showed no discernible difference in the friction forces for the various tilt settings. The tests demonstrated that the repeated insertion and withdrawal of in-core instrumentation thimble assemblies into the fuel bundle guides can be accomplished with reasonable insertion forces.

Life cycle tests were performed to determine if the frictional forces increase as a result of 40 insertions and withdrawals. An automatic timer was installed in the crane electrical circuitry to automatically cycle the thimble assembly between the fully inserted and withdrawn position. The instrument plate was set for five degrees tilt and the assembly was cycled 60 times. The insertion and withdrawal forces were measured during the first and last five cycles. No discernible difference was noticed.

An off-center lift test was performed to determine if the thimble assembly could be withdrawn from the core region while lifting the assembly from an extreme off center position. For a lifting point 11 inches off center, insertion was accomplished without incident. The flexibility of the thimble is such that jamming of the assembly due to off-center lifting does not occur.

Cable insertion tests were performed to determine the forces required to completely insert and withdraw a detector cable from the in-core instrumentation thimble assembly. The guide tube routing included typical bends equal to, or worse than, those found in the reactor. The detector cable was passed through the guide tubing and into a thimble. In all cases, the insertion and withdrawal forces were reasonable for hand insertion.

1.5.8 MATERIALS IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed on the inside wall of the vessel to monitor the change in fracture toughness properties of the material during the reactor operating lifetime. Details of the program are given in Section 4.6.

1.5.9 REFERENCES

1.5-1 Rowe, D. S., “Cross-Flow Mixing Between Parallel Flow Channels During Boiling.” COBRA Computer Program for Coolant Boiling in Rod Arrays, Part 1, BNWL-371, March 1967.

1.5-2 Combustion Engineering Inc., Test Report Number TR-DT-78, dated 8/21/72, "Magnetic Jack Type Control Element Drive Mechanism Design and Test Report."

1.6 IDENTIFICATION OF CONTRACTORS

Originally, The Connecticut Light and Power Company (CL&P), the Hartford Electric Light Company (HELCO), and Western Massachusetts Electric Company (WMECO) (the Owners), and Northeast Nuclear Energy Company (NNECO) were the applicants for the operating license for Millstone Unit 2. At that time NNECO acted as the agent for the owners and was responsible for the design, construction and operation of the plant. However, in 2001, the operating license was transferred to Dominion Nuclear Connecticut, Inc., at which time they became the sole owner and operator of Millstone Unit Number 2.

Combustion Engineering (CE), Inc. was engaged to design, manufacture and deliver the Nuclear Steam Supply System (NSSS) and nuclear fuel for the first core and the first two core reload batches to the site. The NSSS includes the reactor coolant system, reactor auxiliary system components, nuclear and certain process instrumentation, and the reactor control and protective system. In addition, CE furnished technical assistance for erection, initial fuel loading, testing and initial startup of the NSSS.

Bechtel Corporation was engaged as the Engineer-Constructor for this project and as such performed engineering and design work for the balance-of-plant equipment, systems and structures not included under the CE scope of supply. Bechtel was engaged to perform onsite construction of the entire plant with technical advice for installation of the reactor components provided by CE.

The reactor vessel closure head was replaced during refueling outage 16 with a new head assembly fabricated from materials that are less susceptible to Primary Water Stress Corrosion Cracking (PWSCC). The new head assembly was manufactured by Mitsubishi Heavy Industries. Westinghouse/CE was engaged in the design, installation and testing of the head.

The pressurizer assembly was replaced in 2006 with a new assembly fabricated from materials that are less susceptible to PWSCC. AREVA was engaged in the design, fabrication, installation and testing of the replacement pressurizer.

1.6.1 REFERENCES

1.6-1 Millstone Unit 3, Final Safety Analysis Report, Section 13.1 - Organizational Structure.

1.7 GENERAL DESIGN CHANGES SINCE ISSUANCE OF PRELIMINARY SAFETY ANALYSIS REPORT

1.7.1 GENERAL

Since the issuing of the Preliminary Safety Analysis Report (PSAR), a number of changes were made in the design of Millstone Unit 2. These changes improved the operating characteristics and enhance plant safety and reliability. The following reflects changes made up to the time of operating license application.

1.7.2 CONTROL ELEMENT DRIVE MECHANISMS

Magnetic jack drive mechanisms are provided for positioning the control element assemblies (CEA) instead of rack and pinion drive mechanisms. The magnetic jack control element drive mechanism (CEDM) is completely sealed by a pressure boundary, eliminating the need for seals. Motion of the control element drive shaft is accomplished by sequencing five solenoid coils located around the pressure boundary.

Combustion Engineering (CE), Inc., supplied identical CEDMs on previous plants, including Maine Yankee (Atomic Energy Commission (AEC) Docket Number 50-309) and Calvert Cliffs Units 1 and 2 (AEC Docket Number 50-317 and 50-318).

1.7.3 RADIOACTIVE WASTE PROCESSING SYSTEM

1.7.3.1 Clean Liquid Waste Processing System

A closed drains system and a 700 gallon equipment drain sump tank were included in the system to collect liquids containing dissolved hydrogen and fission gases from equipment drains, valve stem leakoffs, and relief valve discharges. The liquid wastes are collected in this tank via the closed drains system. This tank was provided to minimize the release of radioactive gases to the atmosphere without prior processing by the gaseous waste system.

The flash tank was replaced by a packed column-type degasifier utilizing internally generated stripping steam. The degasifier has a better decontamination factor for xenon and krypton than would have been possible with the proposed flash tank.

Plant space and the necessary piping and valves were provided for incorporating two additional demineralizers into the system, if required, based on operating experience.

1.7.3.2 Gaseous Waste Processing System

Four additional waste gas decay tanks were added to the system to allow for a minimum of 60 day decay of all hydrogenated waste gases, including cover gases, collected by the system prior to release to the atmosphere through the Millstone stack.

1.7.4 VITAL COMPONENT CLOSED COOLING WATER SYSTEM

The vital components closed cooling water system was deleted and the components cooled as follows:

Component	Cooling System
Service air compressors and instrument air compressors	Turbine building closed cooling water (interconnecting piping provided to reactor building closed cooling water)
Auxiliary feedwater pump turbine oil cooler	Water being pumped
Diesel generator	Service water
Control room air conditioners	Air

1.7.5 ELECTRICAL

1.7.5.1 AC Power

The station service transformers supply power at 6900V and 4160V via their respective station service busses for large motor loads. Further, the 4160V supplies power to the 480V unit substation transformers for smaller loads.

To preserve redundancy and separation, each motor control center is fed from only one 480 volt load center rather than from two.

1.7.5.2 Diesel Generators

For the change in the diesel engine cooling water supply, see Section 1.7.4.

Additional conditions under which the diesel generators will start automatically are noted in Section 8.3.3.1.

1.7.5.3 DC Supply

A third station battery was added to care for the non safety-related 125 volt DC loads associated with the turbine generator.

Each 125 volt DC distribution panel formerly had a feeder from each of the two station batteries, with diodes to prevent tying the battery buses together. To maintain the independence of redundant sources, the diodes were removed and the DC distribution panels fed from redundant battery buses.

1.7.5.4 Instrument Power

Two 120 volt regulated AC instrument buses were provided (instead of one) to assure redundant power sources for vital instrumentation.

1.7.6 AXIAL XENON OSCILLATION PROTECTION

Automatic initiation of an appropriate protection system for axial xenon oscillation was incorporated into the reactor protective system. This addition provided compliance with the AEC's General Design Criterion 20 as published February 20, 1971, in the Federal Register and as interpreted for preceeding reactors of similar design (see Calvert Cliffs Units 1 & 2 Amendment 15, Question 3.14). The basis for this addition was to provide an automatic protective backup to the operator in the unlikely event he should fail to adjust the full length CEA as required late in core life when axial xenon oscillations may become divergent.

1.7.7 NUMBER OF CONTROL ELEMENT ASSEMBLIES AND DRIVE MECHANISMS

The number of CEAs in the Millstone Unit 2 reactor is 73, compared to 85 CEAs shown in the PSAR design. The number of drive mechanisms was changed from 65 in the PSAR to 69 for Cycle 1. Then, removal of 8 part-length CEAs in 1978 reduced the number of drive mechanisms to 61. This resulted in a net increase in the number of single CEAs (37 to 49) and a net reduction in the number of dual CEAs (40 to 24), thereby providing greater flexibility for optimization of CEA programming and fuel management.

1.7.8 BURNABLE POISON SHIMS

Burnable poison shims were added to the fuel assemblies in Cycle 1, replacing some fuel. These shims permitted lowering of the initial boric acid concentration in the coolant. This provided additional assurance that the moderator temperature coefficient, at power at beginning of life, would not be positive.

1.7.9 STRUCTURES

The following changes have been made:

a. The post-tensioning tendons were encased in galvanized rather than ungalvanized semi-rigid sheaths.

b. The bearing plate material was changed from A-36 to VNT steel.

c. The warehouse area and turbine building were designated Class I structures.

d. All concrete reinforcing steel larger than number 11 was mechanically spliced.

e. Dye penetrant and magnetic particle inspection were not used for liner plate weld quality control.

1.7.10 HIGH PRESSURE SAFETY INJECTION PUMPS

High Pressure Safety Injection (HPSI) pump P-41B (Figure 6.1–1) (Sheet 2) was connected to each of the two suction headers but is normally isolated by valving. This HPSI pump served as a spare and was aligned, process wise and electrically, for operation only when either of the other two HPSI pumps is taken out of service. Two operable HPSI pumps satisfy redundancy requirements for core cooling.

1.7.11 CONTAINMENT PURGE VALVE ISOLATION ACTUATION SYSTEM

Containment Purge Valve Actuation System was changed from two-out-of-four to one-out-of-four logic. See Sections 7.3.2.3 and 7.5.6.3 for details.

1.7.12 CONTROL ELEMENT DRIVE SYSTEM

The Control Element Drive System (CEDS) was modified to include a CEA Motion Inhibit feature which acts to help the operator assure that limits on CEA position are not exceeded. The CEDS is described in Section 7.4.2.

1.8 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SPECIAL INTEREST ITEMS [THIS SECTION PROVIDES HISTORICAL INFORMATION PROVIDED TO THE ACRS AT THE TIME OF INITIAL LICENSING AND WAS NOT INTENDED TO BE UPDATED.]

1.8.1 GENERAL

This section describes the status of programs initiated to investigate the items which were identified by the Advisory Committee on Reactor Safeguards (ACRS) as being of special interest and pertaining to all large water-cooled power reactors up to the time of application for an operating license.

In carrying out these programs, information derived from research and development activities of the Atomic Energy Commission (AEC) and other organizations in the nuclear power industry were considered.

1.8.1.1 Ability of Fuel to Withstand Transients at End of Life and Experimental Verification of Maximum Linear Heat Generation Rate

The fuel cladding was designed to limit the transient stresses to two-thirds of the unirradiated value of the yield stress even during a depressurization transient near the end of life, when the internal gas pressure is highest.

Experimental verification of the maximum linear heat generation rate employed in the Millstone Unit 2 design was discussed in the original FSAR submitted at the time of application for an Operating License. Numerous irradiation tests, which bracket the design of these units, were performed, including those in the Westinghouse test reactor, the Shippingport blanket irradiations, the mixed oxide irradiations in the Saxton reactor, the zirconium clad UO_2 fuel rod evaluations in the Vallecitos boiling water reactor, the large speed blanket reactor rod irradiations, the center melting irradiations in Big Rock, Peach Bottom 2 irradiations, and NRX irradiations (AECL-Canada). In these tests, fuel rods similar to those employed in the design of the Millstone Unit 2 core were successfully irradiated to fuel burnups varying from very short term tests up to 60,000 MWD/MTU and at linear heat rates ranging from 5.6 up to 27.0 kW/ft.

1.8.1.2 Fuel Integrity Following a Loss-of-Coolant Accident

The ACRS had asked that information be developed to show that the “...melting and subsequent disintegration of a portion of fuel assembly...will not lead to unacceptable conditions.” They referred specifically to the “...effects in terms of fission product release, local high pressure production, and the possible initiation of failure in adjacent fuel elements...”.

Inquiry was made as to whether accident conditions that might occur which cause clad temperatures to reach such high temperatures that embrittlement occurs, and whether subsequent quenching operations will cause the embrittled portions to disintegrate and thereby prevent a sufficient flow of emergency core coolant to the remainder of the core.

Fuel damage of the magnitude indicated is prevented by the inherent nuclear and thermal characteristics of the UO_2 core and by the provision of engineered safety features (ESF).

With regard to the nonexcursion mechanisms leading to the conditions described by ACRS, the following two conditions might be conjectured:

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 Number 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

A comprehensive quality assurance program has been established to assure that Millstone Unit 2 is designed, fabricated, and constructed in accordance with the requirements of applicable specifications and codes. The 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 Quality Assurance Program (QAP) 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 Quality Assurance Program Description Topical Report. That information is incorporated herein by reference.

Baseline inspection and subsequently in-service inspections are performed and are further discussed in Section 4.6.6.

1.8.1.4 Separation of Control and Protective Instrumentation

In addition to any redundancy and separation provided for control or for protective instrumentation, the control and protective instrumentation are independent of each other. Control action and protective action derived from the same process variable are generated by separate instrumentation loops. Malfunction of a single control instrumentation loop cannot impair the operation of the protective instrumentation loop and conversely malfunction of the protective instrumentation loop does not affect operation of the control loop. The instrumentation for a single protective and a single control channel may be located adjacent to one another, and their circuits may be routed in the same cable tray, but each is capable of performing its function independently of the other. Further discussion is provided in Chapters 7 and 8.

1.8.1.5 Instrumentation for Detection of Failed Fuel

Early detection of the gross failure of fuel elements permits early applications of action necessary to limit the consequences.

Based on a study of the expected fission and corrosion product activities in the reactor coolant, it was concluded that the gross gamma plus specific isotope monitor provides a simple and reliable means for early detection fuel failures.

The design bases of the detection system include the following:

a. Trends in fission product activity in the reactor coolant system (RCS) (specifically Rb-88) are used as an indication of fuel element cladding failures.

b. There is a time delay of less than five minutes before the activity, emitted from a fuel element cladding failure, is indicated by the instrumentation. This time delay is a function of the location of the monitor.

c. The information obtained from this system will not be used for automatic protective or control functions or detection of the specific fuel assembly (or assemblies) which has failed.

d. The high activity alarm will be supplemented with radiochemical analysis of the reactor coolant for fission products to provide positive identification of a fuel element failure.

The location and operation of the detector, designated as a process radiation monitor, is described in Sections 7.5.6.3 and 9.2.2.

Note: This section provides historical information provided to ACRs at the time of initial Licensing and was not intended to be updated.

1.8.1.6 Effects of Blowdown Forces on Core and Primary System Components

The dynamic response of reactor internals resulting from hydrodynamic blowdown forces under a postulated LOCA condition was the subject of a CE topical report which contained a complete description of the theoretical basis for methods of analysis for the various reactor components, as well as documentation of computer programs and the respective analytical structural models.

Reactor vessel internal structures were analyzed to ensure the required structural integrity during abnormal operating conditions, including the effects of blowdown, pressure drop and buckling forces. For the LOCA, the CEFLASH-4 computer program was used to define the flow transient and the WATERHAMMER program determines the corresponding dynamic pressure load distribution. The dynamic response of the reactor vessel internals to the space and time-dependent pressure loads were obtained through the use of a number of structural dynamic analysis codes. Lateral and vertical dynamic response of the internals were considered, as well as the transient response and dynamic buckling of a core support barrel in shell modes. Both the CEFLASH-4 and WATERHAMMER models were evaluated against the LOFT program results.

The loads resulting from the LOCA condition were added to the loads resulting from normal operation and the design basis earthquake (DBE) for each critical component and the component deflections and stresses analyzed to ensure compliance with the criteria specified in Section 4.2.

1.8.1.7 Reactor Vessel Thermal Shock

Sufficient emergency core cooling water is available to flood the core region in the event of a major LOCA. The Millstone Unit 2 design uses a section of each of the RCS cold legs to conduct the water from the safety injection nozzles to the reactor vessel. This water then flows into the downcomer annulus and into the lower plenum of the reactor vessel before flooding the core itself. Analytical investigations were performed to provide assurance that the resultant cooling of the irradiated inner surface of the thick-walled reactor vessel will not induce or propagate cracks sufficient to cause the reactor vessel to fail.

An analytical evaluation of pressurized thermal shock effects in CE’s NSSS was issued by CE in December 1981 (CEN-189). The limiting case is a small break LOCA with the assumption of concurrent loss of all feedwater. For Millstone Unit 2, it was found that crack initiation would not occur during this limiting transient throughout the unit's design life (32 EFPY).

Subsequently, the Pressurized Thermal Shock Rule (10 CFR 50.61, 1986) was used for embrittlement shift prediction. The results confirmed that the reactor vessel was fully able to withstand a postulated pressurized thermal shock imposed by the ECCS through the unit's design life.

1.8.1.8 Effect of Fuel Rod Failure on the Capability of the Safety Injection System

CE conducted experimental and analytical investigations of fuel-rod failures under simulated LOCA conditions. The analytical work provided indications of the actual conditions to be expected in the core during a transient, in terms of potential clad heating rates, internal pressures and transient duration. The experimental work applied these parameters in various combinations to establish the nature of fuel-rod deformation which might occur under accident conditions. This subject 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 Systems for Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-1.

1.8.1.9 Preoperational Vibration Monitoring Program

A preoperational vibration monitoring program (PVMP) was completed for the Palisades reactor internals. Results of this program were submitted to the AEC by CE Report CENPD-36. Additional PVMPs were developed for both the Maine Yankee and Fort Calhoun reactor internals. In keeping with the requirements for prototype vibration test programs, predictions of hydraulic forcing functions and structural response were made for the Maine Yankee and Fort Calhoun reactor internals and correlated to test program measurements. Vibration test data from all three reactors was used in demonstrating the adequacy of the Millstone Unit 2 reactor vessel internals to sustain flow-induced vibration effects. The vibration test data available, together with appropriate analyses, permitted the assessment of design or fabrication differences existing among the subject reactors as they related to the vibrational response characteristics of the Millstone Unit 2 reactor internals. A comparison of applicable design parameters for the Palisades, Fort Calhoun, Maine Yankee and Millstone Unit 2 reactors as of the time of application for operating license is presented in Table 1.8-1.

The analytical methods which formed the basis for the CE vibration response predictions were provided in the Maine Yankee and Fort Calhoun vibration monitoring programs submittals. Palisades, Maine Yankee and Fort Calhoun Flow Model Test reports and a description of the methodology utilized to relate these data to in-reactor forcing functions were provided, as well as a description of the structural response computer code.

1.8.1.9.1 Basis of Program

The suitability of using PVMP data from Palisades, Omaha and Maine Yankee as a composite prototype was based on the following:

a. Reactor internals structural response and LOCA hydraulic loadings could be adequately predicted with computer programs available, and the methods and procedures will be provided and justified.

b. The hydraulic forcing function predicting method was provided and justified. The forcing function method was verified by measurements in the prototype(s).

c. Additional instrumentation to measure or derive forcing functions was added to the Fort Calhoun reactor in accordance with Regulatory Guide 1.20 (formerly Safety Guide 20).

The prediction methods and procedures were used to predict the responses (amplitude and frequency) for the Fort Calhoun PVMP.

d. The Maine Yankee and Fort Calhoun PVMP results were satisfactory, satisfying AEC licensing requirements for all CE reactor plants which had either construction or operating permits, providing the configuration and flow modes were similar as specified in Regulatory Guide 1.20 (formerly Safety Guide 20).

e. CE provided predictive methodology and predicted and limiting values of response (acceptance criteria) on the Maine Yankee program. The program results were provided on a timely basis in accordance with the Regulatory Guide 1.20 (formerly Safety Guide 20).

f. CE submitted a report on the LOCA dynamic analysis methods and procedures.

1.8.1.9.2 Millstone Unit 2 Program

The PVMP to be conducted for Millstone Unit 2 reactor internals was consistent with those portions of the former Safety Guide 20 (after replaced by Regulatory Guide 1.20), which addressed nonprototype reactors.

The following was the PVMP plan for Millstone Unit 2. As noted above, this program was contingent upon the results to be obtained from Maine Yankee and Fort Calhoun PVMP.

1. The reactor internals important to safety were be subjected during the preoperation functional testing program to all significant flow modes of normal reactor operation and under the same test conditions conducted on the Palisades, Fort Calhoun, and Maine Yankee designs.

   The test duration was at least as long as that conducted on the Palisades, Fort Calhoun and Maine Yankee designs.

2. Following completion of the preoperational functional tests, the reactor internals were removed from the reactor vessel and visual and nondestructive examination of the reactor internals was conducted. The areas examined included:

   a. All major load bearing elements of the reactor internals relied upon to retain the core structure in place;

   b. The lateral, vertical, and torsional restraints provided within the vessel;

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals;

d. Those other locations on the reactor internal components which were examined on the Palisades, Fort Calhoun, and Maine Yankee designs;

e. The interior of the reactor vessel for evidence of loose parts or foreign material.

A summary of the PVMP inspections described above was submitted after the completion of the inspection and tests in a report.

It should be pointed out that the reactor thermal shield was removed from the lower internals assembly because of the damage suffered due to excessive vibratory movement. An evaluation was performed to assess the effects of thermal shield removal on the vibratory response of the rest of reactor internals. It was concluded that the effect would be minimal and that the conclusions of the PVMP were still valid.

1.8.2 SPECIAL FOR MILLSTONE UNIT 2

1.8.2.1 Release of Radioactivity in Case of Damaged Fuel Assemblies in Spent Fuel Pool

In the event of release or radioactivity resulting from damaged fuel in the spent fuel pool, the auxiliary exhaust system (AES) which is described in Section 9.9.8, diverts the effluent through the enclosure building filtration system (EBFS) charcoal filters prior to release through the Millstone stack. The AES maintains the fuel handling area under a negative pressure to limit uncontrolled release of radioactivity.

1.8.2.2 Hydrogen Control

The independent systems in the hydrogen control systems monitor and mix hydrogen in the containment following a LOCA (see Section 6.6). Each is a full capacity, completely redundant, independent system. Air to operate the hydrogen monitoring system CIV’s is provided by the instrument air system with a backup air bottle system that is designed to meet single failure criteria. Two, full capacity hydrogen purge systems not credited in accident analyses are provided. The hydrogen recombiner system has no mitigating function.

1.8.2.3 Common Mode Failures and Anticipated Transients Without Scram

CE analyzed the response of pressurized water reactors which are typical of Millstone Unit 2 to demonstrate the diversity of the reactor protective system in mitigating common mode failures and the response of the plant to anticipated transients without scram (ATWS). Results of these studies were submitted to the AEC as topical reports.

CE Report CENPD-11, entitled “Reactor Protection System Diversity” was submitted on March 4, 1971. This report evaluated systematic, nonrandom, concurrent failures, (i.e., common mode

failures) of redundant devices not considered credible based on quality assurance in design, qualification testing, and periodic testing that common mode failure could disable all instrument channels which measure a given process parameter, the report, nevertheless, addresses this type of failure. Monitoring of the condition by diverse means or principles enables a protection system to withstand common mode failures. The evaluations included the following accidents: control element assembly (CEA) withdrawal, CEA drop, loss of reactor coolant flow, excess load, loss of load and loss of feedwater. The results of the study demonstrated that the diversity of the reactor protective system is such that gross fuel damage or consequential failures in the RCS or in the main steam system will not occur for any of the accidents analyzed.

A draft of the CE report, entitled “Topical Report on Anticipated Transients Without Scram” (Proprietary) was submitted to the AEC on January 10, 1972. Evaluations were performed in this report based upon the assumption that no CEA are inserted into the core during the course of the following transients: CEA withdrawal, CEA drop, idle loop startup, loss of flow, boron dilution, excess load, loss of load, loss of feedwater, sample line break, and pressurizer safety valve failure. The transient resulting from loss of normal onsite and offsite power was also analyzed but with a conservative one percent negative reactivity insertion assumed following reactor trip signal generation, since for this case the failures which initiate the transient would also remove power from the control element drive mechanism (CEDM), allowing the CEAs to insert. The final report, with results and their applicability to Millstone Unit 2, was submitted to the AEC.

1.8.3 REFERENCES

1.8-1 Millstone Unit 3, Final Safety Analysis Report, Section 13.1 - Organizational Structure.

TABLE 1.8-1 COMPARISON OF PREOPERATIONAL VIBRATION MONITORING PROGRAM DESIGN PARAMETERS

<Parameter>	Palisades	Fort Calhoun	Maine Yankee	Millstone Unit 2
R_{mean}, inches	75-7/8	61-5/16	75.25	75.25
Upper CSB: t, inches	2	2	2.5	2.5
Upper CSB: L, inches	109.25	101-3/8	135-5/8	141.75
Upper CSB: $R_{mean,}$ inches	75-5/8	61-1/16	74-7/8	74-7/8
Middle CSB: t, inches	1.5	1.5	1.75	1.75
Middle CSB: L, inches	166.75	166-1/8	144.75	148.75
Middle CSB: $R_{mean,}$ inches	75-3/8	60-11/16	74-5/8	74-5/8
Lower CSB: t, inches	2	2.25	2.25	2.25
Lower CSB: L, inches	38.5	35-5/8	38	38
Lower Cylinder ID, inches	Integral	Integral	141	141
Core Cylinder OD, inches	Integral	Integral	145	145
Support Cylinder L, inches	Integral	Integral	42	42
Structure Supported	Integral	Integral	CSB Flange	CSB Flange
Core Shroud Support	Bolted to CBS	Bolted to CBS	Bolted to CBS	Bolted to CBS
Core Shroud: $R_{mean,}$ inches	73.5	59-1/16	72-5/8	72-5/8
Core Shroud: Cylinder t, inches	2	1.5	2	2
UGS: L, inches	15	24	24	24
UGS: Beams inches	18 by 1.5	24 by 1.5	24 by 1.5	24 by 1.5
UGS: Plate t, inches	3	3.25	4	4
Thermal Shield	No	Yes	Yes	Yes

TABLE 1.8-1 COMPARISON OF PREOPERATIONAL VIBRATION MONITORING PROGRAM DESIGN PARAMETERS

<Parameter>	Palisades	Fort Calhoun	Maine Yankee	Millstone Unit 2
Number of Loops	2	2	3	2
Design Minimum. Flow, 10^6 lbm/hr	125	71.7	122	139
Inlet Design Temperature, F	548	547	546	544
Inlet ID, inches [(a)]	35-1/8	28.75	39	35-3/16
Outlet ID, inches [(a)]	48-5/8	37	40	48-1/8
Inlet Pipe Velocity, ft/sec	37.7	33.7	39.2	41.6
Downcomer Velocity, ft/sec	19.6	25.2	24.9	26.7
Core Inlet Velocity, ft/sec	12.2	12.4	13.0	15.4
Outlet Pipe Velocity, ft/sec	41.4	41.3	42.6	46.5

(a)These IDs are measured at the inside wall of the reactor vessel as shown for the Millstone 2 reactor vessel in Figure 4.3–1.

CSB = Core Support Barrel

UGS = Upper Guide Structure

Velocity = Design Minimum Velocity

1.9 TOPICAL REPORTS

In support of the Final Safety Analysis Report, various “topical reports” prepared by Combustion Engineering, Inc., and Bechtel Corporation were referenced throughout this document. A list of “topical reports” as of the time of application for operating license is given in Table 1.9-1.

TABLE 1.9-1 TOPICAL REPORTS

Combustion Engineering, Inc.

Title	Millstone Unit 2 Original FSAR Section
ASME paper 68-WA/HT-34, December 1968 Winter Annual Meeting	1.8.1.2
Statement of Affirmative Testimony and Evidence of Combustion	1.8.1.2
Engineering in the matter of Rulemaking Hearing for the Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-1	1.8.1.8
Dynamic Analysis of Reactor Vessel Internals Under Loss of Coolant Accident CENPD-42-3 (Submittal to AEC in July 1972)	1.8.1.6
Thermal Shock Analysis of Reactor Vessels Due to Emergency Core Cooling System Operation, A-68-9-1, March 15,1968, submitted as part of Amendment 9 to the Maine Yankee license application	1.8.1.7
Experimental Determination of Limiting Heat Transfer Coefficients During Quenching of Thick Steel Plates in Water, A-68-10-2, December 13, 1968	1.8.1.7
Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel During Emergency Core Cooling, A-70-19-2, January 1970	1.8.1.7
Palisades Precritical Vibration Monitoring Program, CENPD-36	1.8.1.9
Precritical Vibration Monitoring Program, CENPD-55	1.8.1.9
Reactor Protective System Diversity, CENPD-11, February 1971	1.8.2.3
Topical Report on Anticipated Transients Without Scram, CENPD-41	1.8.2.3
INTHERMIC, A Computer Code for Analysis of Thermal Mixing, CENPD-8	3.5.3
COSMO IV, A Thermal and Hydraulic Steady State Design Code for Water Cooled Reactors, CENPD-9	3.5.3
Seismic Qualification of Category I Electric Equipment for Nuclear Steam Supply Systems, CENPD-61	7.2.6.3

TABLE 1.9-1 TOPICAL REPORTS (CONTINUED)

Bechtel Corporation

Title	Millstone Unit 2 Original FSAR Section
Consumer Power Company Palisades Nuclear Power Plant Containment Building Liner Plate Design Report, B-TOP-1 (submitted to AEC in October, 1969)	5.2.4.5
Full-Scale Buttress Test for Prestressed Nuclear Containment Structures, BC-TOP-7	5.2.3.3.3
Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants, BN-TOP-1	5.2.9.1
Design for Pipe Break Effects, BN-TOP-2 (REV. 1)	Question 4.16

1.10 MATERIAL INCORPORATED BY REFERENCE

The following is a list of material incorporated by reference in the Final Safety Analysis Report [(1)]:

1. Millstone Unit 2 Technical Requirements Manual (TRM).

2. As identified in the List of Figures, the engineering controlled plant drawings that are, coincidentally, MPS-2 FSAR Figures.

3. The Quality Assurance Program Description (QAPD) Topical Report.

(1) Information incorporated by reference into the Final Safety Analysis Report is subject to the update and reporting requirements of 10 CFR 50.71(e) and change controls of 10 CFR 50.59 unless separate NRC change control requirements apply (e.g., 10 CFR 50.54(a)).

1.A AEC GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

10 CFR PART 50 APPENDIX A

On February 20, 1971, the Atomic Energy Commission published in the Federal Register the General Design Criteria for Nuclear Power Plants. Prior to this date, proposed General Design Criteria for Nuclear Power Plants as issued on July 11, 1967, in the Federal Register were in effect. Before issuance of the construction permit for Millstone Unit 2, discussions reflecting the design intent in consideration of the 1967 proposed criteria were submitted in the PSAR. Design and construction was thus initiated and has been completed based upon the 1967 proposed criteria.

Since February 20, 1971, the applicants have attempted to comply with the intent of the newer General Design Criteria to the extent possible, recognizing previous design commitments. The extent to which this has been possible is reflected in the discussions of the 1971 General Design Criteria which follow.

CRITERION 1 - QUALITY STANDARDS AND RECORDS

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

Discussion of the quality standards for those structures and components which are essential to the prevention of incidents which would affect the public health and safety or to mitigation of their consequences are presented in appropriate sections of the FSAR. The quality assurance program in effect to assure that these structures, systems, and components will satisfactorily perform their safety functions is discussed in Section 12.8.

For example, components of the safety injection and containment cooling systems are designed and fabricated in accordance with established codes and/or standards as required to assure that their quality is in keeping with the safety function of the component. It is not intended, however, to limit quality standards requirements to this list.

High Pressure Injection, Low Pressure Injection, and Containment Spray Pumps

a. Surfaces of pressure retaining materials for the high and low pressure safety injection pumps were examined by liquid penetrant techniques in accordance with

the provisions of ANSI-B31.1, Paragraph 136.5.3(d). Surfaces of pressure retaining materials for the containment spray pumps were examined by dye penetrant techniques in accordance with the provisions of Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968. Casings for all three types of pumps have been hydrostatically tested to at least 1.5 times the design pressures.

b. Pressure containing butt welds for the safety injection pumps have been radiographed in accordance with Section VIII of the ASME Code, Paragraph UW-51.

c. The pump supplier submitted certified mill test reports of pressure containing materials.

d. At least one pump of each type has been hydraulic-performance tested for capacity and head, in accordance with the requirements of the Hydraulics Institute.

e. The pump seals have been designed to provide a high degree of assurance of their proper operation, including compatibility of seal materials with water chemistry conditions and minimum dependence on externally supplied cooling water.

f. Pump drive motors conform to NEMA Standards, MG-1.

Safety Injection Tanks

ASME Code, Section III, Class C.

Safety Injection and Containment Spray System Motor-Operated Valves and Control Valves

a. The design criteria for pressure containing parts is in accordance with ANSI B16.5.

b. Radiographic inspection of pressure containing butt welds has been performed in accordance with the requirements of ASME Code, Section VIII.

c. Certified mill test reports of pressure containing materials were provided by the supplier.

d. Pressure containing parts were hydrostatically tested in accordance with MSS-61.

e. Isolation valves are designed, fabricated, and tested in accordance with Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968. Control valves are designed, fabricated, and tested in accordance with ASME Code Section III, Nuclear Power Plant Components, Class II, 1971.

Containment Coolers

a. The cooling coils are similar to a representative section of a coil which was tested under the maximum environmental conditions which would exist following a loss-of-coolant accident (LOCA). The test results demonstrated that the full size coil assembly would be capable of removing the required heat load. These data are filed with the AEC in Topical Report W-CAP-7336-L.

b. The cooling coils are tested in accordance with ASME Code, Section VIII.

c. Air moving equipment, including fan motors, were designed to standards of the Air Moving and Conditioning Association, AMCA-211A.

d. A fan and motor combination were satisfactorily tested to prove their ability to operate under the conditions which would exist within the containment following a LOCA. These data will be presented to the AEC in Topical Report W-CAP-7829. The motor insulation and internal cable splice are filed in Topical Reports W-CAP-7343-L and W-CAP-9003, respectively.

e. Piping from the fan coolers to the containment penetrations was designed in accordance with the provisions of ANSI B31.1.0. The penetrations piping was designed to ANSI B31.7, Class II and the penetration isolation valves to the ASME Pump and Valve Code, Class II.

f. Valves, other than the penetration isolation valves, were designed in accordance with ANSI B31.1.0 and ANSI B16.5. Manually operated butterfly valves were in accordance with AWWA-C504.

Shutdown Heat Exchangers

a. Pressure containing materials were tested and examined per ASME Code, Section III, Class C.

b. Heat transfer design and physical design are in accordance with TEMA standards.

c. Certified mill test reports of pressure containing materials were provided by the supplier.

d. Radiographic inspection of pressure containing butt welds was performed in accordance with the requirements of ASME Code, Section III, Class C.

e. Pressure containing parts were hydrostatically tested in accordance with ASME Code, Section III, Class C.

All tests and inspections are reviewed during material procurement and fabrication of the components to assure conformance with the quality control techniques of the applicable codes and standards.

The appropriate sections in the FSAR discuss the specific codes and standards invoked in fabricating or erecting the structures, systems, and components important to safety.

Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained for the life of the plant. (See Section 12.8).

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

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

All structures, systems, and components important to safety have been designed to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The most severe natural phenomena which are considered and discussed in other sections of this FSAR are as follows:

a. Earthquakes / SeismologySection 2.6

a. Wind and Tornadoes / MeteorologySection 2.3

a. Floods / HydrologySection 2.5.4

Appropriate natural phenomena are considered in the designs of structures, systems, and components. Accepted standards for the forces imposed by natural phenomena are used in the design.

A general description of the seismic analysis program is found in Section 5.8. Additional information on major structure design against the effects of natural phenomena is included in the following sections:

Containment Structure Section 5.2
Enclosure Building Section 5.3
Auxiliary Building Section 5.4

Turbine Building Section 5.5
Intake Structure Section 5.6
Reactor Vessel Internals Appendix 3.A
Reactor Coolant System Appendix 4.A

CRITERION 3 - FIRE PROTECTION

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

Millstone Unit number 2 structures, systems, and components important to safety are designed and located to minimize the probability and effects of fires. Fire protection systems (active and passive) have been provided to assure that all possible fires are detected, controlled, and extinguished.

Fire protection and detection systems and components are designed and installed in accordance with applicable requirements of the National Fire Protection Association (NFPA). In areas where combustible material may exist, fixed fire detection and suppression are generally provided (Section 9.10).

Fire detection and fire suppression systems of appropriate types and capacities are designed to minimize the adverse effects of fires on structures, systems, and components important to safety. In some areas, portable extinguishers are used in lieu of water suppression systems. In areas such as the D.C. equipment rooms, a Halon suppression system is used in lieu of fixed water suppression to assure that sensitive electronics are not affected by water spray.

Fire fighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the capabilities of any structure, system, or component important to safety.

In areas where water may cause damage to safety equipment, such as vital electrical panels or the emergency diesel generators, either shielding is provided or the water suppression system is designed such that its actuation does not affect the safety systems it protects (pre-action sprinkler system, manual activation, shielding, etc.).

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 Section 5.1.2) for the loading conditions stated in Sections 5.2.2, 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 Sections 5.2.2.1.6, 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 5.2.5.1.

The design loads for the containment and major component supports to ensure a safe shutdown after a loss-of-coolant accident are described in Section 5.2.2.1.3.

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 Section 5.2.4.3. Missile protection and pipe whipping protection criteria for these systems and components are given in Sections 5.2.5.1 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, were reviewed and approved by the commission. Subsequent to the commission review and approval, weld overlays were applied to dissimilar metal welds (DMWs) at the shutdown cooling, the safety injection and the pressurizer surge nozzles. A revised LBB analysis was performed for these nozzles (see Reference A.30). 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:

Core barrel snubbers, core barrel stabilizer blocks
Reactor vessel core support ledge
Reactor Cavity Seal, Neutron Shielding
Pressurizer Blockhouse
Protection of Closed Systems
RBCCW piping

Steam Generator Blow Down piping
Steam Generator Blow Down sampling piping

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

Both the auxiliary and the turbine buildings of Millstone Unit 2 are structurally connected to their respective Millstone Unit 1 buildings. The combined buildings are isolated in the lateral direction as discussed in Section 5.4.1 (auxiliary building) and Section 5.5.1 (turbine building). All vertical loads which may interact between Millstone Unit 1 and Millstone Unit 2 portions of the buildings were investigated to ensure that they will function safely under all design conditions.

The Millstone Unit 2 Condensate Polishing Facility is located in Warehouse Number 5, which is situated North of the Millstone Unit 2 Turbine Building and South of the Millstone Unit 3 Condensate Polishing Facility and Auxiliary Boiler Building.

A list of shared facilities appears in Section 1.2.13.

The safe shutdown of any unit will not be impaired by the failure of the facilities and systems which are shared.

CRITERION 10 - REACTOR DESIGN

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

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

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

a. The value of the departure from nucleate boiling ratio (DNBR) will not be less than its design limit to ensure that fuel failure does not occur.

b. The peak temperature in the fuel will be less than the melting point of irradiated UO_2 (considering effects of irradiation on melting point).

c. The maximum primary stresses in the zircaloy fuel clad shall not exceed two-thirds of the minimum yield strength of the material at the operating temperature.

d. Net unrecoverable circumferential strain shall not exceed 1 percent as predicted by computations considering clad creep and fuel-clad interaction effects.

e. Cumulative strain cycling usage, defined as the sum of the ratios of the number of cycles at a given effective strain range (E) to the permitted number (N) at that range shall not exceed 1.0.

f. The fuel rod will be designed to prevent gross clad deformation under the combined effects of external pressure and long term creep.

The thermal margins during normal operation ensure that the minimum thermal margins during anticipated operational occurrences do not exceed the design basis. The DNBR limit ensures a low probability of occurrence of DNB.

The occurrence of DNB does not necessarily signify cladding damage; it represents a local increase in temperature which may or may not cause thermal damage, depending upon severity and duration.

The design is adequate to satisfy the design bases in the event of a reactor coolant system depressurization transient at the end of a fuel cycle.

Limitation of fuel burnup will be determined by material rather than nuclear considerations. See references in Chapter 3. Sufficient margin is provided in this core design to allow for the ratio of peak-to-average burnup.

CRITERION 11 - REACTOR INHERENT PROTECTION

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

The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operating range will be a decrease in reactivity; i.e., the inherent nuclear feedback characteristics will not be positive.

The reactivity coefficients for this reactor are listed in Table 3.4-2 and are discussed in detail in Section 3.4.3.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

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

The reactor core is designed not to have sustained power oscillations. If any power oscillations occur, the control system is sufficient to suppress such oscillations.

The basic stability of a pressurized water reactor with UO_2 fuel is due to the fast acting negative contribution to the power coefficient provided by the Doppler effect.

Any trend toward xenon oscillations which may occur in the core are controlled and suppressed by movement of the control element assemblies (CEAs) so that the thermal design bases are not exceeded. Xenon oscillations are characterized by long periods and slow changes in power distribution. The nuclear instrumentation will provide the information necessary to detect these changes.

Xenon stability analysis for Millstone Unit 2 is discussed in Section 3.4.5. The reactor protective system is discussed in Section 7.2.

The reactor protective system automatically trips the reactor if axial xenon oscillations are permitted to approach unsafe limits (Sections 7.2.3.3.10 and 1.7.6).

CRITERION 13 - INSTRUMENTATION AND CONTROL

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

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

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

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

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

b. Two independent CEA Position Indicating Systems.

c. Manual control of reactor power by means of CEA's.

d. Manual regulation of coolant boron concentrations.

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

Instrumentation measures temperatures, pressures, flows, and levels in the main Steam System and Auxiliary Systems and is used to maintain these variables within prescribed limits.

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

The containment pressure and temperature instrumentation is designed to monitor these parameters during normal operation and the full range of postulated accidents.

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

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

Reactor coolant system components are designed in accordance with the ASME Code, Section III, Pump and Valve Code (reactor coolant system pumps), and ANSI B31.7 (see Section 4 for codes and effective dates). Quality control, inspection, and testing as required by these standards and allowable reactor pressure-temperature operations ensure the integrity of the reactor coolant system.

The reactor coolant system components are considered Class I for seismic design.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

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

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

The operating conditions established for the normal operation of the plant are discussed in the FSAR and the control systems are designed to maintain the controlled plant variables within these operating limits, thereby ensuring that a satisfactory margin is maintained between the plant operating conditions and the design limits.

The reactor protective system functions to minimize the deviation from normal operating limits in the event of certain anticipated operational occurrences. The results of analyses show that the design limits of the reactor coolant pressure boundary are not exceeded in the event of such occurrences.

CRITERION 16 - CONTAINMENT DESIGN

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

The reactor containment structure, described in Section 5.2, consists of a prestressed concrete cylinder and dome with a reinforced concrete base. A one-quarter inch thick welded steel liner plate is attached to the inside face of the concrete to provide a high degree of leak tightness. Designed as a pressure vessel, the containment structure is capable of withstanding all design postulated accident conditions including a loss-of-coolant accident (LOCA). All containment penetrations are sealed as described in Section 5.2.6. Isolation valves are provided for all piping systems which penetrate the containment, as described in Section 5.2.7.

As an extra measure of safety, an enclosure building completely surrounds the containment. In the event of an accident, the enclosure building filtration region (EBFR), described in Section 6.7.2, is maintained at a slightly negative pressure to preclude leakage to the environment. Potential leakage from the containment is channeled into the enclosure building filtration system as described in Section 6.7. Throughline leakage that can bypass the EBFR is discussed in Section 5.3.4.

CRITERION 17 - ELECTRIC POWER SYSTEMS

> An on site electric power system and an off site electric power system are provided to permit functioning of structures, systems, and components important to safety. The safety

function for each system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded as a result of anticipated operational occurrences; and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

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

Electric power from the transmission network to the on site electric distribution system is supplied by two physically independent circuits (not necessarily on separate rights-of-way), designed and located so as to minimize to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits is designed to be available in sufficient time following a loss of all on site AC power supplies and the other off site electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits is designed so it is available within a few seconds after a loss-of-coolant accident (LOCA) to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions are included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the transmission network, or from the on site electric power supplies.

The off site power supplies system is described in Sections 8.1 and 8.2. The preferred source of auxiliary power for unit shutdown is from or through the reserve station service transformers. System interconnection is provided by four 345 kV circuits. These transmission lines are on a single right-of-way with each line installed on an independent set of structures. A description of the structure routing configuration is described in Section 8.1.2.1.

The combination breaker-and-a-half and double breaker-double bus switching arrangement in the 345 kV substation includes two full capacity main buses. Primary and backup relaying are provided for each circuit along with circuit breaker failure protection. These provisions permit the following:

a. Any circuit can be switched under normal or fault conditions without affecting another circuit.

b. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.

c. Short circuits on any section of bus are isolated without interrupting service to any element other than those connected to the faulty bus section.

d. The failure of any circuit breaker to trip within a set time initiates the automatic tripping of the adjacent breakers and thus may result in the loss of a line or generator for this contingency condition; however, power can be restored to the good element in less than eight hours by manually isolating the fault with appropriate disconnect switches.

Overhead lines from the switchyard to the reserve station service transformers are separated at the switchyard structure and are carried on separate towers. These transformers are located near each Unit, and are physically isolated from the normal station service transformers and from the main transformers.

In the event of loss of power from the normal station service transformer, there is an immediate automatic transfer of auxiliary loads to the Unit 2 reserve station service transformer. In the unlikely event that power is not available from this source, and from the On site Emergency Diesel mentioned below, the operator can manually connect emergency bus A-5 (24E) to Unit 3 bus 34A or 34B. By means of interlocked circuit breakers, the Unit 2 post accident loads can be fed from this source.

The on site power supply system is described in Sections 8.3 and 8.5. Two full capacity, separate and redundant batteries are provided for all DC loads and for 120 volt AC vital instrument loads.

In the event that off site power is not available when needed, a “start” signal is given to both emergency diesel generators (DG). These generators and their auxiliaries are entirely separate and redundant, and each generator feeds one 4,160 volt emergency bus. A generator is automatically connected to its bus only if there is no bus voltage and only if the dead bus did not result from protective relay action.

The electric power distribution system is described in Section 8.7. The redundancy of the power sources is enhanced by separate and redundant auxiliary power and control distribution systems. A single failure and any possible related failures in that channel cannot adversely affect equipment and components on the other redundant channel.

Due to the redundancy and separation of power supplies, distribution and control required for vital functions, all components can be readily inspected and tested. Similarly, most subsystems can be tested in their entirety.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

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

protection system, and the transfer of power among the nuclear power unit, the off site power system, and the on site power system.

The operability and functional performance of the components of these systems are verified by periodic inspections and tests as described in Chapter 8.

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

a. Manually initiated demonstration of the ability of the diesel-generators to start, synchronize and deliver power up to 2750 kW continuous, when operating in parallel with other power sources. Normal unit operation will not be affected.

b. Demonstration of the readiness of the on site generator system and the control systems of vital equipment to automatically start, or restore to operation, the vital equipment by initiating an actual loss of all normal AC station service power. This test will be conducted during each refueling interval.

   Demonstration of the automatic sequencing equipment during normal unit operation. This test exercises the control and indication devices, and may be performed any time, as the sequencing equipment is redundant to normal operations. If there is a safety injection actuation signal while the test is underway, it takes precedence and immediately cancels the test. The equipment then responds to the safety injection actuation signal in the manner described in Section 8.3.

Since operation of the protective system will be infrequent, each system is periodically and routinely tested to verify its operability. Each channel of the protective systems, including the sensors up to the final protection element, is capable of being checked during reactor operation. The output circuit breakers are provided to permit individual testing during plant operation. See Chapters 7 and 8 for further details.

CRITERION 19 - CONTROL ROOM

A control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCA). Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

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

The control room is provided with two separate air conditioning systems and two particulate, absolute, charcoal filter unit assemblies, an airborne radioactivity detector in the fresh air supply line and dampers which act to shunt the intake air through the filters in the event of a high airborne radioactivity level. The dampers are automatically actuated from the control room monitors. Acting on a high radiation level indication, the fresh air dampers close and recirculation dampers open to provide a complete closed cycle ventilation mode with a portion of the air stream being drawn through the HEPA-charcoal filter assembly. In addition, an area radiation monitor is provided to indicate and alarm on high radiation level.

In the event the operator is forced to abandon the control room, a hot shutdown panel (C21) provide the instrumentation and control necessary to maintain the plant in the hot shutdown condition (see Section 7.6.4). The potential capability for bringing the plant to a shutdown is also provided outside the control room.

Fire Shutdown System Panels located outside the control room contain the instruments and controls necessary to achieve a hot shutdown condition should the control room become uninhabitable due to fire (see Section 7.6.5). The Fire Shutdown Panel can be utilized for any emergency event which requires control room evacuation.

Not all indicators and controls provided on the Fire Shutdown Panel are available for all fires. Alternate methods of compliance are documented in the Millstone Unit 2 10 CFR 50 Appendix R Compliance Report.

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

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

The reactor is protected by the Reactor Protective System from reaching a condition that could result in exceeding acceptable fuel design limits as a result of anticipated operational occurrences (ANS-N18.2, Condition II). The Protective System is designed to monitor the reactor operating conditions and initiate a reactor trip if any of the following measured variables exceeds the operating limits:

a. High power level (variable, highest of thermal or neutron flux).

b. High pressurizer pressure.

c. Thermal margin (variable low pressure).

d. Turbine trip (equipment protection only).

e. Low reactor coolant flow.

f. Low steam generator level.

g. Low steam generator pressure.

h. Local power density.

i. High containment pressure.

The Engineered Safeguards Actuation System detects accident conditions and initiates the Safety Features Systems which are designed to localize, control, mitigate, and terminate such accidents. The Engineered Safeguards Actuation System protects the general public from the release of radioactivity by actuating components that cool the reactor core, depressurize the containment, isolate the containment, and filter any containment leakage (see Section 7.3). The following parameters are continuously monitored;

a. Low pressurizer pressure.

b. High/high-high containment pressure.

c. Containment gaseous and particulate radiation.

d. Low steam generator pressure.

e. High fuel handling area radiation.

f. Low refueling water storage tank level.

g. Emergency bus undervoltage.

The Auxiliary Feedwater Automatic Initiation System (AFAIS) provides a dedicated source of feedwater of sufficient capacity to remove decay heat and sensible heat following casualty situations. Automatic initiation of auxiliary feedwater occurs in response to a low Steam Generator level in a two out of four (2 of 4) channel auctioneered matrix (see Section 7.3.2.2.h).

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

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

The protective system is designed to provide a high functional reliability and inservice testability. No single failure will result in the loss of the protective function. The protective channels are independent, e.g., with respect to piping, wire routing, mounting and supply of power. This independence permits testing and the removal from service of any component or channel without loss of the protection function.

Each channel of the protective system, including the sensors up to the final protective element, is capable of being checked during reactor operation. Measurement sensors of each channel used in protective systems are checked by observing outputs of similar channels which are presented on indicators and recorders on the control board. Trip units and logic are tested by inserting a signal into the measurement channel ahead of the trip units and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays. The logic output relays are tested individually for initiation of trip action. See Chapter 7.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

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

The reactor protective systems conform to the provisions of the Institute of Electrical and Electronic Engineers (IEEE) Criteria for Nuclear Power Plant Protection Systems, IEEE-279, 1971. Two to four independent measurement channels, complete with sensors, sensor power supplies, signal conditioning units and bistable trip units, are provided for each protective parameter monitored by the protective systems. The measurement channels are provided with a high degree of independence by separate connection of the channel sensors to the process systems. Power to the channels is provided by independent vital power supply buses. See Section 7.2.

Combustion Engineering Topical Report CENPD-11 (“Reactor Protection System Diversity,” W. C. Coppersmith, C. I. Kling, A. T. Shesler, and B. M. Tashjian CENPD, February 1971) demonstrates that functional diversity has been incorporated in the protective system design.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

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

Protective system instrumentation has been designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply or disconnection of the system, Redundancy, channel independence, and separation are incorporated in the protective system

design to minimize the possibility of the loss of a protection function under adverse environmental conditions. See Sections 7.2 and 7.3.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

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

The reactor protective systems are separated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protective system. See Section 7.2.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

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

Reactor shutdown with CEA's is accomplished completely independent of the control functions since the trip breakers interrupt power to the full length CEA drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of controlling groups without exceeding acceptable fuel design limits. An analysis of these accidents is given in Section 14.4. The reactor protection system will prevent specified acceptable fuel design limits from being exceeded for any anticipated transients.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

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

Two independent systems are provided for controlling reactivity changes. The Control Element Drive System (CEDS) controls reactivity change required for power changes and power distribution shaping, and is also used for reactor protection. The boric acid shim control compensates for long term reactivity changes such as those associated with fuel burnup, variation

in the xenon and samarium concentrations, and plant cooldown and heatup. See Sections 7.4.2 and 9.2.2.1.

Either system acting independently is capable of making the core subcritical from a hot operating condition and holding it subcritical in the hot standby condition at 532°F.

Either system is able to insert negative reactivity at a sufficiently fast rate to prevent exceeding acceptable fuel design limits as the result of a power change (i.e., the positive reactivity added by burnup of xenon).

The boron addition system is capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

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

The combined capability of the reactor control systems in conjunction with dissolved boron addition by the safety injection system is such that under postulated accident conditions, even with the CEA of highest worth stuck out of the core, the core would be maintained in a geometry which assures adequate short and long term cooling. See Criteria 26 and 28.

CRITERION 28 - REACTIVITY LIMITS

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

The basis for selecting the number of control element assemblies in the core includes assuring that the reactivity worth of any one assembly is within a preselected maximum value. The control element assemblies have been separated into sets: a shutdown set and a regulating set further subdivided into groups as necessary. Administrative procedures and interlocks are used to permit only one shutdown group to be withdrawn at a time, and to permit withdrawal of the regulating groups only after the shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits that prevent the rates of reactivity change and the worth of individual assemblies from exceeding limiting values. See Sections 7.4.2, 14.4.1, 14.4.2, and 14.4.3.

The reactor coolant pressure boundary and reactor vessel internals are designed to be capable of accommodating without rupture, and with limited plastic deformation, the static and dynamic loads associated with an inadvertent and sudden release of energy to the coolant such as that resulting from CEA ejection, CEA drop, steam line rupture or cold water addition. See Sections 14.4.8, 14.4.9, and 14.1.5.

The boric acid system rate of reactivity addition is too slow to cause rupture of the reactor coolant pressure boundary or disturb the reactor pressure vessel internals.

CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

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

Anticipated operational occurrences have been considered in the design of the protection and reactivity control systems. As is demonstrated in the safety analysis in Chapter 14 and the Combustion Engineering Report CENPD-11 (“Reactor Protection System Diversity”, W. C. Coppersmith, Cl. L. Kling, A. T. Shesler, and B. M. Tashjian, CENPD-11, February 1971), the design is adequate to minimize the consequences of such occurrences and assures that the health and safety of the public is protected from the consequences of such occurrences.

The adherence to a detailed program for quality assurance, careful attention to design, component selection and system installation, coupled with the design features of redundancy, independence, and testability will assure that a high probability exists that the protection and reactivity control systems will accomplish their functions. See Criteria 21 through 26.

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

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

The reactor coolant pressure boundary components have been designed, fabricated, erected and tested in accordance with the ASME Code Section III, 1971 through summer 1971 Addenda and ANSI B31.7, 1969 as specified in Criterion 14. Replacement steam generator subassemblies were fabricated in accordance with ASME Code Section III 1983 through summer 1984 Addenda.

The replacement reactor vessel closure head including all nozzles (CEDM, HJTC, ICI and the vent) is constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1998 Edition through 2000 Addenda.

Containment sump instrumentation is used to detect reactor coolant system leakage by providing information on rate of rise of sump levels and frequency of sump pump operation. Flow

instrumentation indicates and records makeup flow rate and volumes from the primary water system. This instrumentation allows detection of suddenly occurring leaks or those which are gradually increasing. The containment air monitoring system (see Section 7.5.6) provides an additional means of reactor coolant system leakage detection.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

Carbon and low alloy steel materials which form part of the pressure boundary meet the requirements of the ASME Code, Section III, paragraph N-330 at a temperature of +40°F. (Ref. Section 4.2.2). The actual nilductility transition temperature (NDTT) of the materials has been determined by drop weight tests in accordance with ASTM-E-208. For the reactor vessel base metals, Charpy tests were also performed and the results used to plot a Charpy transition curve. To address changes in regulations, the original design requirements of N-330 were supplemented and the materials' initial nil-ductility reference temperatures (RT_{NDT}) were conservatively established based upon available or supplemental material toughness testing. In the case of the replacement steam generators, the materials were required to satisfy NB-2331 and RT_{NDT} values were established to satisfy current requirements.

Carbon and low alloy steel materials including weld filler metal which form part of the reactor pressure boundary for replacement reactor vessel closure head satisfy ASME Section III, NB 2000. Actual NDTT was established by drop weight test in accordance with ASTM-E-208 at - 40°F. RT_{NDT} of the replacement head based materials was established by Charpy V-notch test at - 40°F. Charpy transition curves were plotted using test data for the base material of the replacement reactor vessel head.

All the reactor coolant pressure boundary components are constructed in accordance with the applicable codes and comply with the test and inspection requirements of these codes. These test inspection requirements assure that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding code requirements are performed. The tests and inspections performed on the reactor vessel are summarized in Section 4.6.5.

The reactor vessel beltline materials receive sufficient neutron irradiation to cause embrittlement (an increase in RT_{NDT}). To provide conservative margins against nonductile or rapidly propagating failure, several techniques are employed. Operating limits which account for the

RT_{NDT} of all pressure boundary materials, both unirradiated and irradiated, are established in accordance with the requirements of 10 CFR 50 Appendix G (Additional details are provided in Section 4.5.1). In addition, compliance with 10 CFR 50.61 assures that the shift in the transition temperature of the reactor vessel beltline materials provides adequate margins of safety against severe pressurized thermal shock events.

To assure that the reactor vessel beltline materials are behaving in the predicted manner, a reactor vessel material surveillance program is conducted (See Criterion 32 and Section 4.6.2). Toughness testing of unirradiated reactor vessel materials was performed to establish the baseline, and the irradiated surveillance materials are periodically tested as surveillance capsules are removed during the plant's design life, in accordance with the requirements of 10 CFR 50, Appendix H.

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

The thermal stresses induced by the injection of cold water into the vessel have been examined. Analysis shows the there is no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Boiler and Pressure Vessel Code, Section III. (Ref. Section 4.5.4).

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

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

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

Provisions are made for inspection, testing, and surveillance of the Reactor Coolant System boundary as required by ASME Boiler and Pressure Vessel Code, Section XI.

The Reactor vessel surveillance program was designed in accordance with ASTM E185. It complies with ASTM E185-73 and 10 CFR 50, Appendix H. Section 4.6.3 presents the details of the reactor surveillance program. Sample pieces taken from the same shell plate material used in fabrication of the reactor vessel are installed between the core and the vessel inside wall. These samples will be removed and tested at intervals during vessel inside wall. These samples will be removed and tested at intervals during vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests will be performed on the samples to develop a Charpy transition curve. By comparison of this curve with the Charpy curve and drop weight tests for specimens taken at the beginning of the vessel life, the change of NDTT will be determined and operating instructions adjusted as required.

CRITERION 33 - REACTOR COOLANT MAKEUP

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

Reactor Coolant System (RCS) makeup during normal operation is provided by the Chemical and Volume Control System (CVCS) which includes three positive displacement charging pumps rated at 44 gpm each. Two operating CVCS pumps are capable of making up the flow loss for leaks in the reactor coolant boundary of up to 0.250 inches equivalent diameter. Two CVCS pumps are sufficient to makeup for a 0.250 inch equivalent diameter RCS break assuming either: 1) minimum letdown with no RCS leakage or 2) letdown isolated with maximum Technical Specification allowed leakage. This CVCS design results in a substantial RCS steady state pressure that is well above the shutoff head of the high pressure safety injection pumps. The above described CVCS capability fulfills the intent of Criterion 33. Information on CVCS is contained in Section 9.2.

CRITERION 34 - RESIDUAL HEAT REMOVAL

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

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

Residual heat removal capability is provided by the shutdown cooling system for reactor coolant temperature less than 300°F (see Section 9.3). For temperatures greater than 300°F, this function is provided by the steam generators (see Section 10.3). Sufficient redundancy, interconnections, leak detection, and isolation capabilities exist with these systems to assure that the residual heat removal function can be accomplished, assuming failure of a single active component. Within appropriate design limits, either system will remove fission product decay heat at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary will not be exceeded.

CRITERION 35 - EMERGENCY CORE COOLING

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

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

The emergency core cooling system is discussed in detail in Chapter 6. It consists of the high pressure safety injection subsystem, the low pressure safety injection subsystem, and the safety injection tanks (see Section 6.3).

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

The ECCS design satisfies the criteria specified in 10 CFR 50.46(b).

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

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

Chapter 6 describes the arrangement and location of the components in the emergency core cooling system. All pumps, the shutdown cooling heat exchangers, and valves and piping external to the containment structure are accessible for physical inspection at any time. All safety injection valves and piping inside the containment structure, and the safety injection tanks, may be inspected during refueling.

The accessibility for inspection of the reactor vessel internals, reactor coolant piping and items such as the water injection nozzles is described in Sections 4.6.3 through 4.6.6.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components,

(2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The Emergency Core Cooling System (Safety Injection System) is provided with testing facilities to demonstrate system component operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the systems or with the initiation of control circuits, as described in Section 6.3.4.2.

The safety injection system is designed to permit periodic testing of the delivery capability up to a location as close to the core as practical. Periodic pressure testing of the Safety Injection System is possible using the cross connection to the charging pumps in the Chemical and Volume Control System.

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

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

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

The operational sequence that brings the Safety Injection System into action, including transfer to alternate power sources, can be tested in parts as described in Chapters 6, 7, and 8.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

The containment spray system (Section 6.4) and the containment air recirculation and cooling system (Section 6.5) are provided as redundant, independent systems, each fully capable of reducing the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintaining them at acceptably low levels.

Sufficient heat removal capability is provided by any of the following combinations of equipment:

a. Two containment spray pumps with associated heat exchangers.

b. Three of the four containment air recirculation and cooling units.

c. One containment spray pump with associated heat exchanger in combination with two containment air recirculation and cooling units.

The containment heat removal systems are provided with suitable interconnections such that each combination of two containment air recirculation and cooling units and one containment spray pump, aligned with the associated shutdown cooling heat exchanger, are provided with cooling water from the same RBCCW header and powered by the same emergency bus. All associated components, such as valves, are likewise powered from the same emergency bus. Each combination of these components is capable of removing heat at a rate greater than required to limit the postaccident containment pressure and temperature. A single failure of any active component does not render the redundant group inoperable.

The containment spray system is provided with containment isolation capabilities in accordance with Criterion 56. The above containment penetration is provided with leak detection capabilities in accordance with Criterion 54.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

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

Major components of the containment spray system are located to permit access for periodic maintenance and inspection. Components of the containment air and recirculation system are located within the containment and are therefore accessible for maintenance and inspection during shutdown.

The containment sump is located in the lowest elevation of the containment at Elevation (-)22-6 and is accessible during reactor shutdown for periodic visual inspections (see Section 6.2).

The containment spray nozzles are accessible for periodic inspection during reactor shutdown.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

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

The spray system and the air recirculation and cooling systems in the containment have provisions for online testing to assure system operation, performance and structural and leaktight integrity of the associated components. Testing procedures are described in Sections 6.4.4.2 and 6.5.4.2, respectively.

The containment heat removal systems undergo preoperational testing prior to plant startup. The test procedure is described in Chapter 13.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEAN UP

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment are provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
>
> Each system has suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

The containment is not provided with an atmosphere cleanup system. However, a second barrier, the enclosure building, is provided around the containment to collect potential leakage from the containment under postaccident conditions.

The enclosure building filtration system (EBFS) is provided to collect and process potential leakage from the containment during postaccident operation. Potential containment leakage is into the enclosure building filtration region (EBFR) which forms the outer barrier in the double containment boundary. The EBFS is described in Section 6.7. Throughline leakage that can bypass the EBFR is discussed in Section 5.3.4.

The hydrogen control system is provided to mix and monitor the concentration of hydrogen in the containment atmosphere following postulated accidents to assure the containment integrity is

maintained. This is discussed in Section 6.6. Reduction of hydrogen concentration is not credited for design basis accidents.

Each of these cleanup systems consist of completely redundant, independent safety function. These are provided with suitable interconnections and separations such that a single failure in any subsystem does not render the redundant subsystem inoperable.

The hydrogen control system is incorporated with containment isolation capabilities for each piping subsystem which penetrates the primary containment. Containment isolation is in accordance with Criterion 56. Provision for leak detection is incorporated in accordance with Criterion 54.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

> The containment atmosphere cleanup systems are designed to permit appropriate periodic inspection of important components, such as filter frames, fans, hydrogen recombiners, analyzers, valves, ducts, and piping to assure the integrity and capability of the systems.

The enclosure building filtration system (EBFS) is located to permit access for periodic inspection and maintenance. The components of the hydrogen control system located outside the containment are accessible for periodic inspection and maintenance. The components located inside containment are accessible for inspection and maintenance during shutdown.

The hydrogen control system and EBFS are incorporated with provisions for online testing to demonstrate system operation, performance and integrity. These tests procedures are described in Sections 6.6.4.2 and 6.7.4.2, respectively.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEM

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

The enclosure building filtration system (EBFS) and hydrogen control system are incorporated with provisions for online testing. The test procedures are described in Sections 6.7.4.2 and 6.6.4.2, respectively.

The containment atmosphere cleanup systems undergo preoperational tests prior to plant startup. Test procedures are described in Chapter 13.

CRITERION 44 - COOLING WATER

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

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

The RBCCW system, described in Section 9.4, and the service water system, described in Section 9.7.2, are provided to transfer heat from structures, systems, and components important to safety to an ultimate heat sink. The systems are designed to transfer the combined heat load of these structures, systems, and components under normal and accident conditions.

The RBCCW supplies cooling water to components important to safety through two independent headers. One header provides adequate heat removal capability to safely shutdown the plant under accident conditions, but at a lesser rate. Service water is supplied to the RBCCW heat exchangers by two independent headers to assure heat removal capability. Two service water pumps are in continuous operation with a spare pump provided. One pump supplies sufficient heat removal capability for the RBCCW heat exchangers to safely shut down the plant and for accident mitigation.

The RBCCW and service water systems are provided with suitable redundancy in components and suitable interconnections to assure heat removal capability. The systems are designed to enable isolation of system components or headers and to detect system maloperation.

The RBCCW and service water systems are designed to operate with onsite power (assuming offsite power is not available) and with offsite power (assuming onsite power is not available).

The systems are designed such that a single failure in either system will not adversely affect safe operation, accident mitigation, or safe shutdown of the plant.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

The RBCCW system and service water system, excluding underground piping, are designed to permit periodic inspection of important components, such as pumps, heat exchangers, valves and piping to assure the integrity and heat removal capability of the system. The components of the RBCCW system located outside the containment are located in a low radiation area, which

permits access for periodic inspection and maintenance during operation. Components of the RBCCW system located inside the containment are accessible for inspection and maintenance during plant shutdown. Inspection of RBCCW system components is described in Section 9.4.4.2. Major service water system components, such as pumps and strainers, are accessible for periodic inspection during normal operation. Inspection of the service water system is described in Section 9.7.2.5.

CRITERION 46- TESTING OF COOLING WATER SYSTEM

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

Online testing provisions are incorporated in the RBCCW and service water systems to demonstrate the operability, performance, structural and leaktight integrity of the systems. The RBCCW and service water systems are designed so that under conditions as close to design as practical, the performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, and the transfer between normal and emergency power sources. Testing of the RBCCW and service water systems are described in Sections 9.4.4.2 and 9.7.2.5, respectively.

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 Section 5.2.2. 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.

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 ASME Boiler and Pressure Vessel Code, Section III (Nuclear Vessels) 1968 Edition through the summer 1969 addenda Paragraph N-1211. Further description of the liner plate is contained in Section 5.2.3.

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 Section 5.2.8.1.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

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

The containment consists of a prestressed reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete slab. A one-quarter inch thick steel liner plate is attached to the inside surface of the concrete containment and its penetrations. Consideration has been given to both design and construction techniques to assure the containment pressure boundary behaves in a ductile manner and the probability of a rapidly propagating fracture is minimized.

The liner plate is designed to carry no load, and serves only as a leaktight barrier. Analytical calculations of the strains under an extreme and most improbably set of load conditions indicate the strains are well within the ductile limits of the material. The analytical approach to liner design is presented in the Bechtel Corporation Proprietary Report B-TOP-1.

At all penetrations the liner plate is thickened using the 1968 ASME Code, Section III for Class B Vessels as a guide to limit stress concentrations.

Provisions, as described in Section 5.2.5.1.1, are made to prevent a potential internally generated missile from rupturing the liner plate.

Materials for the penetrations require satisfactory Charpy V-notch impact test results. All penetrations are stress relieved. The construction materials selected for the liner plate and penetrations are given in Section 5.2.1.

Additional details concerning the construction techniques and inspection provisions are outlined in Section 5.9.3.5.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

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

The reactor containment and other equipment which is subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. The test procedure is described in Section 5.2.8.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

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

The reactor containment is designed to permit appropriate periodic testing of all important areas. Details of the containment testing and inspection are discussed in Section 5.2.8.

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

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

Piping systems penetrating containment are provided with suitable redundancy to assure the systems function adequately during postulated accidents such that failure of a portion of a system will not create a hazard to safe unit operation. Piping systems are provided with containment isolation valves in accordance with the requirements of Criterion 55, 56, and 57. Containment isolation valves have been selected and tested to provide adequate operation at maximum flow conditions. Provisions are incorporated for leak detection and performance testing of those piping systems penetrating the containment (Section 5.2.7.4.2).

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

> Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can

be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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

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

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

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

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

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

For those piping systems penetrating the containment and connected directly to the reactor coolant pressure boundary, isolation provisions have been incorporated. Section 5.2.7 indicates applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

Provisions are made for leak testing as described in Section 5.2.7.4.2.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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

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

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

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

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

For those piping system penetrating the containment and connected directly to the containment atmosphere, isolation provisions have been incorporated. Section 5.2.7 indicates the applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary not connected directly to the containment atmosphere has at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve is outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

For those piping systems penetrating the containment which are neither part of the reactor coolant pressure boundary nor connected directly with the containment atmosphere, isolation provisions have been incorporated.

Section 5.2.7 indicates applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

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

The nuclear power unit design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity is provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

The radioactive waste processing system (RWS), as described in Section 11.1, is designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes from Millstone Unit 2. The RWS is designed to ensure that the general public and plant personnel are protected against exposure to radioactive material in accordance with 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I.

All liquid and gaseous radioactive releases from the RWS are designed to be accomplished on a batch basis. All radioactive materials are sampled prior to release to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I and to determine release rates. Radioactive materials which do not meet release requirements will not be discharged to the environment. The RWS is designed with sufficient holdup capacity and flexibility for reprocessing of wastes to ensure release limitations are met.

The RWS is designed to preclude the inadvertent release of radioactive material.

All storage tanks in the clean liquid waste and gaseous waste systems are provided with valve interlocks which prevent the addition of waste to a tank which is being discharged to the environment. Each discharge path from the RWS is provided with a radiation monitor which alerts unit personnel and initiates automatic closure of redundant isolation valves to prevent further releases in the event of noncompliance to 10 CFR Part 20, Sections 1301 and 1302, and Appendix B.

Section 11.1.5 describes the plant design for the handling of solid wastes.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

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

Systems for fuel storage and handling, and all systems containing radioactivity are designed to ensure adequate safety under normal and postulated accident conditions. Design of these systems are described in the sections listed below:

System	Section
Reactor Coolant System	4.0
Engineering Safety Features Systems	6.0

System	Section
Auxiliary Systems	9.0
Radioactive Waste Processing System	11.0

All components important to the safety of these systems are located to permit periodic inspection as required. Suitable shielding, as described in Section 11.2, is provided for these components to protect plant personnel and to allow inspection and testing.

To ensure the containment and confinement of radioactivity, all components are designed and tested in accordance with accepted Codes and Standards. All system components are visually inspected and adjusted, if required, to ensure correct installation and arrangement. The completely installed systems were subject to acceptance tests or preoperation tests as described in Chapter 13 to ensure the integrity of the systems.

The spent fuel pool cooling system described in Section 9.5, is designed to ensure adequate decay heat removal from stored fuel. Sections 5.4.3 and 9.5 describe how the spent fuel pool is designed to prevent significant reduction in fuel storage coolant inventory.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

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

New fuel assemblies are stored in dry racks in parallel rows at elevation 38 feet 6 inches of the auxiliary building. The base of the new fuel racks at elevation 38 feet 6 inches minimizes the possibility of flooding the fuel assemblies. Nevertheless, the new fuel racks maintain a center to center distance of 20.5 inches, large enough to prevent criticality in the unlikely event of flooding with unborated water. Additional details of new fuel storage are given in Sections 9.8.2.1.1and 9.8.4.1.1.

Spent fuel assemblies are stored in parallel rows at the bottom of the spent fuel pool. The racks are separated into 4 regions, designated Regions 1, 2, 3, and 4.

Fuel assemblies used at Millstone Unit 2 may include reduced enrichment fuel rods adjacent to guide thimbles and reduced enrichment axial blanket regions. The criticality analyses are performed using a single enrichment in all fuel rods that is the highest initial planar average U-235 enrichment of the axial regions in the fuel assembly. This averaged enrichment is designated as the initial planar average enrichment.

Region 1 can store, in a 2 out of 4 storage pattern, any fuel assembly with a maximum initial planar average enrichment up to 4.85 weight percent U-235. The other two locations in the 2 out of 4 storage pattern are designated as Restricted Locations (shown in Figure 9.8–7). Fuel storage rack locations designated as Restricted Locations in Figure 9.8–7 shall remain empty. No fuel

assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.(1)

Regions 2 and 4 use fuel burnup credit and store fuel assemblies in a 3 out of 4 storage pattern, in which the fourth location in a 2 x 2 storage array is designated as a Restricted Location per Figure 9.8–7.

Regions 1 and 2 contain Boraflex panels which are no longer credited as neutron absorbers.

Region 3 uses fuel burnup credit and has all storage locations available. In addition, fuel assemblies stored in Region 3 must contain either three Borated Stainless Steel Poison Rodlets (installed in the assembly's center guide tube and in two diagonally opposite guide tubes) or a full length, full strength Control Element Assembly (CEA).

There are also Non-standard Fuel Configurations in the spent fuel pool (SFP). A Non-standard Fuel Configuration is an object containing fuel that does not conform to the standard fuel configuration. The standard fuel configuration is a 14 x 14 array of fuel rods (or fuel rods replaced by un-enriched fuel rods or stainless steel rods) with five (5) guide tubes that occupy four lattice pitch locations each. Fuel in any other array is a “Non-standard Fuel Configuration.” Reconstituted fuel in which one or more fuel rods have been replaced by either un-enriched fuel rods or stainless steel rods is considered to be a standard fuel configuration.

Note that each of the Non-standard Fuel Configurations must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require Borated Stainless Steel Poison Rodlets or a CEA if stored in Region 3.

GDC 62 states that the “Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.” As detailed above, the Region 1, 2, 3, and 4 storage racks, require more than just fuel geometry alone for reactivity control. All four regions credit soluble boron in the spent fuel pool water. Regions 1, 2, and 4 credit Restricted Locations per Figure 9.8–7. Regions 2, 3, and 4 use fuel burnup credit. Region 3 requires that fuel assemblies contain either three Borated Stainless Steel Poison Rodlets or a full length, full strength CEA (note that the criticality analysis of a given Non-standard Fuel Configuration may qualify it for Region 3 storage without these inserts). Administrative controls are used to ensure proper placements of Borated Stainless Steel Poison Rodlets and CEAs, use of soluble boron and fuel burnup credit, and control of Restricted Locations. Further, for accident conditions, soluble boron is credited in the spent fuel pool water. The NRC has concurred that the credit for these neutron poisons, soluble boron, fuel burnup credit, Restricted Locations, and associated administrative controls are acceptable in meeting the requirements of GDC 62.

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(1) Note that Region 1 and 2 SFP rack storage locations contain removable Boraflex panel boxes which house the Boraflex panels. The Boraflex panel boxes were manufactured as an integral part the original SFP racks and as such are NOT stored components in SFP rack storage locations. Criticality analysis has shown that the Restricted Locations are acceptable with or without the Boraflex panel boxes.

Both the spent fuel and new fuel storage racks are designed to preclude any deformation of the racks during earthquake loads that would reduce the center to center spacing to a point where the fuel would approach criticality.

Fuel handling equipment is designed to ensure safe handling of fuel assemblies and to prevent criticality. Section 9.8.4 describes the safety features of the fuel handling equipment.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

Section 9.5.2.1 describes the monitoring and alarm instrumentation provided for the spent fuel storage system to detect conditions that may result in loss of decay heat removal capability and excessive radiation levels. Section 7.5.6 describes the monitoring provisions for radioactive waste handling and storage areas.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

Containment radiation is monitored by gaseous and particulate monitors as described in Sections 7.5.1.2 and 7.5.6.3.

Radiation in effluent discharge paths and the plant environs are monitored as described in Sections 7.5.6.2 and 7.5.6.3.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 2: Site and Environment

CHAPTER 2—SITE AND ENVIRONMENT

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NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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CHAPTER 2 – SITE AND ENVIRONMENT

This section contains information on the geological, seismological, hydrological, meteorological and demographic characteristics of the Millstone site and vicinity to show the adequacy of the site from a safety viewpoint.

2.1 GENERAL DESCRIPTION

Information regarding the general description of the site, location, area, boundaries, exclusion area, authority, control of activities, traffic control and relocation of roads is presented in Sections 2.1.1 and 2.1.2 of the Millstone 3 Final Safety Analysis Report (Reference 2.1-1). With the exceptions given below, that information is incorporated herein by reference.

The Exclusion Area Boundary (EAB) as shown in Figure 2.1-3 of Reference 2.1-1 is drawn for MP3. For the land sectors, the EAB is equivalent for MP2 and MP3 as it is the property line. For the water sectors, the EAB is assumed to be a circle, centered on the release point with a radius equal to the nearest land EAB distance from the release point. The circle shown in the Reference is for a MP3 containment release. The EAB distances used for accident calculations for MP2 are given in Table 2.3-1. The distances used for nearest land for normal effluent dose calculations are also given in Table 2.3-1.

2.1.1 REFERENCES

2.1-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.1 - Geography and Demography.

2.2 POPULATION, LAND USE AND WATER USE

2.2.1 POPULATION

Information regarding population and population distribution is presented in Section 2.1.3 of the Millstone 3 Final Safety Analysis Report (Reference 2.2-1). That information is incorporated herein by reference.

2.2.2 LAND USE

Information regarding land use, explosions, fire, corrosive materials, and toxic gas is presented in Sections 2.2.1, 2.2.2, 2.2.3, and 2.2.4 of the Millstone 3 Final Safety Analysis Report (Reference 2.2-2). That information is incorporated herein by reference.

2.2.3 WATER USE

Information regarding the water use, commercial and recreational is presented in Sections 2.1.2, 2.1.4 of Reference 2.2-1, Sections 2.2.2, 2.2.3, 2.2.4, of Reference 2.2-2, and Section 2.4 of the Millstone 3 Final Safety Analysis Report (Reference 2.2-3). That information is incorporated herein by reference.

2.2.4 REFERENCES

2.2-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.1 - Geography and Demography.

2.2-2 Millstone Unit 3, Final Safety Analysis Report, Section 2.2 - Nearby Industrial, Transportation, and Military Facilities.

2.2-3 Millstone Unit 3, Final Safety Analysis Report, Section 2.4 - Hydrologic Engineering.

2.3 METEOROLOGY

Information regarding meteorology is presented in Section 2.3 of the Millstone 3 Final Safety Analysis Report (Reference 2.3-1). With the exceptions given below, that information is incorporated herein by reference.

2.3.1 REGIONAL CLIMATOLOGY

(See Section 2.3.1 of Reference 2.3-1.)

2.3.2 LOCAL METEOROLOGY

(See Section 2.3.2 of Reference 2.3-1.)

2.3.2.1 Potential Influence of the Plant and Its Facilities on Local Meteorology

Millstone Unit 2 uses a once-through cooling water system, discharging its cooling water into an existing quarry, into which Units 1 and 3 also discharge, and thence into Long Island Sound. Thin wisps of steam fog occasionally form over the quarry and less frequently over the discharge plume during the winter months, depending on tidal conditions and temperature differences between air and water. This fog dissipates rapidly as it moves away from the warm water area. The areal extent of the steam fog is negligible.

2.3.2.2 Local Meteorological Conditions for Design and Operating Bases

2.3.2.2.1 Design Basis Tornado

The design basis tornado for Millstone Unit 2 is defined in Chapter 5. Tornado missile protection is defined in Section 5.2.5.1.2.

2.3.3 ON SITE METEOROLOGICAL MEASUREMENTS PROGRAM

(See Section 2.3.3 of Reference 2.3-1.)

2.3.4 SHORT TERM (ACCIDENT) DIFFUSION ESTIMATES

2.3.4.1 Objective

Accidents could result in short term releases of radioactivity from several possible venting points. Atmospheric diffusion factors (X/Q) based on site meteorological data are calculated at the exclusion area boundary (EAB), and low population zone (LPZ) for each downwind sector for each release point. The diffusion factors are calculated for different release time periods depending on the length of the release. The diffusion factors are used in the calculation of radiological consequences.

2.3.4.2 Calculations

2.3.4.2.1 Venting Point and Receptor Locations

The distances from the various release points to the exclusion area boundary in each sector are listed in Table 2.3-1.

The Low Population Zone (LPZ) as shown in Figure 2.1-11 of Reference 2.3-1 is drawn for Unit 3 but closely approximates the LPZ for Unit 2. The LPZ is taken to be 3860 meters in all sectors from any Unit 2 release point.

2.3.4.2.2 Models

Accident X/Q's were calculated using the basic methods of Regulatory Guide 1.145 for elevated releases; the X/Q's for the first four hours are calculated using a seabreeze fumigation model adapted from Regulatory Guide 1.3. X/Q values for the control room due to ground level releases were calculated using the guidance of Regulatory Guide 1.194, (Reference 2.3-2).

2.3.4.3 Results

The calculated X/Q's used in DBA radiological consequence calculations are presented with the list of assumptions used in each calculation in Chapter 14.

2.3.5 LONG TERM (ROUTINE) DIFFUSION ESTIMATES

2.3.5.1 Objective

Low levels of radioactivity are routinely released from the Millstone stack or the Unit 2 vent. Atmospheric diffusion factors (X/Q) based on site meteorological data are calculated for various downwind receptor locations of interest. The meteorological data is used to calculate the dose consequences to the public from routine airborne effluents. The calculated doses are submitted annually to the Nuclear Regulatory Commission (NRC).

2.3.5.2 Calculations

2.3.5.2.1 Venting Point and Receptor Locations

Routine releases from the Gaseous Waste Processing System and containment ventings are emitted from the Millstone stack. Routine releases of building ventilation and containment purge air are emitted from the Unit 2 enclosure building roof vent. Releases from the Millstone stack are considered elevated. Releases from the Unit 2 vent are considered to be mixed; that is, conditionally either elevated or ground level depending on ambient wind speed. The distances from the Millstone stack and Unit 2 enclosure building roof vent to the nearest site boundary, the nearest land, and to the nearest residence in each sector are listed in Table 2.3-1, and used in X/Q calculations. The distance to the nearest resident in each sector may vary from those shown in

Table 2.3-1 as a result of the annual land use census performed in accordance with the Radiological Effluent Monitoring and Off site Dose Calculation Manual (REMODCM).

2.3.5.2.2 Database

Calculations are performed on a quarterly basis using the actual meteorology for that period.

2.3.5.2.3 Models

All X/Q values for elevated releases from the Millstone stack and mixed releases from the Unit 2 vent are calculated from hourly onsite meteorological data via methods adapted from Regulatory Guide 1.111 using a conventional Gaussian plume model.

2.3.6 REFERENCES

2.3-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.3—Meteorology.

2.3-2 Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

TABLE 2.3-1 DISTANCES FROM RELEASE POINT TO RECEPTORS

Downwind Sectors	Distance (Meters)						
	Millstone Stack to EAB	Turbine Building to EAB	Blowdown Vent to EAB	Unit 2 Vent to Nearest Land	Unit 2 Vent to Nearest Residence	Millstone Stack to Nearest Land	Millstone Stack to Nearest Residence
SSW	496 (2)	680 (2)	620 (2)	14,500	14,500	14,500	14,500
SW	496 (2)	680 (2)	620 (2)	3,430	3,520	3,660	3,820
WSW	496 (2)	680 (2)	620 (2)	3,100	3,140	3,270	3,290
W	496 (2)	680 (2)	620 (2)	2,830	2,930	3,050	3,070
WNW	649	680 (2)	620 (2)	2,550	2,700	2,700	2,760
NW	710	680 (2)	620 (2)	1,930	2,240	997	997
NNW	1,029	886	915	915	915	1,029	1,029
N	1,677	1,138	1,138	1,138	1,138	1,695	1,695
NNE	813	1,726	1,597	997	997	813	813
NE	496 (3)	680 (3)	620 (3)	620	620	496	736
ENE	496 (2)	680 (2)	620(2)	1,070	1,760	1,101	1,560
E	496 (2)	680	620	1,600	1,650	1,410	1,480
ESE	496 (2)	680 (2)	620 (2)	1,900	2,010	1,640	1,760
SE	496 (2)	680 (2)	620 (2)	31,700	31,700	31,700	31,700
SSE	496 (2)	680 (2)	620 (2)	12,390	12,390	12,390	12,390
S	496 (2)	680 (2)	620 (2)	13,100	13,100	13,100	13,100

The distances may vary from those shown as a result of the annual land use census.
Water Sector, so (1) is used when greater than shoreline distance.
Shortest site boundary distance in any landward sector.

TABLE 2.3-2 (DELETED)

2.4 GEOLOGY

Information regarding the geology of the Millstone site is presented in Section 2.5 of the Millstone 3 Final Safety Analysis Report (Reference 2.4-1). With the exceptions given below, that information is incorporated herein by reference.

2.4.1 GENERAL

(See Section 2.5 of Reference 2.4-1.)

2.4.2 REGIONAL GEOLOGY

(See Sections 2.5.1 and 2.5.3 of Reference 2.4-1.)

2.4.3 SITE GEOLOGY

(See Sections 2.5, 2.5.1, and 2.5.4 of Reference 2.4-1.)

2.4.3.1 Site Surficial Geology

The locations of borings relevant to Unit 2 are shown in Figures 2.4–1 and 2.4–2a while representative boring logs are presented on Figures 2.4–2b, 2.4–2c and 2.4–2d. Additional logs of borings 101 through 114 are shown in the Millstone Unit 2 PSAR; Amendment 1, Appendix 2B, AEC Docket Number 50-336.

Site contours constructed on the upper surface of the basal till have been also presented in the Millstone Unit 2 PSAR, Amendment 1, Plate 2.

2.4.3.1.1 Westerly Granite

An old granite quarry is located at the site. This quarry was worked for over 100 years until its abandonment in 1960. The quarry excavation, some 1,200 feet in length and 300 to 400 feet in width, extends in a northwest-southeast direction in the southerly part of Millstone Point. The quarry lies in a belt of the Westerly granite.

The rock in the quarry is jointed. At the northwesterly end, the joints strike N15E and dip 85 NW. At the southeasterly end of the quarry are two sets of joints, one striking N70W and dipping 85 NW, the other striking N-S with an 85W dip. On the easterly side of the quarry is a possible very small, geologically ancient shear, of no importance, which strikes N50E and dips 50 SE. Sheet jointing, resulting from the relief of rock stress, is prominent throughout the quarry. Prior to construction of the quarry outlet channel there were seeps of water at the south end of the quarry. They appeared to have come from the soil above the bedrock. This indicates that the rock is impervious and allows little, if any, water to seep in from Long Island Sound.

2.4.4 SEISMIC REFRACTION SURVEYS

Seismic surveys, both onshore and offshore, were performed by Weston Geophysical Engineers, Incorporated, to provide complimentary and subsurface data to evaluate site geology as well as furnish supplementary soil and rock velocities. Seismic lines were positioned to detect any large anomalous features that might exist at the site. The land seismic lines and the locations of test borings 1 through 5 are shown on the seismic line location map, Figure 2.4–3. An additional land seismic survey was conducted during October 1971, by Weston Geophysical Engineers, Inc., (Reference 2.4-2). This survey was initiated to determine the dynamic moduli of the compacted backfill underlying the warehouse building, primary water storage tank and refueling water storage tank. Data obtained were then utilized in the dynamic structural analysis of these facilities. Due to their remoteness with respect to Unit 2, offshore seismic lines have not been depicted.

Depths to bedrock, seismic velocity data, and inferred geologic correlation are shown on the profile section, Figure 2.4–4. Geophysical refraction survey compressional wave velocity in the bedrock has been calculated at 13,500 to 14,000 fps. with the shear wave velocity in the same medium estimated to be approximately 5500 to 7500 fps.

The seismic results indicated (and subsequently confirmed by additional borings) that the bedrock surface is covered by dense glacial overburden material and/or saturated alluvium. Since the water level in the quarry prior to excavation of the quarry discharge canal was approximately 17 feet lower than that of Long Island Sound, the material would appear to be dense and relatively impermeable.

Bedrock rises and depressions appear to trend in a north-south direction, as would be anticipated in a normal erosional pattern.

Additional information concerning site seismic refraction surveys is contained in Section 2.5.4 of Reference 2.4-1.

2.4.5 REFERENCES

2.4-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.5, “Geology, Seismology, and Geotechnical Engineering.”

2.4-2 Weston Geophysical Engineers, Inc., Report to Bechtel Corporation, “Seismic Velocity Measurements of Compacted Fill Material, Millstone Nuclear Projects, Unit Number 2,” December 1971.

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.4–1 GEOLOGICAL FEATURES

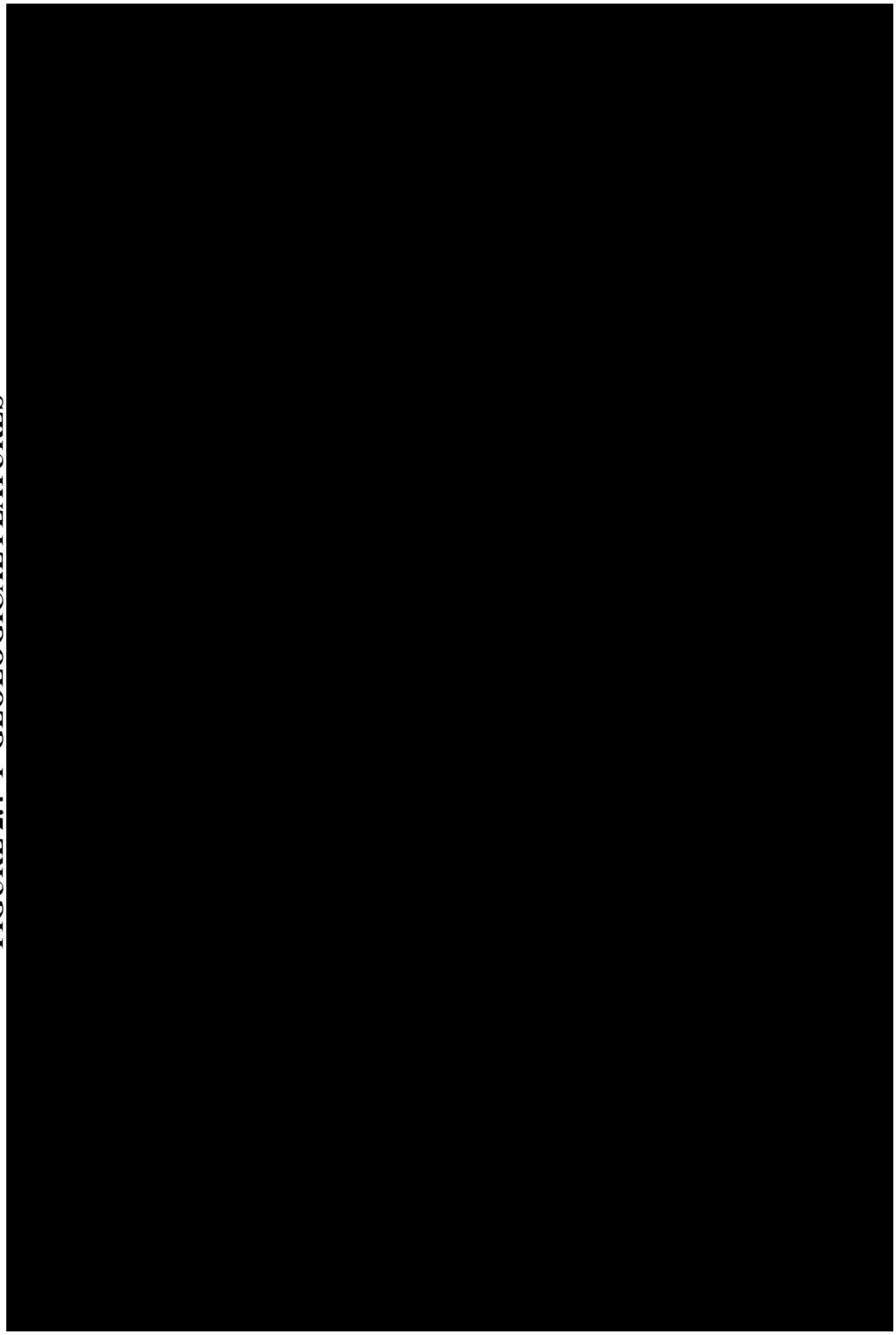


FIGURE 2.4–2A BORING PLAN

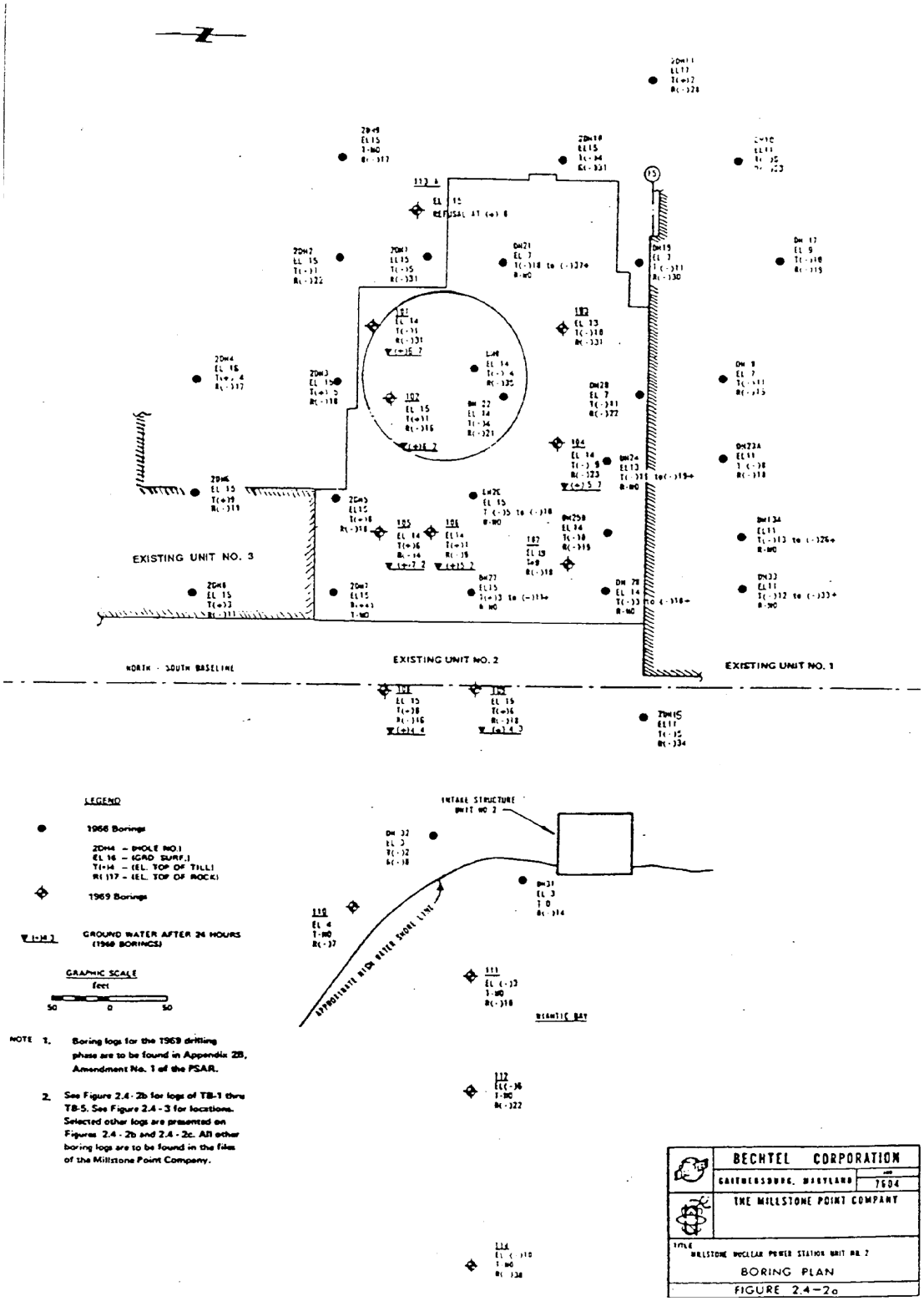
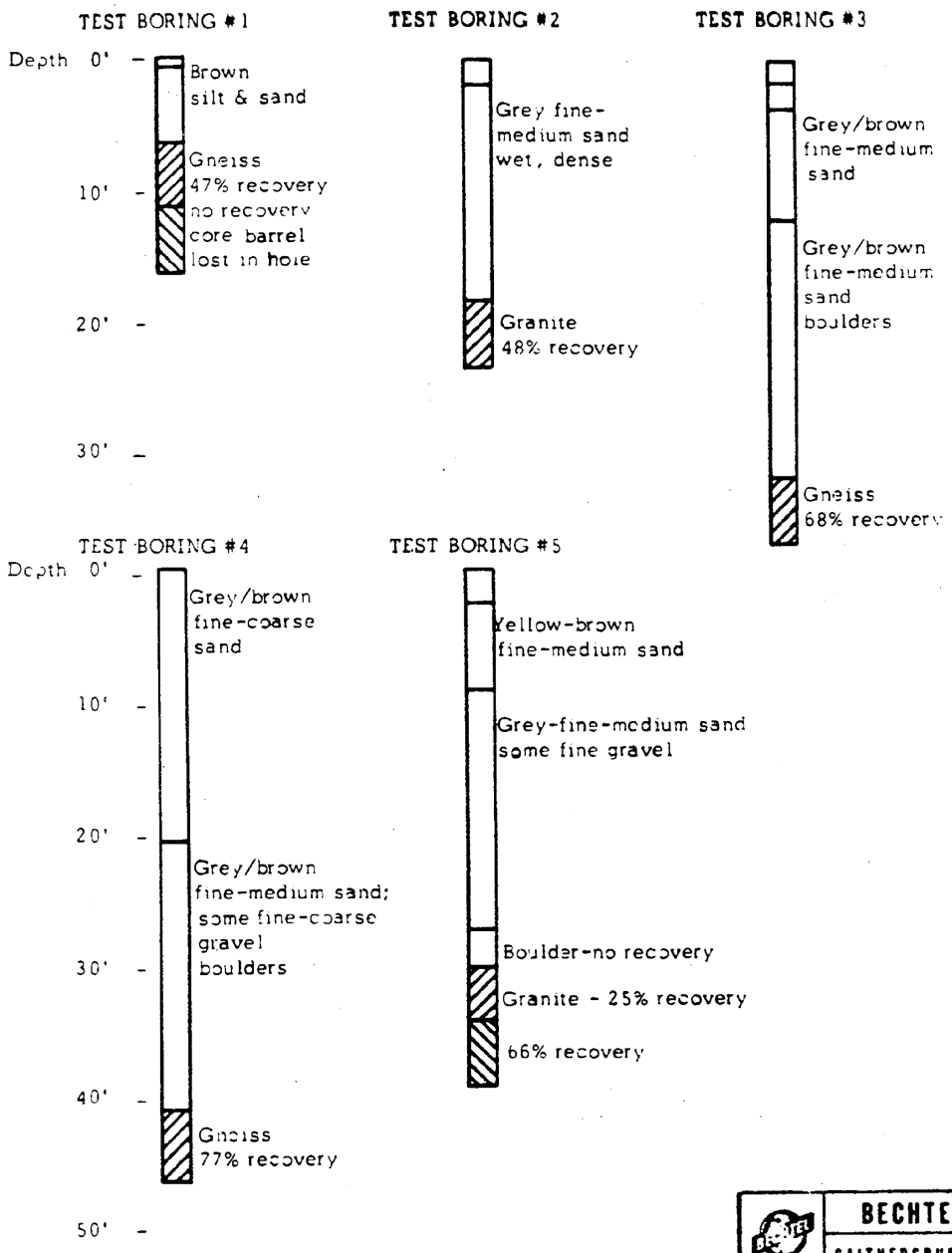


FIGURE 2.4–2B TEST BORINGS



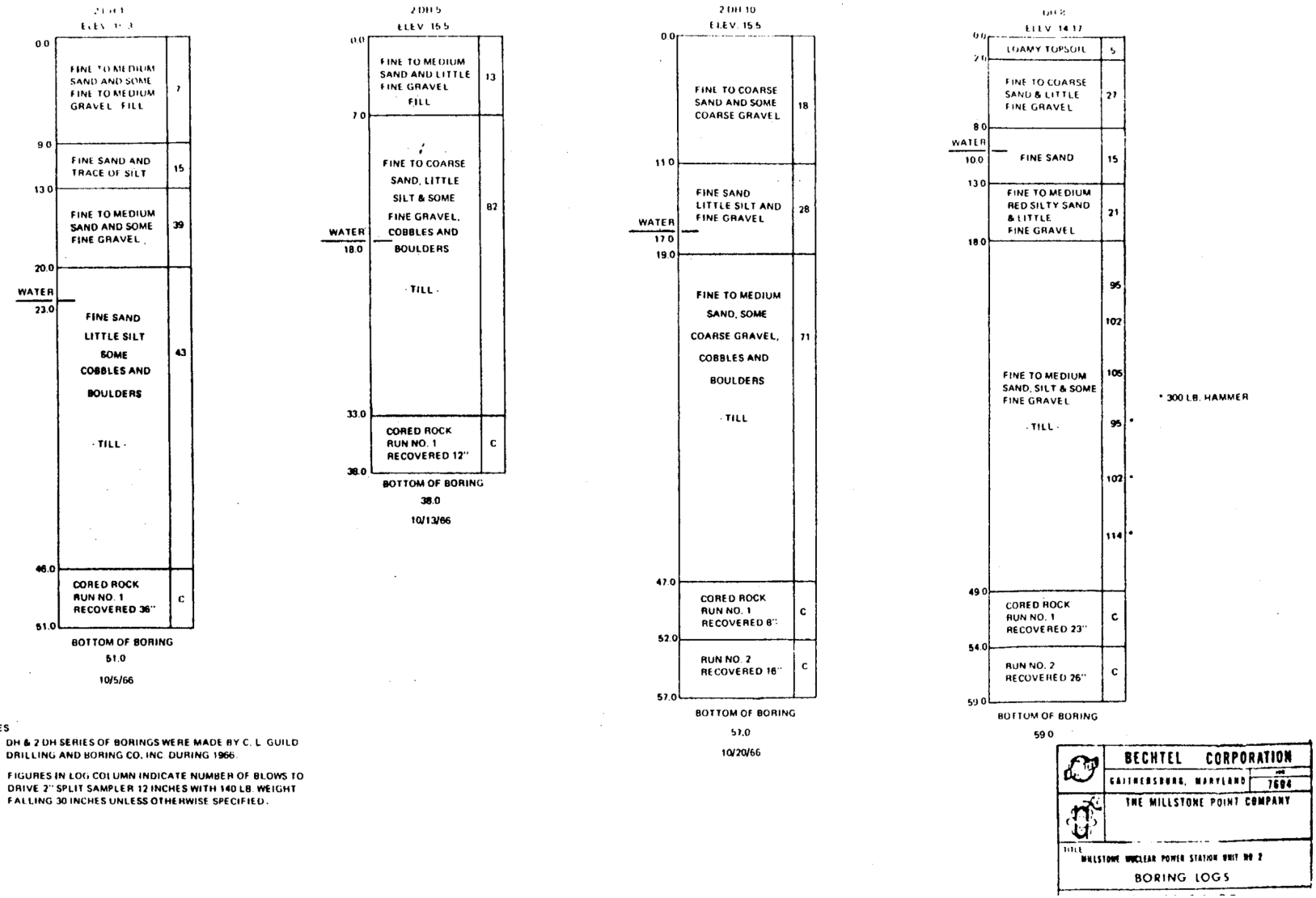
NOTE:

TEST BORINGS MADE BY SOIL TESTING, INC.

BECHTEL CORPORATION	
GAITHERSBURG, MARYLAND	JOB 7604
THE MILLSTONE POINT COMPANY	
TITLE: MILLSTONE NUCLEAR POWER STATION - UNIT NO. 2 TEST BORINGS	
FIGURE 2.4-2b	

JUN 10 1982

FIGURE 2.4–2C BORING LOGS

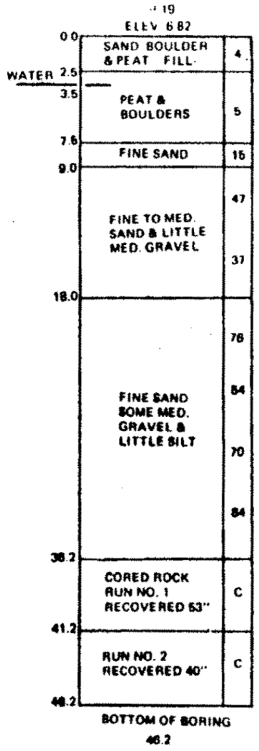


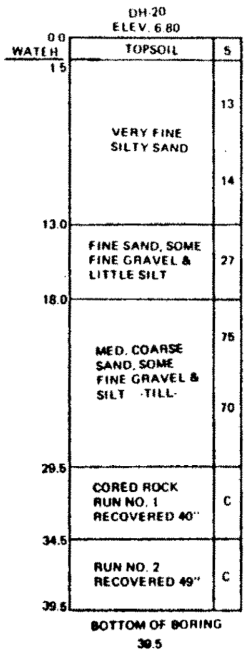
NOTES

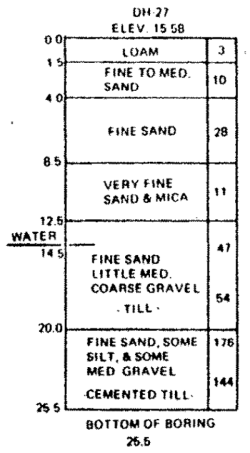
1. DH & 2 DH SERIES OF BORINGS WERE MADE BY C. L. GUILD DRILLING AND BORING CO., INC. DURING 1966.
2. FIGURES IN LOG COLUMN INDICATE NUMBER OF BLOWS TO DRIVE 2" SPLIT SAMPLER 12 INCHES WITH 140 LB. WEIGHT FALLING 30 INCHES UNLESS OTHERWISE SPECIFIED.

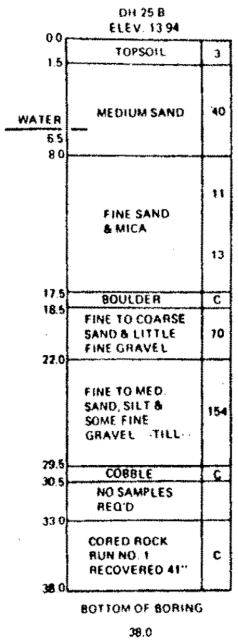
BECHTEL CORPORATION
GAITHERSBURG, MARYLAND 7694
THE MILLSTONE POINT COMPANY
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BORING LOGS

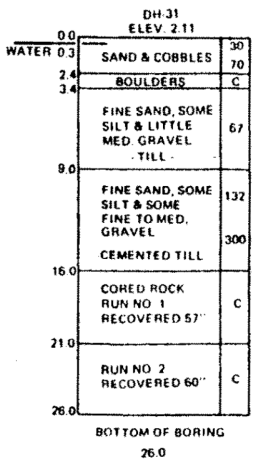
FIGURE 2.4–2D BORING LOGS











NOTE:
SEE FIGURE 2.4 - 2e FOR GENERAL NOTES.

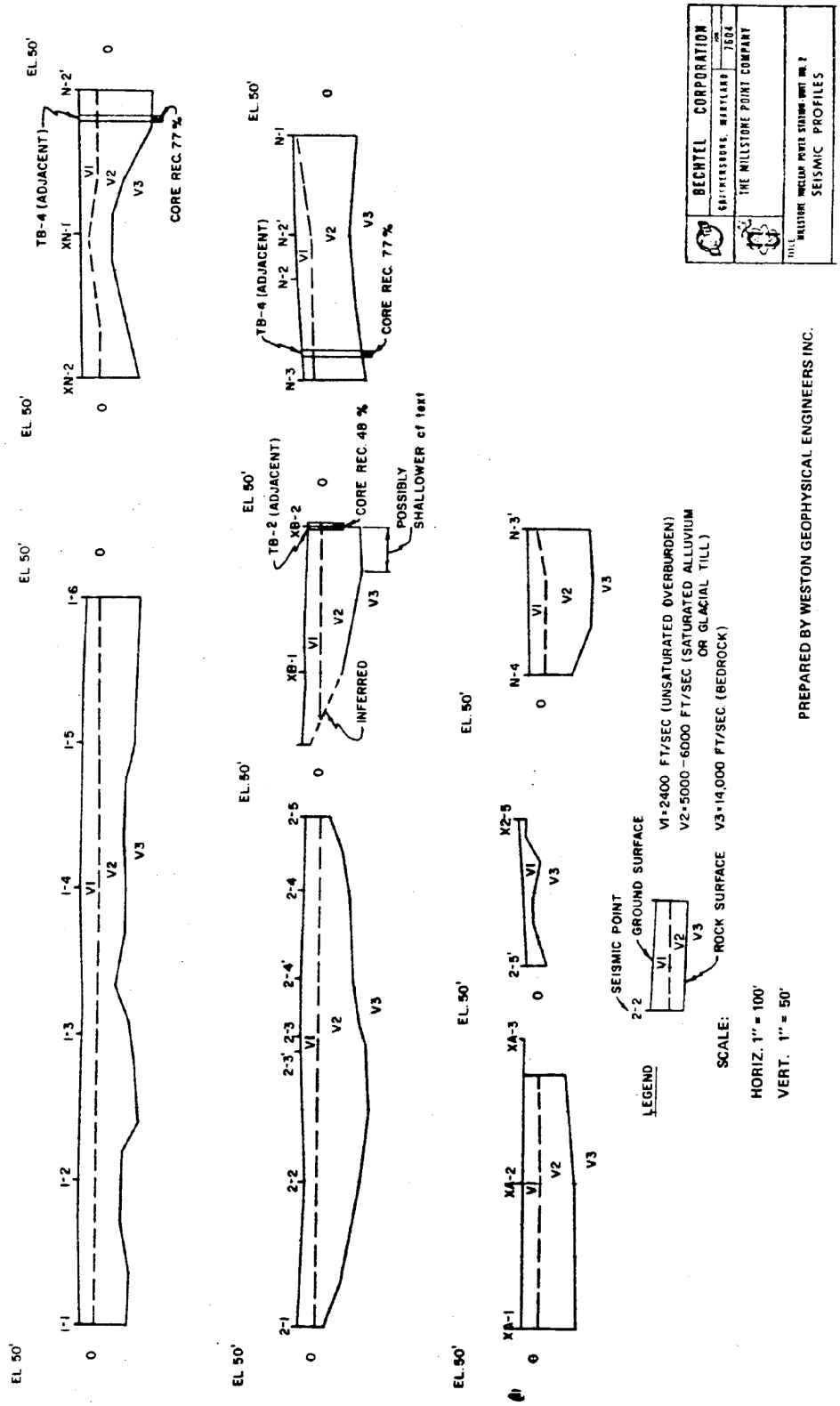
BECHTEL CORPORATION
GAITHERSBURG, MARYLAND
7604
THE MILLSTONE POINT COMPANY
TITLE
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BORING LOGS

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.4–3 SEISMIC LINE LOCATION MAP

[redacted]

FIGURE 2.4–4 SEISMIC PROFILES



2.5 HYDROLOGY

Information regarding hydrology of the Millstone site is presented in Section 2.4 of the Millstone Unit 3 Final Safety Analysis Report (Reference 2.5-1). With the exceptions given below, that information is incorporated herein by reference.

2.5.1 GENERAL

Millstone Point is located on the north shore of Long Island Sound. To the west of the site is Niantic Bay and to the east is Jordan Cove. Figure 2.5–1 shows the general topography of the Millstone area. Section 2.5.4 discusses the probable maximum hurricane used to calculate maximum water levels. Millstone Unit 2 structures are protected by flood walls and gates to elevation 22 feet MSL. Flooding protection of structures is discussed in Section 2.5.4.

The Millstone Site has several shallow wells near it, the nearest being one third of a mile from the station proper (see Figure 2.4-32 of Reference 2.5-1). None of these provide water for domestic purposes but one is used to water a nearby baseball field and to supply a drinking fountain at the field. Due to the relatively impervious bedrock base, it is improbable that any water accidentally released at the site surface could reach the wells. The larger communities near the site derive most of their water from reservoirs which are outside the surface drainage from the site.

Sea water will be pumped through the condenser to remove the residual heat from the steam exhausted by the turbine. This sea water will be taken from and discharged to Long Island Sound through lines which will be physically separated.

Since the site is a peninsula which juts into Long Island Sound, the high tides associated with hurricanes could produce flooding. This problem has been studied extensively, and plant buildings have been designed to withstand flooding.

2.5.2 PUBLIC WATER SUPPLIES

(See Section 2.4.1.2 (Hydrosphere) and 2.4.13 (Groundwater) of Reference 2.5-1.)

2.5.3 REGIONAL AND SITE WATER FLOW

The groundwater environment at the Millstone site is characterized by generally impermeable bedrock acquicludes overlain by soil masses of varying permeabilities. The bedrock is mostly Monson gneiss, with a Westerly granite dike intruding the gneiss in the quarry area. Neither rock is permeable, and there appears to be little movement of water through fissures in either formation since the quarry did not fill with either fresh or salt water after its abandonment in 1960. The overlying soil is composed of relatively dense glacial till of low permeability surmounted by ablation soils (drift) which are generally more porous and permeable. Both soils, as well as the rock formations, show sharp irregularities in depth at different locations. Gneiss outcrops in the higher terrain in the north part of the site indicate that this terrain is underlain by rock, creating a drainage divide, with groundwater transport to the east, west and south. This transport is probably accomplished mostly in the upper layers of the soil (the ablation soils and upper glacial tills being

more permeable than the underlying glacial tills), with a hydraulic gradient from high terrain to low lying areas or to Long Island Sound occurring when sufficient recharge from precipitation is available. The aquifer systems onsite can be best described as unconfined (water table) aquifers, with pronounced variations in depth at different locations. Locally perched water table conditions occur in some areas of soil stratification and shallow ponded water is frequently visible in localized bedrock troughs. The configuration of the water table for the site, thus, cannot be drawn with any degree of accuracy, but some limited information does exist. In May 1965, water levels in seven dug or driven wells were measured; the level in the west part of the site (Bay Point area) was +5 feet MSL; in the east part of the site (Jordan Cove) it was +2 feet MSL. A series of borings in 1969 in the area now taken up by Millstone Unit 2 structures indicated water table elevations between +4.3 feet. and +7.2 feet MSL with an average of +5.6 feet. MSL (Figure 2.5–2). These levels may have been higher than normal due to heavy rain preceding the test borings. The average influx rate of water into two test pits located at the north end of the Millstone Unit 2 turbine building (in August 1969) was approximately 8 gallons per hour over a 48 hour period (Figures 2.5–3 and 2.5–4 show the pit locations and dimensions). The sandy soils of the shore near the Millstone Unit 2 intake structure are highly permeable, since water levels in one test hole (number 110 on Figure 2.5–2) varied with water levels in Niantic Bay; however, these coarser sands are generally restricted to the shore area, and the information derived from the less permeable till material of the test pits is more characteristic of the overall permeabilities of the site soils.

The bedrock surface is exposed at the south end of the site but covered with a dense glacial till at the north end. Since both are quite impervious, precipitation does not sink into it readily, and much of it runs off on the surface directly into Niantic Bay or Jordan Cove. Some surface water collects in depressions in the northern part of the site.

In general, because of the complexities of achieving reliable data, the USC&GS does not attempt to gauge the flow in estuaries or embayments. Consequently, the fresh water discharge into Niantic Bay and Jordan Cove is not measured. These are the two bodies of water adjacent to the Millstone Point Site to the west and east, respectively. There are no established U.S. Geological Survey gauging stations on either the Niantic River or Jordan Brook, nor has the survey conducted any short term, special gauging studies.

An estimate has been made of the average flow into and out of the Niantic River due to tidal action. Assuming a mean cross-section of 1,620 square feet for the channel leading from Niantic Bay to the Niantic River and a mean tidal current of 0.77 knot, there would be a total flow of 45,489,600 cubic feet per each ebb or flood tide over a six hour period (average flow about 13,000 cu ft/sec). This figure should be taken as an approximation only. It is based on the best data available. (Cross-section estimate from the USC&GS, Chart 214; current velocity is from observations from the survey ship, MARMER during a 24 hour period in August, 1965 when it was located near the entrance to the Niantic River.) It is noted that only a small fraction of the 120,000 cfs tidal flow into which the discharge canal flow is diluted could possibly enter the Niantic River. There are no dams on the Niantic River and due to the limited drainage area, it is estimated that the most serious flood effects will come from the bay.

2.5.4 TIDES AND FLOODING

2.5.4.1 Normal Tides

(See Section 2.4.1.2 of Reference 2.5-1.)

2.5.4.2 Tides and Flooding Due to Storms

Information regarding tides and flooding due to storms is presented in Sections 2.4.2, 2.4.3, and 2.4.5 of Reference 2.5-1. With the exceptions given below, that information is incorporated herein by reference.

The various surge levels and associated most severe wind generated wave action is included in Table 2.5–1. Wind directions from 5 hours before maximum surge (T = -5) to ten hours after maximum surge (T = 10) are shown in relation to the plant layout in Figure 2.5–5.

Before maximum surge conditions occur, wave action is directed on the eastern shore, which is a sufficient distance from the plant such that local shore damage, if it should occur, would not affect the safe operation of the plant.

At T = -2 hours, the postulated surge level exceeds plant grade. However, both the Millstone Unit 2 intake structure and adjacent shoreline are protected from direct wave action by both the Millstone Units 1 and 2 plant structures until the surge reaches its probable maximum. Wind oriented waves directed between azimuth 130° and aximuth 160°, corresponding to T = 0 (probable maximum surge) and T = 3 hours, respectively, pass between the Millstone Unit 1 plant structure and intake structure. During probable maximum conditions, the south wall of the intake structure would be exposed to a maximum surge level of 18.11 feet and a runup of 3.69 feet such that the maximum water level would be elevation 21.80 feet MSL.

Between azimuth 160° and 170°, the Millstone Unit 1 intake structure protects the Millstone Unit 2 intake structure and adjacent shoreline. At approximately azimuth 170° (corresponding to T = 5 hours after peak conditions), direct wave action could impinge against the shoreline and the Millstone Unit 2 structure but at a severe oblique angle. It is not until T = 8 hours that waves are so directed as to effectively impinge on the intake structure. However, by this time, the surge height is reduced to elevation 6.8 feet MSL with wave heights of 7.5 feet, resulting in a total runup to elevation 10.6 feet MSL. Analytical studies indicate that this wave action has no detrimental effect on the rip-rap.

A typical profile through the plant and intake structure is shown in Figure 2.5–6. Additional profiles between the plant and shoreline are shown in Figure 2.5–7. Rip-rap placed in a 1-to-1 slope up to approximately elevation 13 feet MSL (and higher) extends from the Millstone Unit 1 intake structure to approximately 120 feet north of the Millstone Unit 2 intake structure. The area to the east and south of the Millstone Unit 2 intake structure, between the rip-rap and the access road, is covered with bituminous paving material and graded at approximately a 1 percent slope toward the shore.

The only possibility of a clapotis to form in front of the intake structure would occur at some time beyond the tenth hour after surge conditions. At this time, the wind velocity would be considerably lower than 40 mph and the fetch limited by the Niantic Bay such that a clapotis would be insignificant. However, the intake structure was conservatively designed to withstand the external forces of a clapotis based upon probable maximum conditions, with the winds reoriented to provide perfect reflection of waves.

In those instances when wind directed waves would impinge upon the front of the intake structure, the maximum runup (at 8 hours after probable maximum conditions) is limited to elevation 10.6 feet MSL. Since this does not exceed the bottom elevation of the operating floor, the operating floor will not be exposed to the effects of hydrodynamic surging. Such possibilities are further limited since water fluctuations within the intake structure would be damped by the energy lost in passage through the restricted openings in the trash racks and traveling screens. Internal water levels would be further attenuated due to the fact that the water must enter the structure through a submerged opening (elevation (-)10 to (-)30 feet MSL) through which the pressure response factor would be less than unity.

Slots covered by checkered plates are provided in the operating floor to allow for the passage of the storm surge (which has no dynamic effect) and to allow for the effective venting of air due to the remote possibility of hydrodynamic surging.

All affected plant structures are designed to accommodate the effects discussed above.

Note: Information concerning Resonance Phenomena is contained in Section 2.4.5.4 of Reference 2.5-1.

2.5.4.2.1 Study of Flooding Potential from Design Basis Hurricane

A maximum probable hurricane study for the Millstone plants was presented in Docket 50-245, Amendment 15, using the Hydrometeorological Section ESSA, HUR 7-97 Interim Report “Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coast of U.S.” The conclusions reached in this study showed a stillwater level slightly in excess of 16 feet and a maximum runup to elevation 17-18 feet MSL. This stillwater surge was calculated by combining the rise due to atmospheric reduction, wind set up, and an astronomical tide.

The stillwater surge level analysis was performed for three possible PMH configurations as follows: large radius, slow speed of translation (LR/ST); large radius, medium speed of translation (LR/MT); large radius, high speed (LR/HT). The PMH parameters and associated stillwater surge levels for these configurations are as follows:

PMH configuration	LR/ST	LR/MT	LR/HT
Central Pressure (inches Hg)	27.26	27.26	27.26
Peripheral Pressure (inches Hg)	30.56	30.56	30.56

PMH configuration	LR/ST	LR/MT	LR/HT
Radius to Maximum Winds (nm)	48	48	48
Angle of Radius to Direction of Translation (degrees)	115	115	115
Translational Speed (knots/hr)	15	34	51
Maximum Gradient Wind (mph)	124	124	124
Maximum (over water) Surface Wind (mph)	116	127	137
Stillwater Surge Level (feet MSL)	18.1	16.8	16.5

Therefore, it is observed that the large radius, slow speed of translation PMH yields the most severe design conditions.

An investigation was subsequently made to determine the effects of increasing the astronomical tide by one foot with a coincidental two foot forerunner on sea level anomaly. A recheck has also been made using a wind stress factor of 1.10 and a slightly modified storm track. It was determined that with a stillwater level of 19.17 feet MSL, the storm surge would exceed elevation 19 feet MSL for 36 minutes, elevation 18 feet MSL for 108 minutes, elevation 17 feet. MSL for 172 minutes, and elevation 16 feet MSL for 236 minutes.

The PMH storm track and surge traverse line are shown in Figure 2.5–10. For purposes of maximizing storm surge effects at Millstone, the PMH is required to track northwestward from the open ocean across Long Island and Long Island Sound, with the storm center impacting the Connecticut coast just east of New Haven. Historically, most hurricane movement in coastal areas at these latitudes is more to the north or northeast (and, consequently, the surge is less), but a northwestward movement is meteorologically feasible under appropriate circumstances. With storm movement in this direction, the direction of the maximum surge is approximately the same; the maximum surge would follow a fetch of surge traverse line running from the open ocean near Montauk Point to the Millstone site, passing between Little Gull Island and Fishers Island. The belt of maximum winds during the time of maximum surge is assumed to encompass this fetch area with wind direction parallel to the surge traverse line. The assumed bottom profile under the surge traverse line is shown in Figure 2.5–9. This profile was actually taken along a line from Westerly, Rhode Island SSE to the open ocean in order to satisfy the bathystrophic model constraint that the surge traverse line be perpendicular to the bottom contours; the profile location is shown in Figure 2.5–10.

The pertinent PMH parameters selected were as follows:

central pressure:	27.26 inches
radius to maximum wind:	48 nm
forward (translational) speed:	15 knots
maximum gradient wind:	123 mph
maximum surface wind (over water):	124 mph
peripheral pressure:	30.56 inches

Other combinations of storm size (radius to maximum wind) and forward speed were evaluated, but the above parameters produced the maximum surge.

At the time of maximum surge, the calculated components of the surge were as follows:

wind setup	12.47 feet
water level rise due to pressure drop	2.20 feet
astronomical tide	2.50 feet
initial rise	2.00 feet
total surge stillwater increase	19.17 feet MSL

Other factors used were as follows:

bottom friction factor	0.0025
wind stress coefficient	1.10

The total stillwater surge height of 19.17 feet was presented in Amendment 19 to Millstone Unit 1 FSAR (Docket No. 50-245); however, the 2.00 feet initial rise postulated to account for a forerunner anomaly is too high for the New England Coast, and the AEC has accepted a total stillwater surge height of 18.2 feet. (i.e., with a 1.00 feet initial rise as recommended by the Coastal Engineering Research Center, U. S. Army) as appropriate for the Millstone area (Appendix E, Docket Number 50-336, Safety Evaluation of Millstone Nuclear Power Station Unit Number 2 by the Division of Reactor Licensing, USAEC, August 7, 1970).

A 1.00 feet initial rise, giving a maximum stillwater level of 18.11 feet MSL is used as the basis for the surge and wave calculation presented in this amendment. The components of this maximum surge stillwater level are as follows:

wind setup	12.41 feet
water level rise due to pressure drop	2.20 feet
astronomical tide	2.50 feet
initial rise (forerunner)	1.00 feet
total surge stillwater level increase	18.11feet

Other factors used were:

bottom friction factor	0.0025
wind stress coefficient	1.10

The surge hydrograph for the site for T = -5 to T = +10 hours is presented in Figure 2.5–8.

Estimates of waves occurring from T = -5 to T = +10 hours are shown in Table 2.5–1. Wind directions and approach angles of these waves with respect to plant structures are shown in Figure 2.5–5. These wave estimates are based on the stillwater levels shown in the foregoing surge hydrograph (answer to Question 2.7.2); that is, they are based on a surge with a 1 feet forerunner anomaly and a maximum stillwater level of 18.11 feet MSL.

The maximum wave level of 42.5 feet MSL was obtained at the vertical wall of the intake structure at T = +0.8 with a large radius (LR) high speed (HT) PMH. The resulting stillwater level with the incident was calculated to be 18.1 feet MSL. The basis for this value can be found in the

response to Question 2.10, Amendment 14, to the PSAR for Millstone Nuclear Power Station, Unit No. 3, License Application, Docket No. 50-423, dated December 14, 1973.

Flood design considerations are discussed above and in Section 2.4.2.2 of Reference 2.5-1. Plant procedures address necessary precautions and actions to take in the event of anticipated hurricane, tornado, or flood conditions.

Meteorological conditions are determined from weather service forcasts and / or site meteorological instrumentation.

A. Action By Plant Personnel

Upon receiving hurricane warnings from weather service providers, Millstone operators implement appropriate procedures. Actions taken on Millstone Unit 2 include:

1. All plant flood gates are closed.

2. Traveling 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 miles per hour be measured at the meteorological tower, 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 committee(s) as described in the Quality Assurance Program Description Topical Report. 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 feet MSL in accordance with the Technical Requirements Manual.

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 water from shorting pump motors.

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.

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 approaches 16.5 feet MSL which is just below the level of the Circulating Water Pump Motor Variable Frequency Drives (VFD). At this time, the plant would initiate a shutdown and be 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 feet 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 feet 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 14 feet 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 feet 6 inches, sample/remove/verify no water in diesel fuel oil storage tank or transfer piping, and disassemble, inspect and restore transfer pump motors.

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.

   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 3, one electric-driven fire pump from Unit 2 and a diesel-driven fire pump. 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 feet MSL is, therefore, sufficient to protect the pumps against the maximum PMH standing water level of 18.11 ft MSL. Although power for the electric-driven fire pumps may not be available, the diesel-driven fire pump remains available since both the pump and its auxiliaries are located within the flood protected fire pump house.

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 until the protected Service Water Pump Motor is restored.

3. Because one diesel's cooling system is modified such that it can be cooled by fire water, and especially because emergency diesel-generated power is not necessary for decay heat removal, cooling water from the service water system is not essential for decay heat removal. Nevertheless, the operability of the service water pumps is as specified and justified below:

a. Two service water pumps operating until the level of water in the intake structure reaches 22 feet MSL.

b. When the criteria specified in the Technical Requirements Manual is met, the motor for the third service water pump is secured and protected from flooding by lowering and tying down an open bottomed, closed top, "can" over the motor. The length of this "can" is designed to prevent water from rising above elevation 22.0 feet MSL inside. This treatment will assure that this service water pump will be readily available when the water recedes. The criteria specified will give four hours lead time before water crosses plant grade. The entire operation of protecting the motor can be accomplished within two hours.

c. Thus, by the measures previously described, service water will be maintained, except for a maximum period of 9.1 hours. This is based on the duration of the intake water level above 22 feet MSL which is calculated to be 2.3 hours plus an additional 4.8 hours for the water level to recede to elevation 14 feet MSL, plus the time to recommission the previously protected motor which is 2 hours.

d. Additionally, the other service water pump motor(s) will be removed, disassembled, steam cleaned, dried, reassembled and reinstalled in the field. Thus, at least one additional service water pump could be placed in service.

4. Section 8.5 of the FSAR describes the loads which will be placed on the DC system on a loss of AC power. Table 8.5-1 lists the specific loads and battery discharge rate, assuming one battery is out of service and the DO1-DO2 bus tie breakers closed. Since the Technical Specifications require both batteries to be in

service for plant operation, the discharge rate from the two batteries is such that necessary equipment can be supplied for a minimum period of 8 hours without charger support. Thus, although action has been taken to maintain AC power, the capacity of the batteries provides ample time to restore AC power if it was lost.

C. Conclusion

By the action described and the justification provided, it is concluded that the plant can readily be maintained in a safe condition throughout a PMH.

As a matter of record, no such phenomenon has been evident in any of the surges along the New England coast as plotted in the Weather Bureau Technical Paper Number 48. No mention has been made of the forerunner in the Corps of Engineers" hurricane studies for New London and Stratford, Connecticut. In Technical Paper Number 48, it is stated that, "...the data presented in this report give little support for the concept of a forerunner heralding the approach of a hurricane" and also stated that, "...short period anomalies in mean sea level not related to the hurricane, but not fully explained, may account for some of the reported forerunners." To our knowledge, the only area where a forerunner or anomaly of two feet has been used by the Corps of Engineers is in the Galveston area where the monthly mean sea level from 1919-1961 period varied by a little more than two feet. Along the New England coast during this same period, the monthly mean sea level has varied less than one foot.

At the time of the peak surge, the wind is from the southeast direction, and the wave attack would be along the large axis of the point. Millstone Unit 1 would, thus, "shield" Millstone Unit 2 from the direct wave attack.

The significant exposures of the site are to the southwest where the plant is located closest to the open shoreline and to the southeast where it is located closest to the quarry. On the southwest exposure, the minimum distance from the plant to the shoreline is about 220 feet, and the minimum distance to the top of the slope of plant fill is about 180 feet. On the southeast exposure, the minimum distance from the plant to the quarry is about 400 feet. The area between the plant and the quarry has been filled with high point graded to elevation 14.5 feet MSL and low point to elevation 14.0 feet MSL. The pertinent elevations of the Millstone Unit 2 plant are shown in Figure 2.5–6.

Plans for protection of the shores immediately north and south of the Millstone Unit 2 Intake Structure are shown in Figure 2.5–12. Sectional views of protective structures and bottom profiles are shown in Figure 2.5–13. Wave forces for design are conservatively based upon these profiles and in accordance with the procedures presented in the "Shore Protection Manual," considering the PMH conditions for the site. Breaking wave forces are computed as a function of the water depth, wave period, and bottom profile. The design of the protective structures is based on the criteria for stability as contained in "Technical Report Number 4, Shore Protection Planning and Design" and the "Shore Protection Manual" by CERC. Maximum breaking wave heights are determined in accordance with the method presented by Weggel, ASCE Journal of Waterways, Harbors and Coastal Engineering, November 1972, and the Shore Protection Manual, 1974.

Analysis of the dynamic wave forces on the wall between the Millstone Units 1 and 2 intakes shows that a storm surge level of 14 feet MSL and the associated wave height results in the maximum dynamic forces on the structure. The wall between the two intakes is divided into three zones for design purposes. The wave force diagrams for the three zones and their corresponding bottom profiles near the walls are shown in Figure 2.5–14. The breaking wave height for the shore reach north of the Millstone Unit 2 intake, excluding the area of the wing wall, is based on the maximum allowed breaker controlled by bottom topography as obtained in the PMH study for the site. However, the bottom topography allows a non-breaking wave to reach the wall for the reach immediately north of the Millstone Unit 2 intake. Wave force diagrams for the wall and the wing wall north of the Millstone Unit 2 intake are also shown in Figure 2.5–14.

Table 2.5–2 presents a summary of surge levels and wave conditions which were used in the design. Wave periods, other than the design period of 9.4 seconds, were considered and were determined to yield maximum effect on the dynamic force of ± 10 percent.

The shores are protected by post-tensioned, reinforced concrete walls which are founded upon bedrock. In consideration of the large magnitude of the computed wave forces due to the PMH conditions, the post-tensioned system was selected as a feasible and positive method by which these extreme forces could be sustained. The anchorage system consists of five to eleven strands, consisting of seven wires per strand, which are anchored into bedrock by drilling and grouting. Upon completion of the post-tensioning operation, the anchorages are also encased in concrete. A typical anchorage detail is shown in Figure 2.5–15.

During installation, each anchorage is prestressed to 80 percent of the guaranteed minimum ultimate material strength and relaxed to 70 percent and allowed to remain in place for twelve hours, at which time the anchorage is checked at lift-off to insure integrity of the anchorage load. Design value for each anchorage is 60 percent of guaranteed minimum ultimate material strength.

The walls are designed to be stable against all forces, including buoyancy, which may possibly occur from extreme low water to the PMH condition. The area immediately back of the walls is protected with quarry stones having a gradation designed to prevent scour that may possibly result from overtopping waves. Wall stability is not dependent on fill material, i.e., the wall is stable in a free standing condition under wave attack. The wing walls (transition areas), which are located immediately north and south of the Unit 2 intake, were analyzed in consideration of all additional loads imposed by the added sea walls and are stable as composite sections under all loadings. The post-tensioning anchorage system in these areas extends through all previously placed concrete, and the structural anchorage is developed in bedrock as shown in the typical anchorage details.

All existing stones within the immediate vicinity of the intake are removed.

The walls are provided with drainage pipes located approximately fifteen feet apart at the base of the walls. A filter system is provided behind the walls, having a gradation designed to prevent loss of soil material, and to provide relief against permanent hydrostatic pressure build up (even though the walls are designed to resist such pressure as stated previously). The filter system and the gradation of the filter materials are shown in Figure 2.5–16. The graded materials are quality controlled in accordance with the method as outlined in ASTM Specification D-2940-71T.

Protection is provided in the area behind the walls to prevent scour due to PMH conditions. The extent of the area to be protected is shown in Figure 2.5–17. The extent of scour behind the walls is based on the results of laboratory studies reported in Beach Erosion Board (CERC) Technical Memorandum No. 134, entitled "Beach Profiles as Affected by Vertical Wall." The maximum scouring velocity is estimated using solitary wave theory. The size of stone as shown in Figure 2.5–16 is obtained from the Shore Protection Manual (CERC). For the corners around Millstone Unit 2 and north corner of Millstone Unit 1 intake, stones having a weight of 600 pounds minimum and 1,000 pounds maximum are provided.

Construction materials for the sea walls are as follows:

1. Concrete shall have a specified minimum compressive strength of 5,000 psi at 28 days.

2. Reinforcing steel shall conform to Specification ASTM A615, Grade 60, except that bent ties shall meet the provisions of Specification ASTM A615, Grade 40.

3. Post-tensioning anchorage material shall conform to Specification ASTM A-416, Grade 270, "Uncoated stress-relieved strand for prestressed concrete."

4. Screens at drainage pipes shall conform to Specification ASTM A240, Type 304.

5. Gradation of fill material is shown in Figure 2.5–16.

Storms approaching the site from (1) the east, (2) the north, or (3) the southeast could conceivably provide the mechanisms to produce setdown at the site. However, because of the location of Millstone Point in a relatively open area of Long Island Sound and because of prior effects that the PMH would have to have caused before getting into a position where it could produce setdown, the phenomenon appears unlikely to occur to any significant extent at Millstone.

The following is a general discussion of the three storm tracks and possible setdown effects. There is no specific or definitive information available in the form of mathematical models giving numerical values for setdown in an open coast situation, nor can be bathystrophic surge prediction model used to predict maximum stillwater level be applied in reverse to this kind of phenomenon.

Case 1: A hurricane approaching from the east, tracking along the Atlantic coast of Long Island, from east to west, is not common but can be postulated. Its maximum wind field, oriented from east to west over Long Island, would build a surge along the length of the Sound directed toward the New York City area. However, any water outflow from the Millstone area to support this surge would be balanced by an inflow from the open ocean simply due to level difference. This inflow would be augmented by a local water rise due to low pressure and by concurrent onshore transport of water by southeast winds, all of which would tend to keep the stillwater level at Millstone elevated even in the event of some loss of water to a westward moving surge. Also, the prior locations of the storm would have favored

the net transport of water into the Sound; that is, east and northeast winds would have preceded the storm.

Case 2: A hurricane approaching from the north is unlikely. The upper wind structure guiding hurricanes at this latitude is generally strong enough so that the looping tracks occasionally seen in lighter (tropical) wind regimes do not occur. However, postulating a loop and a northerly approach to Millstone from the Boston area, with passage of the storm from north to south approximately over Providence, strong northwesterly offshore winds could occur at Millstone. Because of the friction effects of an overland trajectory and a partial cutoff of the hurricane from its energy source (warm water), these northwesterly winds would no longer be at hurricane force and would probably be similar in intensity to those of a strong wintertime extratropical coastal cyclone passing to the east of the site. Since several of these storms pass up the coast in a normal winter, available historical records of minimum low water should be indicative of the setdown effects of a hurricane with this orientation.

In addition, water outflow from the Millstone area would be partially compensated for by inflow from the western portion of Long Island Sound, and the pressure effect on stillwater level would be opposition to the wind effect, both factors mitigating setdown.

Case 3: A hurricane approaching from the southeast would have to move northwestward over Providence so that the maximum offshore winds would occur at Millstone. However, in this orientation, prior locations of the storm would have favored net transport of water into the Sound; that is, east and northeast winds would have preceded the storm. In addition, the pressure effect in favor of stillwater level increase is operative, and a net inflow of water to the Millstone area from the western part of the Sound would occur, both factors working against setdown. The offshore winds might be stronger than in Case 2, but the attenuating effects of overland friction would still result in considerable reduction of wind speeds. As in Case 2, available historical records of minimum low water should be indicative of the setdown effects of a hurricane with this orientation.

As discussed in Section 2.5.4.2.3 of the FSAR the only safety-related equipment housed in the intake structure is the service water cooling pumps. The historical low tide at New London was about minus 4.37 feet MSL on December 11, 1943. Additionally, extreme low water as shown on USC&GS Chart 214 is minus 3.5 feet MLW (minus 4.47 feet MSL). The Applicants conclude that sufficient margin exists between historical low water and the service water pump design low water level of minus 7 feet MSL to assure continuous operation under maximum setdown conditions.

2.5.4.2.2 Flood Protection for Plant Structures

The design of Millstone Unit 2 reflects the decision to provide flood protection up to elevation 22 feet MSL minimum for the containment, turbine, and auxiliary buildings. The containment is

flood protected to an elevation in excess of elevation 22 feet MSL by its reinforced, poured concrete walls and its normally closed, airtight penetrations. The poured concrete walls of the auxiliary and turbine buildings are to elevation 22 feet MSL.

All penetrations into the auxiliary and turbine buildings are provided with hinged flood gates to elevation 22 feet MSL to assure water tightness against both the water and any debris in the water.

Flood gates, with sealing compressive membranes, are used wherever possible. If the gates at any opening pose an operational encumbrance, stop logs with equivalent sealing capability are used in those locations. The openings which have protection are shown in the plan review, Figure 2.5–18.

In the event of severe local flooding or when standing water crosses the plant, appropriate procedures will be implemented.

The drainage system throughout the Millstone Unit 2 site, including the roof and yard drainage, is designed based on an intensity of three inches per hour rainfall.

The roof area is divided into four subareas, relative to the place where the header drains are connected to the underground storm drain system as shown in Figure 2.5–19. The protected roof surface area, number and size of roof drains are given in Table 2.5–3.

To prevent an accumulation of water which could exceed the design load of the roof, if roof drains are clogged, the parapet walls are provided with scuppers.

The yard, immediately adjacent to Millstone Unit 2 buildings, is similarly divided into ten subareas, each one draining into a catch basin as shown in Figure 2.5–20. All catch basins, except CB number 1, are connected by a storm sewer system draining into Niantic Bay next to the north side of Millstone Unit 2 intake structure. Catch Basin number 1 is connected to the existing Catch Basin number 1 of Millstone Unit 1.

Table 2.5–4 includes all catch basins, top elevation, area draining into each and runoff flowing in and out of each catch basin.

The rational method is used to determine the runoff within the Millstone Unit 2 site.

With a rainfall intensity of 9.4 inches per hour, the total runoff is 60.0 cfs. Inlet times are very small, therefore, their effect on the rainfall intensity is neglected. The runoff coefficient is estimated to be equal to 1.0, considering the entire area is paved and the infiltration losses are nil. The storm sewer from Catch Basin CB number 9 to its outfall into Niantic Bay adjacent to the intake structure is twenty-four inches in diameter and has a maximum flow capacity of 8.8 cfs. During the maximum rainfall, the capacity of the sewer will be exceeded. Excess runoff will be accumulated in the yard area until it reaches elevation 14.5 feet MSL and will overtop the road. Water will then flow out into Jordan Cove on the east and Niantic Bay on the west.

All drain connections from the buildings to the Millstone Unit 2 storm drain system are provided with backwater valves, preventing water to backflow into the buildings at any time; also, stoplogs and flood gates are provided at all entrances to prevent water from flowing in.

In the winter, combined precipitation and ice accumulation in excess of the design load are prevented by the provision of scuppers in the parapet walls. Therefore, it is concluded that all the safety-related structures and equipment are capable of withstanding PMP without loss of safety-related functions.

At the interface between Millstone Units 2 and Unit 1, the flood protection capability of Millstone Unit 1 was originally credited in the evaluation of Millstone Unit 2 flood protection capability. The entire periphery of Millstone Unit 1 was flood protected. However, with the decommissioning of Millstone Unit 1, the flood boundary was revised to support the decommissioning effort. As shown in Figure 2.5–18 a flood wall is provided to a minimum elevation of 22 feet 0 inches along the common area between Unit 1 and Unit 2 Turbine Buildings. Protection for the Millstone 2 Auxiliary Building is provided by the adjacent Millstone Unit 1 Control Building which is provided with flood protection on the south and east walls to a minimum elevation of 22 feet 0 inches and at the 14 feet 6 inches floor elevation. Therefore the entire southern interface between Millstone Units 2 and 1 is flood protected to a minimum elevation of 22 feet 0 inches.

Weather service forecasts provide adequate hurricane status such that there will be sufficient time to secure the plant against flooding. Securing the plant will be done at the discretion of the shift supervisor and can be accomplished by one man who would close and lock all of the hinged gates and install the flood logs. The flood logs will be specifically designated as pieces of flood protection equipment and stored in the vicinity of the place where they will be used.

During a hurricane, the plant operations will be in accordance with normal, abnormal and/or emergency operating procedures.

Millstone Unit 2 can be brought safely from an operating condition to either a hot or cold shutdown condition with the equipment listed in Tables 7.6-1 and 7.6-2. The philosophy of flood protection has evolved from the need to protect this required equipment which is shown at its various elevations, in Figures 2.5–21 and 2.5–22.

It has been seen that Millstone Unit 2 is extremely well protected against flooding, even on the side which adjoins Millstone Unit 1. All emergency shutdown equipment, including the auxiliary power sources are flood protected.

The average ground elevation around the plant buildings is at elevation 14.0 feet MSL. With the maximum surge stillwater elevation of 18.1 feet MSL, based on three possible PMH configurations, the maximum depth near the building (except for the intake structure) is equal to 4.1 feet. A wave height equal to 78 percent of the maximum available depth or 3.2 feet could be generated anywhere around the buildings (except for the intake structure). These waves could produce runup to elevation 25.1 feet MSL. The containment building, auxiliary building and the warehouse building have exterior concrete walls up to elevation 54.5 feet MSL minimum. The

turbine building and the enclosure building have metal siding above the top of the flood wall at elevation 22.0 feet MSL. The metal siding is continuous over the exterior flood wall and is connected to the flood wall with waterproof caulked connections. Although runup to elevation 25.1 feet MSL could be generated, the siding would prevent water resulting from splashing effects from entering the building. It is concluded that there can be no adverse effects on any of the safety-related buildings due to the design waves.

2.5.4.2.3 Intake Structure Flood Protection

The intake structure, as shown in Figure 2.5–23, is constructed of reinforced concrete with an invert Elevation of minus 27 feet MSL, operating deck at elevation 14 feet MSL, and a cutoff wall to Elevation minus 10 feet MSL, all based on mean sea level being at zero feet.

The only safety-related system in the intake structure is the service water system. The service water pump motors and associated electrical and control equipment are protected to elevation 22 feet MSL. The service water cooling pumps are also designed for a low water of elevation minus 7 feet MSL which is 2.2 feet lower than the extreme low water shown on USC&GS Chart 214.

The intake structure was analyzed both statically and dynamically for the standing wave effects. The maximum and minimum pressures at the foot of the intake structure 30 feet below the mean sea level were calculated to be 3,960 and 2,220 psf, respectively. The net uplift pressure on the operating floor was found to be 930 psf. The stability of the structure was studied and found stable under these conditions. The louvers in front of the structure are protected by specially designed structural frames which are shown in Figure 2.5–24 and are capable of withstanding a maximum pressure of 1,120 psf due to pressure from a non-breaking wave.

The maximum water level inside the intake structure caused by the standing wave condition is calculated to reach elevation 26.5 feet MSL. The analysis was performed utilizing an unsteady state mathematical model that takes into account the profile of the incident wave, inleakage through the louvers and system headloss. One service water pump motor is protected.

Probable minimum low water level at the intake structure resulting from an occurrence of a probable maximum hurricane (PMH) oriented so as to cause maximum depression of the water surface (setdown) at the site is calculated to be minus 5.85 feet MSL as indicated in Section 2.4.11 of the Millstone Unit 3 Final Safety Analysis Report (Reference 2.5-1).

The design low water level of the circulating water pump is minus 3.5 feet MSL, and the service water pump level is minus 7.0 feet MSL. The fire water pumps are supplied from two 245,000 gallon storage tanks connected to the public water system of the town of Waterford. The probable minimum low water has no effect on the service water pump.

2.5.4.2.4 Flood Protection of Electrical Equipment

With the provision for concrete flood protection walls around the enclosure building, turbine building and auxiliary building up to elevation 22 feet MSL, and with flood gates at any external openings in these walls, overflow into and subsequent accumulation of flood water within these

buildings is expected to be negligible for the maximum given stillwater level to elevation 19.17 feet MSL and a 2.5 foot wave runup. All essential electrical MCCs, switchgear and panels located below elevation 22 feet MSL are in close proximity to stairways or other openings to lower floors to render accumulation of water at a given location impossible. Where this is not the case the equipment is mounted on a four inch raised pad.

Power and control cables, cable terminations and any electrical devices required for the trouble-free operation of the service water pumps in the intake structure are located above elevation 22 feet MSL where possible. Any of these located below this point are of tight construction.

Physical separation is provided between redundant essential electrical equipment and circuits required for safe shutdown.

Entry of cables connecting outdoor equipment to equipment within the flood protected areas is so designed to preclude leakage or overflow of flood water into these areas by provision of proper seals and/or by carrying cable raceways to elevation 22 feet MSL. Outdoor transformers and switchgear are protected to elevation 19.5 feet MSL.

2.5.4.2.5 Underground Tanks

There are no underground storage tanks to supply fuel at Millstone Unit 2.

2.5.4.3 Prevention from Icing

It is highly unlikely that the formation of ice would occur in front of the intake structure in such a manner as to obscure flow. The design of the intake structure, its entrance velocity, and rakes on the trash racks prevent ice formations from obscuring flow. Typical types of ice that could be present at the intake structure are: surface ice and frazil ice.

Surface ice is prevented from obscuring flow due to the submerged opening entrance to the intake structure. The entrance is located at an elevation of -10 to -27 ft MSL. In order to obscure flow, weather temperatures would have to be cold enough to freeze the external sea water in the area of the intake greater than 10 feet at MSL. This has not happened in the history of operation of Millstone Unit 2.

Frazil ice (needle-shaped ice crystals suspended in water) occurs in the presence of supercooling when turbulence is too great to allow surface ice to form. Frazil ice can build on surfaces, such as trash racks and walls, which have a temperature at or below freezing thereby obscuring flow into downstream components and systems such as the SW and CW Systems.

However, frazil ice only remains suspended and submerged in a flow of water if velocity is greater than 2 fps. At the entrance of the intake structure water velocity is significantly below this value. Since the water velocity is below the minimum velocity needed to keep frazil ice suspended and submerged, it will raise to the surface prior to contact with the trash racks.

2.5.5 OCEANOGRAPHY

2.5.5.1 Water Temperature

(See Section 2.4.11.6 (Heat Sink Dependability Requirements) of Reference 2.5-1.)

2.5.5.2 Current Velocity and Volume Flow in Channel

(See Section 2.4.1.2 (Hydrosphere) of Reference 2.5-1.)

2.5.5.3 Effluent Dilution

(See Section 2.4.12 (Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters) of Reference 2.5-1.)

2.5.6 REFERENCES

2.5-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.4-Hydrologic Engineering.

TABLE 2.5–1 EFFECT OF HURRICANE GENERATED SURGE AND WAVES ON UNIT NUMBER 2 STRUCTURES

Time Hour	Wind Direction Azimuth	Wind Speed MPH	Surge Stillwater Level				Total feet	Significant Wave			Maximum Wave			Runup to Elevation feet	Breaking Location	Impact Point of Runup	Safety Related Structure Affected (Unit 2 Only)
			Wind Setup feet	Pressure Drop feet	Astrotide feet	Fore-Runner feet		Height feet	Period feet	Length feet	Height feet	Period feet	Length feet				
T - 5	076°	67	5.98	0.86	2.50	1.00	10.34	2.4	2.7	37	4.0	4.3	95	11.3	Jordan Cove Shore	Shore Line	None
T - 4	079°	75	7.34	1.02	2.50	1.00	11.86	2.7	2.9	43	4.5	4.5	104	12.9	Jordan Cove Shore	Shore Line	None
T - 3	081°	84	9.06	1.23	2.50	1.00	13.79	3.0	3.0	46	5.0	4.8	118	15.1	Jordan Cove Shore	Shore Line	None
T - 2	084°	93	10.86	1.54	2.50	1.00	15.90	1.1	1.6	13	1.5	2.4	29	16.6	East Faces of Plant Buildings	East Walls of Unit 1 Rad. Waste Storage Building & Unit 2 Auxiliary Building	Diesel Generator
T - 1	102°	94	12.10	1.92	2.50	1.00	17.52	2.3	2.8	40	3.1	3.2	52	18.9	East Faces of Plant Buildings	East Walls of Unit 1 Rad. Waste Storage Building & Unit 2 Auxiliary Building	
T - 0	132°	90	12.41	2.20	2.50	1.00	18.11	2.8	2.9	43	3.7	3.8	74	21.8	Unit 1 Buildings. & Discharge Structures	South Wall of Unit 1 Reactor Building Warehouse and Unit 1 & 2 Discharge Structures	None for Unit 2
T + 1	140	86	12.20	2.00	2.50	1.00	17.70	2.5	2.6	35	3.3	3.4	59	19.9	Unit 1 Buildings & Discharge Structures	South Wall of Unit 1 Reactor Building Warehouse and Unit 1 & 2 Discharge Structures	None for Unit 2
T + 2	148	83	11.19	1.62	2.50	1.00	16.31	1.4	2.3	27	1.9	2.6	35	17.3	Unit 1 Buildings & Discharge Structures	South Wall of Unit 1 Reactor Building Warehouse and Unit 1 & 2 Discharge Structures	None for Unit 2
T + 3	157	81	9.88	1.29	2.50	1.00	14.67	0.2	0.8	4	0.3	1.1	6	14.8	Unit 1 Buildings & Discharge Structures	South Wall of Unit 1 Reactor Building Warehouse and Unit 1 & 2 Discharge Structures	None for Unit 2
T + 4	168	81	8.50	.05	2.50	1.00	13.05	10.5	6.2	197	17.5	9.0	415	18.4	Unit 1 Warehouse and Intake Structure	South Wall of Unit 1 Warehouse and Intake Structure	None for Unit 2
T + 5	179	75	6.72	0.88	2.50	1.00	11.10	9.5	6.1	191	19.9	8.5	370	15.9	Unit 1 Warehouse and Intake Structure	South Wall of Unit 1 Warehouse and Intake Structure	None for Unit 2
T + 6	191	66	5.21	0.77	2.50	1.00	9.48	9.0	6.0	184	5.0	8.3	353	14.0	Unit 1 & 2 Intake Structure	South Wall of Unit 1 Warehouse and Intake Structure	Intake Structure

TABLE 2.5–1 EFFECT OF HURRICANE GENERATED SURGE AND WAVES ON UNIT NUMBER 2 STRUCTURES (CONTINUED)

Time Hour	Wind Direction Azimuth	Wind Speed MPH	Surge Stillwater Level				Total feet	Significant Wave			Maximum Wave			Runup to Elevation feet	Breaking Location	Impact Point of Runup	Safety Related Structure Affected (Unit 2 Only)
			Wind Setup feet	Pressure Drop feet	Astrotide feet	Fore-Runner feet		Height feet	Period feet	Length feet	Height feet	Period feet	Length feet				
T + 7	203	58	4.04	0.67	2.50	1.00	8.21	8.5	6.1	191	14.2	8.1	336	12.5	Unit 1 & 2 Intake Structure	South Wall of Unit 1 Warehouse and Intake Structure	Intake Structure
T + 8	209	51	2.74	0.59	2.50	1.00	6.83	7.5	5.8	172	12.5	7.5	288	10.6	Unit 1 & 2 Intake Structure	Southwest Corners of Unit 1 & 2 Intake Structure	Intake Structure
T + 9	211	46	1.59	0.53	2.50	1.00	5.62	6.8	5.7	166	1.3	7.2	265	9.3	Unit 1 & 2 Intake Structure	Southwest Corners of Unit 1 & 2 Intake Structure	Intake Structure
T + 10	213	41	0.52	0.49	2.50	1.00	4.51	6.0	5.5	155	10.0	6.8	237	8.0	Unit 1 & 2 Intake Structure	Southwest Corners of Unit 1 & 2 Intake Structure	

TABLE 2.5–2 DESIGN WAVE CONDITIONS

Profile Designation	Reach	Type of Wave	Storm Surge Level (feet) MSL	Wave Height (feet)	Wave Period (Seconds)
A-A	Between Units 1 and 2	Breaking	14	14	9.4
B-B	Thru South Wing Wall Unit 2	Non Breaking	12.7	19.3	9.4
C-C	Thru North Wing Wall Unit 2	Non Breaking	12.7	19.3	9.4
D-D	North of Unit 2 Approximately 120 feet North of the Wing Wall	Breaking	12.7	19.3	9.4

TABLE 2.5–3 ROOF SURFACE AREA AND NUMBER AND SIZE OF ROOF DRAINS

Subarea Number	Surface Area (square feet)	Number of Roof Drains	Size of Roof Drains
1	25980	29	4 inches ϕ
2	21890	8	4 inches ϕ
3	16098	4	4 inches ϕ
4	22085	12	4 inches ϕ
Totals	86053	53	

TABLE 2.5–4 ALL CATCH BASINS FOR ELEVATION AREA DRAINING INTO AND RUNOFF FLOWING IN AND OUT OF A CATCH BASIN

Catch Basin Number	Top Elevation Feet (MSL)	Drainage Area Acres	Runoff CFS	Total Runoff CFS [a]	Outlet Capacity (CFS)
Catch Basin Number1	13.5	0.26	2.44	2.44	2.86
Catch Basin Number 2	13.5	0.20	1.88	1.88	2.33
Catch Basin Number 3	13.5	0.39	3.67	5.55	4.57
Catch Basin Number 4	13.25	0.57	5.36	10.91	6.87
Catch Basin Number 10	13.25	1.21	11.37	22.28	5.69
Catch Basin Number 5	13.25	1.52	14.29	36.57	8.69
Catch Basin Number 6	13.5	1.17	11.00	47.57	11.24
Catch Basin Number 7	13.5	0.39	3.67	51.24	13.44
Catch Basin Number 8	13.5	0.25	2.35	53.59	8.77
Catch Basin Number 9	13.5	0.69	6.49	60.08	8.77

a. Includes runoff from upstream and any possible roof drain connection.

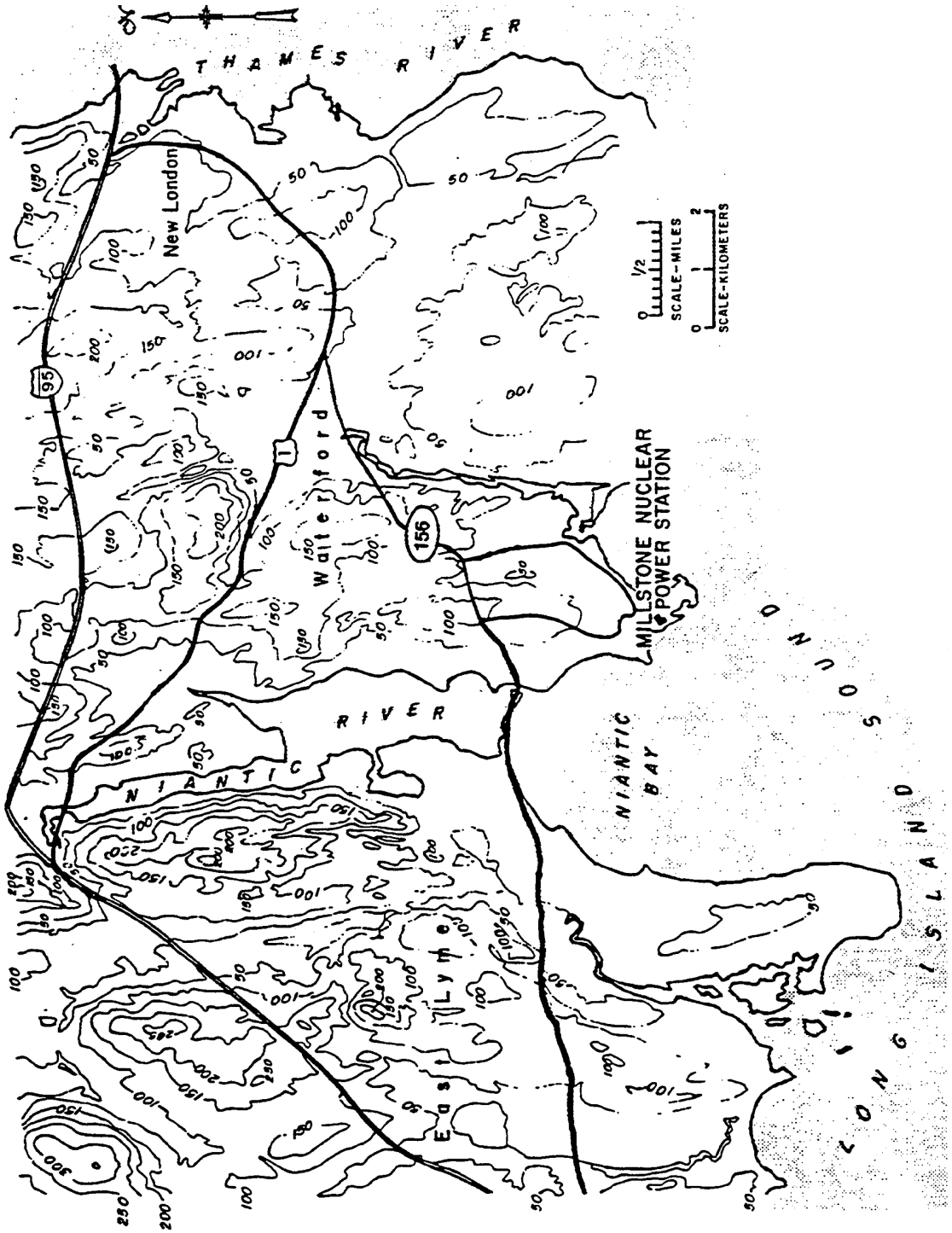


FIGURE 2.5–1 TOPOGRAPHY IN THE VICINITY OF MILLSTONE POINT

FIGURE 2.5–2 BORE HOLE AND TEST PIT LOCATIONS

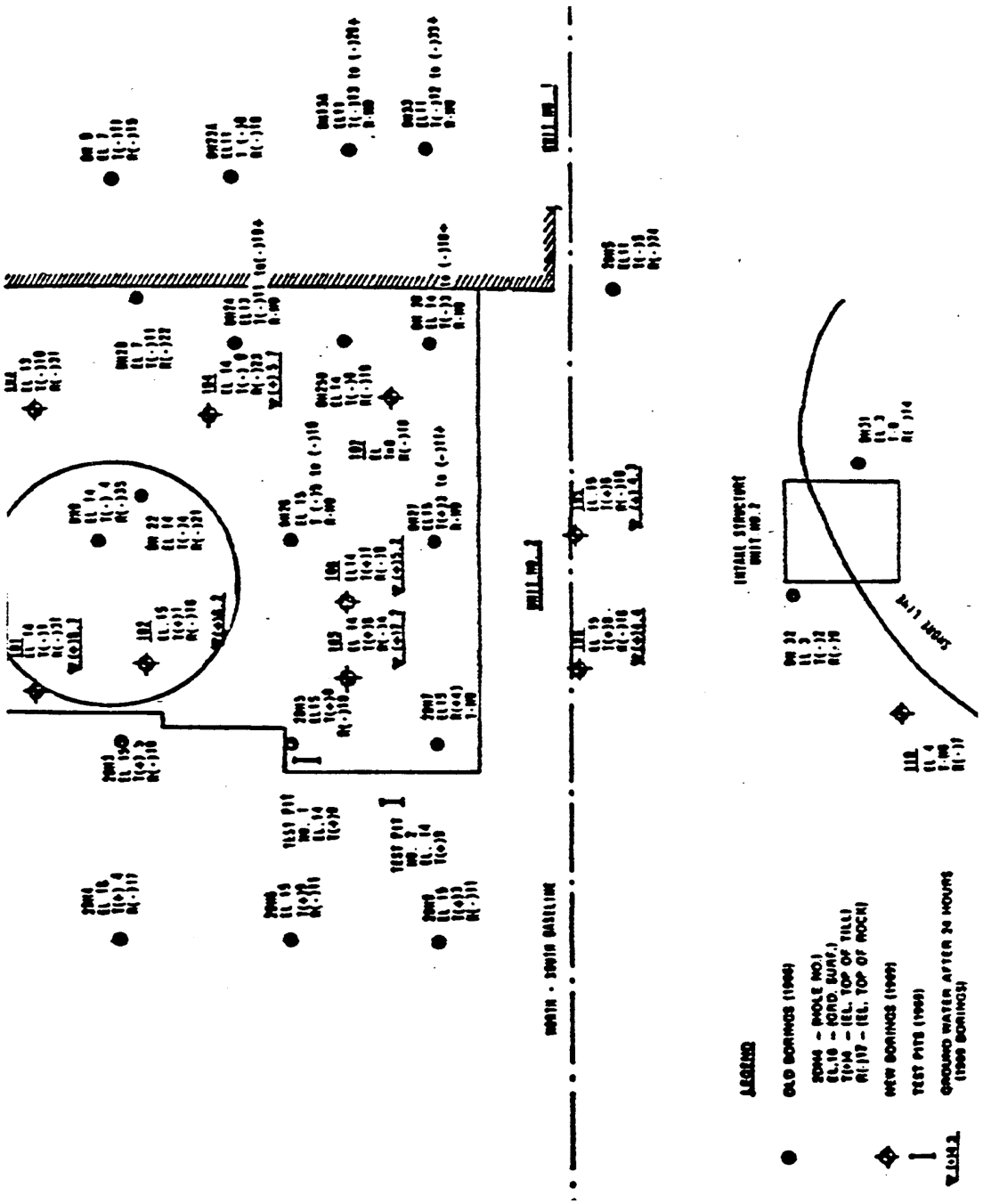
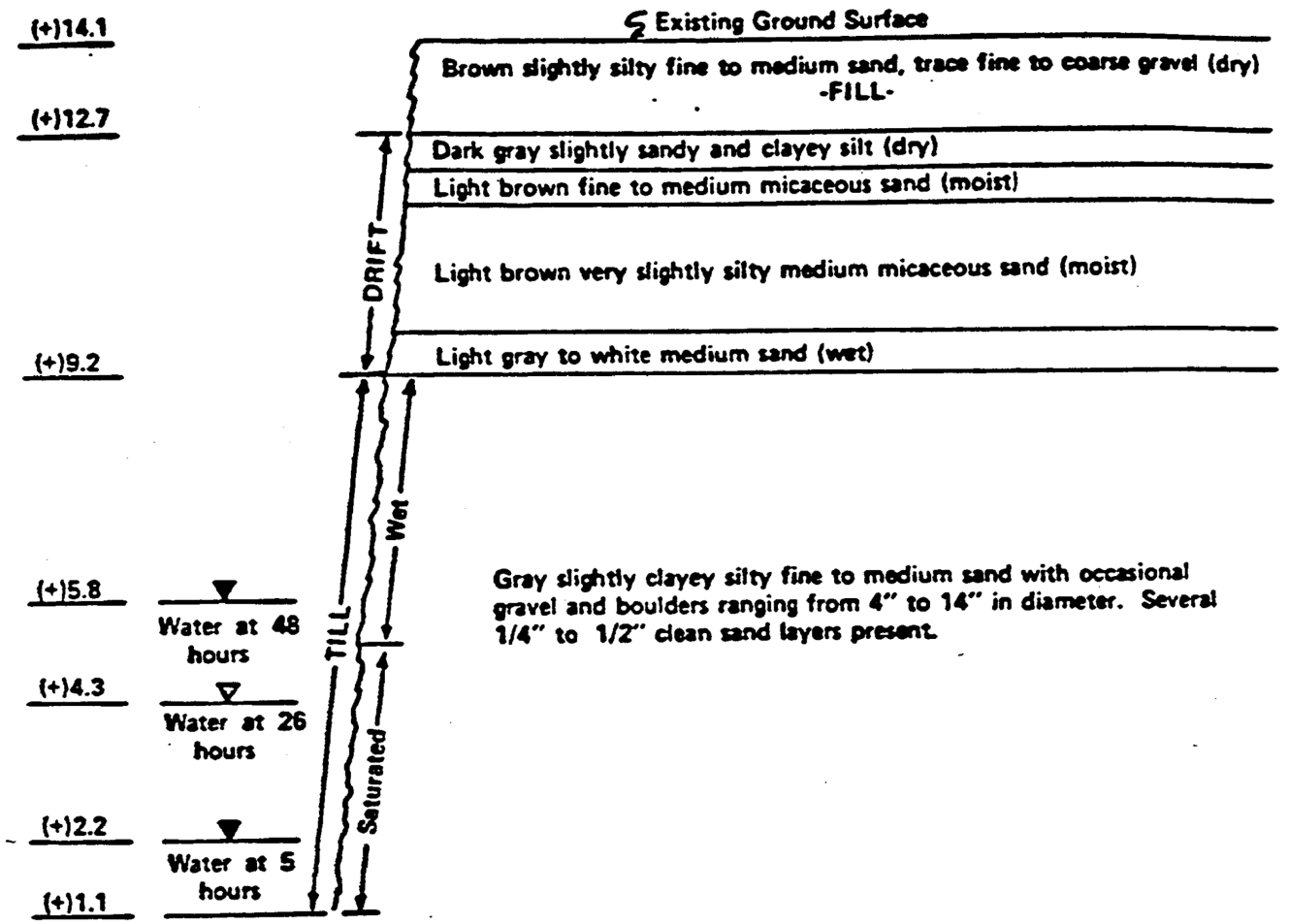


FIGURE 2.5–3 TEST PIT NUMBER 1



1. See Figure 2.5-2 for test pit location.
2. Ground surface hole dimensions 5.0 feet by 15.0 feet. Bottom hole dimensions 2.0 feet by 5.5 feet.

FIGURE 2.5–4 TEST PIT NUMBER 2

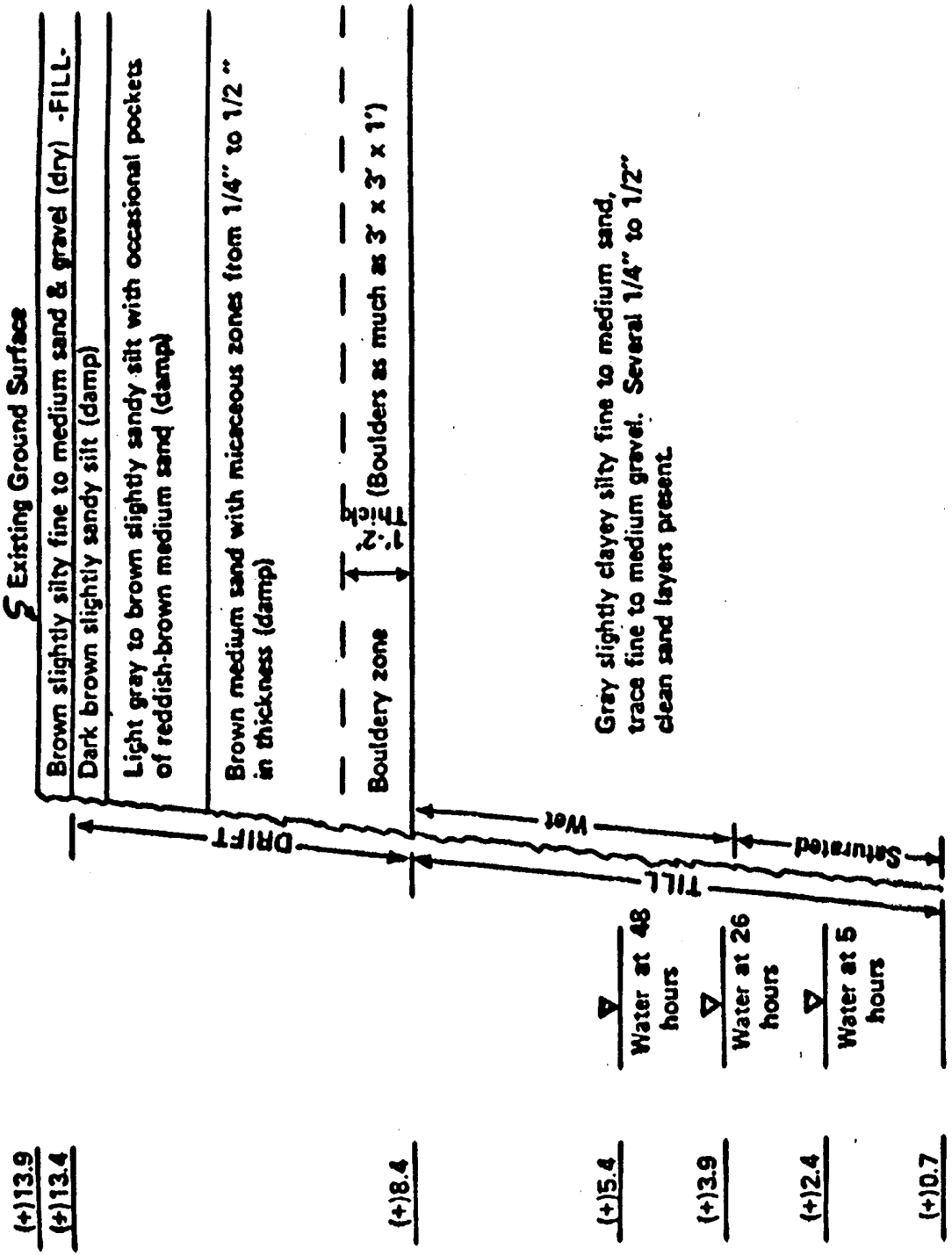
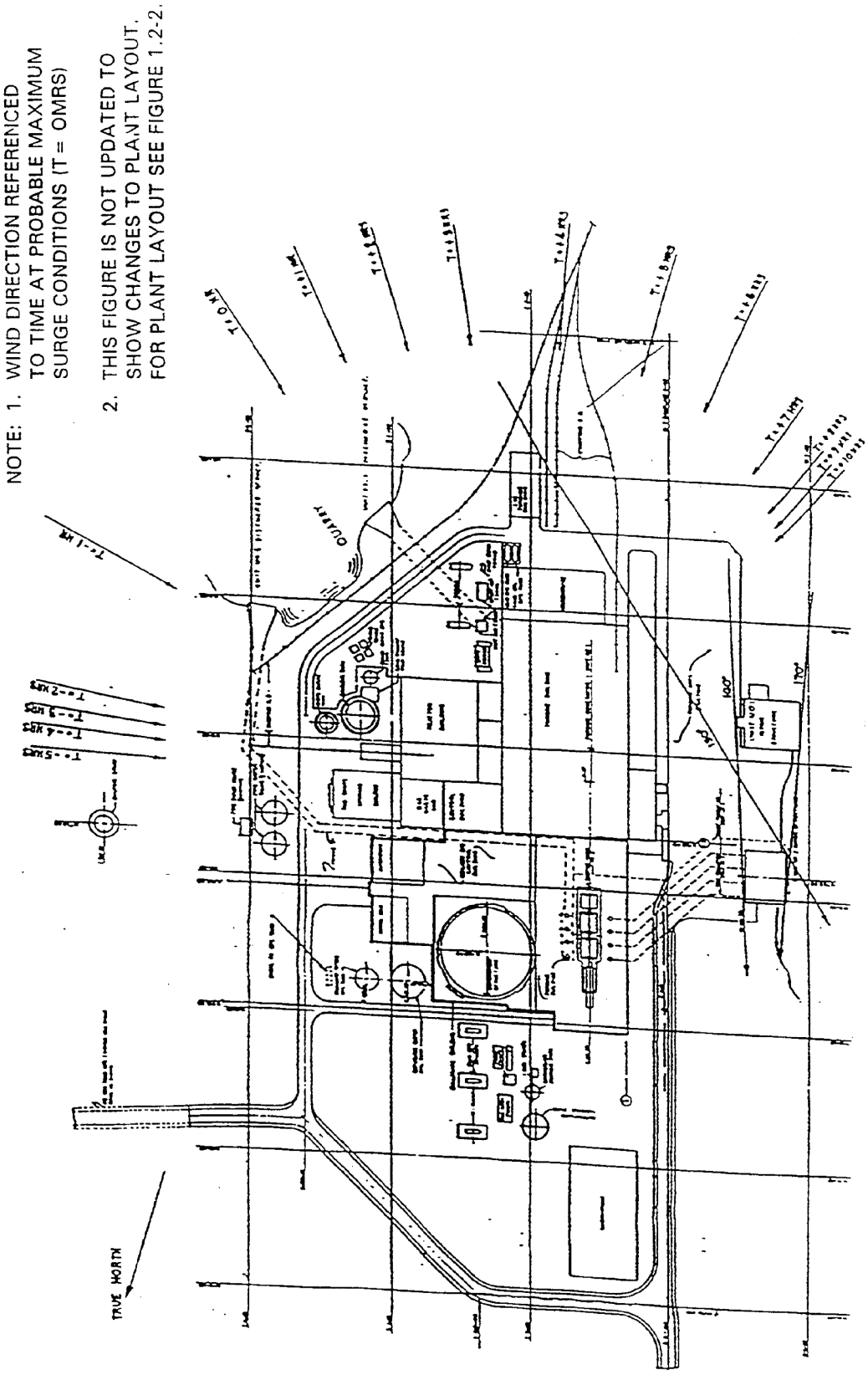


FIGURE 2.5–5 RELATIONSHIP BETWEEN HURRICANE WIND DIRECTION AND PLANT LAYOUT



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.5–6 SECTION THROUGH PLANT

[redacted]

FIGURE 2.5–7 PROFILES OF PLANT

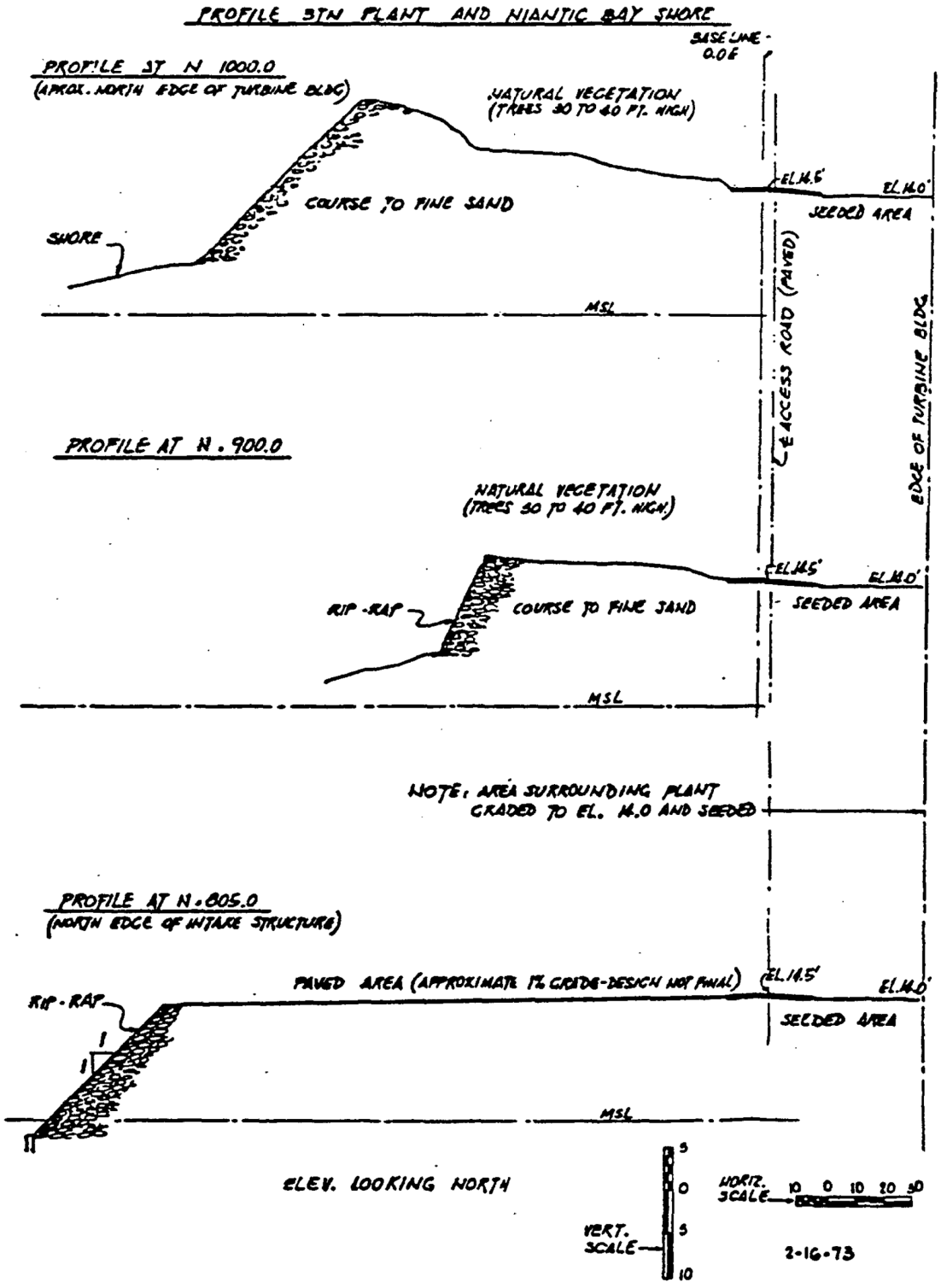
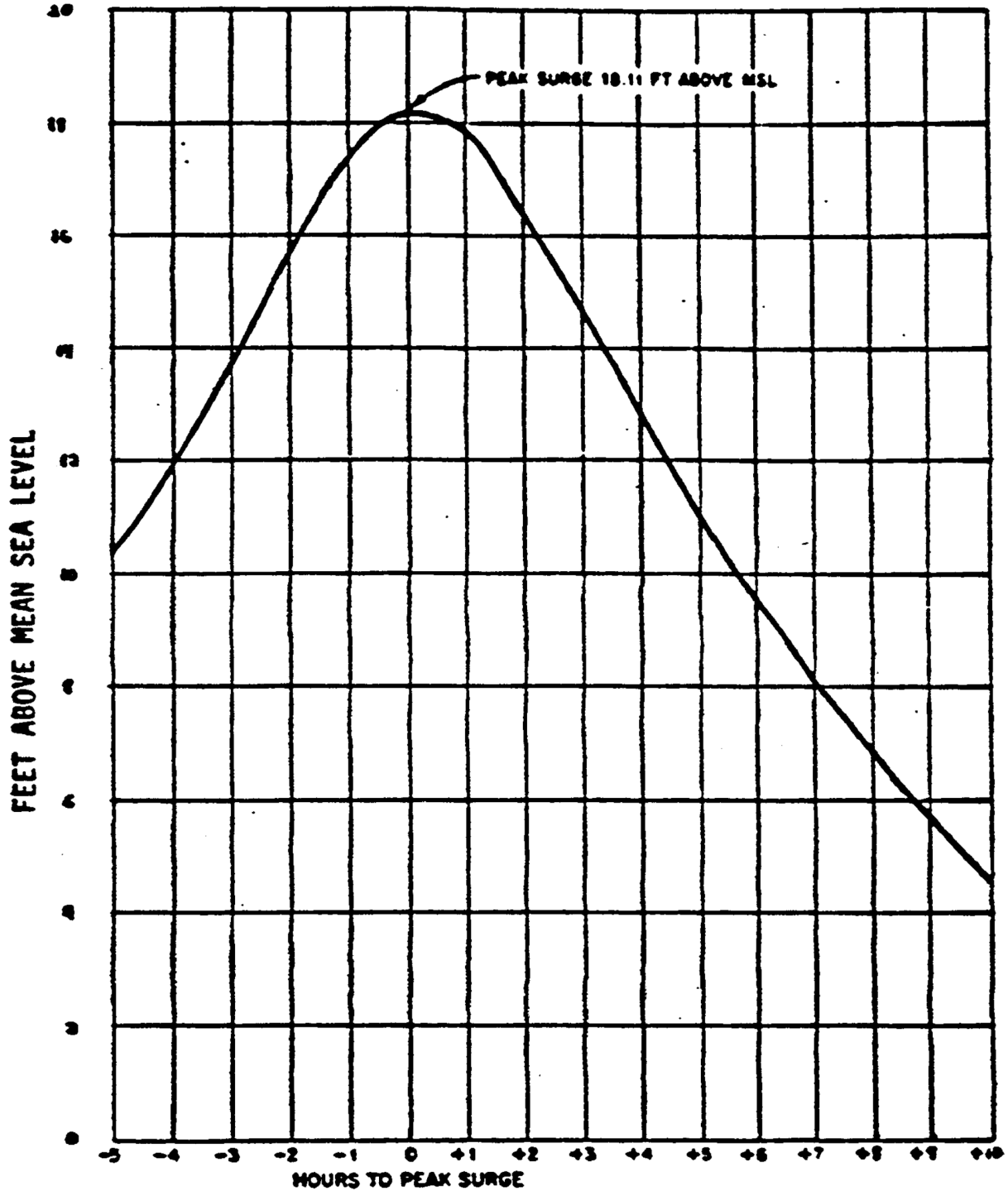


FIGURE 2.5–8 SURGE HYDROGRAPH FOR PMH



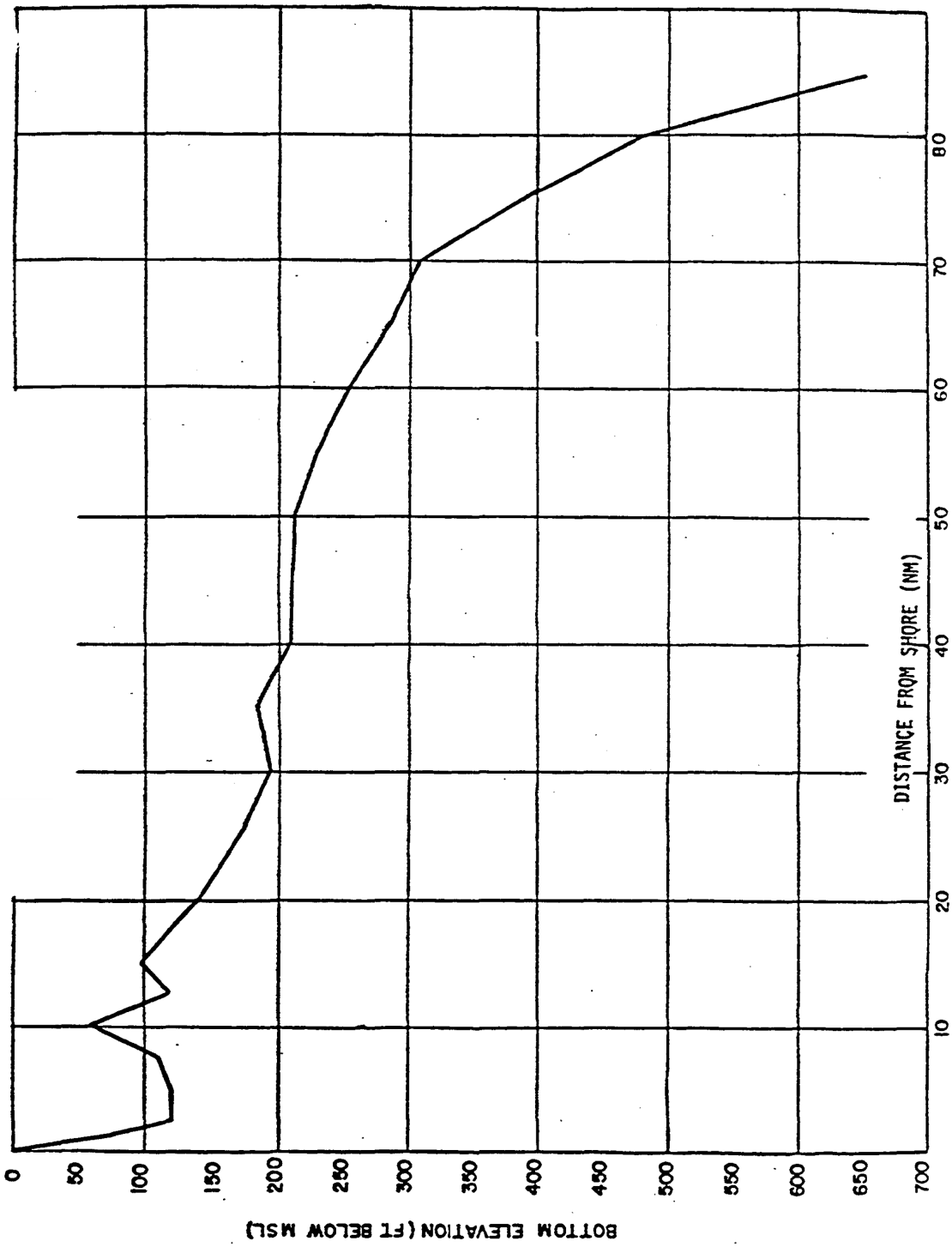


FIGURE 2.5–9 BOTTOM PROFILE FOR SURGE TRAVERSE LINE

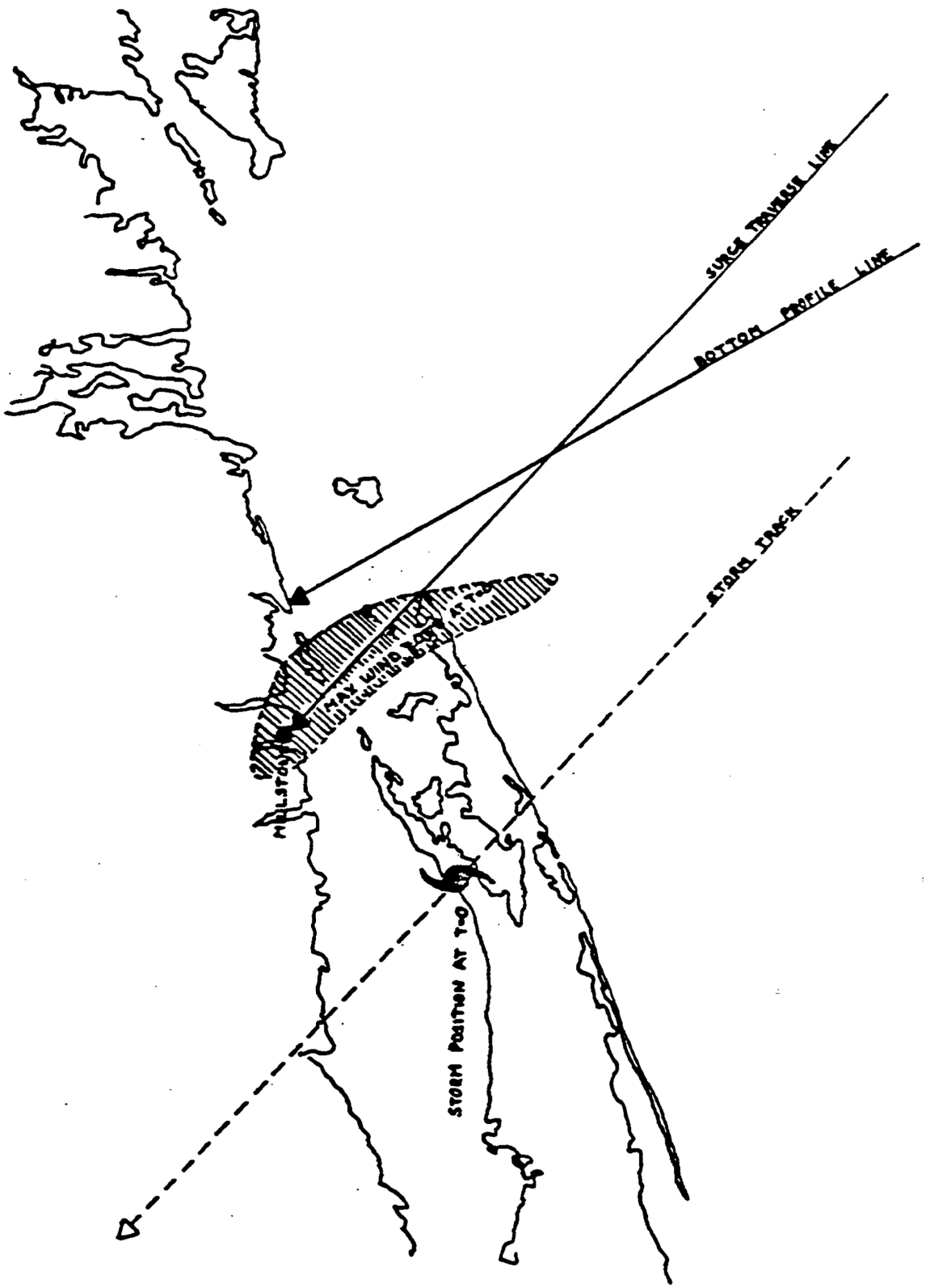


FIGURE 2.5–10 STORM TRACKER FOR PROBABLE MAXIMUM HURRICANE

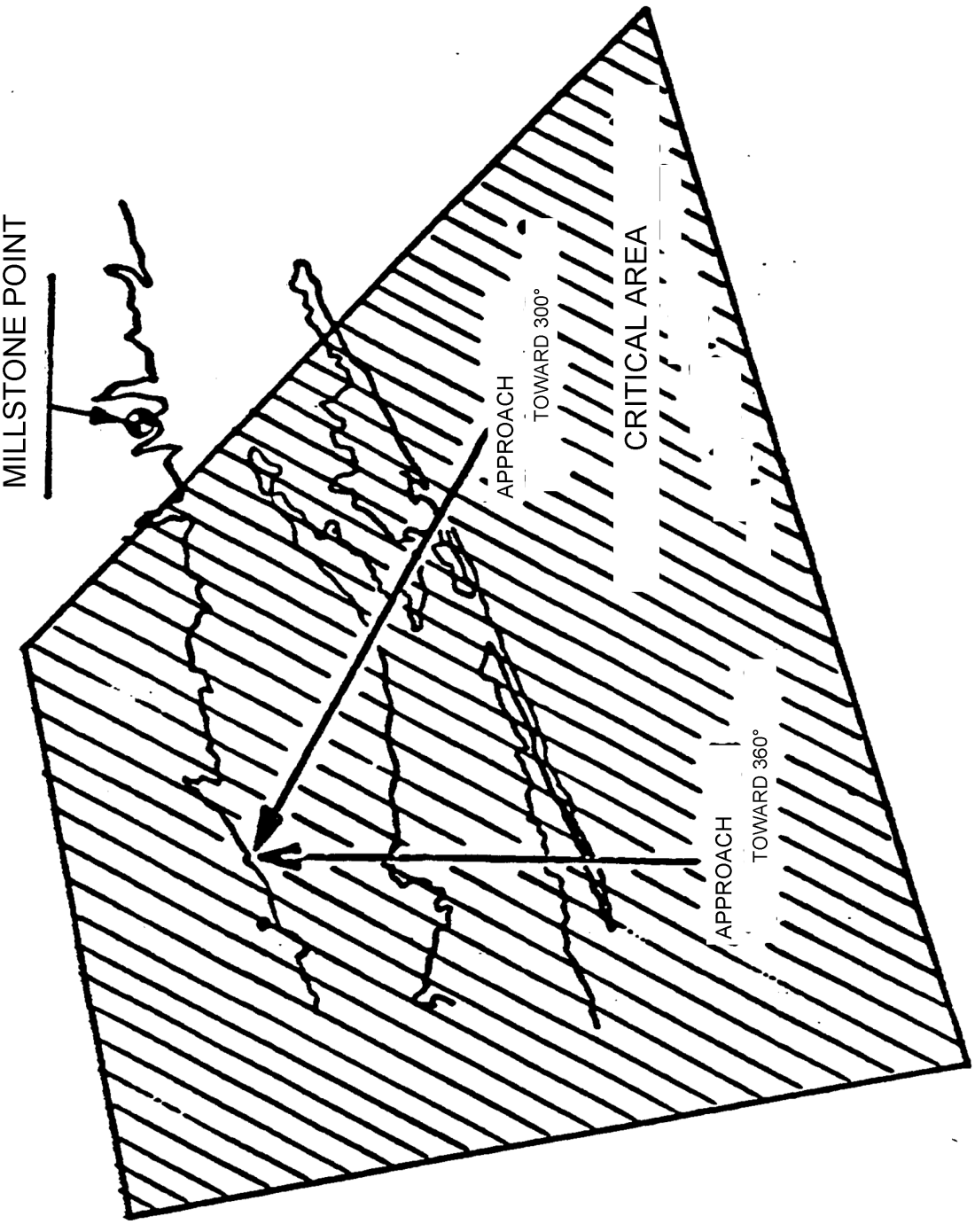


FIGURE 2.5–11 LOCATION OF THE CENTER OF THE STORM WITHIN THE CRITICAL AREA

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.5–12 SHORE PROTECTION

[redacted]

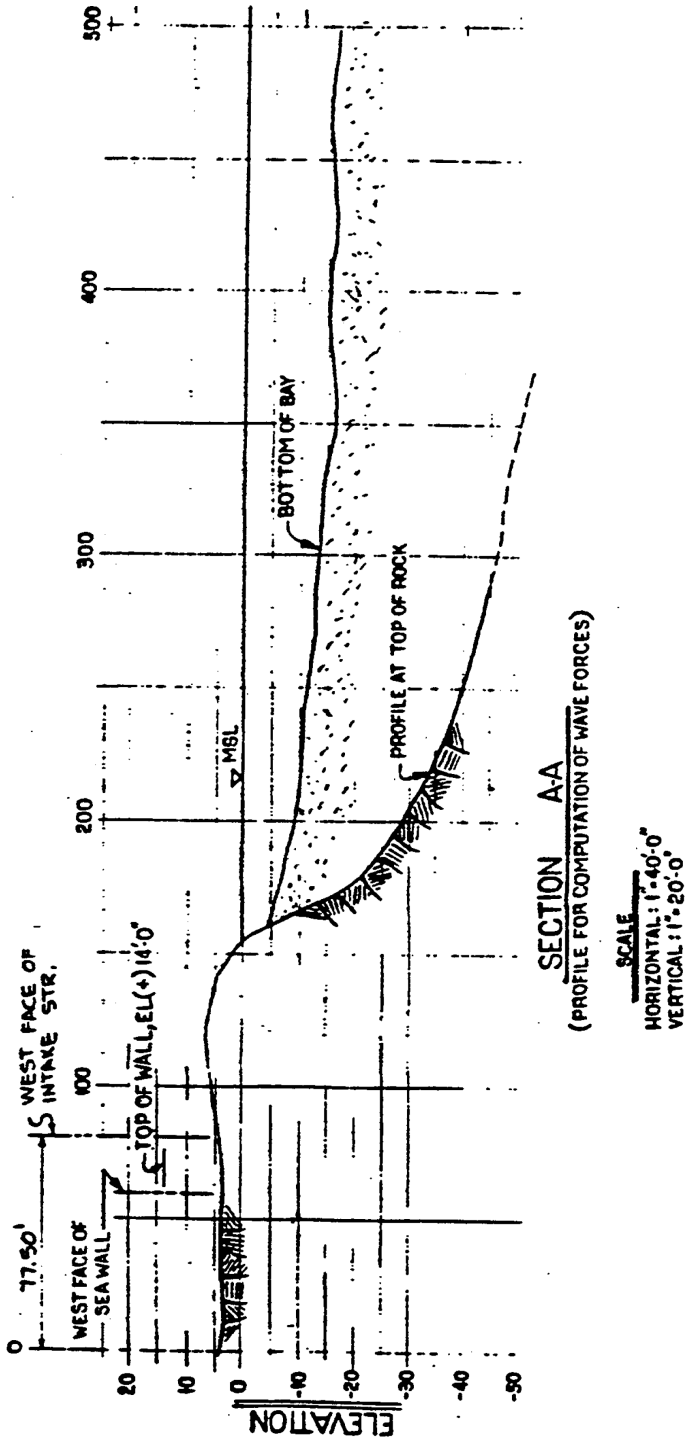


FIGURE 2.5–13 SHORE PROTECTION

FIGURE 2.5–14 WAVE FORCE DIAGRAMS FOR THE THREE ZONES AND THEIR CORRESPONDING BOTTOM PROFILES NEAR THE WALLS

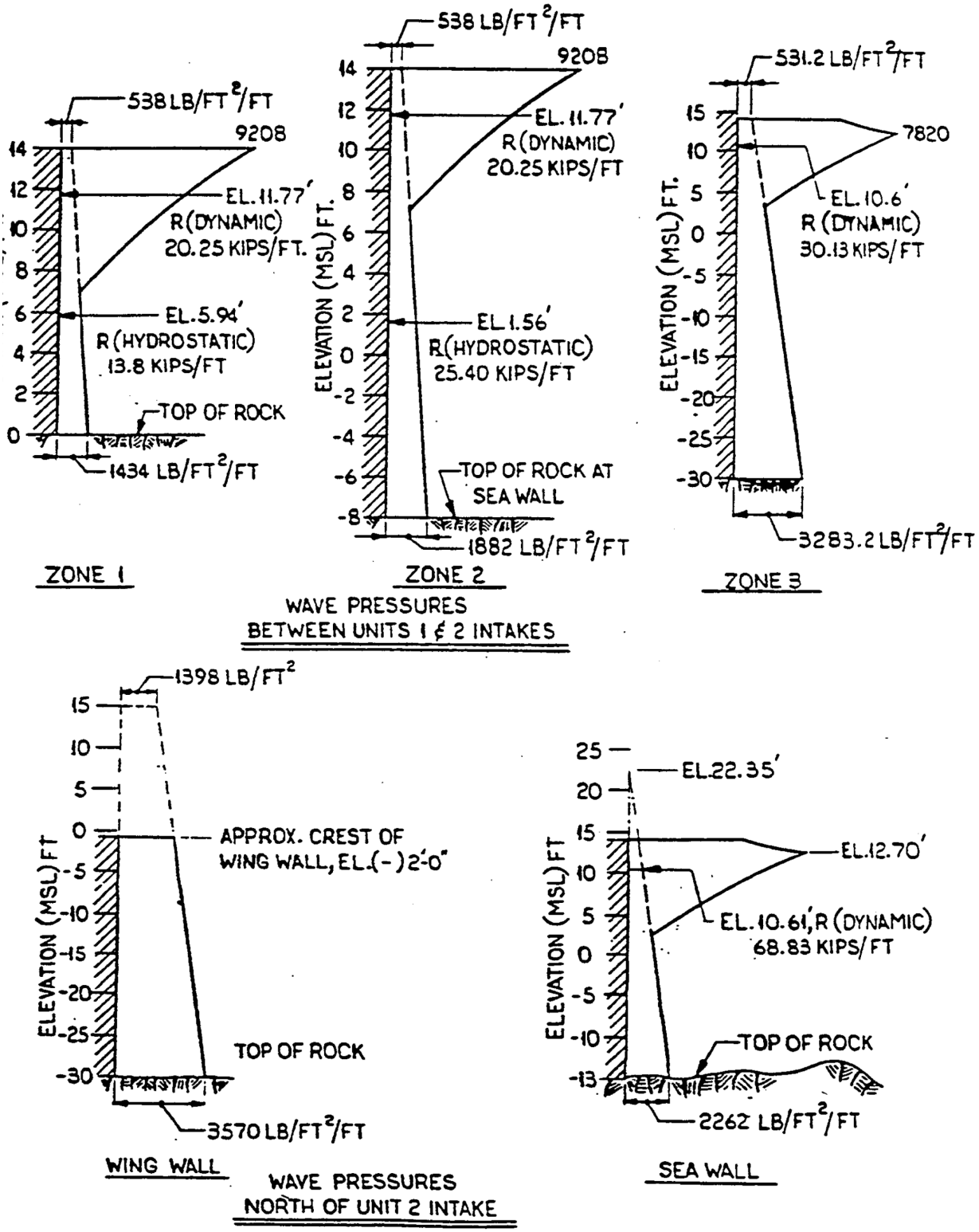
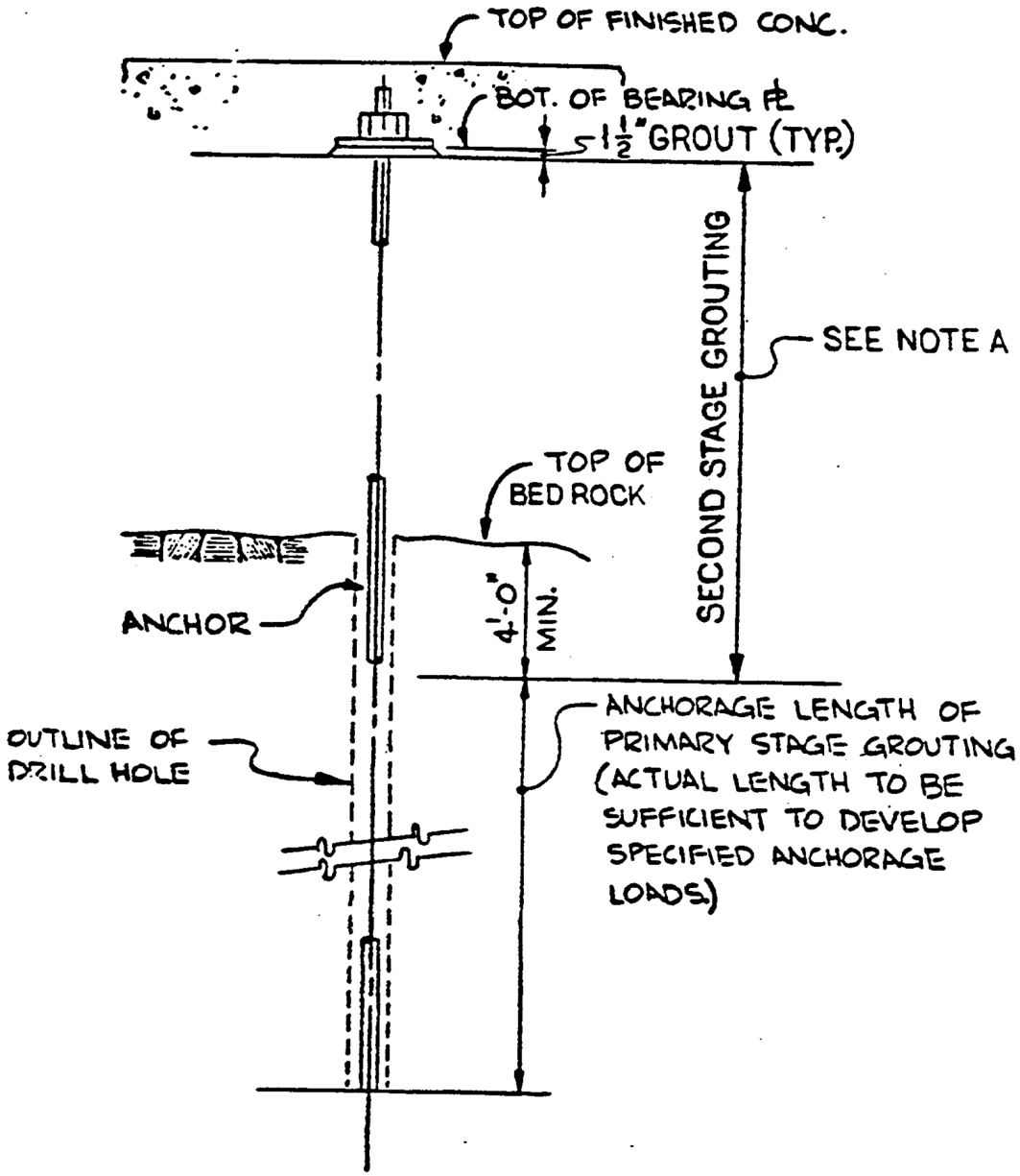


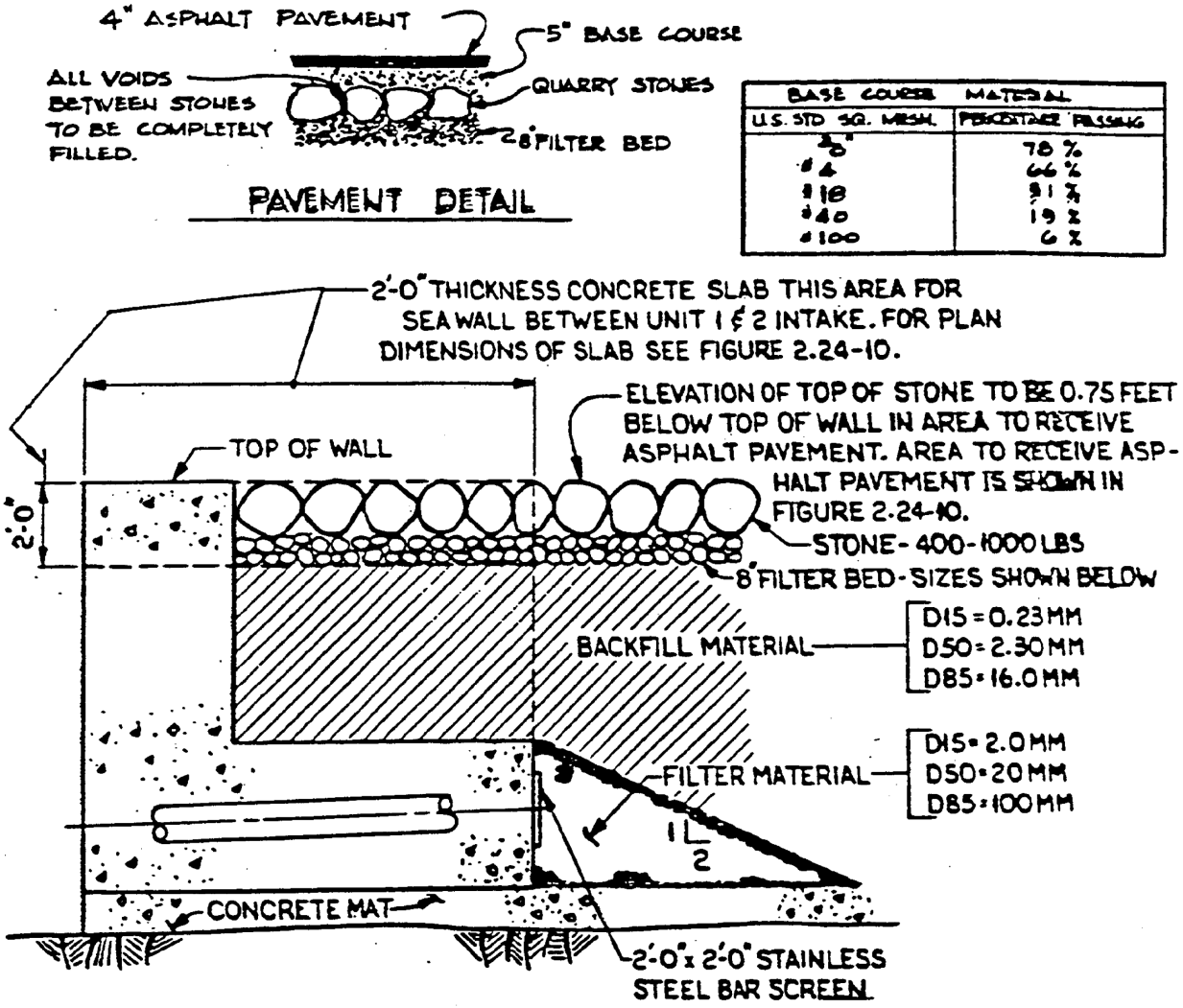
FIGURE 2.5–15 TYPICAL ANCHORAGE DETAIL



NOTE A:

The grouting operation shall be performed in two stages. The primary stage shall be the grout in bedrock required to develop the specified post-tensioning anchorage loads. Secondary grouting is defined as the grouting operation required to fill the remainder of the drilled or cased holes to provide a continuous, non-corrosive environment for theanchorage system.

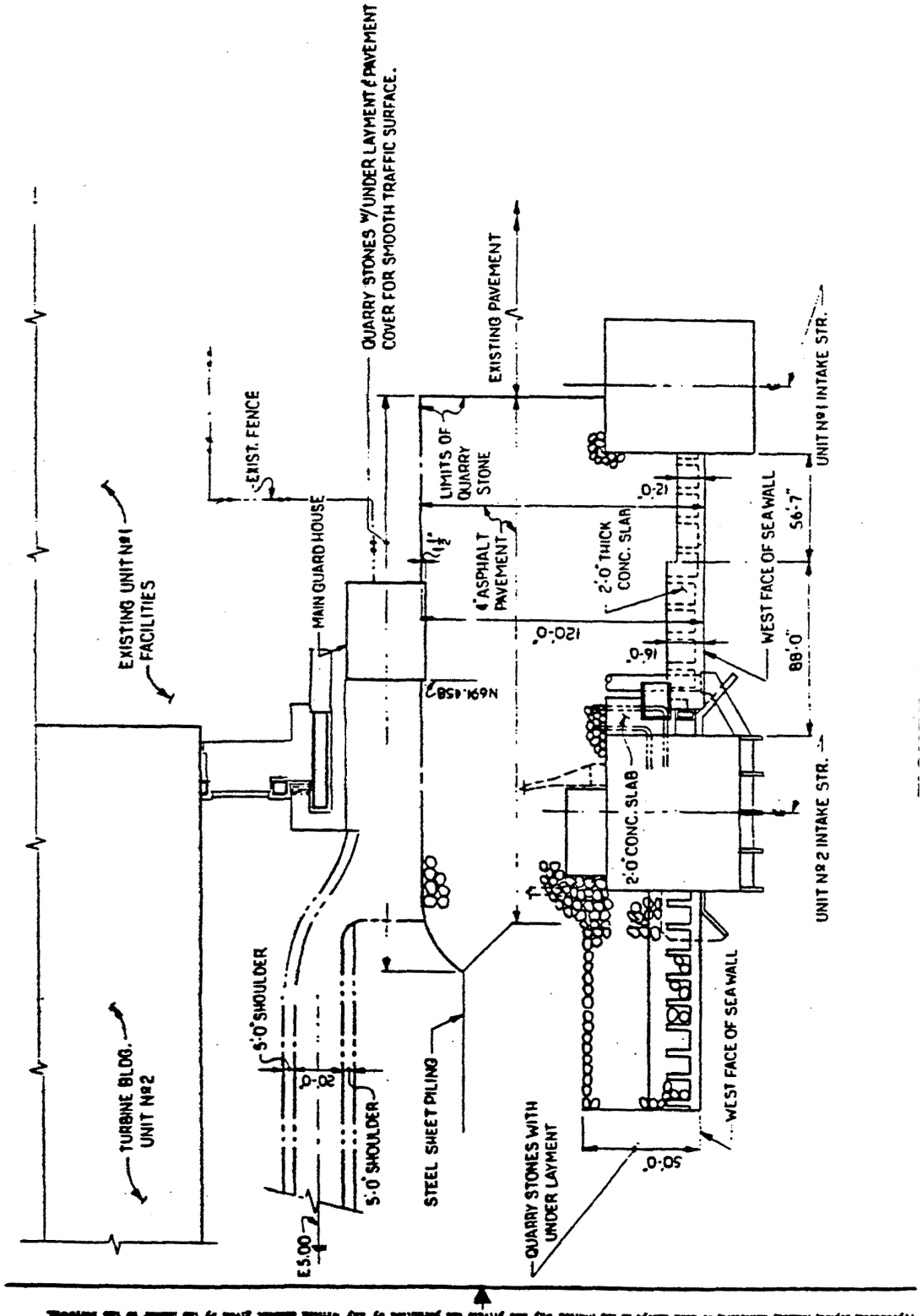
FIGURE 2.5–16 FILTER SYSTEM AND THE GRADATION OF THE FILTER MATERIALS



TYPICAL SECTION THRU SEA WALL SHOWING FILTER SYSTEM AT DRAINS AND QUARRY STONE PROTECTION FOR SCOUR PREVENTION (TYPICAL FOR WALLS NORTH & SOUTH OF INTAKE STRUCTURE NO. 2)

FILTER BED MATERIAL	
U.S. STD SQUARE MESH	PERCENTAGE PASSING
2"	100%
1 1/2"	90-100
1"	20-55
3/4"	0-15
3/8"	0-5

FIGURE 2.5–17 SCOUR PROTECTION



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.5–18 FLOOD PROTECTION

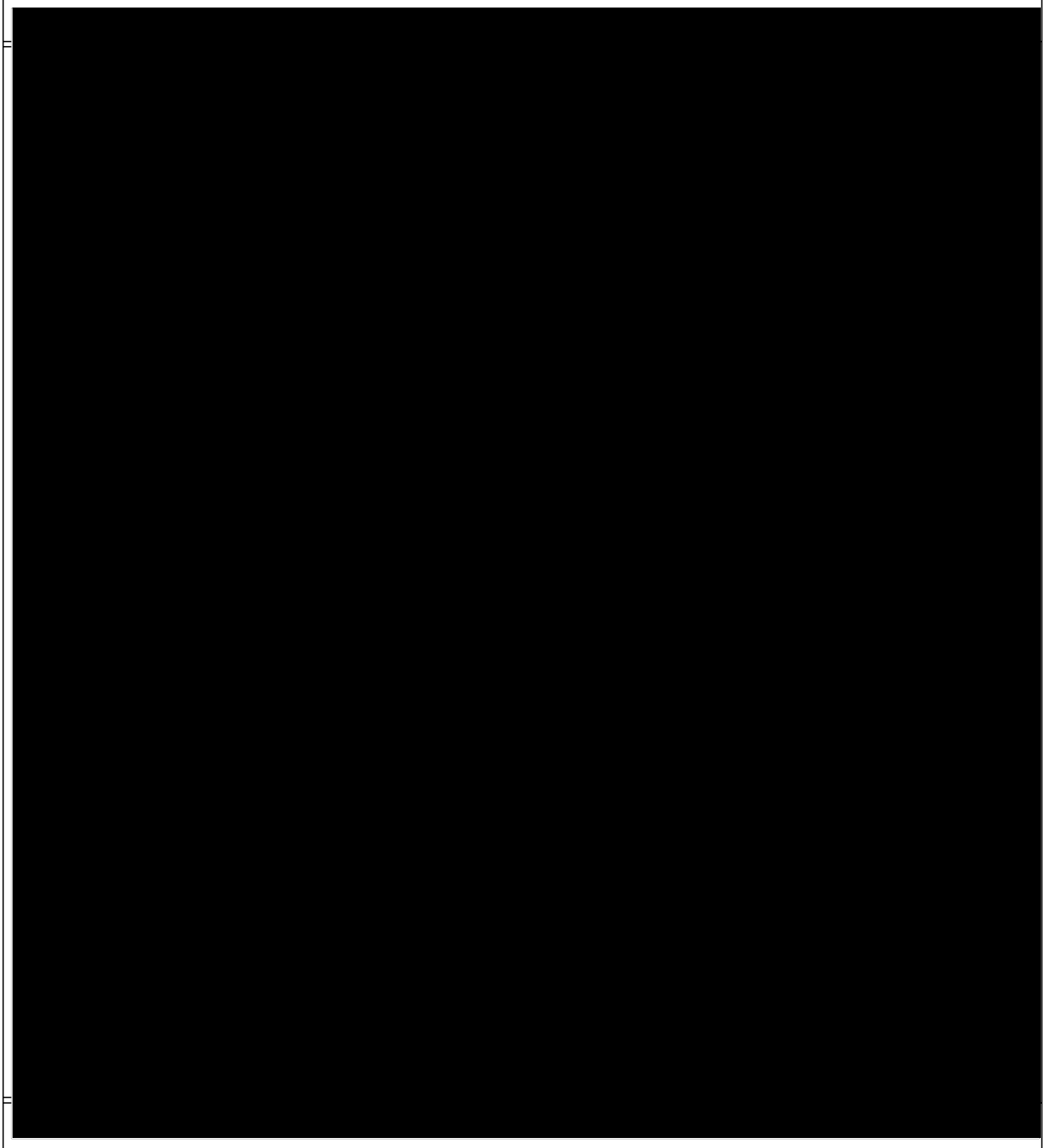


FIGURE 2.5–19 GENERAL ROOF PLAN

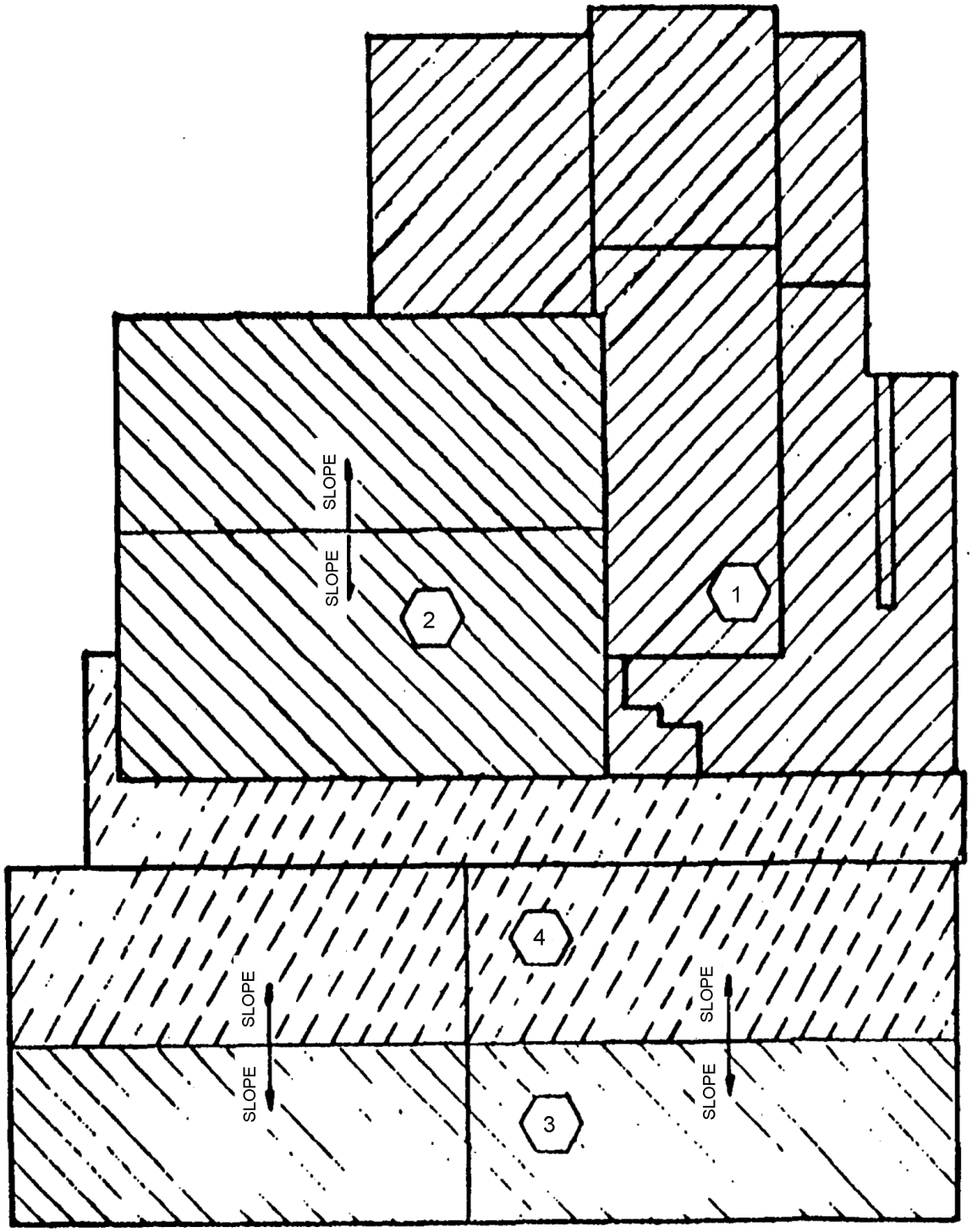
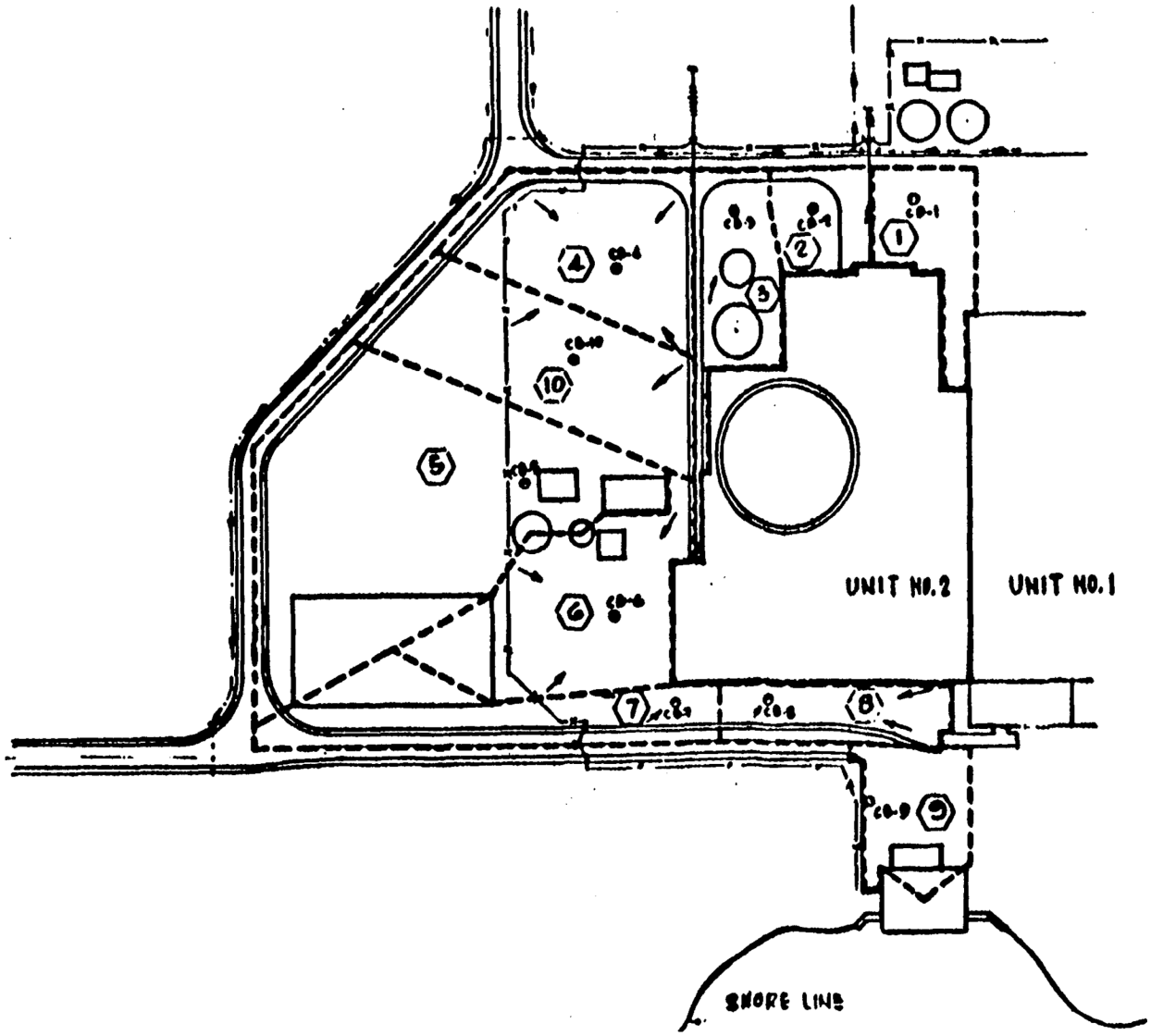


FIGURE 2.5–20 DRAINAGE PLAN



Note: Millstone Unit 3 is not shown. See Figure 2.4-7 of Reference 2.4-1 for current layout.

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 2.5–21 TURBINE AND AUXILIARY BUILDING PLANS AT ELEVATION 14 FEET 6 INCHES



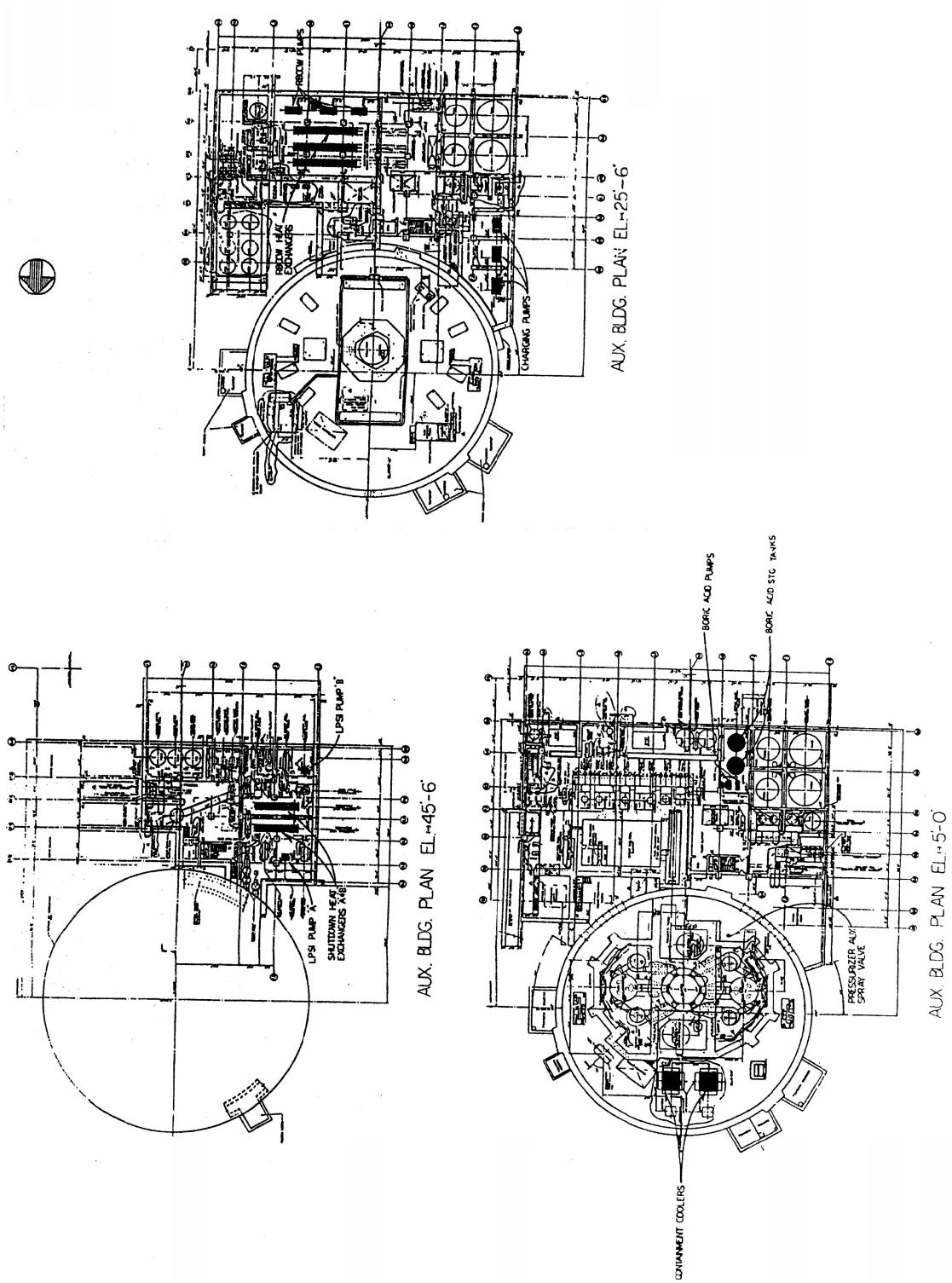


FIGURE 2.5–22 AUXILIARY BUILDING BASEMENT PLANS

FIGURE 2.5–23 INTAKE STRUCTURE

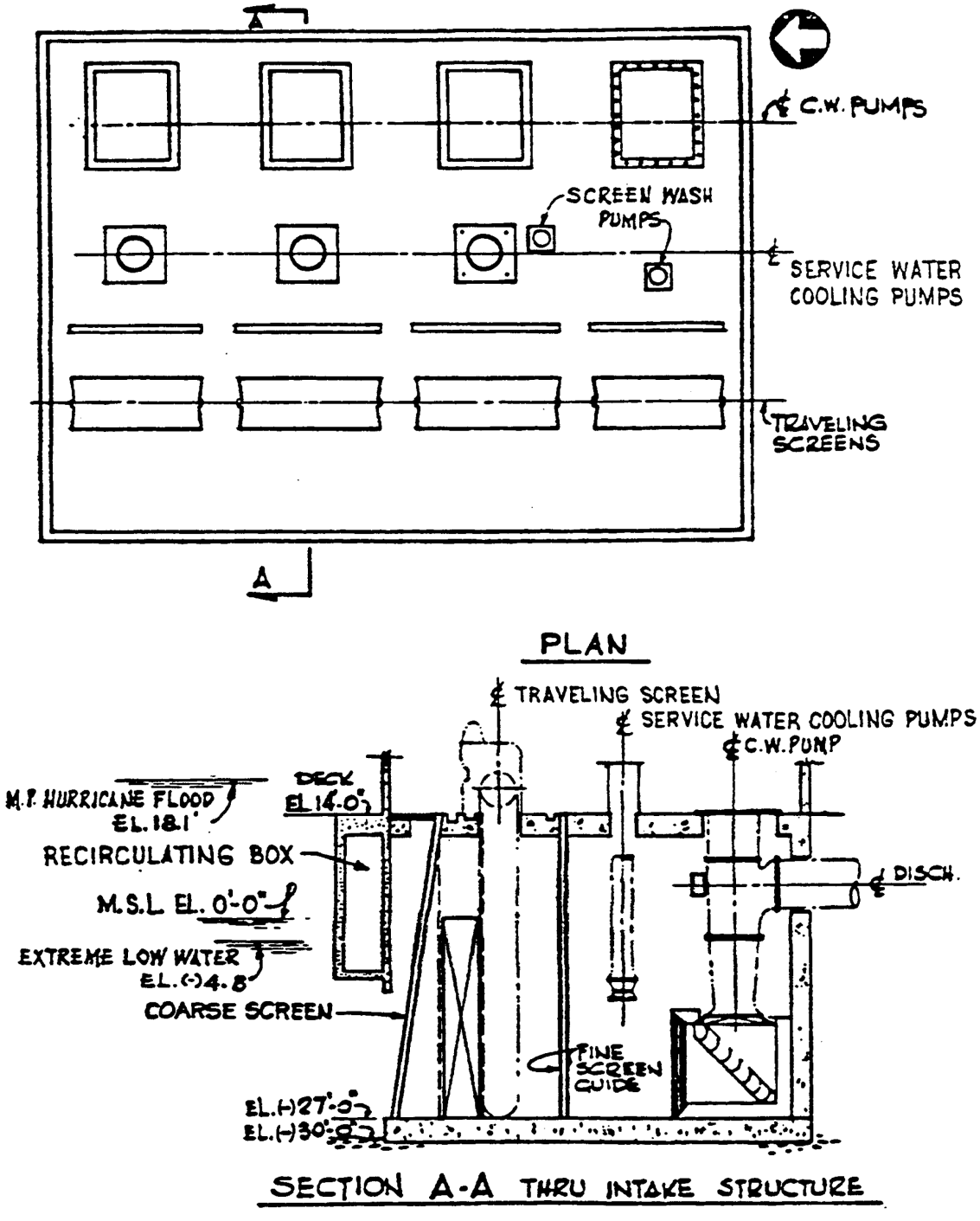
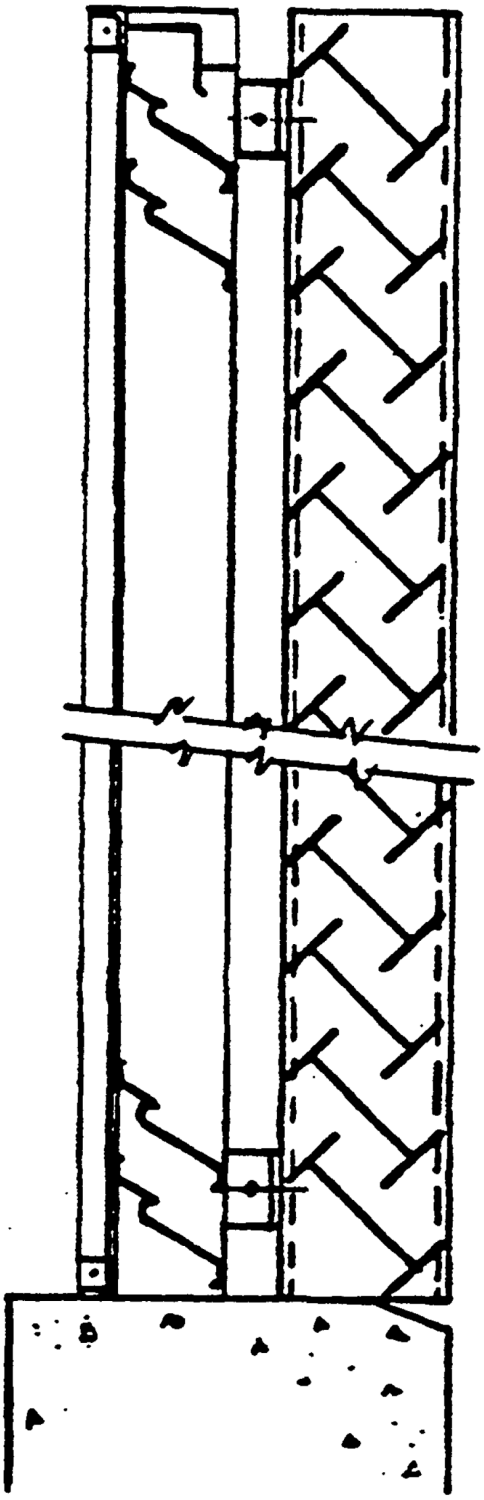


FIGURE 2.5–24 LOUVER PROTECTION - INTAKE STRUCTURE



2.6 SEISMOLOGY

Information regarding the seismology of the Millstone Site is presented in Section 2.5 of the Millstone 3 Final Safety Analysis Report (Reference 2.6-1). That information is incorporated herein by reference.

2.6.1 REFERENCES

2.6-1 Millstone Unit 3, Final Safety Analysis Report, Section 2.5, “Geology, Seismology, and Geotechnical Engineering.”

2.7 SUBSURFACE AND FOUNDATIONS

2.7.1 GENERAL

This summarizes the results, analysis and evaluation of the subsurface and foundation investigations. The field exploration and the laboratory testing conducted for Unit 2 were done under the supervision and direction of Bechtel Corporation. These studies included site and area reconnaissance, field supervision of the boring operations, a review of pertinent literature, and the foundation analysis and evaluation. Graphic boring logs, laboratory test data, and a list of references cited were previously presented in the Millstone Unit 2 PSAR, Docket Number 50-336, subsequent submittal, or are referenced here. Additional information is also presented in Section 2.5.4.3 of the Millstone 3 Final Safety Analysis Report (Reference 2.7-1).

2.7.2 EXPLORATION

In addition to general area and site reconnaissance work, the site was examined through an extensive subsurface drilling program. Previous investigations (in conjunction with Unit 1) included a preliminary subsurface investigation made by Soil Testing, Inc., of Ansonia, Connecticut, in May, 1965, and a geological-geophysical survey conducted by Weston Geophysical Research, Inc., Weston, Massachusetts, in October 1965. These efforts were followed in 1966 by two series of additional borings by the C. L. Guild Drilling and Boring Company, Inc., of Braintree, Massachusetts.

Additional test borings test pits, geophysical refraction surveys, and a plate load test study were made to further define the actual soil, rock, and groundwater conditions (both on and offshore), and to verify the previous interpretation of subsurface conditions within the Unit 2 area.

Gridler Foundation and Exploration Company, Clearwater, Florida, completed a preliminary offshore and onshore subsurface investigation during the period of June through August 1969. Also, Weston Geophysical Engineers, Inc., Weston, Massachusetts, conducted an offshore geophysical survey during the same period.

Offshore borings were made by the American Drilling and Boring Company, East Providence, Rhode Island, during the month of July 1970 to obtain additional information in the vicinity of the Unit 2 intake structure (Reference 2.7-2).

The American Drilling and Boring Company concluded the subsurface exploration work in November 1970, with a series of onsite borings for transmission tower structures (Reference 2.7-3).

A geotechnical representative from Bechtel Corporation was continuously in the field during the Unit 2 drilling operations to provide technical review, inspection, direction, and to advise the drilling contractor of the boring locations and alterations made as the data was received. Frequent field visits were made by supervisory personnel from Bechtel Corporation to review boring progress and modify the boring program.

Soil test borings were generally done using rotary drilling methods and a drilling mud fluid to maintain hydrostatic balance in the boring. Penetration tests were made at regular intervals. The number of hammer blows required to drive the sampler one foot was recorded and is designated the penetration resistance. The boring procedure and penetration resistance were taken in accordance with ASTM Specification D 1586-67. The penetration resistance is a simple in-place shear test, and as such is an index of soil strength, consistency and density.

The soil samples obtained in the field were examined by the Bechtel geotechnical representative. Selected samples were shipped directly from the field to the testing laboratory. The jar samples from the split-spoon penetration tests were returned to the Bechtel Corporation Washington Area Engineering Office and the classifications were verified.

The drilling program was supplemented by test pit excavations at selected locations. The test pits provided an additional opportunity for examination of the principal soil types and observations of ground water behavior.

2.7.3 SITE CONDITIONS

2.7.3.1 Area Geology

(See Sections 2.5.1 and 2.5.4 of Reference 2.7-1.)

2.7.3.2 Soil Conditions

(See Section 2.5.4 of Reference 2.7-1.)

2.7.4 LABORATORY TESTING

The laboratory testing program provided foundation engineering data and physical characteristics data necessary to design foundations resting on soil. The testing was conducted in accordance with currently accepted procedures (References 2.7-4 through 2.7-7).

The testing program generally was confined to the determination of the soil parameters of remolded (fill) soil. The laboratory program included grain size and specific gravity tests to determine particle size and distribution.

Atterberg limit tests to determine soil plasticity characteristics, static triaxial shear tests to aid in the evaluation of fill foundation bearing capacity and elastic settlement properties, compaction tests, and numerous moisture-density, void ratio and relative density determinations.

2.7.5 FOUNDATIONS

2.7.5.1 Structural Data

The following is a summary of the major structural components of the plant, foundation type, foundations elevation, supporting material, and foundation static contact pressure for settlement evaluation which represents conservative estimates of long term static loadings:

Structure and Foundation Type	Supporting Material	Foundation Elevation (feet)	Maximum Static Pressure (psf)
Reactor Building Mat	Unweathered rock	(-)33	7,500
Turbine Pedestal Mat	Unweathered rock	(-)5 to (-)18	3,600
Turbine Buildings Columns	Unweathered rock	(-)2 to (-)22	25,000
Auxiliary Building Mat (west of column line M-7)	Unweathered rock	(-)50	20,000
Warehouse Area of Auxiliary Building Mat (east of column line M-7)	Controlled select compacted fill	2 foot 6 inches to 8 foot 6 inches	2,400
Enclosure Building Caissons	Unweathered rock	(-)18 to (-)26	130,000
Intake Structure Mat	Unweathered rock	(-)30	15,660
Refueling Water Storage Tank	Controlled select compacted fill	11	3,000
Condensate Storage Tank	Glacial till	11	3,000
Condensate Surge Tank	Glacial till	11	3,000

2.7.5.2 Foundation Evaluation

The soils and bedrock at this site were suited for construction of the plant without adverse settlements. The bedrock provides excellent support for structures, under both static and dynamic conditions. Applied foundation bedrock contact pressures are all well below the allowable bearing pressure of 200,000 psf.

The in situ glacial till material is uniformly very dense in consistency, with an allowable bearing capacity in excess of 10,000 psf. The applied contact pressures, which are approximately one-quarter of the allowable bearing pressure, resulted in negligible settlement.

The compacted backfill material was a select, processed free draining, offsite borrow material. The soil is a well-graded fine to coarse sand with some gravel. A typical grain size distribution would be:

Sieve	Accumulative (% passing)
2 inch	100.0
1.5 inch	99.0
1 inch	91.2
0.75 inch	84.3
0.5 inch	77.1
3/8 inch	71.9
Number 4	60.1
Number 10	51.7
Number 40	29.4
Number 100	11.8
Number 200	6.6

Optimum Standard Proctor dry density (ASTM D-698) for the material was approximately 132 pounds per cubic foot. Optimum Modified Proctor dry density (ASTM D-1557) was approximately 137 pounds per cubic foot.

The following allowable bearing pressures were assigned to structural fill areas utilizing the select backfill:

Fill Compaction (Minimum)	Type of Backfill and Allowable Bearing Capacity
90% Standard Proctor (ASTM D-698; AASHO T-99)	General, nonstructure supporting, fill areas and plant parking lot
95% Standard Proctor	Structural backfill areas supporting facilities with contact pressures of 3000 psf or less
95% Modified Proctor (ASTM D-1557; AASHO T-180)	Structural backfill areas supporting facilities with contact pressure of 4000 psf or less

These criteria covered the majority of the backfill conditions. Unique conditions of footing size and load were evaluated on an individual basis. Plate load tests and triaxial shear data indicate that the allowable select backfill bearing pressures are conservative. Excavation and backfill requirements are shown in Figures 2.7–1 and 2.7–2 respectively.

Due to the free draining characteristic of the backfill material, settlement was elastic rather than time depended or hydrodynamic settlement, and essentially complete following construction. Settlement of foundations on select backfill was small.

The major structures founded on select backfill material are the warehouse portion of the auxiliary building. Settlement analyses, based on plate load and triaxial test data, showed ultimate settlement would be less than one-half inch. Settlement joints between the warehouse portion and the remainder of the auxiliary building were provided to accommodate this settlement.

2.7.6 LIQUEFACTION

If a loose saturated sandy soil is subjected to ground vibrations, as during an earthquake, it tends to compact and decrease in volume. If the soil cannot drain during the rapid load fluctuations imposed by an earthquake, there is a buildup in pore pressure until it is equal to the overburden pressure. The effective stress then becomes zero, the soil looses its strength, and develops a "quick" or liquefied condition. If this condition is of general extent and the pressure is not otherwise relieved, it can cause a flow or bearing capacity failure.

The phenomenon of liquefaction is generally not applicable to this site as critical structures are founded on bedrock.

For evaluation of the liquefaction potential of those soils supporting foundation loads, the glacial till and select granular backfill, data from standard penetration tests in the glacial till and in-place density tests in the fill were examined.

Standard penetration tests (ASTM D-1586-67) in the glacial till ranged from a minimum of 58 blows per foot with an average value in excess of 100 blows per foot. All compacted fill supporting foundation slabs were compacted to at least 95 percent Standard Proctor which is approximately equal to 80 percent relative density (Reference 2.7-8). Therefore, the soils at the site are not susceptible to liquefaction during the design basis earthquake. Additional information is also presented in Section 2.5.4.8 of Reference 2.7-1.

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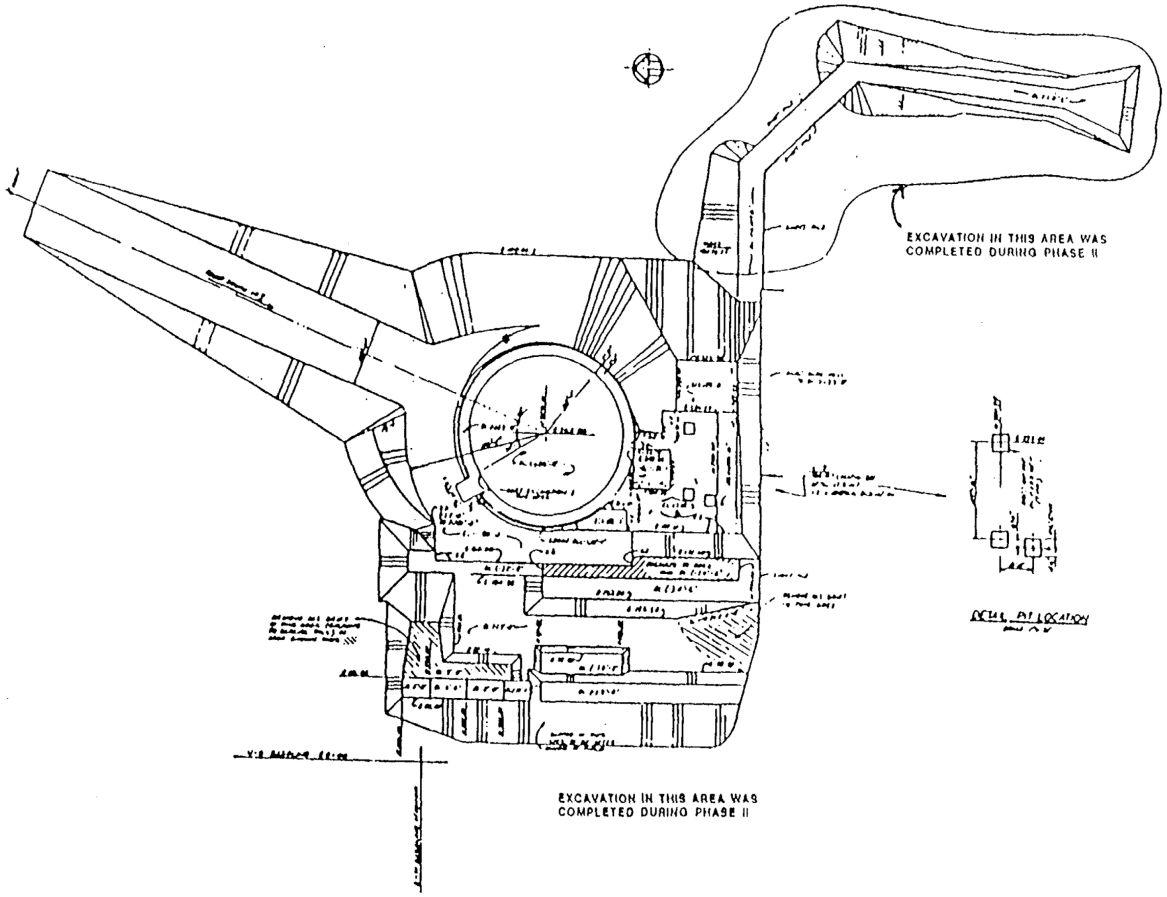
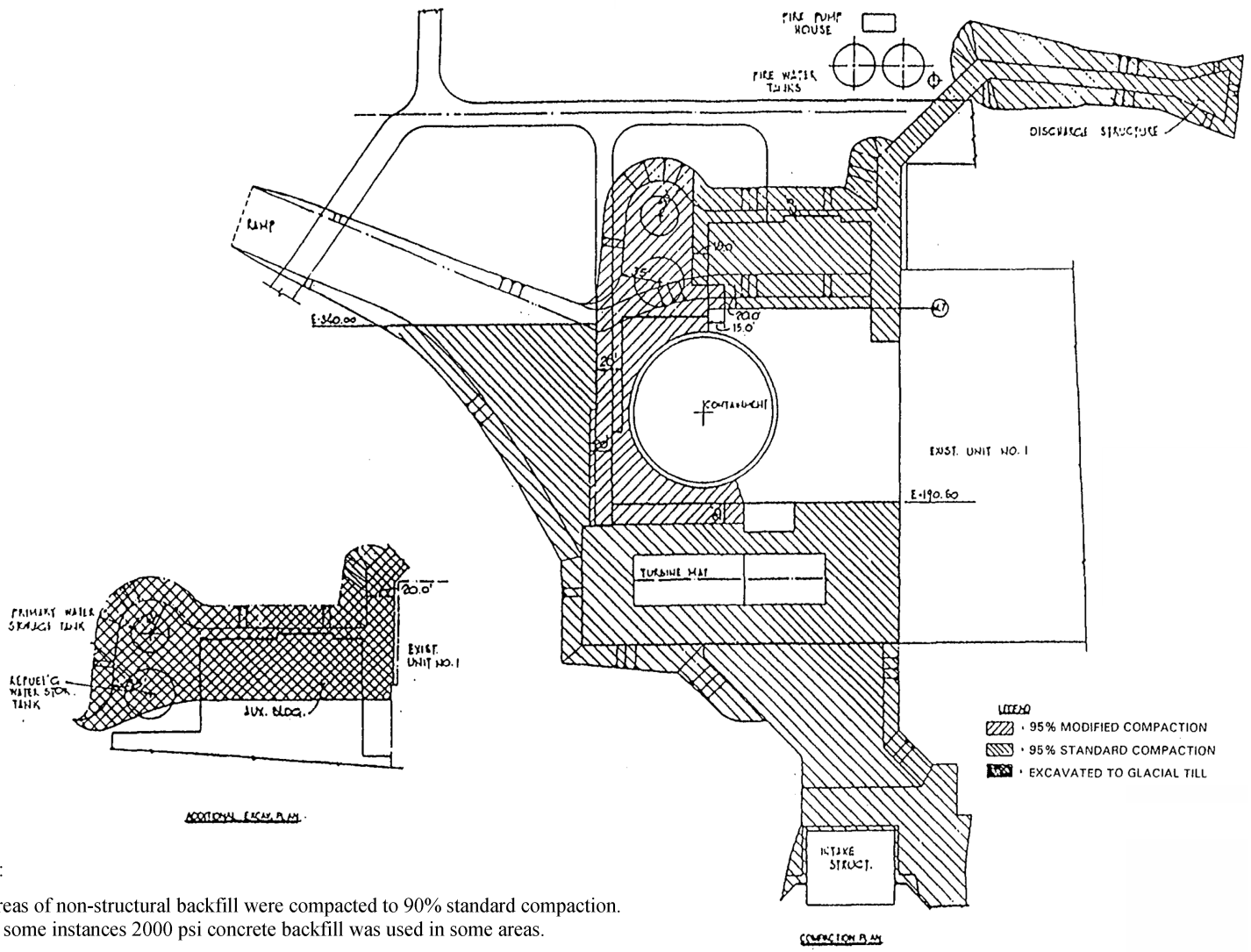
FIGURE 2.7–1 EXCAVATION PLAN



NOTE:

1. Elevations shown are based on MSL datum elevation 0.00 feet.
2. Unless otherwise noted, slopes are 1-1/2:1 in earth and 1:3 in rock.
3. Locations and elevations of excavation limits in containment and lower auxiliary building allow 6 inches between outside face of concrete walls or bottom of slabs and rock to allow for placing concrete forms or mud mat for waterproofing membrane. The coordinates location limits of excavation in other areas are approximate and may have required some local excavation when construction began.
4. Construction slopes are protected against surface drainage and excessive erosion.

FIGURE 2.7–2 BACKFILL AND COMPACTION REQUIREMENTS



Notes:

1. Areas of non-structural backfill were compacted to 90% standard compaction.
2. In some instances 2000 psi concrete backfill was used in some areas.

2.8 ENVIRONMENTAL MONITORING PROGRAM

A site oriented multifaceted environmental monitoring program was begun at the Millstone site in 1968 several years prior to the startup of Unit 1. This period provided the base line data necessary to identify and evaluate any changes in the indicator organisms which might occur due to the operation of the nuclear plant or from any other cause. The program now in effect will continue to experience modifications as additional information is gained.

The development of the existing program, the base line data it has obtained and the results are fully described in annual reports entitled “Monitoring the Marine Environment of Long Island Sound at Millstone Nuclear Power Station, Waterford, Connecticut,” submitted to the Connecticut Department of Environmental Protection (CT-DEP) in fulfillment of the National Pollutants Discharge Elimination System (NPDES) requirements.

2.9 ENVIRONMENTAL RADIATION MONITORING PROGRAM

2.9.1 GENERAL

A study of environmental radiation levels was started during April 1967. This program was started before any of the units were or became operational on the site. It provided a baseline from which any changes in radiation, due to station operation or other causes, can be detected and evaluated.

2.9.2 SURVEY PROGRAM

The preoperational radiological monitoring program for Millstone Unit Number 2 was the same as that for Millstone Unit Number 1. This program which began in April 1967 ended when Unit Number 1 became operational in July 1970. The program and the results obtained are described in detail in Section 5.2.1 of Millstone Unit Number 2 Environmental Report Operating License Stage (EROLS).

The Operational Radiological Environmental Monitoring Program is described in detail in the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) and in each Annual Radiological Environmental Operating Report as submitted to the Nuclear Regulatory Commission. This program is designed to detect potential radiological consequences of station operation.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 3: Reactor

CHAPTER 3—REACTOR

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CHAPTER 3 – REACTOR

3.1 SUMMARY DESCRIPTION

The reactor is of the pressurized water type using two reactor coolant loops. A vertical cross section of the reactor is shown in Figure 3.1–1. The reactor core is composed of 217 fuel assemblies, 73 control element assemblies (CEA) and up to four neutron source assemblies. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 136 inches and an active length of 136.7 inches. The fuel assemblies are comprised of a structure and fuel and poison rods. The structure, which provides for 176 rod positions, consists of five guide tubes attached to spacer grids and is enclosed at the top and bottom by end fittings. Each of the guide tubes replaces four fuel rod positions and provides a channel which guides the control element over its entire length of travel. In selected fuel assemblies the central guide tube houses in-core instrumentation. The reactor is currently fueled by assemblies produced by AREVA.

The fuel is low enrichment UO_2 in the form of ceramic pellets and encapsulated in zirconium alloy tubes. These tubes are seal welded as hermetic enclosures.

Figure 3.1–2 shows a view of the reactor core cross section and some dimensional relations between fuel assemblies, fuel rods and CEA guide tubes.

The reactor internals support and orient the fuel assemblies and CEAs, absorb the static and dynamic loads and transmit the loads to the reactor vessel flange, provide a passage way for the reactor coolant, and guide in-core instrumentation.

The internals will safely perform their function during normal operating, upset and emergency conditions. The internals are designed to safely withstand the forces due to dead weight, pressure differential, flow impingement, temperature differential, vibrations and seismic acceleration. All reactor components are considered category 1 for seismic design. The reactor internals design limits deflection where required by function. Where necessary, components have been subjected to fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation. The effects of shock loadings on the internals is included in the design analysis.

Reactivity control is provided by two independent systems: The control element drive system (CEDS) and the chemical and volume control system (CVCS). The CEDS controls short term reactivity changes and is used for rapid shutdown. The CVCS is used to compensate for long term reactivity changes and can make the reactor subcritical without the benefit of the CEDS. The design of the core and the reactor protective system (RPS) prevents fuel damage limits from being exceeded for any single malfunction in either of the reactivity control systems.

The CEAs consist of five poison rods (control elements) assembled in a square array, with one rod in the center. The rods are connected to a spider casting which is coupled to the control element drive mechanism (CEDM) shaft. There are a total of 73 CEAs. Some CEAs are mechanically connected in pairs and are known as dual CEAs.

Both dual and single CEAs are maneuvered by magnetic jack type CEDM's mounted on the reactor vessel head.

The maximum reactivity worth of the CEAs and the associated reactivity addition rate are limited by core, CEA and CEDS design to prevent sudden large reactivity increases. The design restraints are such that reactivity increases will not result in violation of the fuel damage limits, rupture of the reactor coolant pressure boundary (RCPB), or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

The fuel management scheme employed replaces approximately 40 percent of the core each refueling. Sufficient margin is provided to ensure that peak burnups of the individual fuel assemblies are within acceptable limits.

The nuclear design of the core will ensure that the combined response of all reactivity coefficients to an increase in reactor thermal power yields a net decrease in reactivity and that CEAs are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce power distributions that are within the acceptable limits on overall nuclear heat flux factor (F^N_Q) and departure from nucleate boiling ratio (DNBR). The RPS and administrative controls ensure that these limits are not exceeded.

The reactor coolant enters the upper section of the reactor vessel through four inlet nozzles, flows downward between the reactor vessel shell and the core barrel, and passes through the flow skirt and into the lower plenum where the flow distribution is equalized. The coolant then flows upward through the core removing heat from the fuel rods, exits from the reactor vessel through two outlet nozzles and passes through the tube side of the vertical "U" tube steam generators where heat is transferred to the secondary system. The reactor coolant pumps (RCPs) return the coolant to the reactor vessel.

The principal objective of the thermal–hydraulic design is to avoid fuel damage during normal operation and anticipated transients. It is recognized that there is a small probability of limited fuel damage in certain situations as discussed in Chapter 14.

In order to meet the objective of the thermal–hydraulic design the following design limits are established, but violation of either is not necessarily equivalent to fuel damage:

a. There is a high confidence level that departure from nuclear boiling (DNB) is avoided during normal operation and anticipated transients. This is achieved by confirming the minimum DNBR calculated according to the HTP correlation (Reference 3.1-1) is greater than the 95/95 limit for the correlation;

b. The melting point of the UO_2 fuel is not reached during normal operation or anticipated transients.

The RPS and the reactor control system (RCS) provide for automatic reactor trip or corrective actions before these design limits are exceeded.

The core design bases are presented in Section 3.2; the core mechanical design is discussed in Section 3.3; the nuclear design of the core is discussed in Section 3.4; and the thermal and hydraulic design is discussed in Section 3.5.

3.1.1 REFERENCES

3.1-1 EMF-92-153(P)(A) Rev. 1, “HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, January 2005.

FIGURE 3.1–1 REACTOR VERTICAL ARRANGEMENT

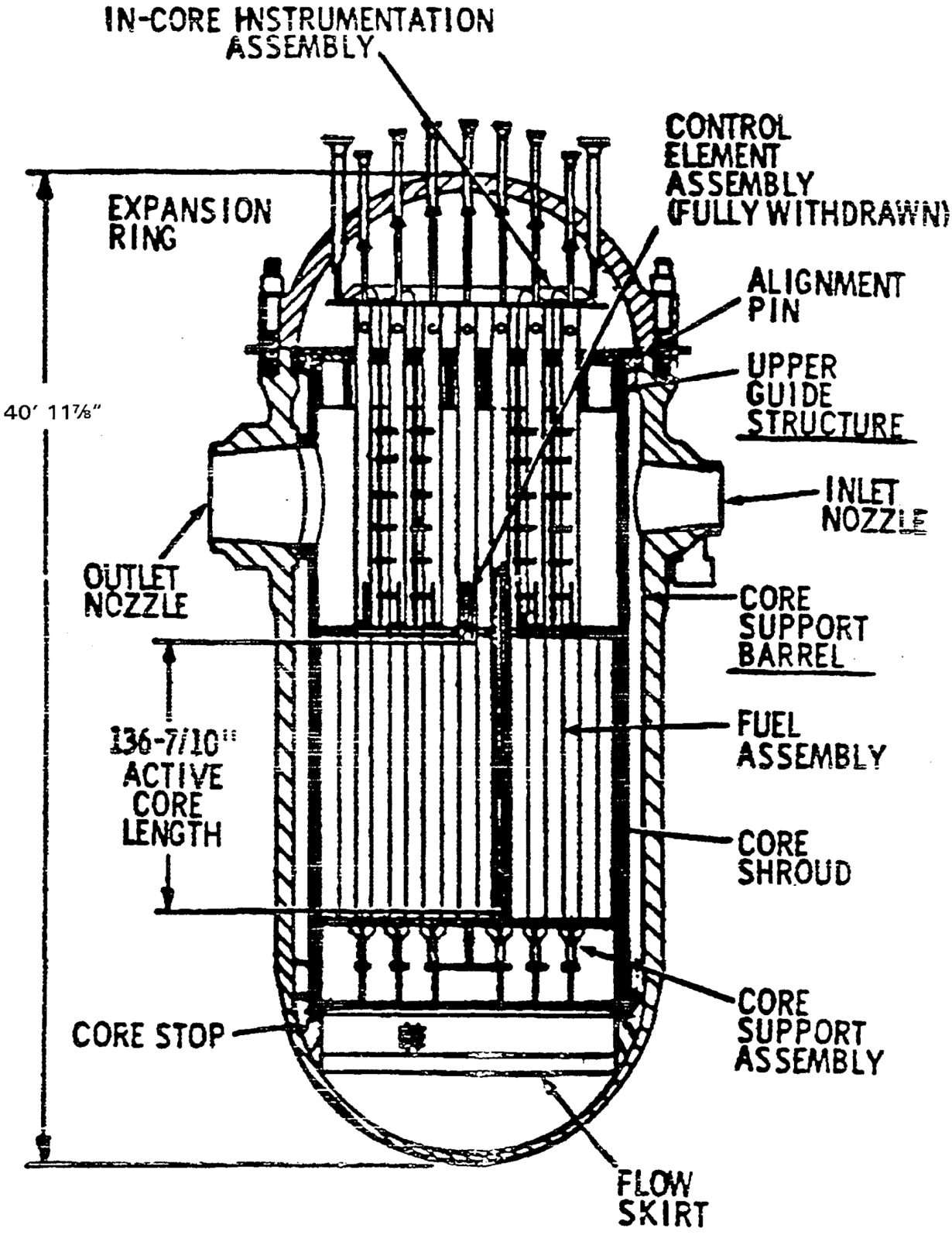
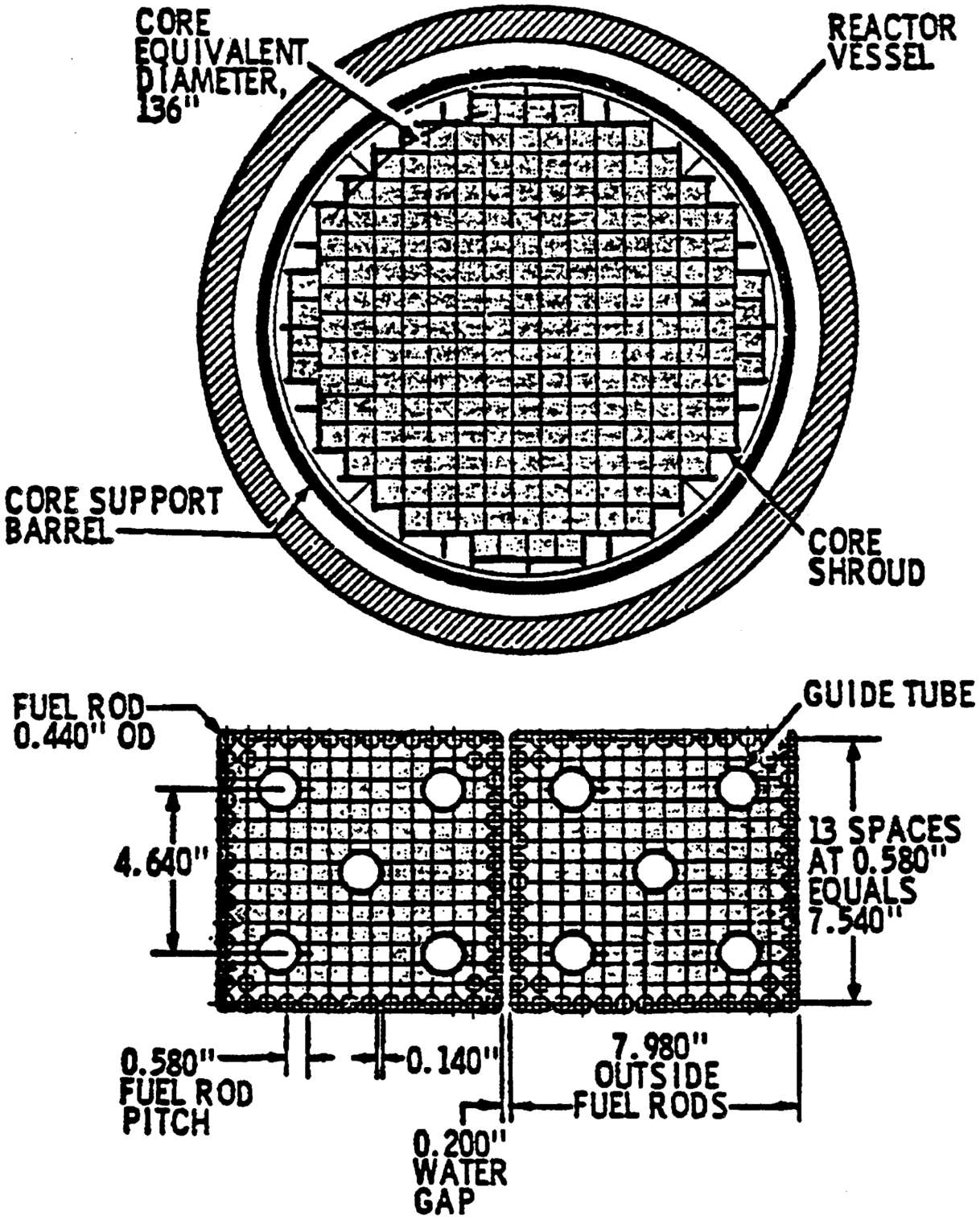


FIGURE 3.1–2 REACTOR CORE CROSS SECTION



3.2 DESIGN BASES

The full power thermal rating of the core is 2,700 MWt. The physics and thermal and hydraulic information presented in this section is based on this core power level.

3.2.1 MECHANICAL DESIGN BASES

3.2.1.1 Fuel Assembly Design Bases

The design bases for evaluating the structural integrity of AREVA fuel assemblies are:

A. Fuel Assembly Handling

The fuel assembly is evaluated for dynamic axial loads of approximately 2.5 times the fuel assembly weight.

B. For All Applied Loads for Normal Operation and Anticipated Operational Events

Fuel assembly component strength is evaluated against either prototype testing or elastic stress analysis. When the stress analysis method is used, the stress limits presented in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, are used as a guide.

The stress design limits for structural components are:

$$P_m \leq 1.0S_m$$
$$P_m + P_b \leq 1.5S_m$$
$$P + Q \leq 3.0S_m$$

where:

P_m is the primary membrane stress intensity
P_b is the primary bending stress intensity
P is the primary stress intensity
Q is the secondary stress intensity

The design stress, S_m is identified in the ASME Boiler and Pressure Vessel Code for austenitic stainless steel as a function of temperature. In the case of Zircaloy, which is not specifically identified in the ASME Boiler and Pressure Vessel Code, the design stress is identified as the lesser of two-thirds the yield stress, S_y, or one-third the ultimate stress, S_u.

The ASME Boiler and Pressure Vessel Code defines the stress intensity based on the maximum shear stress theory. The stress intensity is equal to one-half the largest algebraic difference between two principal stresses.

Primary stresses are developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire wall thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

Secondary stresses are developed by the self-constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the discontinuity conditions due to thermal expansions which cause the stress to occur.

C. Loads during Postulated Accidents

Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods.

The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III. The current methods utilize the limits provided for elastic system analysis.

The design stress intensity value (S_m) is defined the same as for normal operating conditions.

Spacer grid crush load strength is based on the 95% confidence level on the true mean as taken from test measurements on unirradiated production grids at (or corrected to) operating temperature.

3.2.1.2 AREVA Fuel Rod Cladding Design Bases

A discussion of the AREVA fuel rod cladding is given as part of the AREVA fuel rod discussion in Section 3.3.1.1.

3.2.1.3 Control Element Assembly Design Bases

The CEA has been designed to ensure that the stress intensities in the individual structural components do not exceed the allowable limits for the appropriate material established in Section III of the ASME Boiler and Pressure Vessel Code. The exceptions to this criterion are that (a) the Inconel 625 cladding is permitted to sustain plastic strain up to 3 percent due to irradiation induced expansion of the filler materials, and (b) because the ASME Code does not apply to springs, the allowable stresses for the CEA springs are based on values which have been proven in practice.

The CEA stress analyses consider the following load sources:

a. Internal pressure build up due to the effect of irradiation on B_4C (production of helium).

b. External pressure of reactor coolant (in the computation for determining the maximum stress in the cladding due to internal pressure, no internal pressure is assumed).

c. Dynamic stresses produced by seismic loading.

d. Dynamic loads produced by stepping motion of the magnetic jack.

e. Mechanical and hydraulic loads produced during SCRAM.

f. Cladding loads produced by differential expansion between clad and filler materials.

In addition to the comparison of calculated stress levels with allowable stresses, the fatigue damage produced by significant cyclic stresses is also determined. It is a design requirement that the calculated cumulative damage factor for any location may not be equal to or greater than 1.0. The fatigue usage factor calculations are based on the fatigue curves (stress range vs. number of cycles) contained in Section III of the ASME Boiler and Pressure Vessel Code.

3.2.1.4 Reactor Internals Design Bases

The reactor vessel internals are designed to meet the loading conditions and the design limits specified below. The materials used in fabrication of the reactor internal structures are primarily Type 304 stainless steel. The flow skirt is fabricated from Inconel. Welded connections are used where feasible; however, in locations where mechanical connections are required, structural fasteners are used which are designed to remain captured in the event of a single failure. Structural fastener material is typically a high strength austenitic stainless steel; however, in less critical applications, Type 316 stainless steel is employed. Hardfacing, of Stellite material, is used at wear points. The effect of irradiation on the properties of the materials is considered in the design of the reactor internal structures.

A. Categorization and Combination of Loadings

1. Normal Operating and Upset Conditions

The reactor vessel internals are designed to perform their functions safely without shutdown. The combination of design loadings for these conditions are the following:

Normal operating temperature differences

Normal operating pressure differences

Low impingement loads

Weights, reactions and superimposed loads

Vibration loads

Shock loads (including OBE)

Transient loadings of frequent occurrences not requiring shutdown

Handling loads

2. Emergency Conditions

The internals are designed to permit an acceptable amount of local yielding while experiencing the loadings listed above with the SSE load replacing the OBE load.

3. Faulted Conditions

Permanent deformation of the reactor internal structures is permitted. The loadings for these conditions include all the loadings listed for emergency conditions plus the loadings resulting from the postulated LOCA.

B. Design Limits

Reactor internal components are designed to ensure that the stress levels and deflections are within an acceptable range. The stress values for core support structures are not greater than those given in the May 1972 draft of Section III of the ASME Boiler and Pressure Vessel Code, Subsection NG, including Appendix F, “Rules for Evaluation of Faulted Conditions.” Stress limits for the reactor vessel core support structures are presented in Table 3.2-1. In addition, to properly perform their functions, the reactor internal structures will satisfy the deformation limits listed below.

1. Under design loadings plus operating basis earthquake forces or normal operating loadings plus SSE forces, deflections will be limited so that the CEAs can function and adequate core cooling is preserved.

2. Under normal operating loadings plus SSE forces plus pipe rupture loadings resulting from a break of the largest line connect to the primary system piping, deflections will be limited so that the core will be held in place, adequate core cooling is preserved, and all CEAs can be inserted. Those deflections which would influence CEA movement will be limited to less than 80 percent of the deflections required to prevent CEA insertion.

3. Under normal operating loadings plus SSE forces plus the maximum pipe rupture loadings resulting from the full spectrum of pipe breaks, deflections will be limited so that the core will be held in place and adequate core cooling is preserved. Although CEA insertion is not required for a safe and orderly shutdown for break sizes greater than the largest line connected to the primary system piping, calculations show that the CEAs will be insertable for larger breaks except for a

few CEAs located near the vessel outlet nozzle which is feeding the postulated rupture.

3.2.1.5 CEDM/RVLMS (HJTC) Pressure Housing Design Bases

The control element drive mechanism and Reactor Vessel Level Monitoring System (RVLMS) pressure housings form part of the reactor coolant boundary and are, therefore, designed to meet the stress requirements consistent with those of the reactor vessel closure head. The limiting stresses in the CEDM's and RVLMS pressure boundary components due to the design, Level A, Level B, Level C, Level D and Test conditions satisfy ASME Boiler Pressure Vessel Code, Section III, Subsection NB plus Appendix 1 and Section II, Part D, 1998 Edition through 2000 Addenda, including Code Case N-4-12 for the CEDM motor housing material.

The CEDMs and the RVLMs are designed to function normally during and after exposure to normal operating conditions plus the design basis earthquake (DBE). Under normal operating conditions, plus DBE, plus pipe rupture loadings, deflections of the CEDM will be limited so that the CEAs can be inserted after exposure to these conditions. Those deflections, which could influence CEA movement, will be limited to less than 80 percent of the deflections required to prevent CEA movement. The RVLMS and the adjacent CEDMs do not contact each other with maximum lateral displacement of the pressure housings.

Loading Combinations		**ASME Code Subsection**
Design Condition	$P_m \leq S_m$	NB-3221
	$P_l \leq 1.5S_m$	
	$P_l + P_b < 1.5S_m$	
Level A and Level B	$P_l + P_b + Q \leq 3S_m$	NB-3222 and NB3223
(Normal and Upset)	$U \leq 1$	
Level C Condition	$P_m \leq$ greater of $[1.2S_m, S_y]$	NB-3224
(Emergency)	$P_l + P_b \leq$ greater of $[1.8S_m, 1.5S_y]$	
Level D Condition	$P_m \leq$ lesser of $[2.4S_m, 0.7S_u]$	Paragraph F-1330 or F-1340, Appendix F
(Faulted)	$P_l + P_b \leq$ lesser of $[3.6S_m, 1.05S_u]$	
Test Conditions	$P_m \leq 0.9S_y$	NB-3226
	$P_m + P_b \leq 1.35S_y$ when $P_m \leq 0.67S_y$ or $P_m + P_b \leq (2.15\ S_y - 1.2P_m)$ when $0.67S_y < P_m \leq 0.9S_y$	
Design Condition	$P_m \leq S_m$	NB-3221
Shear Stress	$\leq 0.6S_m$	NB-3227.2

where

P_m = General primary membrane stress intensity
P_l = Primary local membrane stress intensity
$P_l + P_b$ = Primary membrane plus bending stress intensity
$P_l + P_b + Q$ = Primary plus secondary stress intensity
S_m = Design stress intensity
S_y = Yield strength
S_u = Tensile strength
U = Cumulative fatigue usage factor

3.2.2 NUCLEAR DESIGN BASES

The initial full power thermal rating of the core is 2700 MWt. It is upon this power level that the physics and thermal and hydraulic information presented in this section are based. The design basis for the nuclear design of the fuel and reactivity control systems are:

a. Excess Reactivity and Fuel Burnup

The excess reactivity provided for each cycle is based on the depletion characteristics of the fuel and burnable poison and the desired burnup for each cycle. The desired burnup is based on the economic analysis of both the fuel cost and the projected operating load demand cycle for the plant. The average burnup in the core is chosen so as to insure that the peak assembly burnup is not greater than 56,000 MWD/MTU for Batch N, 52,500 MWD/MTU for Batch P, and 57,400 MWD/MTU for Batch R and later.

b. Core Design Lifetime and Fuel Replacement Program

The core design lifetime and fuel replacement program are based on a three region core with approximately 40 percent of the fuel assemblies replaced at each refueling.

c. Negative Reactivity Feedback and Reactivity Coefficients

The negative reactivity feedback provided by the design is based on the requirement of General Design Criterion (GDC) 11. In the power operating range, the inherent combined response of the reactivity feedback characteristics (fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), moderator void coefficient (MVC), and moderator pressure coefficient (MPC)) to an increase in reactor thermal power will be a decrease in reactivity.

d. Burnable Poison Requirements

The burnable poison reactivity worth provided in the design will be sufficient to ensure that moderator coefficients of reactivity have magnitudes and algebraic signs consistent with the requirements for negative reactivity feedback and acceptable consequence in the event of postulated accidents or anticipated operational occurrences, viewed in conjunction with the supplied protective equipment.

e. Stability Criteria

The design of the reactor and the instrumentation and control systems is based on meeting the requirements of GDC 12 with respect to spatial oscillations and stability. Sufficient CEA rod worth will be available to suppress xenon-induced power oscillations.

f. Maximum Controlled Reactivity Insertion Rates

The maximum reactivity addition rates are limited by core, CEA, and reactor regulating system (RRS) design based on preventing increases in reactivity which would result in the violation of specified acceptable fuel design limits, damage to the reactor pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency core cooling.

g. Power Distribution Control

Acceptable operation of the reactor in the absence of an accidental transient depends on maintaining a relationship among many parameters, some of which depend on the power distribution. In the absence of an accidental transient the power distribution is controlled such that in conjunction with other controlled parameters, limiting conditions of operation (LCO) are not violated. LCO are not less conservative than the initial conditions used in the accident analyses in Chapter 14. LCO and limiting safety system settings (LSSS) are determined such that specified acceptable fuel design limits are not violated as a result of anticipated operational occurrences and such that specified predicted acceptable consequence are not exceeded for other postulated accidents.

h. Shutdown Margins and Stuck Rod Criteria

The amount of reactivity available from insertion of withdrawn CEAs is required to be sufficient, under all power operating conditions, to ensure that the reactor can be brought to at least 3.6 percent $\Delta\rho$ subcritical from the existing condition, including the effects of cooldown to an average coolant temperature of 532°F, even when the highest worth CEA fails to insert. This criteria is exclusive of any safety allowance and is consistent with the most pessimistic analysis in Chapter 14.

i. Chemical Shim Control

The chemical and volume control system (CVCS) (Section 9.2) is used to adjust dissolved boron concentration in the moderator. After a reactor shutdown, this system is able to compensate for the reactivity changes associated with xenon decay and reactor coolant temperature decrease to ambient temperature. It also provides adequate shutdown margin during refueling. This system also has the capability of controlling long term reactivity changes due to fuel burnup, and reactivity changes during xenon transients resulting from changes in reactor load independently of the CEAs. In particular, any xenon transient may be accommodated at any time in the fuel cycle.

3.2.3 THERMAL AND HYDRAULIC DESIGN BASIS

Avoidance of thermally induced fuel damage during normal steady state and anticipated transient operation is the principal thermal and hydraulic design basis. It is recognized that there is a small probability of limited fuel damage in certain unlikely accident situations discussed in Chapter 14.

The following corollary design basis are established, but violation of them is not necessarily equivalent to fuel damage.

a. A limit corresponding to 95% probability with 95% confidence (Reference 3.2-1) is set on the departure from nucleate boiling ratio (DNBR) during normal operation and any anticipated transients as calculated according to the HTP correlation.

b. The peak temperature of the fuel will be less than the melting point during normal operation and anticipated transients.

The reactor control and protection system will provide for automatic reactor trip or other corrective action before these design limits are exceeded.

The core hydraulic resistance was considered in establishing the operational limits curves provided in Figures 4.5–4 and 4.5–5, and the Low Temperature Overpressure Protection (LTOP) System described in Section 7.4.8. The effect on the RCS flow resistance due to changes in fuel design will be evaluated to determine the impact.

3.2.4 REFERENCES

3.2-1 EMF-92-153(P)(A) Rev. 1, “HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, January 2005.

TABLE 3.2-1 STRESS LIMITS FOR REACTOR VESSEL INTERNAL STRUCTURES

Operating Conditions	Stress Categories and Limits of Stress Intensities
1. Normal and Upset	Figure NG 3221.1 including notes
2. Emergency	Figure NG 3224.1 including notes
3. Faulted	Appendix F, Rules for Evaluating Faulted Conditions

3.3 MECHANICAL DESIGN

The reactor core and internals are shown in Figure 3.3–1A. A cross section of the reactor core and internals is shown in Figure 3.1–2. Mechanical design features of the reactor internals, the control element drive mechanisms (CEDM) and the core are described below. Mechanical design parameters are listed in Table 3.3-1.

3.3.1 CORE MECHANICAL DESIGN

The core approximates a right circular cylinder with an equivalent diameter of 136 inches and an active height of 136.7 inches. It is made up of zirconium alloy clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets and UO_2-Gd_2O_3 pellets. The fuel rods are grouped into 217 assemblies.

Short term reactivity control is provided by 73 control element assemblies (CEA). The CEAs are guided within the core by the guide tubes which are integral parts of the fuel assemblies.

3.3.1.1 AREVA Fuel Rod

The detailed fuel rod design (see Figures 3.3–1A and 3.3–1B) establishes such parameters as pellet diameter and length, density, cladding-pellet diametral gap, fission gas plenum size, and rod pre-pressurization level. The design also considers effects and physical properties of fuel rod components which vary with burnup.

The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures, and excessive cladding stresses and strains. This end is achieved by designing the fuel rods to satisfy the design criteria (Reference 3.3-12) during normal operation and anticipated operational occurrences over the fuel lifetime. For each design criteria, the performance of the most limiting fuel rod shall not exceed the specified limits.

Fuel rods are designed to function throughout the design life of the fuel based upon the reactor operating conditions designated below without loss of mechanical integrity, significant dimensional distortion, or release of fuel or fission products.

The assemblies were evaluated for a peak assembly burnup of 56,000 MWD/MTU for Batch N, 52,500 MWD/MTU for Batch P, and 57,400 MWD/MTU for Batch R and later.

The Millstone Unit 2 Cycle 19 reload core included four fuel assemblies with natural uranium replacement fuel rods with an anti-rotation feature designed to prevent spinning of the rod during operations. The four assemblies containing replacement rods, and the conditions under which they were evaluated for use, are discussed in Section 3.3.1.3.1, "Design Summary".

3.3.1.1.1 Fuel Rod Mechanical Criteria

The cladding primary and secondary stresses shall meet the 1977 ASME Boiler and Pressure Vessel Code Section III (Reference 3.3-1) requirements summarized below:

Stress Intensity Limits

Zircaloy-4 Fuel Rod Cladding		
(*Parameter*)	Yield Strength	Ultimate Tensile Strength
Primary Membrane (P_m)	$< 2/3\ S_y$	$< 1/3\ S_u$
Primary Membrane Plus Primary Bending ($P_m + P_b$)	$< 1.0\ S_y$	$< 0.5\ S_u$
Primary Plus Secondary ($P + Q$)	$< 2.0\ S_y$	$< 1.0\ S_u$
M5® Fuel Rod Cladding		
(*Parameter*)	Yield Strength	
Primary Membrane (P_m)	$< 1.0\ S_y$ (Compression) $< 2/3\ S_y$ (Tension)	
Primary Membrane Plus Primary Bending ($P_m + P_b$)	$< 1.0\ S_y$	
Primary Plus Secondary ($P + Q$)	$< 2.0\ S_y$	
The M5® cladding stress intensity limits are based on hoop yield strength per Reference 3.3-11.		

Primary stresses are developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

Secondary stresses are developed by the self constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the discontinuity conditions due to thermal expansions which cause the stress to occur.

Cladding circumferential strain shall not exceed the design limit through end-of-life (EOL).

The total uniform strain, elastic and plastic shall not exceed the design limit during a transient. The strain analysis was performed with the RODEX2 (Reference 3.3-2) RAMPEX codes benchmarked to available power ramp test data, i.e., INTERRAMP, OVERRAMP, and SUPERRAMP.

The fuel rod shall be designed such that at a rod average burnup when substantial axial consolidation has occurred, the total clad creep deformation shall not exceed the initial minimum diametral fuel cladding gap. This will prevent pellet hangups allowing the plenum spring to close axial gaps until densification is substantially complete, thus preventing the formation of pellet column gaps of sufficient size for clad flattening.

The fuel rod pressure at EOL shall not exceed the criteria approved by the NRC (Ref. 3.3-3). A review of departure from nucleate boiling ratio (DNBR) limits for condition III or IV postulated accidents events is required for fuel rods that exceed nominal system pressure. When fuel rod pressure is predicted to exceed system pressure, the pellet-cladding gap shall not increase for steady or increasing power conditions. Analysis approved by the NRC has shown that the fuel rod gas pressure can safely exceed system pressure without causing any damage to the cladding.

Total cladding wall thinning due to generalized external and internal corrosion shall not exceed a value which will impair mechanical performance over the projected fuel rod design lifetime under the most adverse projected power conditions within coolant chemistry limits recommendations of Table 3.3-2. It will also assure that the metal/oxide interface temperature will remain well below the level where large increases in corrosion, due to the insulating effect of the oxide, would adversely affect the mechanical behavior of the cladding.

The cumulative usage factor for cyclic stresses for all important cyclic loading conditions shall not exceed the design limit.

The clearance between the upper and lower tie plate shall be able to accommodate the maximum differential fuel rod and fuel assembly growth to the designed burnup.

The centerline temperature of the hottest pellet shall be below the melting temperature. Fuel centerline temperature is calculated at overpower conditions to verify that fuel pellet overheating does not occur during normal operation and anticipated operational occurrences.

3.3.1.1.2 Fuel Rod Design Analyses

Each design analysis was performed with Framatome methodology which involves a well defined selection of appropriate data and parameters, and the latest approved versions of computer codes. This methodology, as required, has been submitted to the Nuclear Regulatory Commission (NRC) and approved. The analysis is performed in accordance with the methods described in Framatome’s “Qualification of Exxon Nuclear Fuel For Extended Burnup” (Reference 3.3-3) and “Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU” (Reference 3.3-13).

The impact of fuel thermal conductivity degradation (TCD) with burnup has been considered and included in the fuel rod analyses consistent with the NRC's approval of Framatome treatment of fuel TCD in Reference 3.3-12.

The cladding steady state stress analysis was performed by considering primary and secondary membrane and bending stresses due to hydrostatic pressure, flow-induced vibration, spacer contact, pellet cladding interaction (PCI), thermal and mechanical bow and thermal gradients. Stresses were calculated for the various combinations of the following conditions:

a. beginning of life (BOL) and EOL

b. cold and hot conditions

c. at mid-span and at spacer locations

d. at both the inner and outer surfaces of the cladding

The analysis was performed for the various sources of stress, including pressure, thermal, spacer contact, PCI, and rod bow. The applicable stresses at each orthogonal direction were combined to calculate the maximum stress intensities which are compared to the ASME design criteria. The results of the analysis indicate that all stress values are within acceptable design limits for both BOL and EOL, hot and cold conditions. The EOL stresses have ample margin for both the hot and cold condition stresses.

The cladding steady state strain is evaluated with the RODEX2 code, which has been approved by the NRC (Reference 3.3-2). The code considers the thermal-hydraulic environment at the cladding surface, the pressure inside the cladding, and the thermal, mechanical and compositional state of the fuel and cladding. Pellet density, swelling, densification, and fission gas release or absorption models, and cladding and pellet diameters are input to RODEX2 to provide the most conservative strain calculation or subsequent ramping or collapse calculations for the reference fuel rod design. The major fuel rod performance characteristics modeled by the RODEX2 code are:

a. Radial Thermal Conduction and Gap Conductance

b. Fuel Swelling, Densification, Cracking, and Crack Healing

c. Gas Release and Absorption

d. Cladding Creep Deformation and Irradiation-Induced Growth

e. Cladding Corrosion

f. PCI

g. Free Rod Volume and Gas Pressure

The calculations are performed on a time incremental basis with conditions updated at each calculated increment so that the power history and path dependent processes can be modeled. The axial dependence of the power and burnup distributions are handled by dividing the fuel rod into a number of axial and radial regions. Power distributions can be changed at any desired time, and the coolant and cladding temperatures are readjusted in all the regions. All the performance models, e.g., giving the deformations of the fuel and cladding and gas release, are calculated at successive times during each period of assumed constant power generation. The calculated cladding strain is reviewed throughout the life of the fuel and both the maximum circumferential strain and the maximum strain increment are compared with the design criteria. The calculated strain did not exceed the strain limit. Both the maximum strain and the positive strain increment are below the design limit strain.

The ramping strain and the fatigue evaluation of the fuel rod were evaluated. The ramps are assumed to occur anytime during the irradiation and may reach the maximum peaking factor allowed by the limits of operation. The ramps are analyzed either from cold shutdown or from a variety of hot powered starting conditions. The approach to rated power at the beginning of each reactor cycle is performed to satisfy the AREVA maneuvering and conditioning recommendations. The clad response during ramping power changes is calculated with the RAMPEX code. This code calculates the PCI during a power ramp for one axial node at a time. The initial conditions are obtained from RODEX2 output. The RAMPEX code considers the thermal condition of the rod in its flow channel, and the mechanical interactions that result from fuel and cladding creep at any desired axial section in the rod during the power ramp. As compared to RODEX2, RAMPEX additionally models the pellet cladding axial stress interaction, primary creep with strain hardening, the effects of pellet chips, and localized stresses due to ridging.

The RAMPEX code provides the hoop stress and the stress intensity. The stress results of the ramping analysis are used to evaluate the cladding fatigue damage through life due to the cyclic power variations. The fatigue analysis is based on the O'Donnel and Langer (Reference 3.3-4) design curve. The cyclic amplitudes of the maximum local stress intensity, as determined by RAMPEX over the power cycling range, are compared with this curve to determine the allowed cycles for each stress range. This result is combined with the projected number of duty cycles to determine a fatigue usage factor. All of the reactor cycle (startup) ramp stresses were within the design limit.

Creep collapse calculations are performed with RODEX2 and COLAPX codes. The RODEX2 code determines the cladding temperature and internal pressure history based on a model which accounts for changes in fuel rod volumes, fuel densification and swelling, and fill gas absorption. The reactor coolant, fuel rod internal temperature, and pressure histories generated by the RODEX2 analysis are input to the COLAPX code along with a conservative statistical estimate of initial cladding ovality and the fast flux history. The COLAPX code calculates, by large deflection theory, the ovality of the cladding as a function of time while the uniform cladding creepdown is obtained by the RODEX2 analysis. The cladding ovality increase and creepdown are summed, at a rod average burnup when substantial axial consolidation has occurred, to show that they remain less than the initial minimum pellet clad gap. Measurements of highly densifying irradiated fuel have demonstrated that pellet densification is essentially complete by the time the fuel has attained this burnup so that further creepdown after this phase will not result in significant pellet to pellet gaps. The combined radial creepdown was shown to meet the design criteria. This will prevent pellet hangups due to cladding creep, allowing the plenum spring to close axial gaps until densification is substantially complete, and thus assures that clad collapse will not occur. The pitch of the plenum spring is less than the spacing calculated for stiffening rings in a cylindrical shell under external pressure which will prevent clad collapse in the plenum area.

Calculation of the gas pressure within a fuel rod is performed with the RODEX2 code. The initial fill gas is found by calculating the initial free volume and using the ideal gas law, along with input values for fill gas pressure and reference fill gas temperature. The free gaseous fission product yield is calculated for each axial region and the total yield obtained by summing the axial region contributions. The power of each history used was multiplied for each cycle by a factor required

for the highest projected rod power to reach the F_r limit plus uncertainties. The calculations show that for all power histories analyzed, the rod internal gas pressure will remain below the criteria approved by the NRC (Reference 3.3-3) for use in extended burnup gas pressure analysis.

The waterside corrosion of fuel rods is evaluated with the MATPRO-11 (Reference 3.3-5) correlation. The MATPRO-11 model is a two-stage corrosion rate model which is cubic in dependence on oxide thickness until a transition to a subsequent linear dependence occurs. To calculate the rate changes as a function of both oxide thickness and the operating conditions of the fuel rod, the MATPRO model is incorporated into AREVA's RODEX2 fuel performance code. The RODEX2 code determines the temperature increase of the water along the fuel rod assuming heat balance within a channel for the prescribed mass flow and inlet temperature. The radial temperature drops are evaluated successively between the water, the oxide surface, the metal/ oxide interface, and the inside of the cladding using RODEX2 correlations and methods. To account for the change in corrosion rate due to the changing oxide layer and thermal conditions, the code includes an update in cladding temperature at every calculation step. This is an iterative process due to the continuously changing oxide thickness. Conditions are also revised at times where new power or flow conditions are prescribed. The MATPRO model incorporated in RODEX2 is benchmarked via an overall enhancement factor to oxide thickness data from assemblies in seven separate reactors. Each data point represents the maximum thickness measured along a rod length. The enhancement factor is based on a best fit regression analysis of the data. A final multiplier is also applied which envelopes the data. The waterside corrosion in the cladding was evaluated with RODEX2 for the steady state strain analysis. A best-fit corrosion amplification factor was applied to the MATPRO model along with a final multiplier to bound the measured data on AREVA standard cladding. The maximum calculated oxide thickness was below the design limit.

Fuel rod and fuel assembly growth projected to occur during irradiation was based on conservative design curves established from measured irradiation growth data. The rod growth minus the assembly growth plus tolerances was compared with the clearance within the assembly for fuel rod growth. Differential thermal expansion between the fuel rods and guide tubes was also considered. There is space between the upper and lower tie plates to accommodate the maximum differential growth out to a rod burnup of 62,000 MWd/MTU.

The pellet centerline temperature calculation was performed with the RODEX2 code. Fuel pellet centerline temperatures were calculated at overpower conditions. The high power cycle of each power history was modified to include a spike in each cycle. This spike increased the maximum power of a pellet in the rod up to F^T_Q. Pellet melting temperature is a function of burnup. Considering a conservative peak pellet burnup to determine the minimum pellet melting temperature at EOL, the maximum pellet centerline temperature is well below both BOL and EOL limits.

3.3.1.2 (Deleted)

3.3.1.3 AREVA Fuel Assembly

3.3.1.3.1 Design Summary

The AREVA fuel assemblies are 14 by 14 arrays containing 176 fuel rods in a cage structure of 5 guide tubes and 9 spacer grids. Both the guide tubes and the fuel rod cladding are made of zirconium alloy for low neutron absorption and high corrosion resistance. The fuel assembly upper tie plates are stainless steel castings with Nickel Alloy X-750 holddown springs. The fuel assembly upper tie plate is mechanically locked to the guide tubes and may be easily removed to allow inspection of irradiated fuel rods. For Reload T (Cycle 15) and beyond, lower tie plates are the FUELGUARD™ debris resistant design.

In Reloads M, N, and P (Cycles 10-12), eight of the nine spacers in each fuel assembly are made of a Zircaloy-4 structure with Nickel Alloy 718 springs (i.e., bi-metallic spacer). The ninth spacer, located just above the lower tie plate, is made of Nickel Alloy 718 and, using features of the AREVA High Thermal Performance (HTP) spacer design, has been adapted to provide fuel assembly debris resistance.

The fuel assembly design for Reloads R and S (Cycles 13 and 14) has all nine spacers of the bimetallic design. Additionally, in this design a longer solid fuel rod lower end cap is used. The longer end cap serves to raise the fuel rod cladding above the debris trapping region of the ninth (bottom) spacer.

In Reloads T through X (Cycles 15-18), the High Thermal Performance (HTP) fuel assembly design was implemented in which all nine spacers are of the Zircaloy-4 HTP design. This design retained the longer, solid fuel rod lower end cap.

The fuel assembly design for Reload Y (Cycle 19) and later utilized eight Zircaloy-4 HTP spacers and replaces the ninth, bottom spacer with an Inconel High Mechanical Performance (HMP) spacer. The HMP spacer is similar to the HTP spacer, except that it is constructed of Nickel Alloy 718 and the flow channels are parallel to the fuel.

Drawings of the AREVA fuel assemblies are given in Figure 3.3–2A and Figure 3.3–3A. Fuel assembly drawings for Reload T (Cycle 15) and beyond are included in Figures 3.3–2B and 3.3–3B.

The fuel assembly design for Reload EE (Cycle 25) and later changes to the AREVA Standard CE-14 HTP™ fuel design. This design features M5® clad fuel rods with increased uranium loading (larger diameter pellet and increased theoretical density) and Zircaloy-4 MONOBLOC™ corner guide tubes.

The analysis has shown that the AREVA reload fuel assemblies will meet the design criteria:

a. The maximum steady state cladding strain is well below the design limit.

b. The maximum steady state cladding stress meets the ASME Boiler and Pressure Vessel Code requirements.

c. The transient strain is within the circumferential limit.

d. Cladding creep collapse is precluded.

e. The fuel rod internal pressure at the EOL remains below the criteria approved by the NRC (Ref. 3.3-3).

f. The maximum clad oxidation is below the design limit.

g. The cladding fatigue usage factor is well below the design limit.

h. There is space between the upper and lower tie plate to accommodate fuel rod growth.

i. Pellet centerline temperatures remain below the design criteria.

j. The fuel assembly growth is within the space available between the upper and lower core plates in the reactor core.

k. The assembly holddown springs will prevent bundle lift-off.

The fuel rods consist of short cylindrical UO_2 pellets or UO_2-Gd_2O_3 pellets contained in zirconium alloy tubular cladding. Zirconium alloy end caps are welded to each end to give a hermetic seal.

The fuel rod upper plenum contains a high strength alloy compression spring to prevent fuel column separation during fabrication and shipping, and during incore operation. The rods are pressurized with helium to improve heat transfer and reduce clad creep ovality.

The fuel assembly structure consists of an upper tie plate assembly, lower tie plate, guide tubes and spacer grids, which together provide the support for the fuel rods.

The lower tie plate is a machined stainless steel casting which provides the lower end support for the guide tubes. The zirconium alloy guide tubes are attached to the lower tie plate by means of Nickel Alloy X-750 cap screws. The FUELGUARD™ lower tie plate, included in Reload T and beyond provides protection to the fuel from debris in the primary coolant.

The upper tie plate assembly latches to and provides the upper end support for the guide tubes. The upper tie plate assembly consists of a stainless steel grid structure and reaction plate containing five Nickel Alloy X-750 holddown springs. The springs are located around Nickel Alloy X-750 locking nuts and sleeves which mechanically attach to the guide tubes and pilot into the reactor alignment plate. The springs are partially shrouded on the outside diameter by stainless steel cups to prevent flow induced spring vibration.

The guide tubes, in conjunction with the spacers and tie plates, form the structural skeleton of the fuel assembly and provide channels for insertion of the control rods. The guide tubes are fabricated from Zircaloy-4 tubing and are fully annealed. The center tube is of uniform diameter whereas the outer four guide tubes have a reduced diameter section at the bottom which produces a dashpot action to decelerate the dropped CEAs.

An end plug is welded to the lower end of the guide tube and is drilled and threaded to accept the lower cap screws. At the upper end, the guide tube is crimped into an external stainless steel locking sleeve which engages the upper tie plate assembly. The upper tie plate assembly is locked to the guide tube end fittings and can be unlocked for reconstitution or for fuel examination using special tools.

A stainless steel sleeve assembly with a chrome plated inside diameter is inserted in the top end of the guide tube assembly. This sleeve protects the guide tube from control rod fretting and wear when the rod is in the withdrawn/ready position. The sleeve is mechanically captured by the upper tie plate.

Fuel rod pitch and position is maintained by nine spacer grids. The spacers are axially positioned so that the assemblies will be compatible with existing fuel assemblies.

The bi-metallic spacers used in Reloads M through S (Cycles 10-14) are formed by an interlocking rectangular grid of Zircaloy-4 structural strips (see Figure 3.3–4A). Inconel-718 spring strips are mechanically secured within these strips. The Zircaloy-4 structural strips are welded at all intersections and to the side plates. Dimples formed in the structural strips center the fuel rod within the cell and along with the springs provide a positive but compliant support for each rod, sufficient to prevent fretting vibration.

In Reloads M, N, and P (Cycles 10-12), the debris resistant Nickel Alloy 718 HTP spacer grid in the ninth, bottom location is located just above the lower tie plate. It is formed by an interlocking rectangular grid of Nickel Alloy 718 strips. The strips are welded at all intersections and to the side plates. The spacer is positioned on top of the lower tie plate with the strip intersections directly above the tie plate flow holes. This reduces the size of debris that may pass through the flow holes thereby reducing the possibility of fretting against the cladding. Reloads R and S (Cycles 13 and 14) use a similar debris resistant concept with the Nickel Alloy 718 HTP spacer replaced by a bimetallic spacer coupled with a longer lower end cap on the fuel rods.

The HTP spacers for Reloads T through X (Cycles 15-18) are all Zircaloy-4 (Figure 3.3–4B). The strips are welded at the intersections and side plates. The structure of the Zircaloy-4 strips provides the rod support.

In Reload Y (Cycle 19) and later, all Zircaloy-4 HTP spacers are used in eight locations. The Nickel Alloy 718 HMP spacer is used in the ninth, bottom location. The Nickel Alloy 718 HMP bottom spacer is similar in design to the HTP spacers except for the flow channels, which are not canted.

The Millstone Unit 2 Cycle 19 reload core included four fuel assemblies with natural uranium replacement fuel rods with an anti-rotation feature designed to prevent spinning of the rod during operations. The four assemblies containing replacement rods were installed in symmetric, peripheral core locations against the baffle as shown in Figure 3.3–19 (Reference 3.3-9). The core locations into which the assemblies were placed where P–1, A–8, H–21, and Y–14 (see Figure 3.4–1). The replacement rods installed under these conditions were evaluated against established mechanical, nuclear, and thermal/hydraulic design criteria for Millstone Unit 2 fuel, and were determined to be compliant with their design and licensing bases (Reference 3.3-10).

The fuel assembly design for Reload EE (Cycle 25) and later is updated to the AREVA Standard CE-14 HTP™ design by implementing an M5® clad fuel rod and the MONOBLOC™ corner guide tube design. The M5® clad fuel rod increases the nominal pellet OD from 0.377 inches to 0.3805 inches and increases the nominal pellet theoretical density from 95.35% to 96.00%. The cladding material changes from Zircaloy-4 to M5® for improved corrosion resistance and reduced hydrogen pickup. The cladding ID also changes from 0.384 inches to 0.387 inches to accommodate the larger fuel pellet. The axial position of the fuel column is slightly lowered and the rod length is increased from 146.25 inches to 146.67 inches.

The AREVA Standard CE-14 HTP design utilizes Zircaloy-4 MONOBLOC™ corner guide tubes. The MONOBLOC™ guide tube design maintains a constant outer diameter whereas the inner diameters change between the dashpot region and the non-dashpot region (the wall thickness increases in the dashpot region). As compared to the previous design, the inner diameters of the non-dashpot and dashpot regions are unchanged and the outer diameters of the guide tube in the non-dashpot region are unchanged. The interface with the fuel assembly lower tie plate is also unchanged.

The bottom HMP™ spacer grid on the AREVA Standard CE-14 HTP fuel design is modified at the corner guide tube locations to accommodate the larger MONOBLOC™ guide tube outer diameters. All of the rod positions, interfaces with the fuel rods, and side plates are the same as the previous design. The HTP™ spacer grid design, the FUELGUARD™ lower tie plate, and the reconstitutable upper tie plate are unchanged from the previous design.

3.3.1.3.2 Fuel Assembly Mechanical Criteria

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling operational and accident loads. These limits are applied to the design and evaluation of upper and lower tie plates, grid spacers, guide tubes, holddown springs, and locking hardware.

The design bases for evaluating the structural integrity of the fuel assemblies are:

a. Fuel Assembly Handling - Dynamic axial loads approximately 2.5 times assembly weight.

b. For All Applied Loads for Normal Operation and Anticipated Operational Events - The fuel assembly component structural design criteria are established for the two primary material categories, austenitic stainless steels (tie plates), and Zircaloy (guide tubes, grids, spacer sleeves). The stress categories and strength theory for austenitic stainless steel presented in the ASME Boiler and Pressure Vessel Code, Section III (Reference 3.3-1) are used as a general guide.

   Steady state stress limits are given in FSAR Section 3.3.1.1.1. Stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.

c. Loads During Postulated Accidents - Deflection or failure of components shall not interfere with reactor shutdown or emergency cooling of the fuel rods during postulated seismic and loss of coolant accident (LOCA) occurrences.

   The assembly structural component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III.

The design basis for the guide tube wear sleeves is that the fuel assembly shall not be damaged by CEA induced fretting-wear. Flow tests at reactor conditions of prototypic fuel and guide tube wear sleeve assemblies have been used in establishing the performance of the CEA wear sleeve combination.

The holddown springs, as compressed by the upper core plate during reactor operation, shall provide a net positive downward force during steady state operation, based on the most adverse combination of component dimensional and material property tolerances. In addition, the holddown springs are designed to accommodate the additional load associated with a pump overspeed transient (resulting in possible temporary liftoff of the fuel assemblies), and to continue to ensure fuel assembly holddown following such an occurrence.

The fuel assembly growth plus BOL length shall not exceed the minimum space between the upper and lower core plates in the reactor cold condition (70°F). The reactor cold condition is limiting since the expansion coefficient of the stainless steel core barrel is greater than the coefficient of expansion of the Zircaloy guide tubes.

The spacer assembly is designed to withstand the thermal and irradiation induced differential expansion between the fuel rods and guide tubes and to withstand the design handling and accident loads discussed above. The debris resistant Nickel Alloy 718 HTP spacer used in the ninth, bottom location for reloads M, N and P (Cycles 10-12) was positioned such that the internal strip intersections are directly above the lower tie plate flow holes, thus reducing the size of debris which could pass through the lower tie plate.

In Reloads R and S (Cycles 13 and 14), the Nickel Alloy 718 HTP spacer grid at the ninth, bottom location was replaced with a bimetallic spacer which is raised off the upper surface of the lower tie plate. The gap between the upper surface of the lower tie plate and the lower surface of the bimetallic spacer is spanned by a long fuel rod end cap of solid Zircaloy-4.

The Zircaloy-4 HTP spacer grid is used in all nine locations in Reloads T through X (Cycles 15-18). This design is typically referred to as the 'HTP Fuel Assembly'. This spacer grid design provides improved DNB performance, structural strength, and fretting resistance. The long fuel rod end cap is maintained in the HTP Fuel Assembly.

In Reload Y (Cycle 19) and later, the Zircaloy-4 HTP spacer grid is used in eight locations and a Nickel Alloy 718 HMP spacer grid is used in the ninth, bottom location. This design retains the long fuel rod lower end cap and is typically referred to as the 'HTP+HMP Fuel Assembly'. The HTP+HMP design has improved structural strength, and fretting resistance compared to the HTP design.

In Reload EE (Cycle 25) and later, the fuel design is updated to the AREVA CE-14 HTP™ design by implementing the M5® clad fuel rod design and the MONOBLOC™ corner guide tube design. See Section 3.3.1.3.1 for additional details on the updated design.

The design analysis is based upon reactor operating conditions. Typically, these conditions are:

Nominal Core Thermal Power	= 2700 MW
Nominal Coolant Pressure	= 2250 psia
Maximum Flow for Fuel Assembly Liftoff	= 422,466 gallons per minute (at 380°F)
Maximum Core Coolant Inlet Temperature at Nominal Power	= 549°F
Total Average Linear Power	= 6.206 kW/ft

The power histories used in the design analysis are designed to achieve a peak assembly burnup of 56,000 MWD/MTU for Batch N, 52,500 MWD/MTU for Batch P, and 57,400 MWD/MTU for Batch R and later.

Conservative rod local peaking factors are used which result in a peak rod burnup of 62,000 MWd/MTU. Each of the rod design histories follows the single hottest rod in the first cycle operation, the hottest rod in second cycle operation, etc.

Fuel assembly components must be able to withstand anticipated seismic and LOCA forces. These may result from bundle vibration and impact due to a seismic or LOCA event. An analysis was performed for the previous reloads to determine the maximum bundle displacements and the maximum spacer grid forces expected during postulated accidents for Millstone 2. The loads and displacements analysis, which was performed by CE (Reference 3.3-6), considered the safe shutdown earthquake (SSE) and limiting Branch Line LOCA events, and the dynamic properties of the AREVA reload fuel assemblies. The resulting fuel assembly displacements and the combined seismic and LOCA grid spacer impact forces were provided to AREVA.

The loads and displacements were conservatively adjusted for the Batch R design due to the optimization of the fuel rod. The fuel weight was increased and the assembly stiffness was decreased. The spacer impact loads and the fuel assembly maximum deflections were conservatively recalculated from the reference analysis values. The spacer strength margin, the guide tube stresses, and the fuel rod stresses were calculated for the adjusted loads.

Calculated stresses at the appropriate deflections were combined with the steady state stresses and compared with the ASME Design Criteria for faulted conditions. This limit is 0.7 times the ultimate strength for the primary stress combinations as compared to 0.5 times ultimate for steady state loadings. This criteria was met for both the fuel rods and the guide tubes.

The calculated grid spacer loads during each accident and the combined loads were compared with the allowable grid spacer strength at operating temperature. The loads evaluated were the maximum projected one-sided impact load and the maximum through grid load. The maximum allowable crushing load is the 95 percent confidence lower limit of the true mean of the distribution of crush test measurements. The allowable through grid strength is well above the maximum through grid load. It is also above the maximum one-sided impact load. For Reload R through Reload DD, the seismic/LOCA calculations were reviewed and determined to be bounding. For Reload EE and beyond, the seismic/LOCA calculations were re-evaluated for the upgraded fuel design and produced acceptable margins.

design and produced acceptable margins.

3.3.1.4 Fuel Assembly Holddown Device

A fuel assembly holddown device has been incorporated to prevent the possibility of lifting the fuel assembly by hydraulic forces under all normal flow conditions with temperature greater than 500°F. The holddown device consists of a spring-loaded plate which is integral to the fuel assembly. The springs are compressed as the upper guide structure is lowered into place. The added spring load, together with the weight of the fuel assembly, prevents possible axial motion of the fuel assembly during operating conditions.

The holddown device is incorporated into the upper end fitting and features a movable holddown plate which acts on the underside of the fuel alignment plate (refer to Figure 3.3–5). The movable plate is loaded by coil springs which are located around the upper end fitting posts. The springs are positioned at the upper end of the assembly so that the spring load combines with the assembly weight in counteracting the upward hydraulic forces. The springs are sized and the spring preload selected, such that a net downward force will be maintained for all normal and anticipated transient flow and temperature conditions. It should be noted that the movable plate also serves as the lifting surface during handling of the fuel assembly.

Assembly holddown was previously analyzed in Section 3.6.1 of Reference 3.3-8. The analysis has been reperformed for Batch T and beyond fuel and is conservative.

3.3.1.5 Control Element Assembly

CEAs are provided by Combustion Engineering (CE) and AREVA. The CEA (shown in Figure 3.3–6) is comprised of five Inconel tubes 0.948 inch in diameter. All tubes contain neutron poison materials with the distribution of the poison materials as depicted in Figure 3.3–7. Each tube is sealed by welded end caps. A gas expansion space is provided to limit maximum tube stress due to internal pressure developed by the release of gas and moisture from the boron carbide. The overall length of the CEA is provided in Table 3.3-1. Four tubes are assembled in a

square array around the centrally located fifth tube. The tubes are welded to an upper end fitting. The upper end fittings are attached to a spider hub which couples the CEA to the drive mechanism through the extension shaft.

Mechanical reactivity control is achieved by operational maneuvering of groups of single CEAs. The dual CEA is made up of two single CEAs connected to separate grippers attached to single extension shaft. The arrangement of the CEAs in the core is shown in Figures 3.3–8 and 3.3–9.

There are 49 single CEAs and 12 dual CEAs all operated by a total of 61 CEDMs. Considering the 12 dual CEAs as 24 single CEAs gives an overall number of 73 CEAs in the core.

A buffer (deceleration dashpot) system is used for slowing down the CEAs at the end of a reactor trip. The buffering action is accomplished by guide tubes which have a reduced diameter in the lower section. When the tip of a CEA falls into the buffer region, the pressure buildup in the lower guide tube supplies the force to slow down the CEA. The velocity is decreased to a level which will minimize impact. The final impact is further cushioned by a coil spring arrangement mounted around the center CEA finger.

The four outer guide tubes have the reduced diameter lower section (dashpot). There is no dashpot in the center guide tube. There are four bleed holes above the dashpot region for the four outer guide tubes. For the center guide tube, these four bleed holes are at a lower elevation. For all guide tubes, there is a small drain hole at the bottom. The CEA tip is filled with a Silver-Indium-Cadmium alloy. This replaces the B_4C to avoid the change of buffer characteristics that B_4C radiation-induced swelling might bring about.

The design parameters have been optimized to establish the best combination of buffer stroke and buffer annulus. A significant analytical effort has shown that the pressure buildup and the impact loads are not damaging to the system. In addition, a test program has confirmed the feasibility of the system. It has demonstrated that the buffer will work under the worst expected tolerance condition.

3.3.1.6 Neutron Source Design

For Cycle 18 and beyond, the reactor core will not utilize neutron sources. It has been determined that during startups without neutron sources, there will continue to be a sufficient neutron count rate at each of the four Wide Range (WR) Excore fission detectors due to the high burnup fuel assemblies that will be positioned on the core periphery.

For Cycle 17 and earlier, four neutron sources were installed in the reactor core. They were held in vacant CEA guide tubes by means of an externally loaded spring reacting between the upper fuel alignment plate and the top of the fuel assembly. The cladding of the neutron source rods is of a free standing design. The internal pressure is always less than reactor operating pressure. Internal gaps and clearances are provided to allow for differential expansion between the source material and cladding.

3.3.1.7 In-Core Instruments

The in-core instruments (refer to Section 7.5.4) are located in the in-core instrumentation assembly (Figure 3.3–10). The in-core instrumentated thimble support frame and guide tubes are supported by the upper guide structure (UGS) assembly. The tubes are conduits which protect the in-core instruments and guide them during removal and insertion operations. The thimble support frame supports the 43 in-core thimble assemblies and acts as an elevator to lift the thimbles from the core into the UGS during the refueling operation.

3.3.1.8 Heated Junction Thermocouples

The heated junction thermocouple (HJTC) system is composed of two channels of HJTC instruments. Each HJTC instrument channel is manufactured into a probe assembly consisting of eight HJTC sensors, a seal plug, and electrical connectors (Figure 7.5–6). The eight HJTC sensors are physically independent and located at eight levels from the reactor vessel head to the fuel alignment plate.

The probe assembly is housed in a stainless steel support tube structure that protects the sensors from flow loads and serves as the guide path for the sensors. Figure 3.3–18 describes the locations of the HJTC probe assemblies.

HJTC Probes and Support Tubes in Upper Guide Structure

The HJTC probes and support tubes are installed inside two-part length CEA shrouds which protect the support tubes from normal operating cross-flow loads as well as blowdown loads. The support tubes are latched to the bottom of the CEA shroud and permanently tensioned by means of a threaded spanner nut at the top. Operating loads are far less than the preload developed by the tensioning operation. Therefore, the support tubes will not be affected by thermal or flow loads. The support tubes are designed to account for all tolerance conditions so that proper clearances will be assured. Physically, the support tubes are similar in mass and size to a typical control element assembly drive shaft, which would reside in the same area of the upper guide structure. The presence or absence of the HJTC probes within the support tubes will in no way affect the integrity of the support tubes, the UGS, the pressure boundary, and will have no significant effect upon the hydraulic conditions within the reactor vessel head.

3.3.2 REACTOR INTERNAL STRUCTURES

The reactor internals are designed to support and orient the reactor core fuel assemblies and CEAs, absorb the CEA dynamic loads and transmit these and other loads to the reactor vessel flange, provide flow paths for the reactor coolant, and guide in-core instrumentation.

The internals are designed to safely perform their function during all steady state conditions and during normal operating transients. The internals are designed to safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, vibration and seismic acceleration. All reactor components are considered Class 1 for seismic design. The reactor internals design limits deflection where required by function. In most cases the design of

reactor internals components is limited by stress, not deflection. For the CEA shroud which is the most limiting internal component for deflection, the allowable design deflection limit is 0.5 inch. This limit is two-thirds of the conservatively established loss-of-function deformation limit, 0.75 inch and applies to a break whose equivalent diameter is no larger than the largest line connected to the primary coolant line. The structural components satisfy stress values given in Section III of the ASME Pressure Vessel Code. Certain components have been subjected to a fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation.

The components of the reactor internals are divided into four major parts consisting of the core support barrel, the lower core support structure (including the core shroud), the UGS (including the CEA shrouds, the in-core instrumentation guide tubes and the HJTC support tubes). The flow skirt, although functioning as an integral part of the coolant flow path is separate from the internals and is affixed to the bottom head of the pressure vessel. These components are shown in Figure 3.1–1 and 3.3–11. The in-core instrumentation is described in Section 7.5.4.

Dynamic system analysis methods and procedures which have been used to determine dynamic responses of reactor internals have been provided in CE, Report CENPD-42, “Topical Report of Dynamic Analysis of Reactor Vessel Internals under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class Reactors”.

3.3.2.1 Core Support Assembly

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the lower support structure, and the core shroud. The major materials for the assembly is Type 304 stainless steel.

The core support assembly is supported at its upper end by the upper flange of the core support barrel which rests on a ledge in the reactor vessel flange.

The lower flange of the core support barrel supports and positions the lower support structure. The lower support structure provides support for the core by means of a core support plate supported by columns resting on beam assemblies. The core support plate provides support and orientation for the fuel assemblies. The core shroud which provides lateral support for the fuel assemblies is also supported by the core support plate. The lower end attaches the core barrel to the pressure vessel.

3.3.2.2 Core Support Barrel

The core support barrel is a right circular cylinder with a nominal inside diameter of 148 inches and a minimum wall thickness of 1.75 inch. It is suspended by a 4 inch thick flange from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Press fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head and upper guide structure assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

Since the core support barrel is over 27 feet long and is supported only at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally spaced double lugs around the circumference and are the grooves of a “tongue-and groove” assembly; the pressure vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. During assembly, as the internals are lowered into the vessel, the pressure vessel tongues engage the core support grooves in an axial direction. With this design, the internals may be viewed as a beam with supports at the furthest extremities. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted by this design. The pressure vessel tongues have bolted, lock welded Inconel X shims and the core support barrel grooves are hardfaced with Stellite to minimize wear. The snubber assembly is shown in Figure 3.3–12.

3.3.2.3 Core Support Plate and Support Columns

The core support plate is a 147 inch diameter, 2 inch thick, Type 304 stainless steel plate into which the necessary flow distributor holes for the fuel assemblies have been machined. Fuel assembly locating pins (four for each assembly) are shrunk-fit into this plate. Columns and support beams are located between this plate and the bottom of the core support barrel in order to provide support for this plate and transmit the core load to the bottom flange of the core support barrel.

3.3.2.4 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud (Figure 3.3–13) consists of two Type 304 stainless steel ring sections, aligned by means of radial shear pins and attached to the core support plate by Type 348 stainless steel tie rods. A gap is maintained between the core shroud outer perimeter and the core support barrel in order to provide some coolant flow upward between the core shroud and core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets.

3.3.2.5 Flow Skirt

The Inconel flow skirt is a right circular cylinder, perforated with 2-11/16 inch diameter holes, and reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced machined sections which are welded to the bottom of the pressure vessel.

3.3.2.6 Upper Guide Structure Assembly

This assembly (Figure 3.3–14) consists of the upper support structure, 69 CEA shrouds, a fuel assembly alignment plate and an expansion compensating ring. The UGS assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel

assemblies from being lifted out of position during a severe accident condition and protects the CEAs from the effect of coolant crossflow in the upper plenum. The UGS is handled as one unit during installation and refueling.

The upper end of the assembly is a structure consisting of a support plate welded to a grid array of 24 inch deep beams and a 24 inch deep cylinder which encloses and is welded to the ends of the beams. The periphery of the plate contains four accurately machined and located alignment keyways, equally spaced at 90 degree intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of CEA shrouds.

The CEA shrouds extend from the fuel assembly alignment plate to an elevation about three feet above the UGS support plate. There are 57 single-type shrouds. These consist of cylindrical upper sections welded to integral bottom sections, which are shaped to provide flow passages for the coolant passing through the alignment plate while shrouding the CEAs from cross-flow. There are also 12 dual-type shrouds which in configuration consist of two single-type shrouds connected by a rectangular section shaped to accommodate the dual CEAs. The shrouds are bolted to the fuel assembly alignment plate. At the UGS support plate, the single shrouds are connected to the plate by spanner nuts which permit axial adjustment. The spanner nuts are tightened to proper torque and lockwelded. The dual shrouds are attached to the upper plate by welding.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds.

Precision machined and located holes in the fuel assembly alignment plate align the fuel assemblies. The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hardfaced pins protruding from the core shroud to limit lateral motion of the UGS assembly during operation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the UGS support plate and hence to the expansion compensating ring.

The expansion compensating ring bears on the flange at the top of the assembly to accommodate axial differential thermal expansion between the core barrel flange, UGS flange and pressure vessel flange support edge and head flange recess.

The UGS assembly also supports the in-core instrumentation thimble support frame, guide tubes, and HJTC support tubes.

All integral connections in the reactor internals are designed within the stress intensity limits listed in Tables N-422 and N-416.1 of Section III of the ASME code for normal and upset conditions. For emergency and faulted conditions, the design limits are as given in Table 3.2-1.

3.3.3 CONTROL ELEMENT DRIVE MECHANISM

3.3.3.1 Design

The CEDM is of the magnetic jack type drive. Each CEDM is capable of withdrawing, inserting, holding or tripping the CEA from any point within its 137-inch stroke. The design of the CEDM is shown in Figure 3.3–15 and is identical to that for Maine Yankee (AEC Docket Number 50-309) and Calvert Cliffs Units 1 and 2 (AEC Docket Numbers. 50-317 and 50-318).

The CEDM drives the CEA within the reactor core and indicates the position of the CEA with respect to the core. The speed at which the CEA is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM coils of the shutdown and regulating CEAs are deenergized, allowing the CEA and the supporting CEDM components to drop into the core by gravity. The CEA drop time is 2.75 seconds, where drop time is defined as the interval between the time power is removed from the CEDM coils and the time the CEA has reached 90 percent of its fully inserted position. The reactivity is reduced during such a drop at a rate sufficient to control the core under any operating transient or accident condition. The CEA accelerates to about 11 ft/sec and is decelerated at the end of the drop by the buffer section of the CEA guide tubes. Drive down capability following a reactor trip is not required for safety purposes. The safety analyses of Chapter 14 assume the CEA of highest reactivity worth sticks in the fully withdrawn position. A drive down feature would introduce the possibility of a failure which would prevent power from being removed from the CEDMs during a trip, which would lead to a reduction in plant safety.

There are 69 CEDM nozzles on top of the reactor vessel closure head. Eight of the 69 nozzles were used for the part length CEAs in Cycle 1, six of which are no longer used, and two of which are used for HJTC/RVLMS instrumentation. There are 61 CEDMs in current use. The six spare nozzles are capped with adapters. Each CEDM is connected to a CEA by a locked coupling. The weight of the CEAs and CEDMs is carried by the vessel head.

The CEDM is designed to handle dual, single or part length CEAs. The maximum operating speed capability of the CEDMs is 40 inches per minute for single CEAs and 20 inches per minute for dual CEAs.

3.3.3.2 Control Element Drive Mechanism Pressure Housing

The CEDM housing is attached to the reactor vessel head nozzle by means of a threaded joint and seal welded. The CEDM nozzles are made of Inconel Alloy 690 to minimize Primary Water Stress Corrosion Cracking. The CEDM pressure housings including the magnetic coil jack assemblies were replaced as part of the replacement reactor vessel closure head project.

The CEDM upper housing design and fabrication conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda. The housing is designed for steady state conditions as well as all anticipated pressure and thermal transients.

Once the CEDM housing is seal welded to the head nozzle, it need not be removed since all servicing of the CEDM is performed from the top of the CEDM housing. This opening is closed by means of an upper housing and an omega seal weld. The CEDM pressure housing is capable of being vented after major coolant refills of the reactor coolant system (RCS), such as after a refueling and after reactor coolant pump (RCP) maintenance. However, venting of the CEDM pressure housing is no longer necessary after major refills of the Reactor Coolant System (RCS), since a vacuum refill method is used. The vacuum refill process involves a partial vacuum in the RCS while at mid-loop level and then slowly refilling the RCS.

3.3.3.2.1 Heated Junction Thermocouple Pressure Boundary

The HJTC probe assemblies are located at the two original locations (CEDMs 11 and 13) on the replacement reactor vessel closure head. The HJTC pressure boundary also known as the Reactor Vessel Level Monitoring System (RVLMS) pressure housing assembly consists of upper pressure housing tube, upper flange type Grayloc connection and lower housing. The lower housing is joined to the reactor vessel head nozzle by means of a threaded joint and an omega seal weld. The pressure boundary at the top of the RVLMS pressure housing is maintained by a quick disconnect Grayloc type flange (See Figure 3.3–17). The components are designed to ASME Section III, B&PV Code 1998 Edition through 2000 Addenda.

The pressure and thermal loads associated with normal operation and transient conditions have been included in stress analyses performed in accordance with ASME BPVC criteria. All stresses are within allowable limits.

3.3.3.3 Magnetic Jack Assembly

The magnetic jack motor assembly is an integral unit which fits into the CEDM housing through an opening in the top of the housing. This unit carries the motor tube, lift and hold pawls and magnets. The drive power is supplied by electrical coils positioned around the CEDM housing. The CEDMs are cooled by air supplied at 900 CFM at 95°F (maximum) to each CEDM. The design of the control element drive mechanism is such that loss of cooling air will not prevent the CEDM from releasing the CEA. The ability of the CEDM to release the rods is not dependent on the cooling flow provided by the CEDM Cooling System. Cooling function is only to ensure reliability of the CEDM coil stack. Following insertion of the CEDM motor assembly, the upper pressure housing is threaded into the CEDM motor housing and seal welded. This upper pressure housing encloses the CEDM extension shaft and supports the shroud assembly. The reed switch assembly is supported by the shroud assembly.

The lifting operation consists of magnetically operated step movements. Two sets of mechanical latches (one holding, one lifting) are utilized engaging a notched drive shaft. To prevent excessive latch wear, a means has been provided to unload the lifting latches during the engaging and disengaging operations.

The magnetic force is obtained from large DC magnet coils mounted on the outside of the motor tube.

Power for the electromagnets is obtained from one of two separate supplies. A control programmer actuates the stepping cycle and obtains the CEA location by a forward or reverse stepping sequence. CEDM hold for shutdown and regulating CEAs is obtained by energizing a hold coil at a reduced current while all other coils are deenergized. The full length CEAs are tripped upon interruption of electrical power to all coils.

3.3.3.4 Position Indication

Three separate means are provided for transmitting CEA position indication.

The first method utilizes the electrical pulses from the magnetic coil power programmer. The second method utilizes reed switches and a voltage divider network mounted on the CEDM to provide an output voltage proportional to CEA position. The third method utilizes three pairs of reed switches spaced at discrete locations within a position transmitter assembly. A permanent magnet built into the drive shaft actuates the reed switches one at a time as it passes by them. CEA position instrumentation is discussed in detail in Section 7.5.3.

3.3.3.5 Control Element Assembly Disconnect

The CEA is connected to the drive shaft extension with an internal collet-type coupling at its lower end. (Coupling is performed before the vessel head is installed). In order to disengage the CEA from the drive shaft extension, a tool is attached to the top end of the drive shaft when the reactor vessel head has been removed.

By pulling up on the spring-loaded operating rod in the center of the drive shaft, a tapered plunger is withdrawn from the center of the collet-type gripper causing it to collapse due to axial pressure from the CEA, thus permitting removal of the coupler from the CEA. Releasing the operating rod plunger after the coupler has been withdrawn from the CEA expands the coupler to a diameter that prevents recoupling to the CEA.

3.3.3.6 Test Program

A test program has been conducted to verify the adequacy of the magnetic jack CEDM. The program is described in Section 1.5.4.

3.3.4 REFERENCES

3.3-1 ASME Boiler and Pressure Vessel Code, Section III, 1977 Edition, ASME New York, NY.

3.3-2 K. R. Merckx, “RODEX2 - Fuel Rod Thermal-Mechanical Response Evaluation Model,” XN-NF-81-58 (NP)(A), Revision 2, March 1985 and Supplements.

3.3-3 “Qualification of Exxon Nuclear Fuel for Extended Burnup (PWR),” XN-NF-82-06 (NP)(A), Revision 1, Supplements 2, 4, 5, October 1985.

3.3-4 W. J. O'Donnell and B. F. Langer, “Fatigue Design Basis for Zircaloy Components,” Nuclear Science and Engineering, Volume 20, January 1964.

3.3-5 MATPRO Version, “A Handbook of Material Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior,” TREE-NUREG 1008, December 1976.

3.3-6 J. C. Winslow (CE) to T. J. Honan (NU), CE Letter, “Seismic and Branch Line LOCA Analysis of SPC Reload Fuel for Millstone 2,” NU-88-043 (March 31, 1988).

3.3-7 “PWR Primary Water Chemistry Guidelines,” Revision 2, Electric Power Research Institute (EPRI) Final Report, EPRI NP7077, dated November 1990.

3.3-8 ANF-88-88(P), Rev. 1, “Design Report for Millstone Point Unit 2 Reload ANF-1,” August 29, 1988.

3.3-9 AREVA Contract Requirements Document Number 89-9070921-001-AREVA Contract No. J37MIL219B, January 28, 2008.

3.3-10 AREVA Document 51-9074000-000, “Compliance Document - Replacement Fuel Rod - Millstone 2 Fuel Failure Mitigation,” March 5, 2008.

3.3-11 BAW-10240(P)-A, Revision 0, “Incorporation of M5™ Properties in Framatome ANP Approved Methods,” Framatome ANP, Inc., May 2004.

3.3-12 EMF-92-116(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs,” AREVA Inc., February 2015.

3.3-13 ANF-88-133(P)(A) and Supplement 1, “Qualification of Advanced Nuclear Fuels' PWR Design Methodolody for Rod Burnups of 62 GWd/MTU”, Advanced Nuclear Fuels Corporation, December 1991.

TABLE 3.3-1 MECHANICAL DESIGN PARAMETERS *

Fuel Assembly	
Geometry	14 x 14
Assembly Pitch, inches	8.180
Assembly Envelope, inches	8.160
Rod Pitch, inches	0.580
Number of Grids per Assembly	9
Approximate Assembly Weight, lb.	1280 (batches N and P) 1313 (batches R through DD) 1337 (batches EE and subsequent)
Fuel Rod to Fuel Rod Outside Dimension, inches	7.980
Fuel Rod and Pellet	
Active Stack Length, Cold, inches	136.7
Pellet Diameter, inches	0.3700 (batches N and P) 0.3770 (batches R through DD) 0.3805 (batches EE and subsequent)
Pellet Length, inches	0.425 (batches N and P) 0.435 (batches R through DD) 0.476 (batches EE and subsequent)
Pellet Density (% Theoretical)	94.0 (batches N and P) 95.0 (batches R and S) 95.35 (batches T through DD) 96.00 (batches EE and subsequent)
Clad Material	Zircaloy-4 (batches N through DD) M5™ (batches EE and subsequent)
Clad OD, inches	0.440
Clad Thickness, inches	0.031 (batches N and P) 0.028 (batches R through DD) 0.0265 (batches EE and subsequent)
Diametrical Gap, nominal, cold, inches	0.008 (batches N and P) 0.007 (batches R through DD) 0.0065 (batches EE and subsequent)

TABLE 3.3-1 MECHANICAL DESIGN PARAMETERS * (CONTINUED)

Control Rod Guide Tube	
Number per Assembly	4
Tube ID, above dashpot, inches	1.035
Wall Thickness, above dashpot, inches	0.040
Wall Thickness, dashpot, inches	0.040 (batches N through D) 0.0735 (batches EE and subsequent)
Instrumentation Tube	
Number per Assembly	1
Tube ID, inches	1.035
Wall Thickness, inches	0.040
Spacer Grid	
Material	Zircaloy-4 / Nickel Alloy 718 (bottom grid)
Number per Assembly	8 / 1 (batches N and P) 9 / 0 (batches R through X) 9 / 1 (batches Y and subsequent)
Sleeves (Wear)	
Material	SS, Chrome Plate
Burnable Poison Rod	
Active Length, inches	124.7 + UO_2 blankets
Pellet Material	Gd_2O_3 / UO_2
Pellet Diameter, inches	0.3700 (batches N and P) 0.3770 (batches R through DD) 0.3805 (batches EE and subsequent)
Pellet Length, inches	0.435 (batches R through DD) 0.476 (batches EE and subsequent)
Clad Material	Zircaloy-4 (batches N through DD) M5™ (batches EE and subsequent)
Clad ID, inches	0.378 (batches N and P) 0.384 (batches R through DD) 0.387 (batches EE and subsequent)
Clad OD, inches	0.440

TABLE 3.3-1 MECHANICAL DESIGN PARAMETERS * (CONTINUED)

Clad Thickness, nominal, inches	0.031 (batches N and P) 0.028 (batches R through DD) 0.0265 (batches EE and subsequent)
Diametral Gap, nominal, cold, inches	0.008 (batches N and P) 0.007 (batches R through DD) 0.0065 (batches EE and subsequent)
Control Element Assembly	
Number	73
Number of Absorber Elements per Assembly	5
Type	Cylindrical Rods
Clad Material	Nickel Alloy 625
Clad Thickness, inches	0.036
Clad OD, inches	0.948
Poison Material	B_4C / Ag-In-Cd
Corner Element Pitch, inches	4.64
Total CEA Length, inches	161.31- CE 161.25 - AREVA
Poison Length, inches	132 -CE 133.5 - AREVA
CEA Dry Weight, lb.	95 - CE 85 - AREVA
Total Operating Assembly Dry Weight, lb.	
Single	210 - CE 200 - AREVA
Dual	334 - CE 314 AREVA
Core Arrangement	
Number of Fuel Assemblies in Core Total	217
Number of Single CEAs	49
Number of Dual CEAs	12
CEA Pitch, minimum, inches	11.57

TABLE 3.3-1 MECHANICAL DESIGN PARAMETERS * (CONTINUED)

Spacing Between Fuel Assemblies, Fuel Rod Surface to Surface, inches	0.200
Spacing, Outer Fuel Rod Surface to Core Shroud, inches	0.18
Hydraulic Diameter, Nominal Channel, feet	0.04445
Total Flow Area (Excluding Guide Tubes), square feet	53.5
Total Core Area, square feet	101.1
Core Equivalent Diameter, inches	136
Core Circumscribed Diameter, inches	143.1
Core Volume, liters	32,526
Total Fuel Loading, MTU (Typical)	83.65
Total Heat Transfer Area, square feet	50,117

TABLE 3.3-2 PRESSURIZED WATER REACTOR PRIMARY COOLANT WATER CHEMISTRY RECOMMENDED SPECIFICATIONS

Conductivity (μS/cm at 25°C)	Relative to Lithium and Boron concentration.
pH at 25°C	Determined by the concentration of boric acid and lithium present. Consistent with the Primary Chemistry Control Program.(4)
Dissolved Oxygen, at power	< 0.1 ppm (1) (2) (3)
Chloride	< 0.15 ppm
Fluoride	< 0.10 ppm
Hydrogen	25-50 cc (STP)/KgH_2O
Suspended Solids	≤ 0.35 ppm prior to reactor startup
Li	Consistent with the Primary Chemistry Control Program.(4)
Boron, as boric acid	0-2620 ppm (5)

NOTES:

(1) The temperature at which the Oxygen limit applies is > 250°F.

(2) The at power operation residual Oxygen concentration control value is ≤ 0.005 ppm.

(3) During plant startup, Hydrazine may be used to control dissolved Oxygen concentration at ≤ 0.1 ppm.

(4) During power operation lithium is coordinated with boron to maintain a $pH_{(t)}$ of ≥ 7.0, but ≤ 7.4, consistent with the Primary Chemistry Control Program. Lithium is added to the RCS during plant startup, but prior to reactor criticality, and is in specification per the Primary Chemistry Control Program within 72 hours after criticality. Lithium may be removed from the reactor coolant immediately before, or during, shutdown periods to aid in the cleanup of corrosion products. By evaluation, a maximum lithium concentration of 4.5 ppm is permissible with a target lithium concentration of 4.3 ppm for 100% power operations.

(5) RCS boron concentration is maintained as necessary to ensure core reactivity or shutdown margin requirements are met. Although the RCS and related auxiliary systems containing reactor coolant are designed for a maximum concentration of 2620 ppm boron, it should be noted the design basis for the TSP baskets in the containment sump assumes the RCS, SITs, and RWST are at a maximum boron concentration of 2400 ppm.

FIGURE 3.3–1A FUEL ROD ASSEMBLY (BATCH "DD" AND PRIOR)

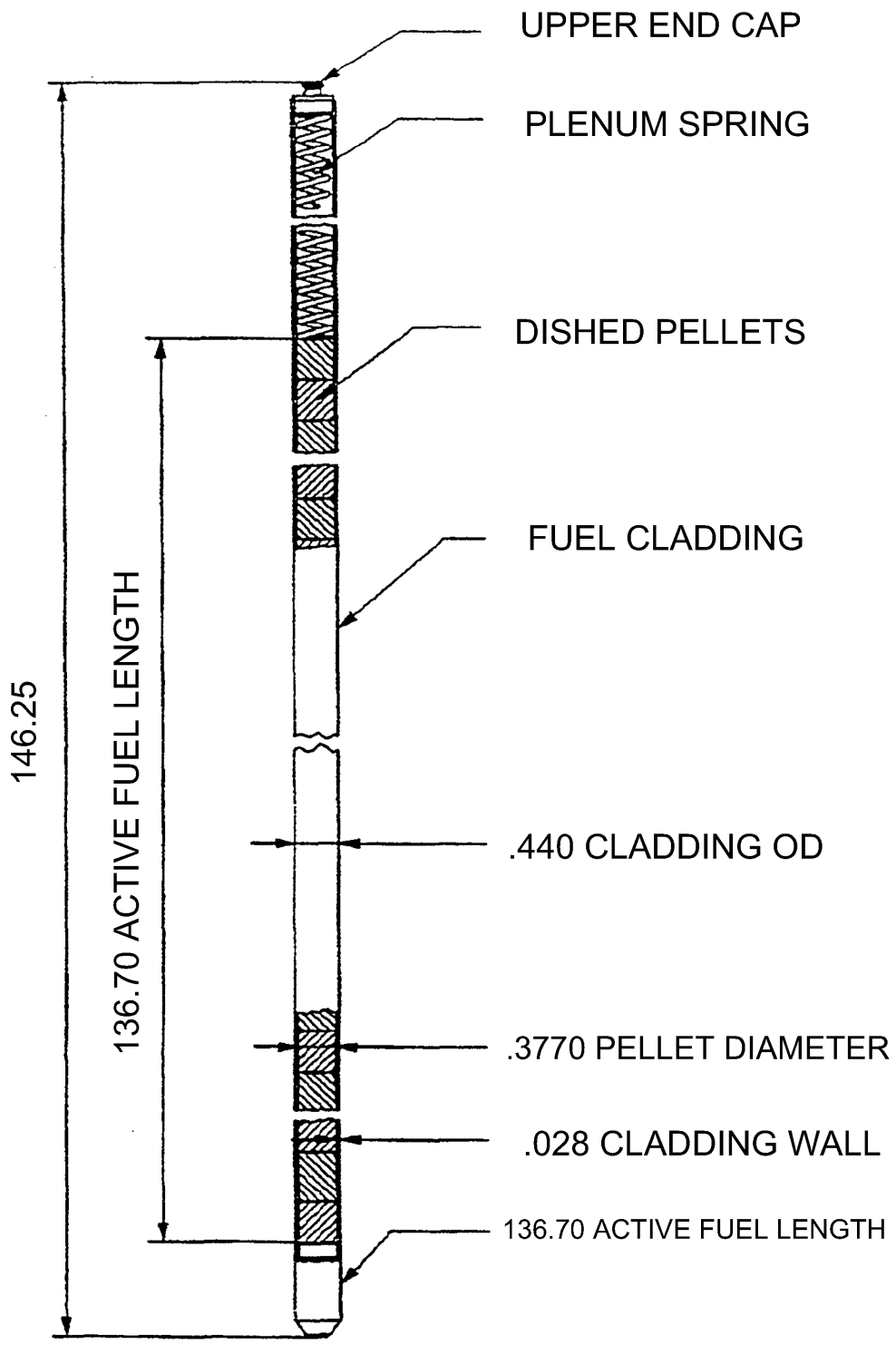
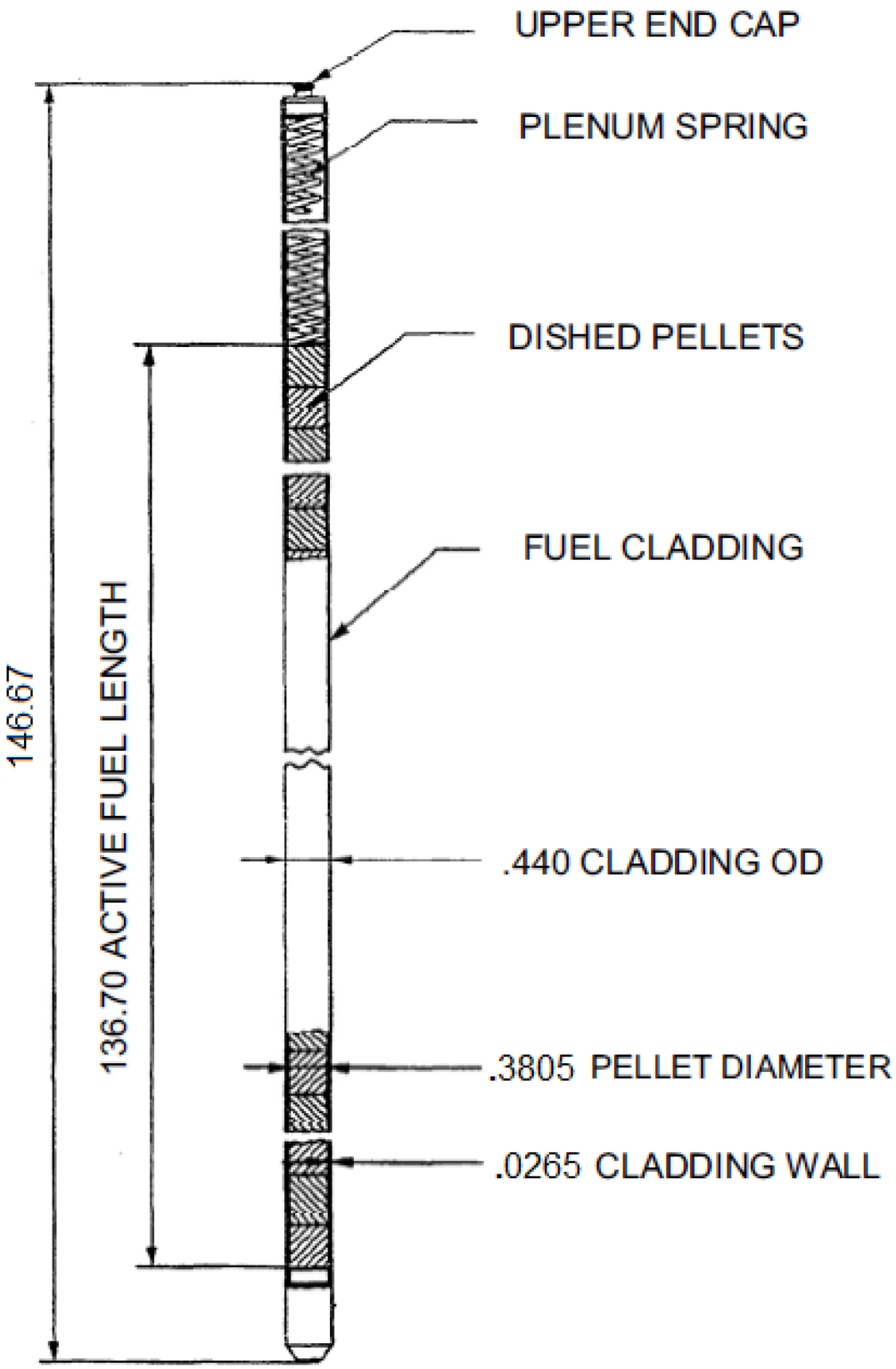


FIGURE 3.3–1B FUEL ROD ASSEMBLY (BATCH "EE" AND LATER)



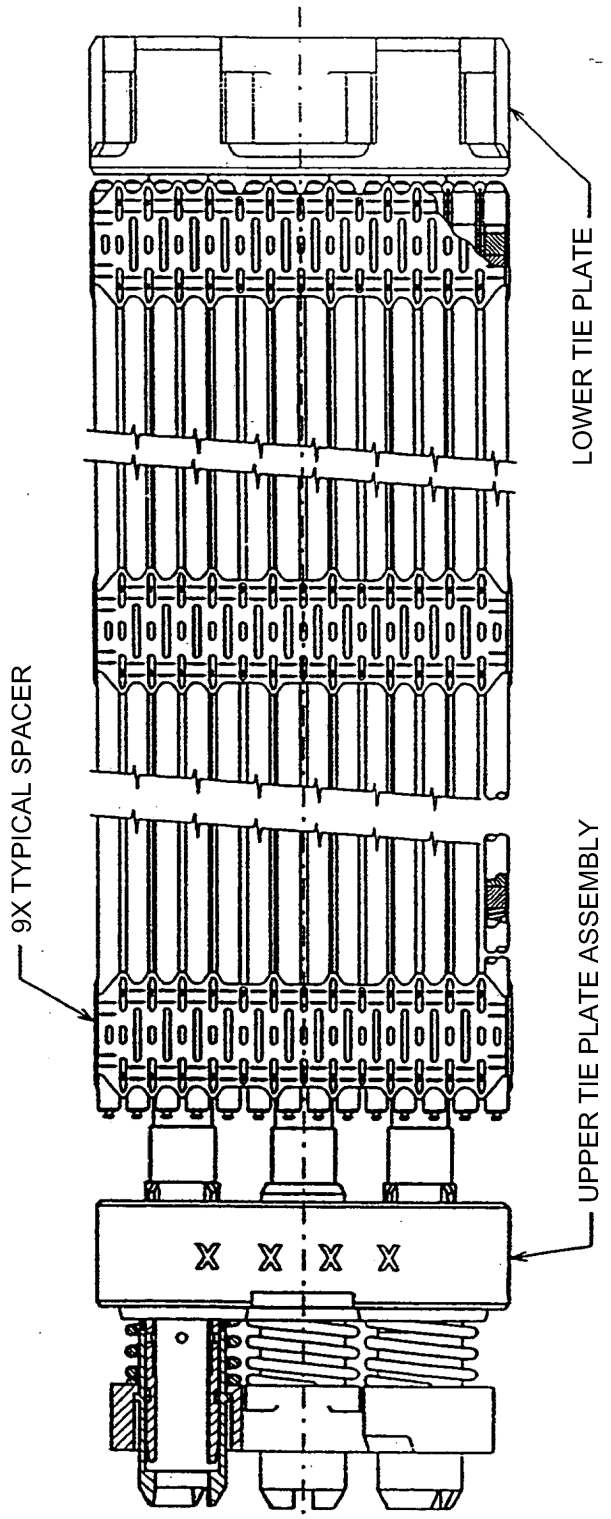
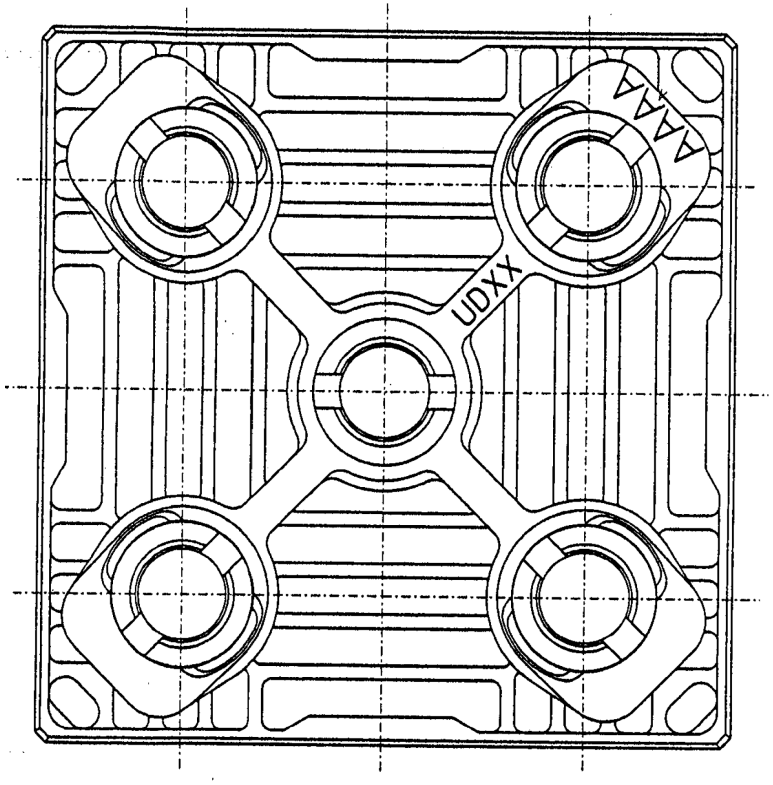
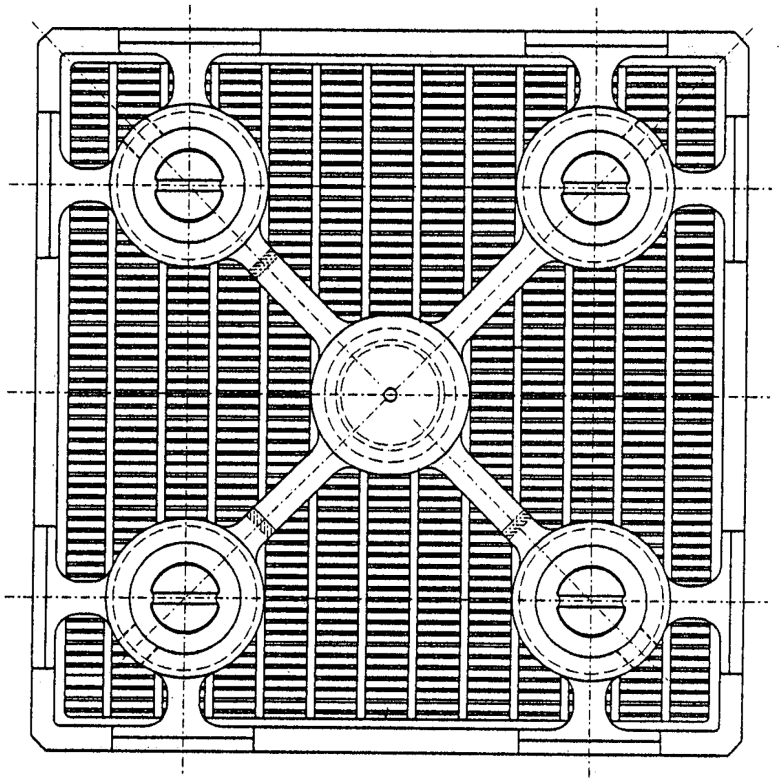


FIGURE 3.3–2A AREVA - RELOAD FUEL ASSEMBLY BATCH "S" AND PRIOR

FIGURE 3.3–2B AREVA - RELOAD FUEL ASSEMBLY BATCH "T" AND LATER



UPPER TIE PLATE



LOWER TIE PLATE

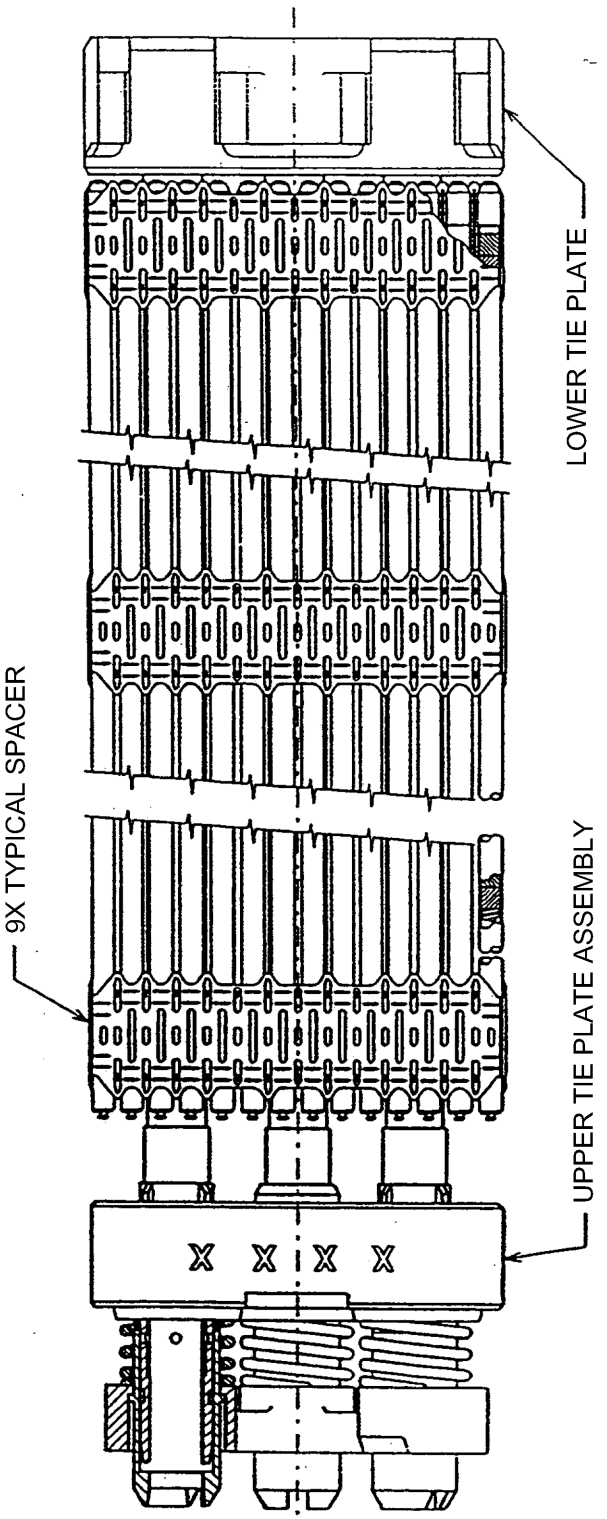


FIGURE 3.3–3A AREVA - RELOAD FUEL ASSEMBLY COMPONENTS BATCH "S" AND PRIOR

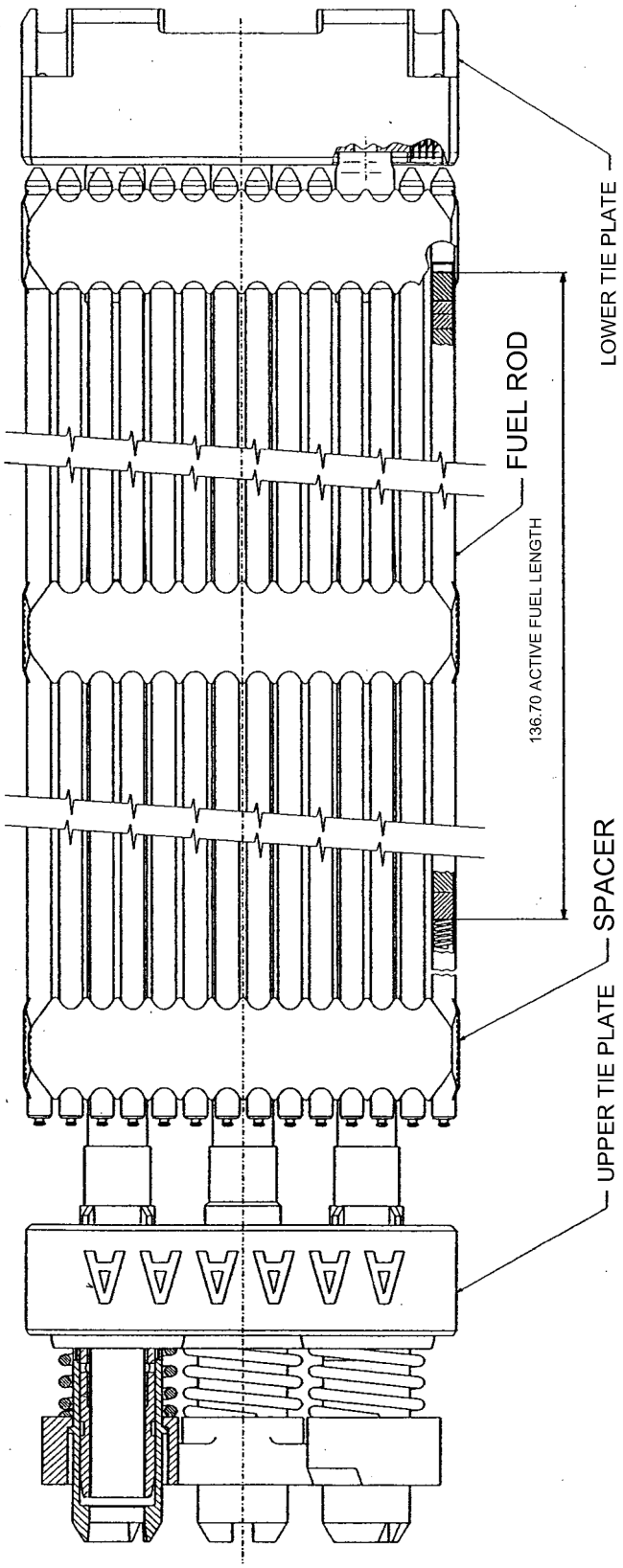
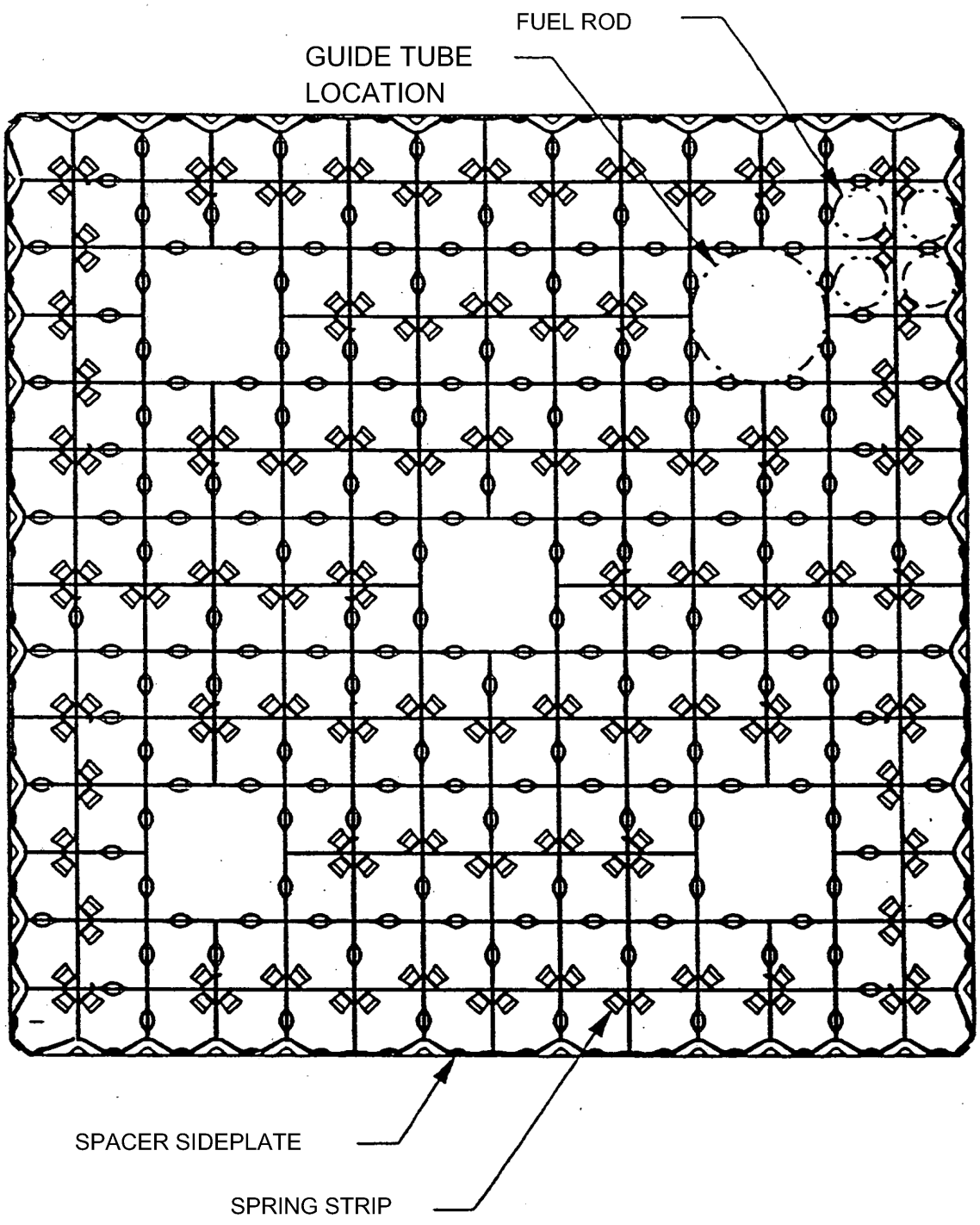


FIGURE 3.3-3B AREVA - RELOAD FUEL ASSEMBLY COMPONENTS BATCH "T" AND LATER

FIGURE 3.3–4A BI-METALLIC FUEL SPACER ASSEMBLY



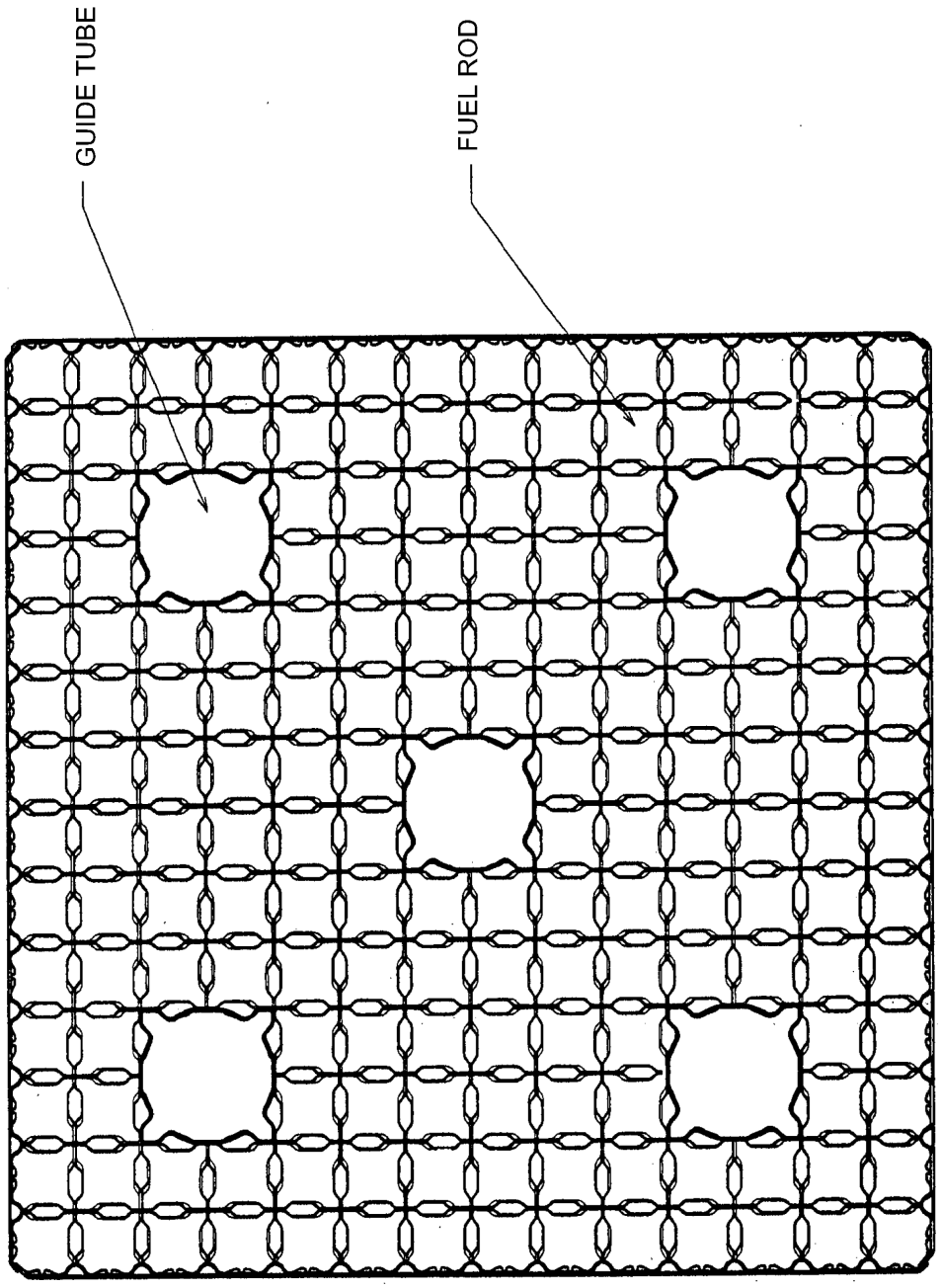


FIGURE 3.3–4B HTP FUEL SPACE ASSEMBLY

FIGURE 3.3–5 FUEL ASSEMBLY HOLD DOWN DEVICE

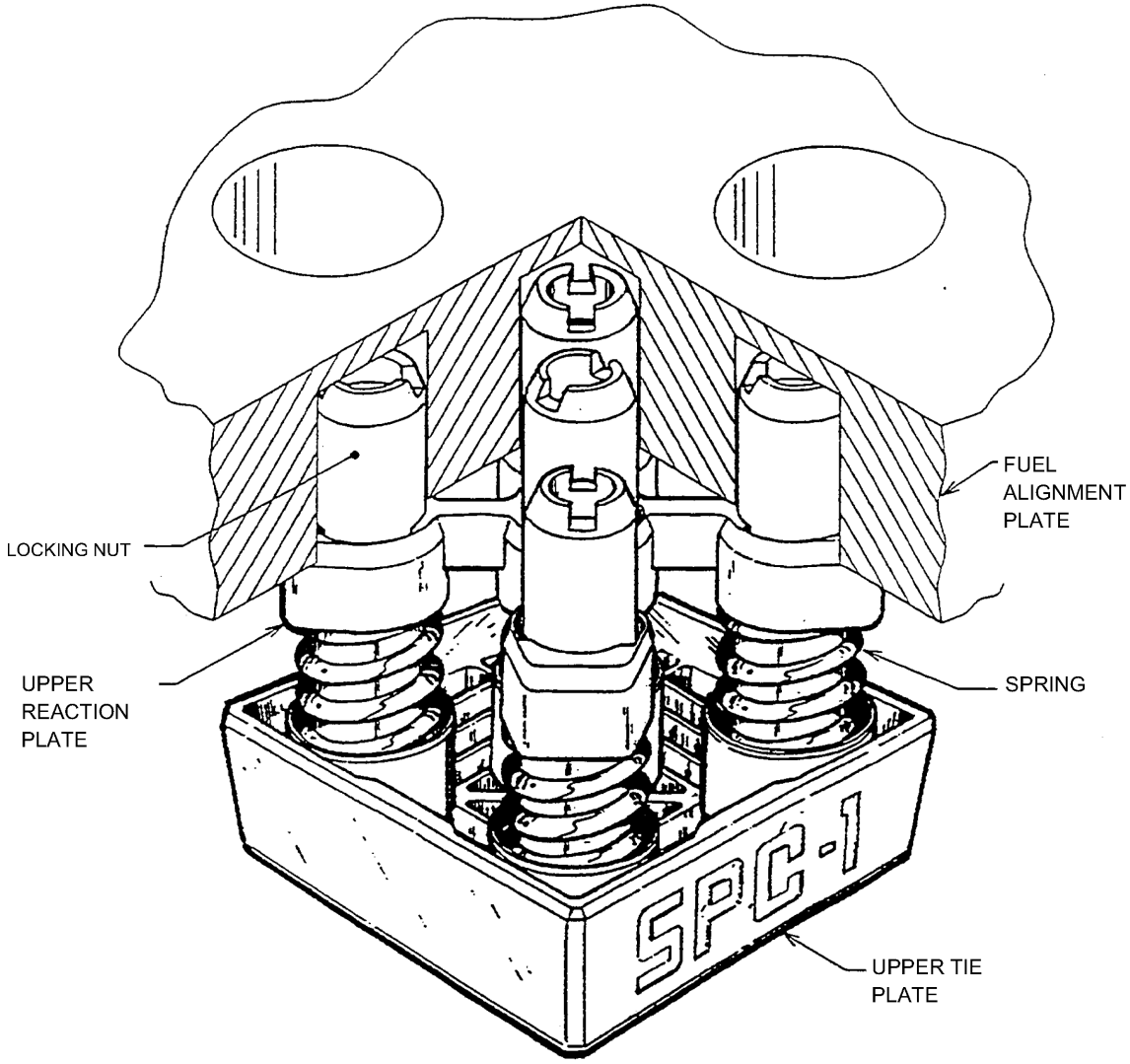


FIGURE 3.3–6 CONTROL ELEMENT ASSEMBLY

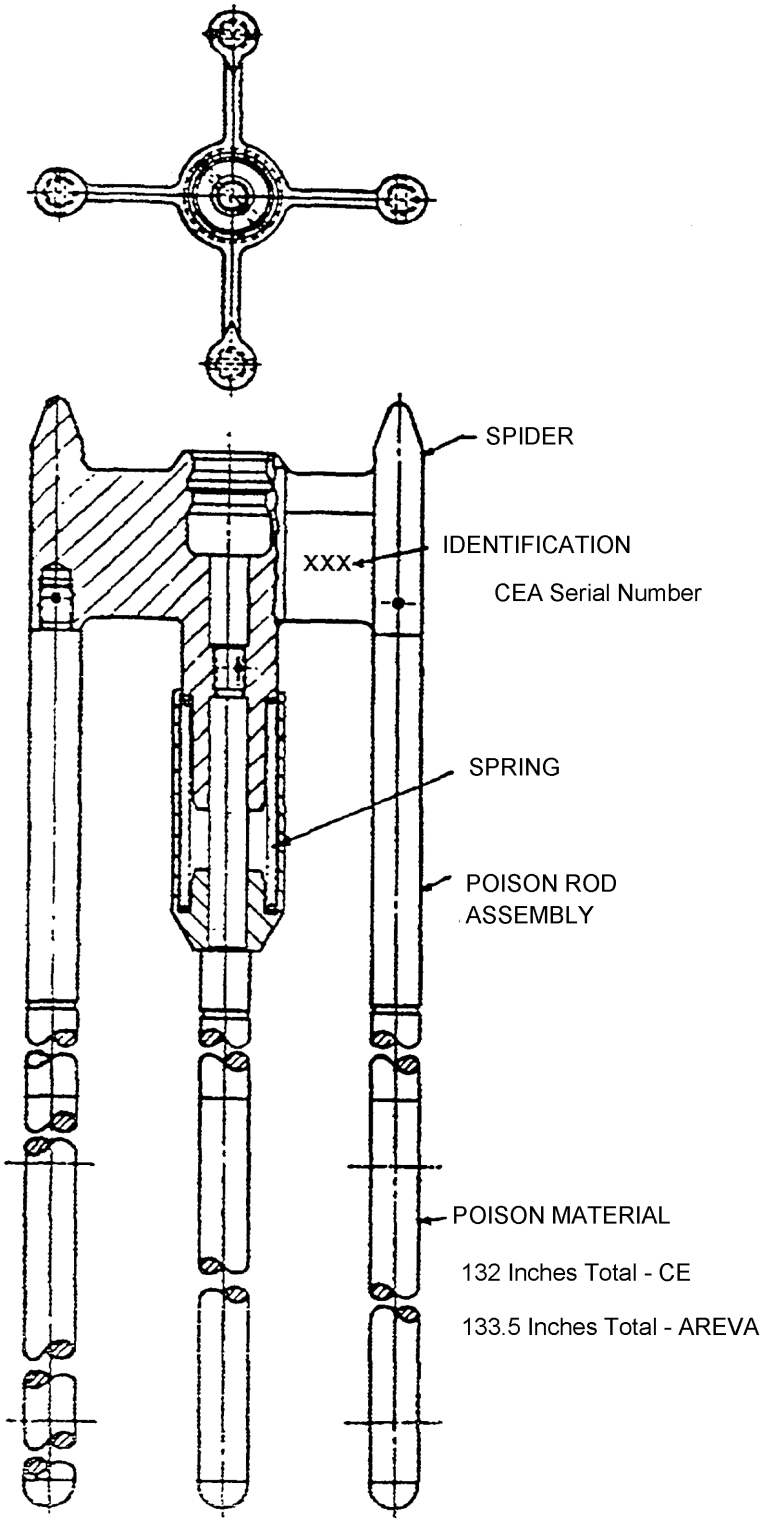
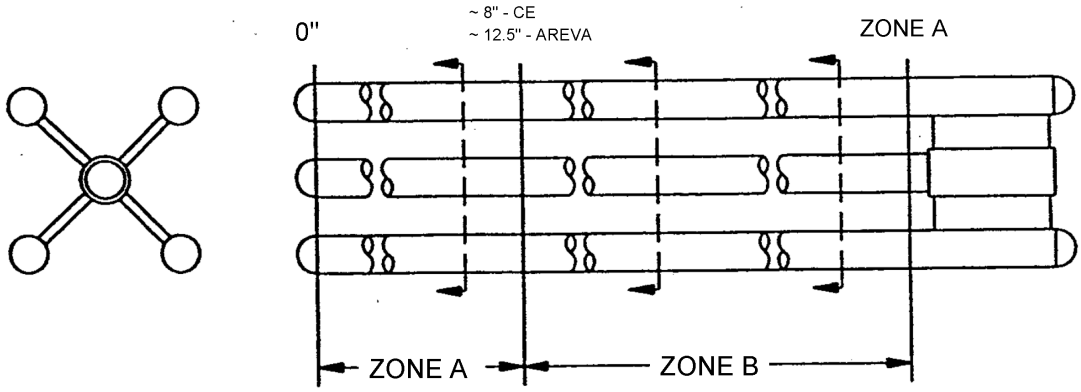
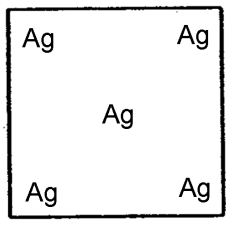
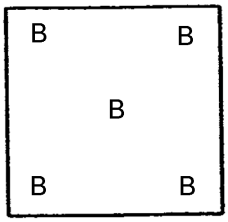


FIGURE 3.3–7 CONTROL ELEMENT ASSEMBLY MATERIALS



NUMBER	MATERIALS ZONE A	ZONE B
73		

NOTES:

$B = B_4C$ $Ag = Ag - In - Cd$

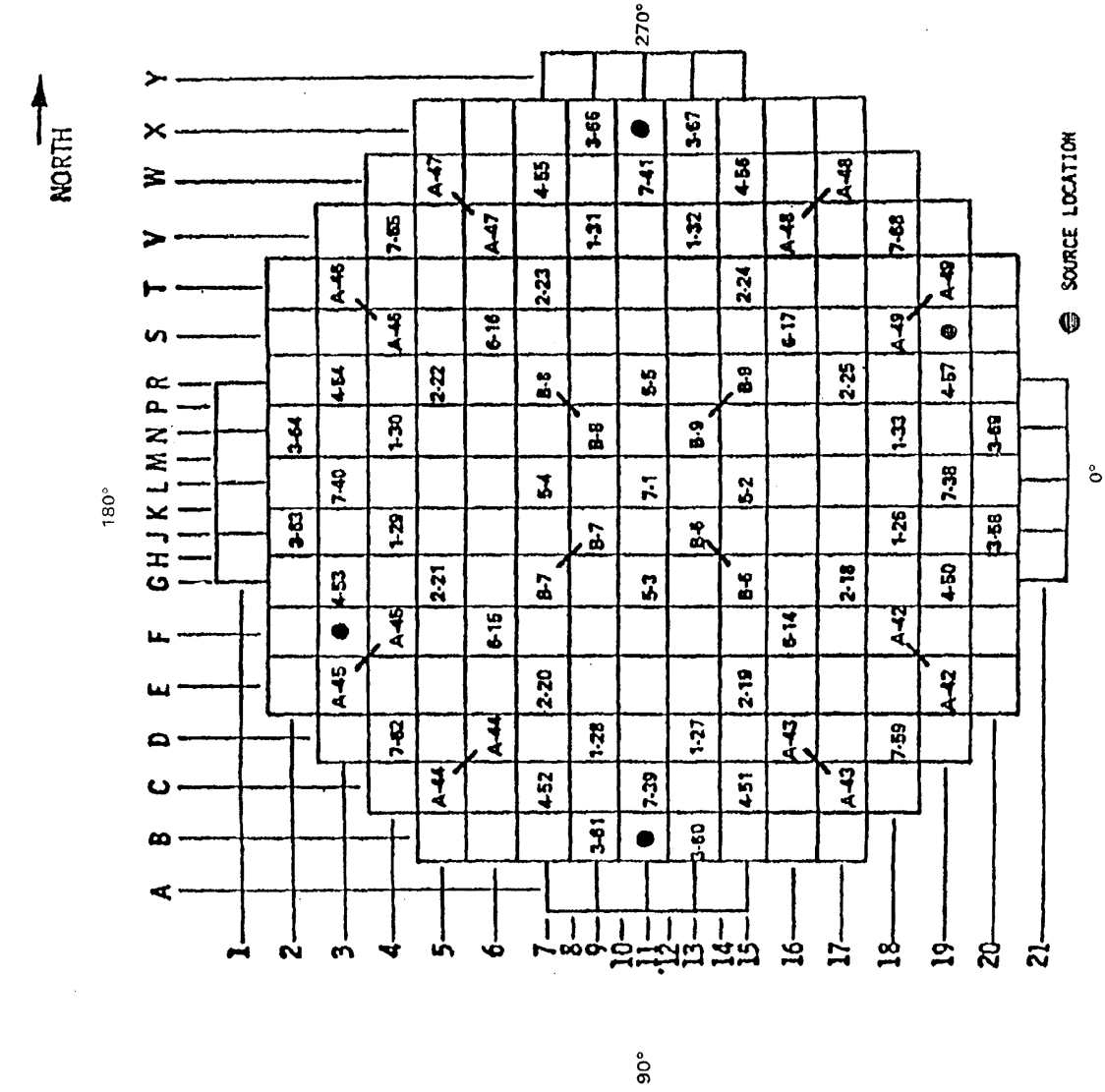


FIGURE 3.3–8 CONTROL ELEMENT ASSEMBLIES GROUP AND NUMBER DESIGNATION

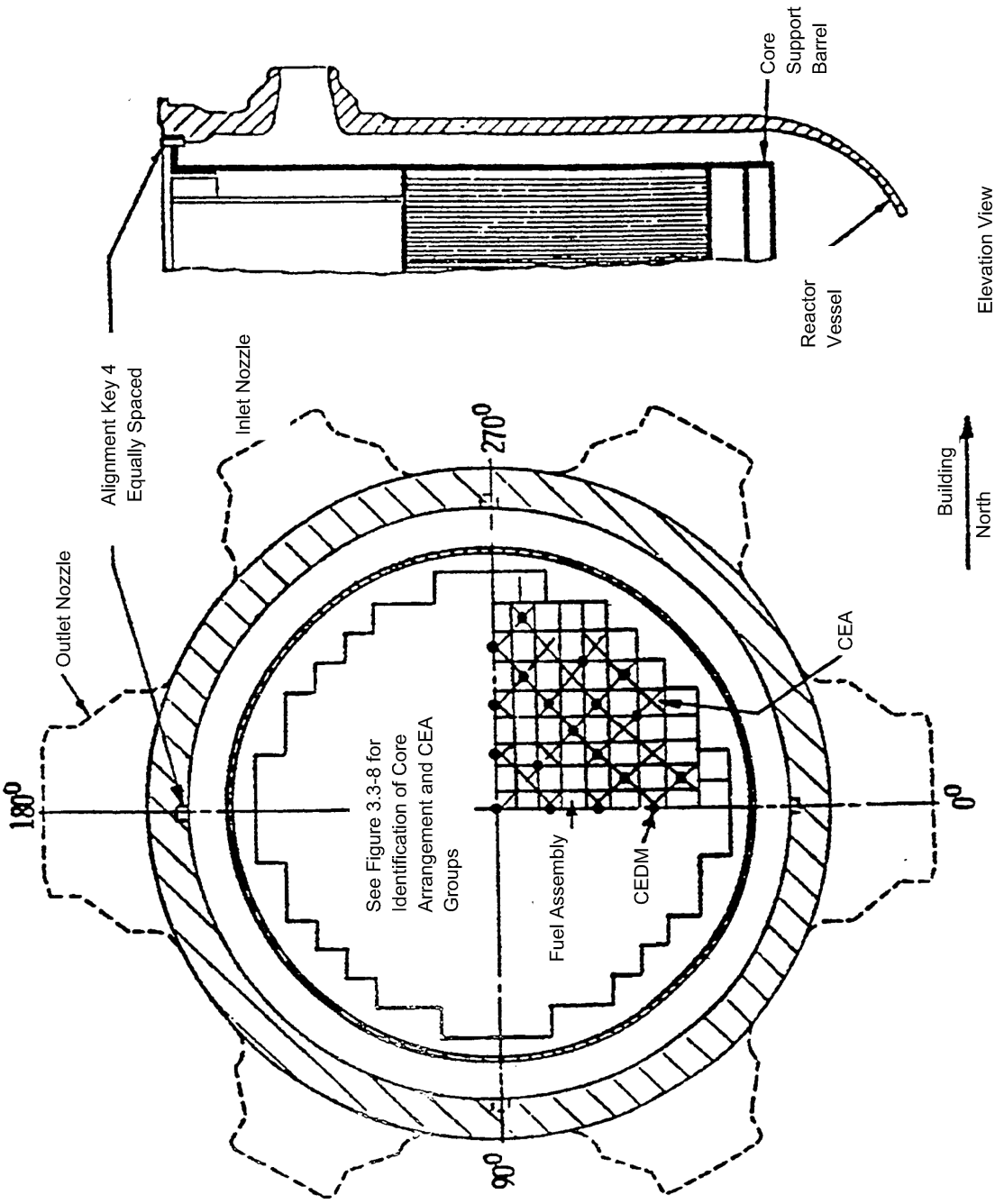


FIGURE 3.3–9 CORE ORIENTATION

FIGURE 3.3–10 IN-CORE INSTRUMENTATION ASSEMBLY

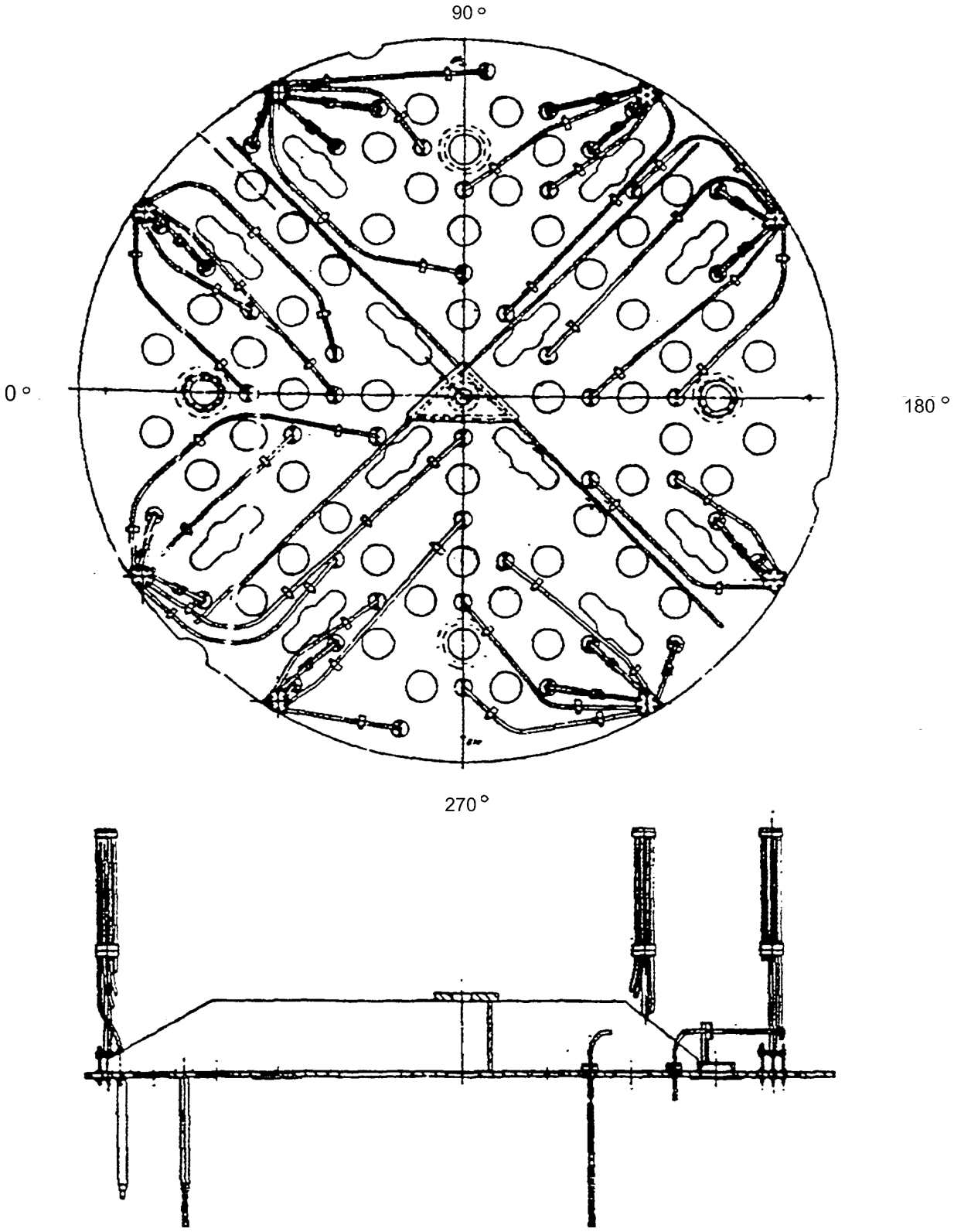
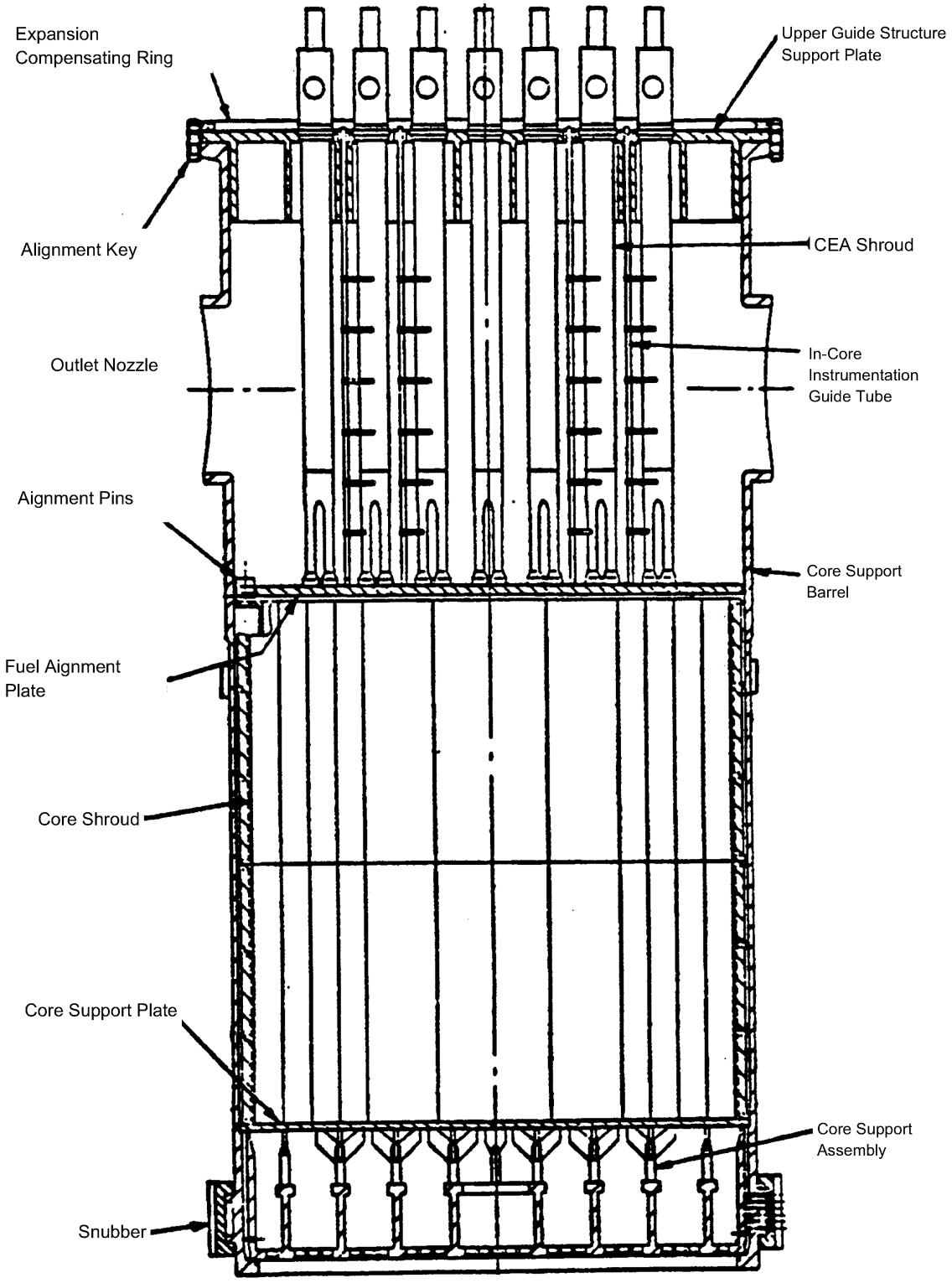


FIGURE 3.3–11 REACTOR INTERNALS ASSEMBLY



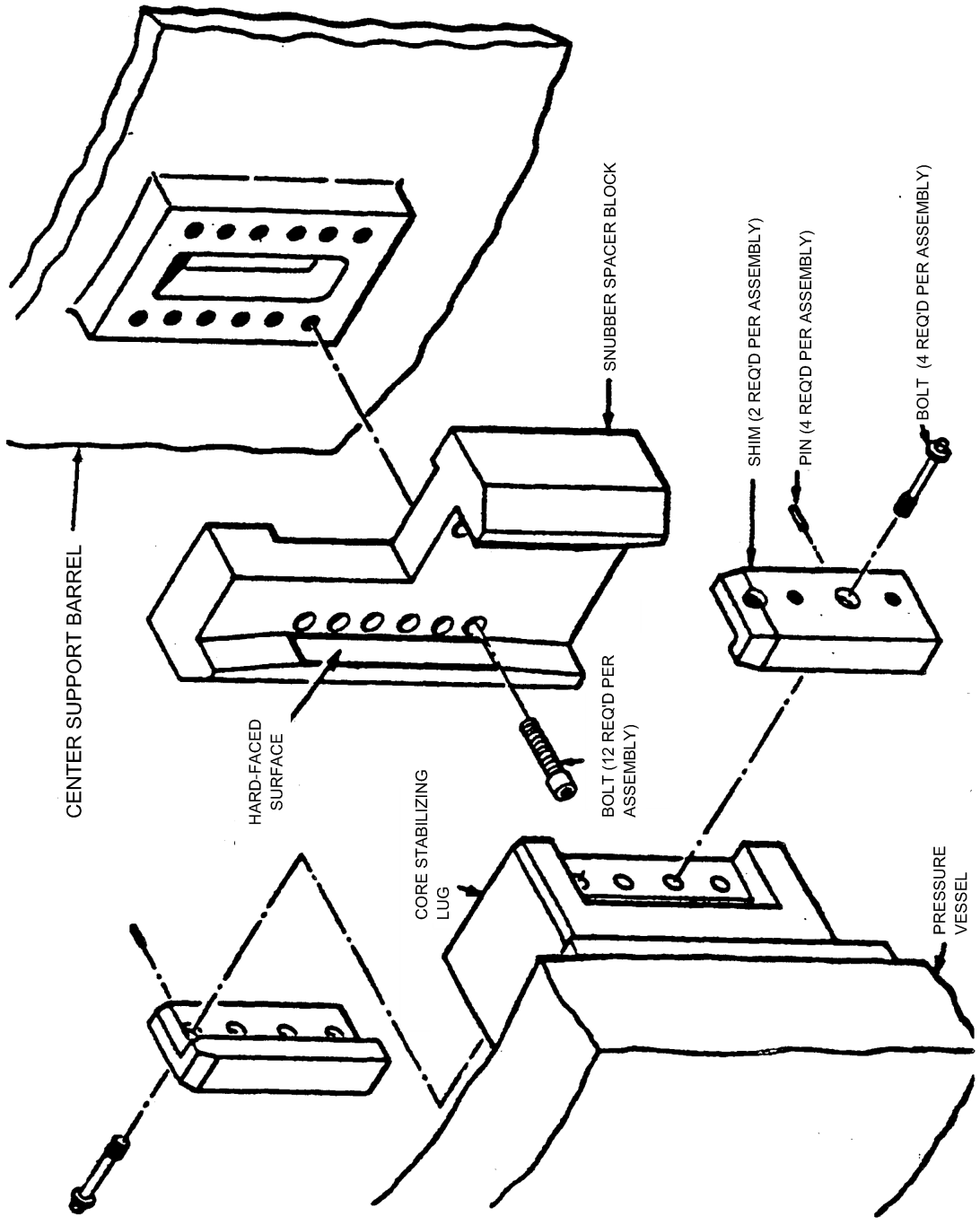


FIGURE 3.3–12 PRESSURE VESSEL-CORE SUPPORT BARREL SNUBBER ASSEMBLY

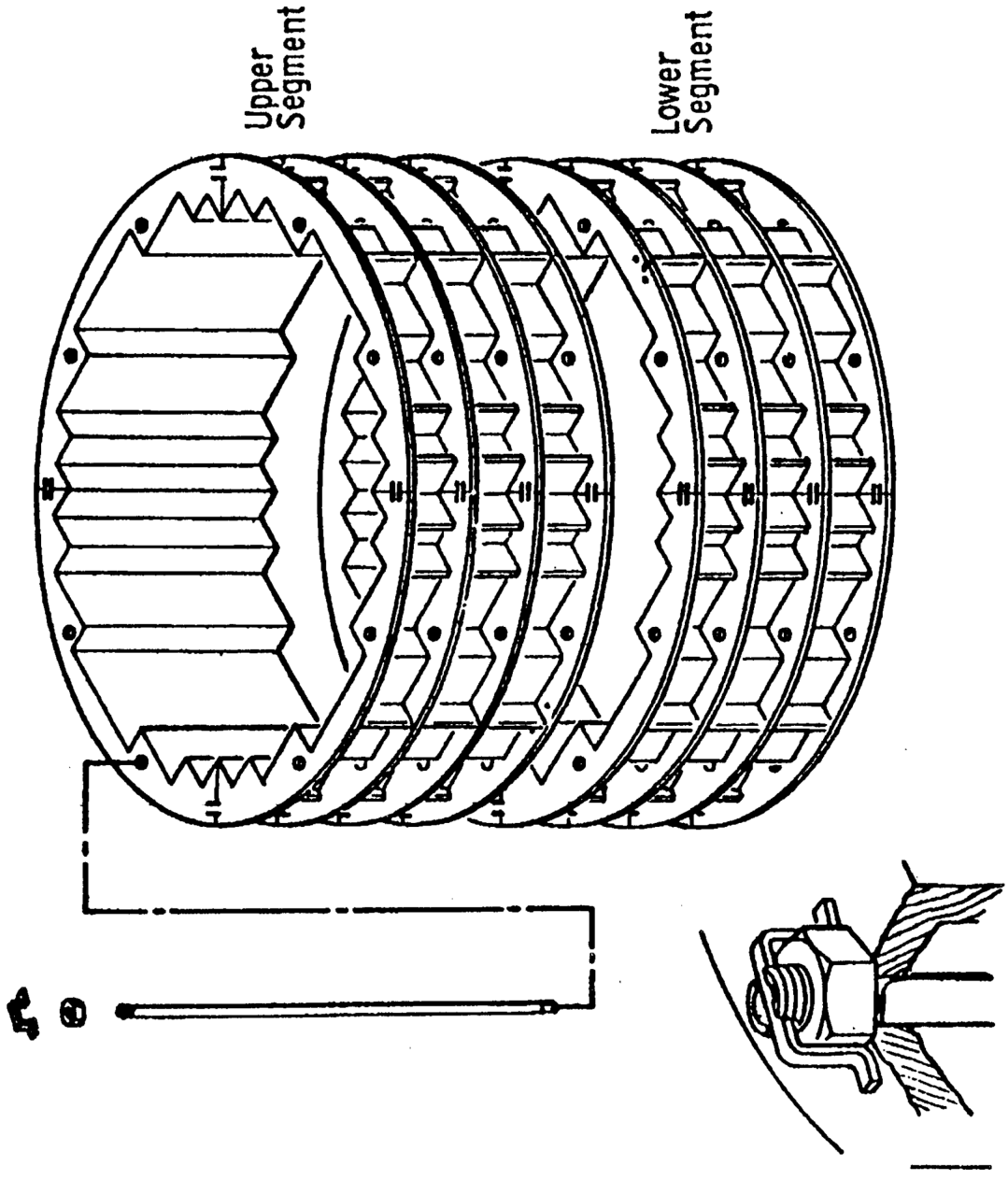


FIGURE 3.3–13 CORE SHROUD ASSEMBLY

FIGURE 3.3–14 UPPER GUIDE STRUCTURE ASSEMBLY

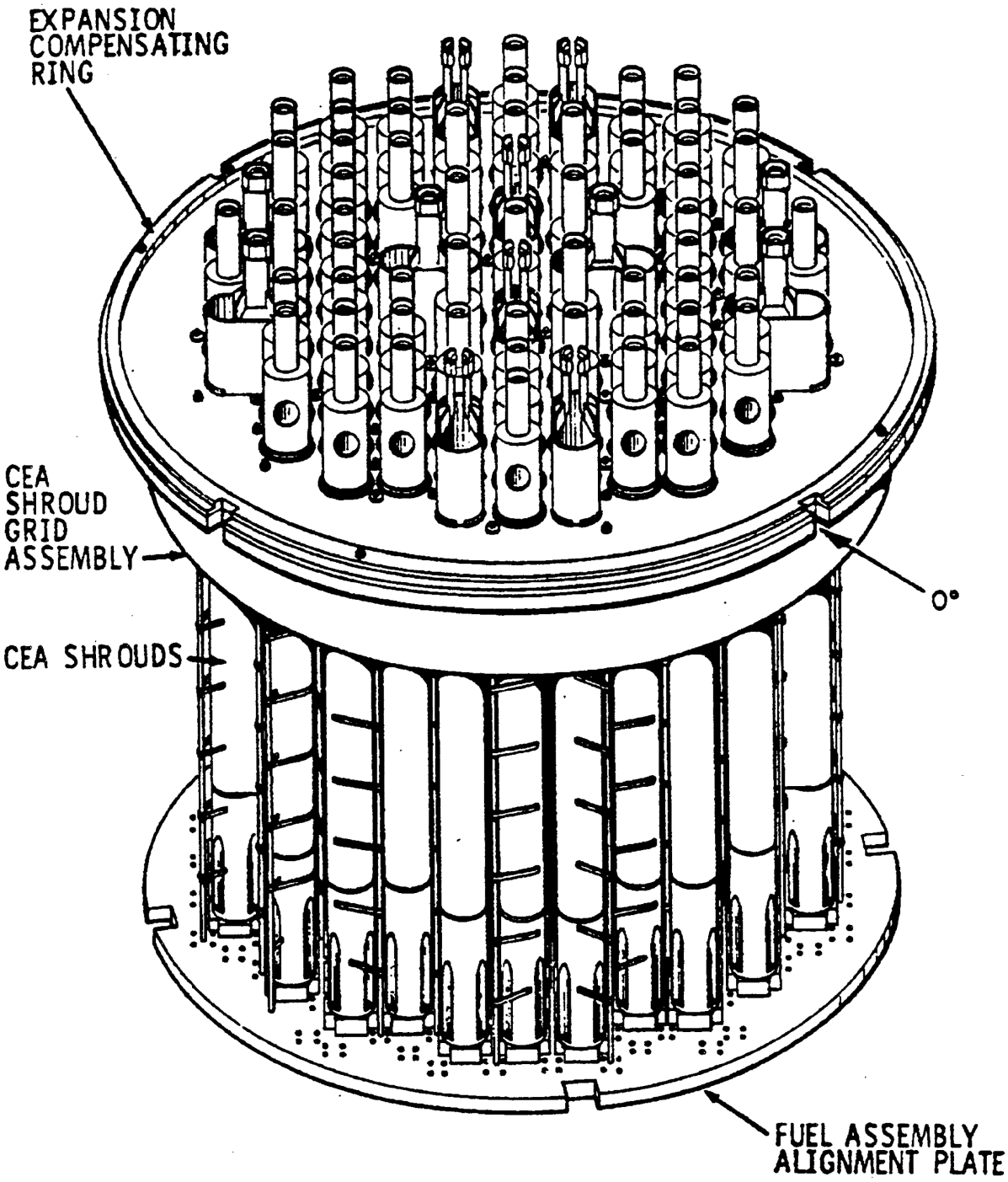


FIGURE 3.3–15 CONTROL ELEMENT DRIVE MECHANISM (MAGNETIC JACK)

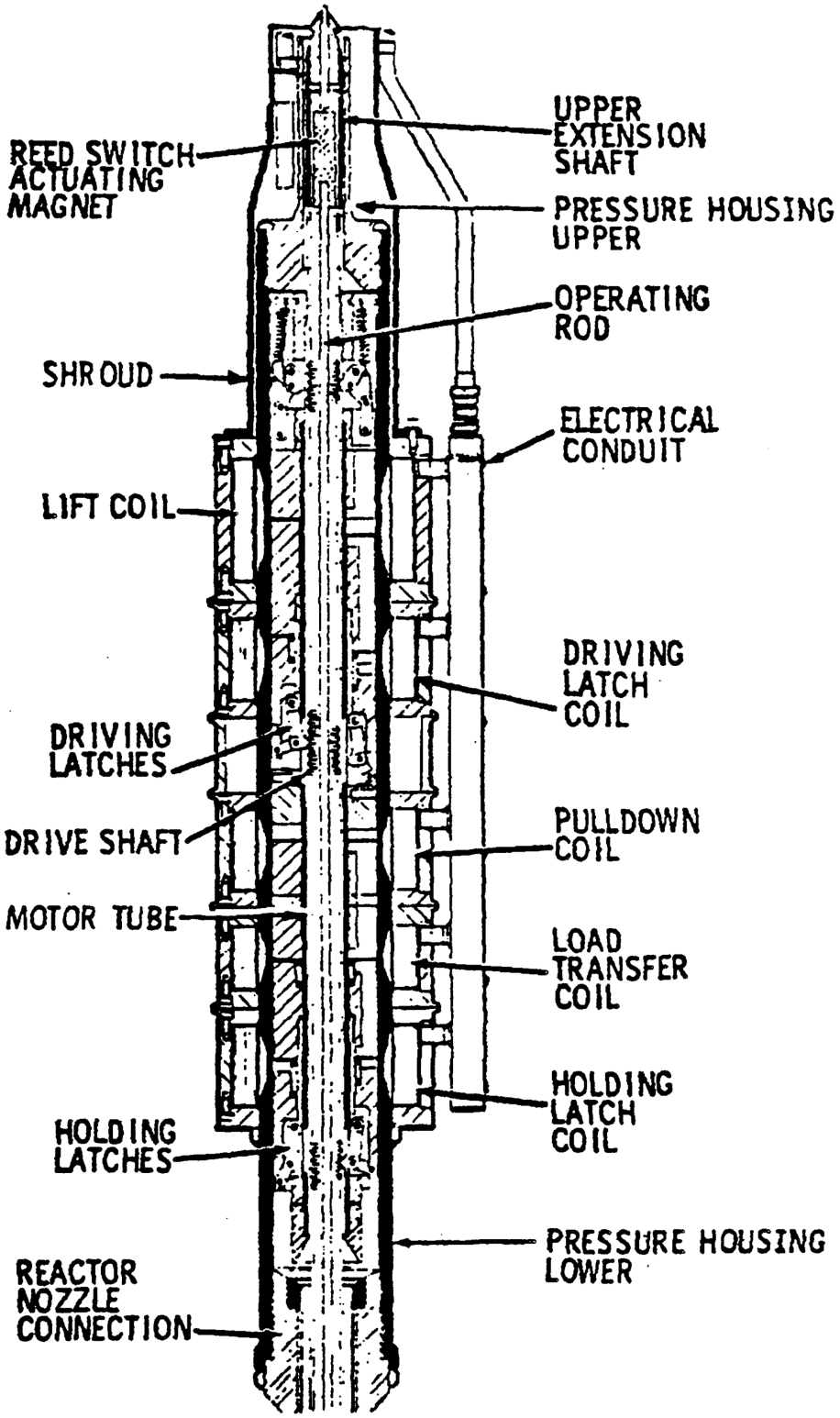


FIGURE 3.3–16 (LEFT BLANK INTENTIONALLY)

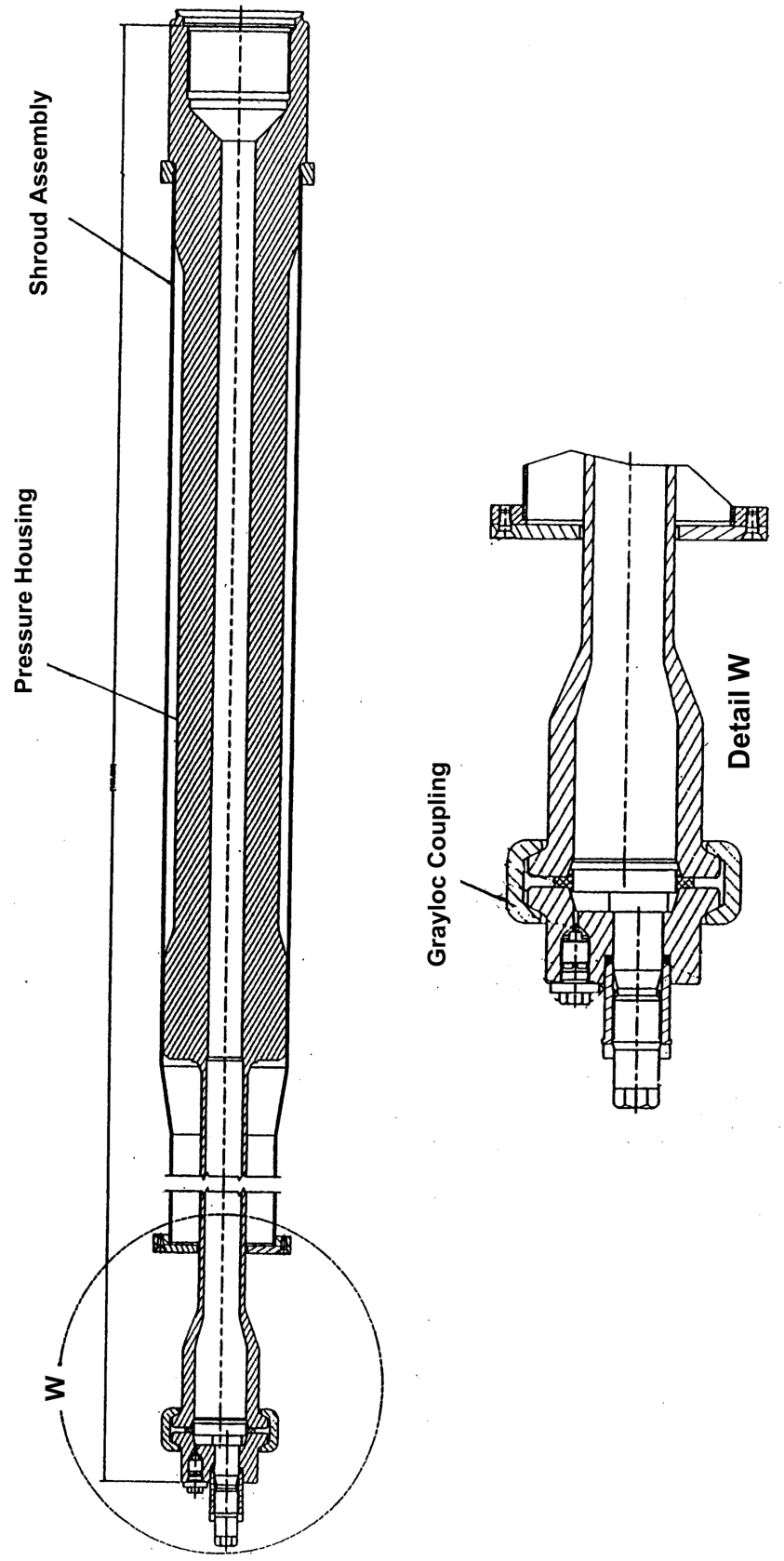


FIGURE 3.3–17 HEATED JUNCTION THERMOCOUPLE PROBE PRESSURE BOUNDARY INSTALLATION

FIGURE 3.3–18 TYPICAL HEATED JUNCTION THERMOCOUPLE PROBE ASSEMBLY INSTALLATION

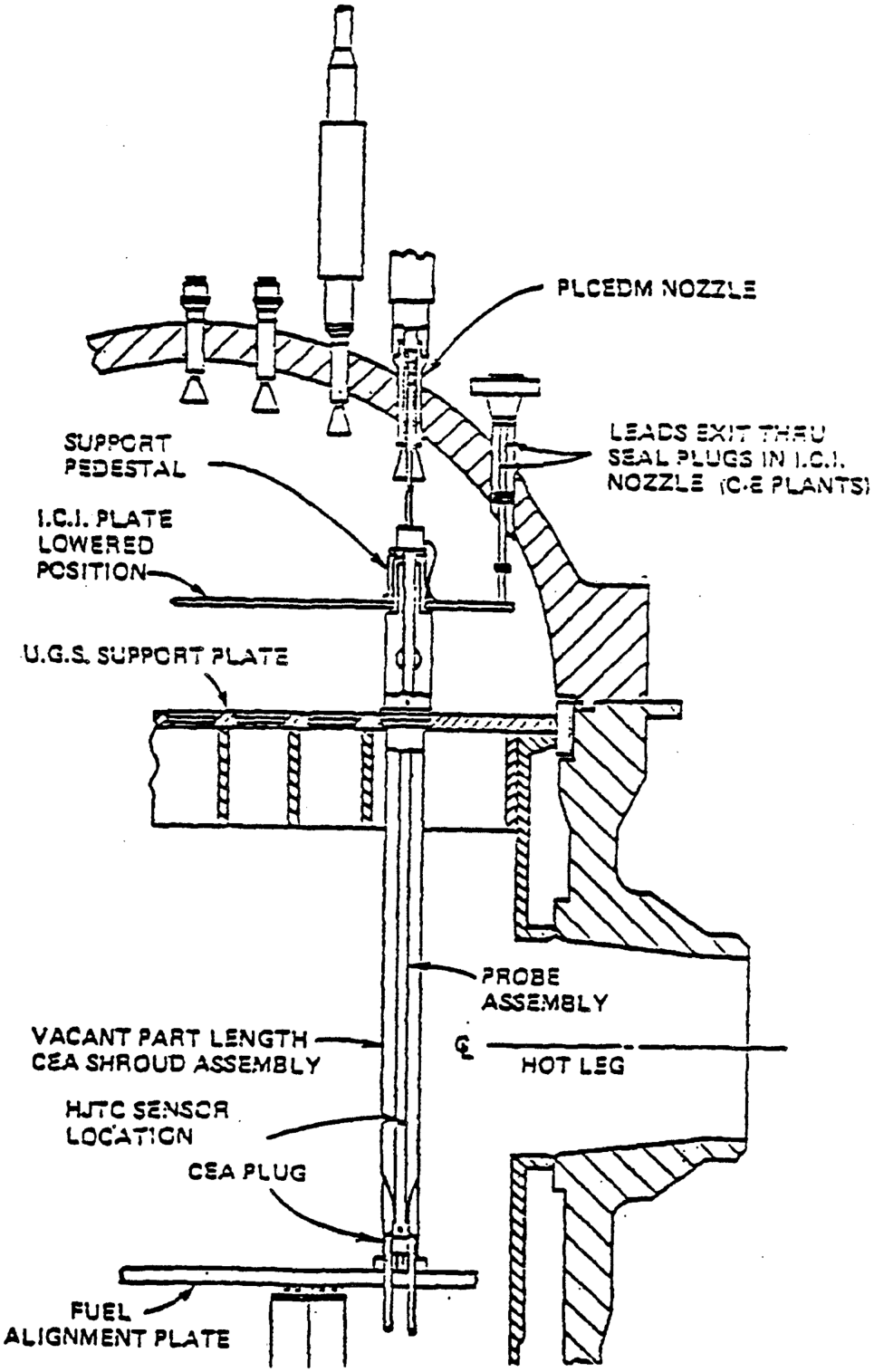
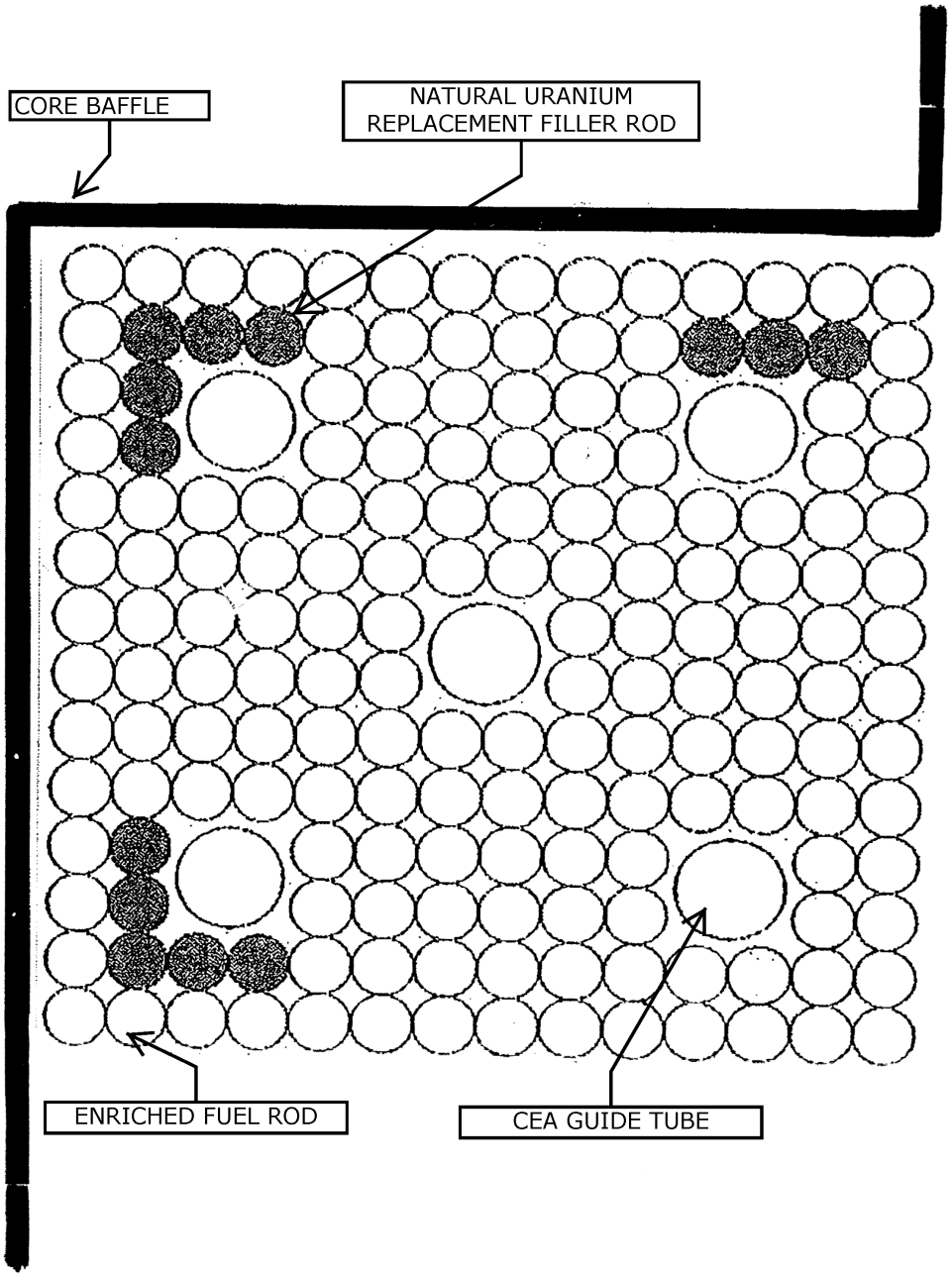


FIGURE 3.3–19 PLACEMENT OF NATURAL URANIUM REPLACEMENT FUEL RODS AND FUEL ASSEMBLY ORIENTATION RELATIVE TO THE CORE BAFFLE FOR CYCLE 19



3.4 NUCLEAR DESIGN AND EVALUATION

3.4.1 GENERAL SUMMARY

This section summarizes the nuclear characteristics of the core and discusses the design parameters which are of significance to the performance of the core in normal transient and steady state operational conditions. A discussion of the nuclear design methods employed and comparisons with experiments which support the use of these methods is included.

The numerical values presented are based on a representative core design. Sufficient analyses are completed each cycle to ensure that actual reload batches keep operating parameters within design limits, accommodate essential reactivity requirements with the control system provided, and meet other requirements for safe operation.

3.4.2 CORE DESCRIPTION

The Millstone Unit 2 reactor consists of 217 assemblies, each having a 14 by 14 fuel rod array. The assemblies are composed of up to 176 fuel rods, four control rod guide tubes, and one center control rod guide tube/instrument tube. The fuel rods consist of slightly enriched UO_2 or UO_2-Gd_2O_3 pellets inserted into Zirconium Alloy tubes. The control rod guide and instrument tubes are made of Zircaloy-4. Each AREVA assembly contains nine spacers. A description of the AREVA supplied fuel design and design methods is contained in References 3.4-1, 3.4-2 and 3.4-3.

A representative loading pattern is shown in Figure 3.4–1 and is expressed in terms of previous cycle core locations and fuel assembly identifiers. A summary of fuel characteristics for a representative core design is presented in Table 3.4-1. Figure 3.4–2 presents representative quarter core assembly movements. Representative beginning of cycle (BOC) and end of cycle (EOC) assembly exposures are shown in a quarter core representation in Figure 3.4–3.

A representative low radial leakage fuel management plan results in scatter loading of the fresh fuel throughout the core. Some fresh assemblies loaded in the core interior contain gadolinia-bearing fuel in order to control power peaking and reduce the initial boron concentration to maintain the moderator temperature coefficient (MTC) within its Technical Specification limit. The exposed fuel is also scatter loaded in the center in a manner to control the power peaking.

3.4.3 NUCLEAR CORE DESIGN

The nuclear design bases for core design are as follows:

a. The design shall permit operation within the Technical Specifications for Millstone Unit 2 Nuclear Plant.

b. The design Cycle length (EFPD) shall be determined on the basis of an estimated Cycle energy and previous Cycle energy window.

c. The loading pattern shall be designed to achieve power distributions and control rod reactivity worths according to the following constraints:

1. The peak linear heat rate (LHR) and the peaking factor Fr shall not exceed Technical Specifications limits in any single fuel rod throughout the cycle under nominal full power operating conditions.

2. The SCRAM worth of all rods minus the most reactive rod shall exceed the shutdown requirement.

The neutronic design methods used to ensure the above requirements are consistent with those described in Reference 3.4-4.

3.4.3.1 Analytical Methodology

The neutronics methods used in the core analysis are described in Reference 3.4-4. The neutronic design analysis for each reload core is performed using the PRISM reactor simulator code. Full-core depletion calculations performed with PRISM are used to determine the core wide power distribution in three dimensions and to reconstruct the individual rod power and burnup distributions. Thermal-hydraulic feedback and axial exposure distribution effects are explicitly accounted for in the PRISM calculations. The CASMO/MICBURN assembly depletion model is used to generate the microscopic cross section input to the PRISM code.

3.4.3.2 Physics Characteristics

The neutronics characteristics of a representative reload core are presented in Table 3.4-2. The safety analysis for each cycle is applicable for a specified previous cycle energy window. A representative HFP letdown curve is shown in Figure 3.4–4.

3.4.3.2.1 Power Distribution Considerations

Representative calculated power maps 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 neutronics model with moderator density and Doppler feedback effects incorporated. The Technical Specification limits on Fr and LHR are 1.69 and 15.1 kW/ft, respectively.

3.4.3.2.2 Control Rod Reactivity Requirements

A representative shutdown margin evaluation is given in Table 3.4-3. The Millstone Unit 2 Technical Specifications require a minimum shutdown margin of 3,600 pcm.

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 -30 pcm/°F at 100 percent of rated thermal power. Representative MTC calculation results are presented in Table 3.4-2.

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. The reload startup physics test program is based on ANSI-19.6-1 (Reference 3.4-9). The Startup Test Activity Reduction (STAR) Program (Reference 3.4-10) provides an alternative to the ANSI-19.6-1 test program provided that specific criteria for the reload core design and construction are satisfied. The STAR Program criteria are established in station procedures and include additional applicability requirements for core design, fuel and control element assembly (CEA) fabrication, CEA lifetime monitoring, refueling and startup testing.

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. [1]

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. [1]

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.

j. HZP to full power reactivity difference.

1.This test may be eliminated if performing the STAR Program per Reference 3.4-10.

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.

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 exactor 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 exactor 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.

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 to 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

The classic method for assessing spatial xenon oscillations is that developed by Randall and St. John (Reference 3.4-5) 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 3.4-6 and 3.4-8. 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 minus 0.06 per hour. When this damping coefficient is corrected for a finite time mesh by the formula in Reference 3.4-7, 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 minus 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 minus 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 3.4-7, 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.

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 minus 1.36 x 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 to 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.

3.4.6 REFERENCES

3.4-1 "Generic Mechanical Design Report Exxon Nuclear 14 x 14 Fuel Assemblies for Combustion Engineering Reactors," XN-NF-82-09(A), Exxon Nuclear Company, Richland, WA 99352, November 1982.

3.4-2 "Design Report for Millstone Point Unit 2 Reload ANF-1," ANF-88-088(P), Rev. 1, Advanced Nuclear Fuels Corporation, Richland, WA 99352, August 1988.

3.4-3 "Millstone Unit 2 Mechanical Design Report for Increased Peaking" EMF-91-245(P), Siemens Nuclear Power Corporation, January 1992.

3.4-4 EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Mehodology Description, Volume 2 - Benchmarking Results", Siemens Power Corporation, January 1997.

3.4-5 Randall, D., "Xenon Spatial Oscillations," Nucleonics, 16, 3, pages 82-86 (1958).

3.4-6 Stacey, Jr., W. M., "Linear Analysis of Xenon Spatial Oscillations," Nuclear Sci. Eng., 30, pages 453-455 (1967).

3.4-7 Poncelet, C. G., "The Effect of a Finite Time Step Length on Calculated Spatial Xenon Stability Characteristics in Large PWR's" Trans. ANS, 10, 2, page 571 (1967).

3.4-8 CEND-TP-26., Diatch, P.B.

3.4-9 ANSI/ANS-19.6-1 "Reload Startup Physics Tests for Pressurized Water Reactors," 2005.

3.4-10 WCAP-16011-P-A, Revision 0, "Startup Test Activity Reduction Program," February 2005.

TABLE 3.4-1 FUEL CHARACTERISTICS FOR A REPRESENTATIVE RELOAD CORE

Fuel Types	N1	N2	N3	N4	P1	P2	P3	P4	P5	R1	R2	R3	R4	R5	R6
Central Zone Assembly Average Enrichment (w/o)	3.94	3.90	3.87	3.82	3.87	3.86	3.84	3.81	3.76	4.49	4.49	4.47	4.39	4.33	4.42
Number Gadolinia Bearing Rods	0	6	12	16	0	4	8	12	16	0	4	8	12	16	12
Nominal Density (% TD)	94	94	94	94	94	94	94	94	94	95	95	95	95	95	95
Pellet OD (inches)	0.370	0.370	0.370	0.370	0.370	0.370	0.370	0.370	0.370	0.377	0.377	0.377	0.377	0.377	0.377
Clad OD (inches)	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440	0.440
Diametral Gap (inches)	0.0080	0.0080	0.0080	0.0080	0.0080	0.0080	0.0080	0.0080	0.0080	0.007	0.007	0.007	0.007	0.007	0.007
Clad Thickness (inches)	0.031	0.031	0.031	0.031	0.031	0.031	0.031	0.031	0.031	0.028	0.028	0.028	0.028	0.028	0.028
Rod Pitch (inches)	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580	0.580
Spacer Material	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic	Bimetallic
Fuel Supplier	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA	AREVA
Fuel Stack Height Nominal (inches)	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7	136.7
Number of Assemblies	8	20	8	25	8	8	12	8	36	8	8	8	8	48	4
Regionwise Loading (MTU)	3.04	7.60	3.03	9.43	3.04	3.04	4.55	3.03	13.58	3.19	3.19	3.19	3.17	18.98	1.59

TABLE 3.4-2 NEUTRONICS CHARACTERISTICS FOR A REPRESENTATIVE RELOAD CORE

<characteristic>	BOC	EOC
Critical Boron (ppm): HZP, ARO, No Xenon	1453	---
Critical Boron (ppm): HFP, ARO, Equilibrium Xenon	1024	0
Moderator Temperature Coefficient (pcm/°F): HZP	+2.0	-10.4
Moderator Temperature Coefficient (pcm/°F): HFP	-6.0	-23.3
Doppler Coefficient (pcm/°F)	-1.17	-1.33
Boron Worth (pcm/ppm): HZP	-8.8	-10.8
Boron Worth (pcm/ppm): HFP	-8.4	-10.4
LHR (kW/ft) HFP (a)	12.8	11.6
Delayed Neutron Fraction	0.0064	0.0054
HFP, PDIL Worth (pcm)	157	241
N-1 Rod Worth, HZP (pcm)	6271	7696
Excess Shutdown Margin (pcm): HFP	124	323
Excess Shutdown Margin (pcm): HZP	140	751

(a) Including uncertainties.

TABLE 3.4-3 REPRESENTATIVE SHUTDOWN MARGIN REQUIREMENTS

Control Rod Worth (pcm)

<parameter>	BOC: HZP	BOC: HFP	EOC: HZP	EOC: HFP
ARI	9315	9315	10450	10450
N-1	6271	6271	7696	7696
PDIL	2116	157	2862	241
[(N-1) - PDIL] * 0.9	3740	5503	4351	6710

Reactivity Insertion (pcm)

<parameter>	BOC: HZP	BOC: HFP	EOC: HZP	EOC: HFP
Power Defect	0	1507	0	2515
Void	0	50	0	50
Flux Redistribution	0	222	0	222
Total Requirements	0	1779	0	2787

Shutdown Margin (pcm)

<parameter>	BOC: HZP	BOC: HFP	EOC: HZP	EOC: HFP
[(N-1) * PDIL] * 0.9 - Total	3740	3724	4351	3923
Required Shutdown	3600	3600	3600	3600
Excess Shutdown Margin	140	124	751	323

FIGURE 3.4–1 REPRESENTATIVE FULL CORE LOADING PATTERN

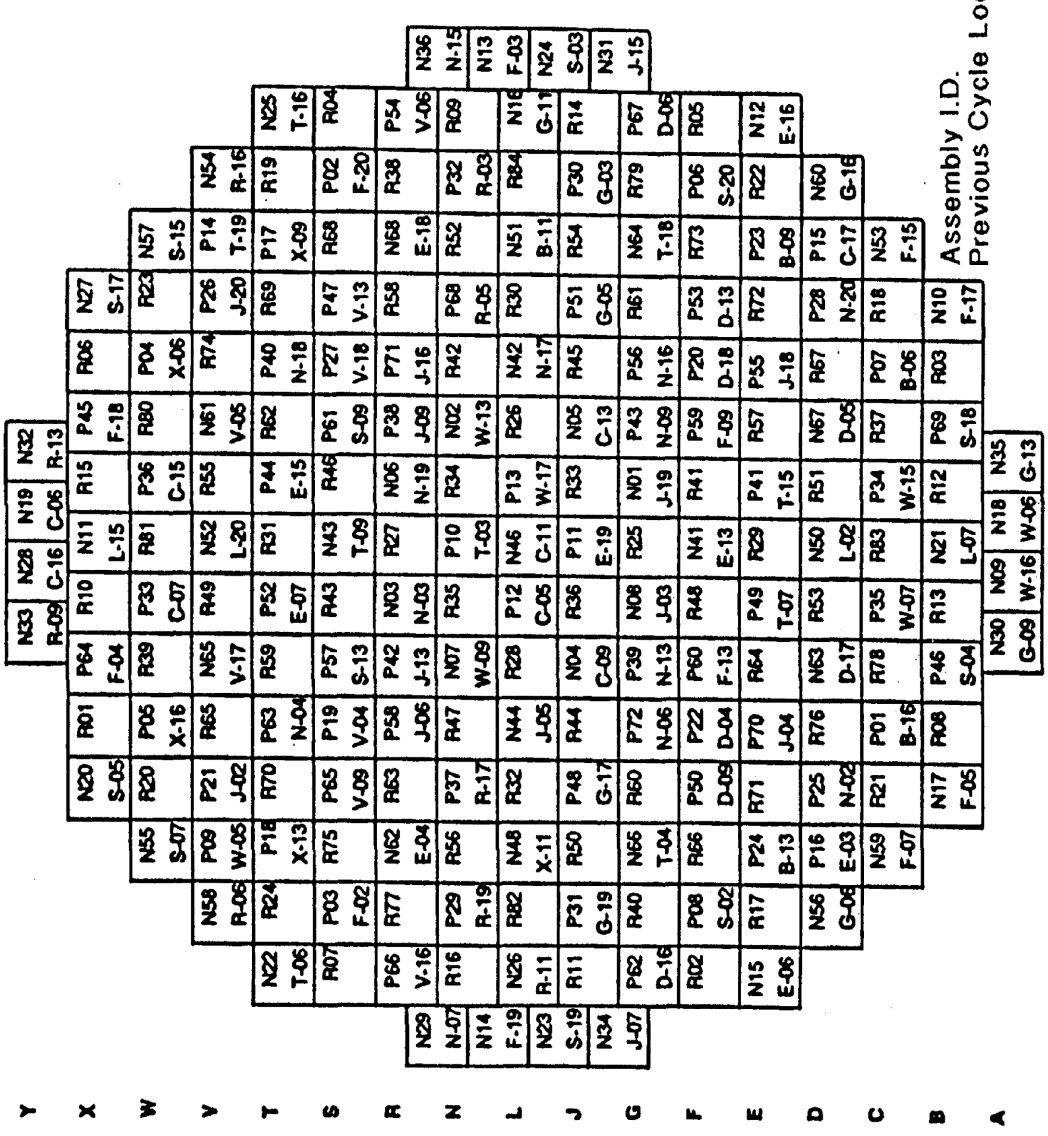


FIGURE 3.4–2 REPRESENTATIVE QUARTER CORE LOADING PATTERN

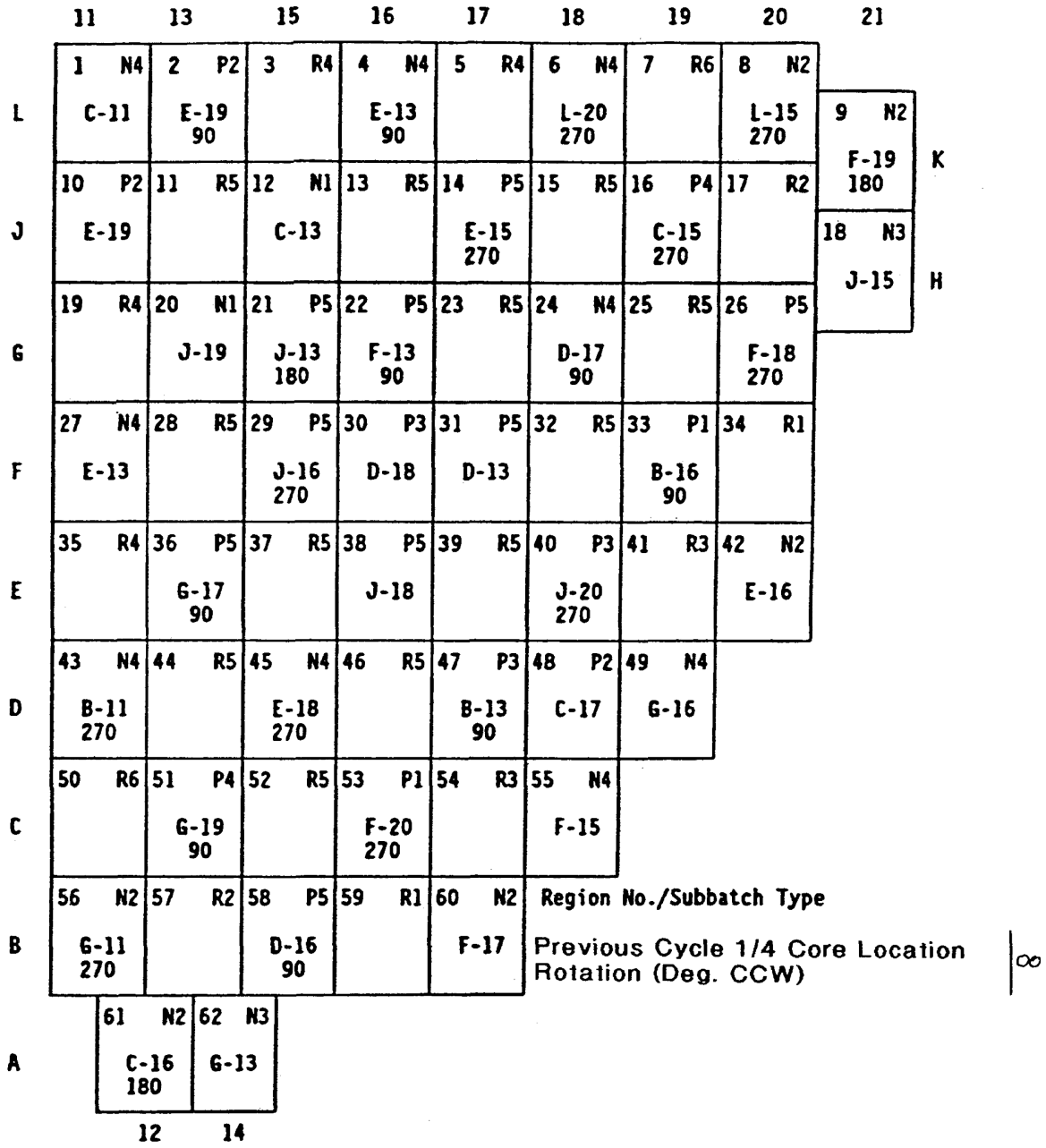


FIGURE 3.4–3 REPRESENTATIVE BOC AND EOC EXPOSURE DISTRIBUTION

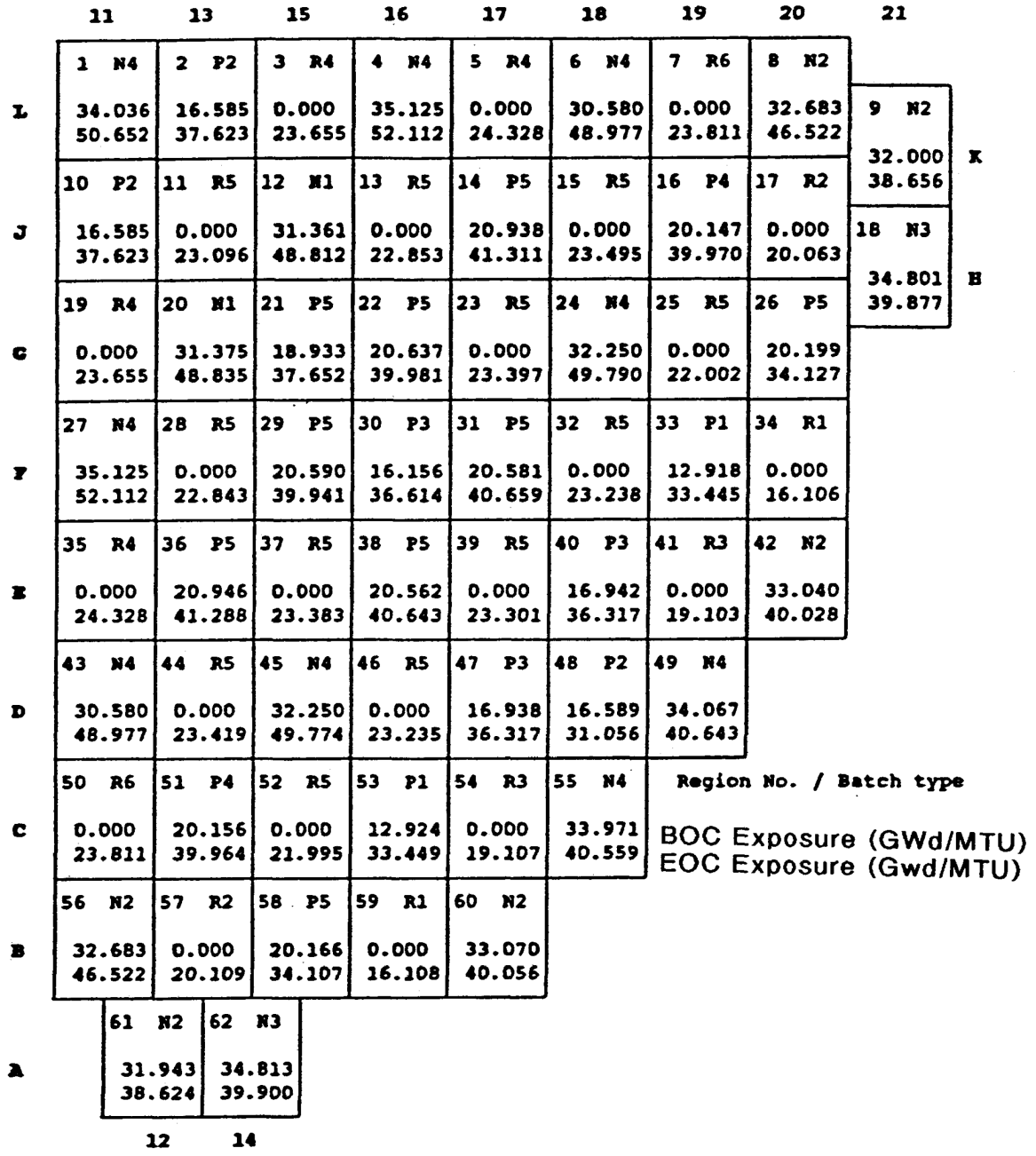


FIGURE 3.4–4 REPRESENTATIVE BORON LETDOWN, HFP, ARO

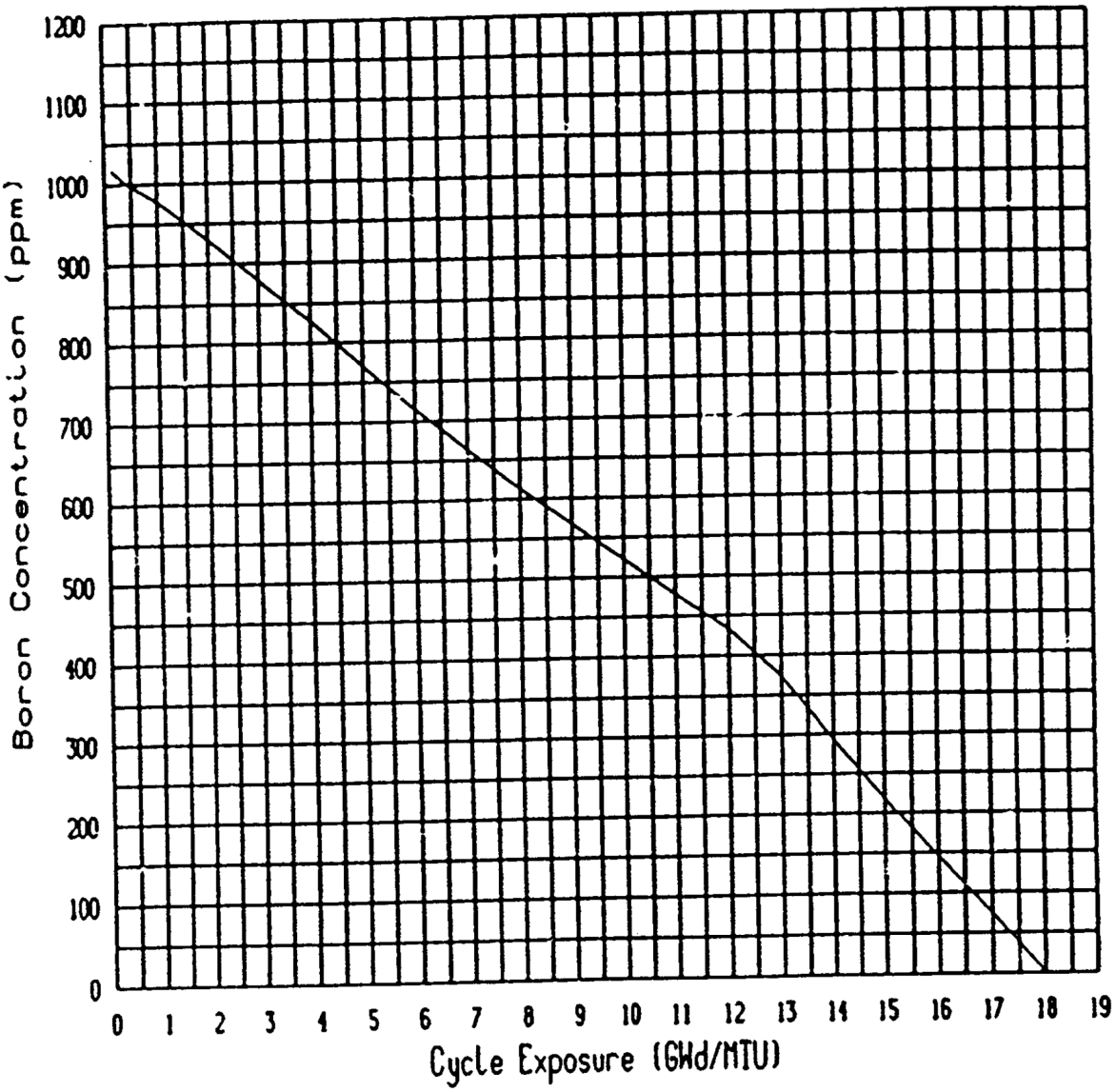


FIGURE 3.4–5 REPRESENTATIVE NORMALIZED POWER DISTRIBUTIONS, HOT FULL POWER, EQUILIBRIUM XENON, 150 MWD/MTU

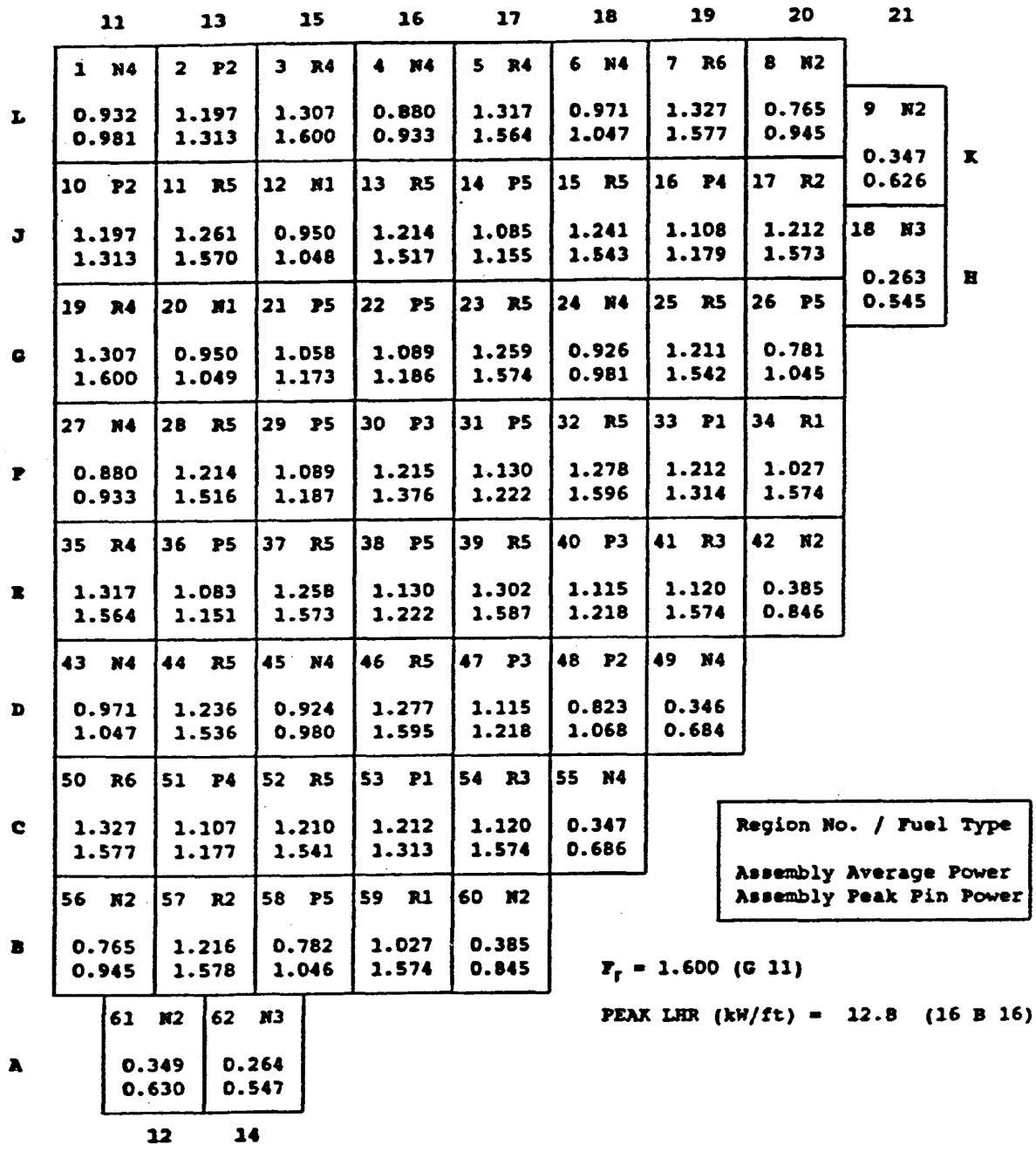
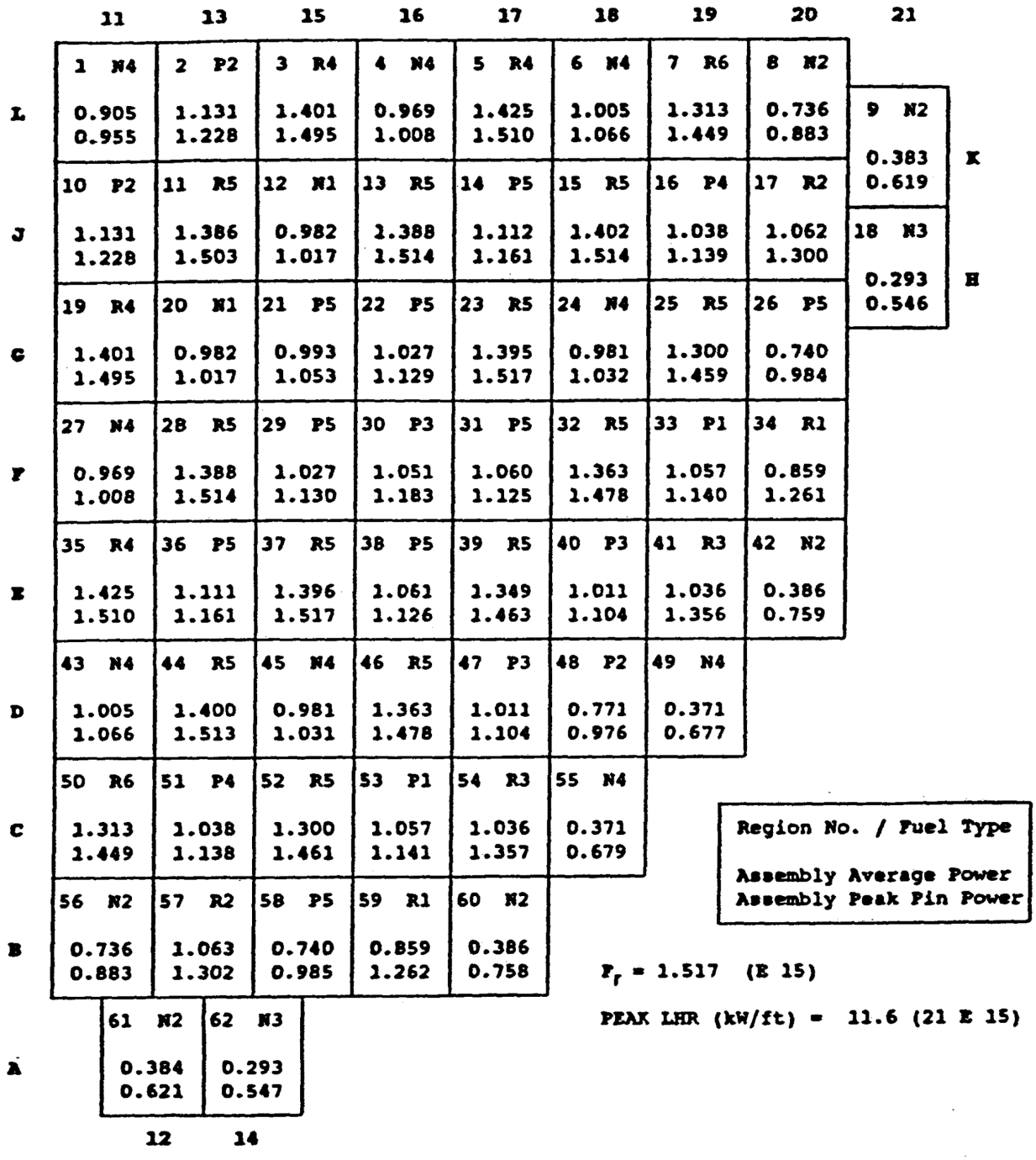


FIGURE 3.4–6 REPRESENTATIVE NORMALIZED POWER DISTRIBUTION, HOT FULL POWER, EQUILIBRIUM XENON, 18,020 MWD/MTU



3.5 THERMAL-HYDRAULIC DESIGN

This section presents thermal and hydraulic analysis of the reactor core, analytical methods utilized, and experimental work supporting the analytical techniques. The prime objective of the thermal and hydraulic design of the reactor is the assurance that the core can meet normal steady state and anticipated transient performance requirements without exceeding the design bases. A summary of the significant reactor and fuel parameters used in the thermal and hydraulic design and analysis is presented in Table 3.5-1.

3.5.1 DESIGN BASES

3.5.1.1 Thermal Design

Avoidance of thermally induced fuel damage during any normal steady state and anticipated transient operation is the principal thermal and hydraulic design basis. The following limits are established, but violation of them will not necessarily result in fuel damage. The Reactor Protection System will provide for automatic reactor trip or other corrective action before these design limits are exceeded.

a. Avoidance of departure from nucleate boiling (DNB) for the limiting rod in the core with 95 percent probability at a 95 percent confidence level.

b. Limitation of the peak temperature of the fuel to less than the melting point during normal operation and anticipated transients.

Since the departure from nucleate boiling ratio (DNBR) criterion ensures that the cladding temperature remains close to the coolant temperature, no additional criteria for cladding temperature are required for normal operation and anticipated transients. For design basis accident conditions (loss of coolant accidents (LOCA)), under which the DNBR criterion does not apply, cladding temperatures are calculated to ensure that they remain below 2200°F, which is the peak clad temperature criterion of 10 CFR 50.46. For other postulated accidents, fuel failure is assumed to occur if the calculated DNBR is below the DNB correlation 95/95 limit.

3.5.1.2 Hydraulic Stability

Operating conditions shall not lead to flow instability during normal steady state and anticipated transient operation.

3.5.1.3 Coolant Flow Rate, Distribution and Void Fraction

A lower limit on the total primary coolant flow rate, called “design” flow, is set to assure that the core is adequately cooled when uncertainties in system resistance, pump head, and core bypass flow are taken in the adverse direction. By design of the reactor internal flow passages, this flow is distributed to the core such that the core is adequately cooled with all permissible core power distributions. The hydraulic loads for the design of the internals are based on the upper limit of the

flow. The upper limit is obtained in a similar manner as the design flow but with the uncertainties taken in the opposite direction.

To ensure that sufficient coolant flow reaches the fuel, the amount of coolant flow which bypasses the core through the guide tubes must not excessively reduce the active core flow. The guide tube coolant flow must, however, be sufficient to ensure that coolant in the guide tubes will not boil and ensure adequate cooling of the CEA fingers. The CEA drop time in the guide tubes must also meet the criterion of 90 percent insertion within 2.75 seconds to ensure that scram performance is in accordance with plant Technical Specifications.

Although the coolant velocity, its distribution, and the coolant voids affect the thermal margin, design limits need not be applied to these parameters because they are not themselves limiting with respect to thermal margin. These parameters are included in the thermal margin analyses and thus affect the thermal margin to the design limits.

3.5.2 THERMAL AND HYDRAULIC CHARACTERISTICS OF THE DESIGN

3.5.2.1 Fuel Temperatures

The RODEX2 code (Reference 3.5-1) incorporates models to describe the thermal and mechanical behavior of the fuel rod in a flow channel including the gas release, swelling, densification, and cracking in the pellet; the gap conductance; the radial thermal conduction; the free volume and gas pressure internal to the fuel rod; the fuel and cladding deformations; and the cladding corrosion as a function of burnup. The calculations are performed on a time-incremental basis with conditions being updated at each calculated increment.

3.5.2.1.1 Fuel Cladding Temperatures

The RODEX2 thermal-hydraulic model (Reference 3.5-1) calculates the lowest cladding surface temperature based on one of two heat transfer regimes; i.e., forced convection and fully developed nucleate boiling. The forced convection and fully developed nucleate boiling heat transfer correlations in RODEX2 were developed by Kays and Thom et al., respectively.

3.5.2.1.2 Fuel Pellet Temperatures

The RODEX2 radial temperature distribution model begins with the standard differential equation of heat conduction (Poisson Equation) for an isotropic solid with internal heat generation. The equation is written in cylindrical coordinates assuming that the thermal conductivity of the fuel is a function of fuel temperature, but is independent of position. With additional assumptions of axial symmetry, negligible heat conduction in the axial direction, and steady state conditions, a one-dimensional (i.e., radial) steady state form of the equation is derived and employed.

The minimum power level required to produce centerline melt in zirconium alloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in kW/ft. This FCMLHR is determined using the methodologies of Reference 3.5-19 and Reference 3.5-22. A conservative cycle specific FCMLHR limit is used for Millstone Unit 2.

The maximum LHR for normal operation and anticipated transients is typically well below the conservative FCMLHR limit. It should be noted that a gadolinia-bearing fuel rod will, for a given LHGR, operate with a higher fuel temperature than an all-uranium-bearing fuel rod. Gadolinia rods are specifically analyzed to centerline melt criteria.

3.5.2.1.3 UO_2 Thermal Conductivity

Lyon's expression for thermal conductivity of the fuel is used in RODEX2. Three corrections are applied: one for density, one to account for burnup-dependent degradation, and one to account for the gadolinia content in the fuel.

3.5.2.1.4 Gap Conductance

The RODEX2 gap conductance model is based on that proposed by Kjaerheim and Rolstad. The total gap conductance has three components: (1) gas conductance, (2) radiation, and (3) fuel/cladding solid-to-solid contact.

3.5.2.2 Departure from Nucleate Boiling Ratio

DNBRs are calculated using approved correlations. An approved core thermal-hydraulic computer code is used to determine the flow and enthalpy distribution in the core and the local conditions in the hot channel for use in the DNB correlation.

3.5.2.2.1 Departure from Nucleate Boiling

The XCOBRA-IIIC (Reference 3.5-2) computer code is employed to evaluate the thermal-hydraulic conditions in the various assemblies and in the subchannels of the limiting assembly. Heat, mass, and momentum fluxes between the inter-rod flow channels are explicitly calculated. Fuel and reactor design conditions employed in these calculations are given in Table 3.5-1.

The calculations include a statistically determined engineering factor to account for manufacturing tolerances, thermal expansion and densification effects. The engineering factor is applied to the local heat flux in the calculation of DNBR.

In-reactor densification results in a shortening of the fuel column. At power levels typical of DNBR-limiting rods, thermal expansion tends to offset the densification effect. The XCOBRA-IIIC model does not specifically model changes in stack length due to thermal expansion and densification.

The HTP DNB correlation, demonstrated to be applicable to the Framatome 14 by 14 reload fuel assemblies for CE reactors, is described in Reference 3.5-3. A minimum allowable limit corresponding to 95% probability with 95% confidence is set on the DNBR during normal operation and any anticipated transients. The XNB CHF correlation, applicable for use in the non-mixing grid region, is described in Reference 3.5-24.

3.5.2.2.2 Hot Channel Factors

Hot channel factors for heat flux and enthalpy rise, F_q and F_r:

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux hot channel factor (F_q) considers the local maximum linear heat generation rate at a point (the hot spot), and the enthalpy rise hot channel factor (F_r) involves the maximum integrated linear heat generation rate along a channel (the hot channel).

Engineering hot channel factor, F_E:

The engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication uncertainties for fuel pellet diameter, density, and enrichment, as well as the effect of densification. A conservative value of 1.03 is used. The effect of variations in fabrication tolerances is considered in the analysis. To account for manufacturing uncertainties and densification, the peak rod heat flux is increased by 3% in the calculation of DNBR.

3.5.2.2.2.1 Nuclear Peaking Factors

Assembly and rod peaking factors and axial power distributions are input into the XCOBRA-IIIC code. Departure from nucleate boiling is dependent on the local rod heat flux and the local fluid conditions within the channel.

The effect of asymmetries in core power distribution (specifically azimuthal power tilt) is not directly taken into account in the XCOBRA-IIIC thermal-hydraulic calculations. The effects of azimuthal power tilt are accounted for in the generation (verification) of the TM/LP trip and LPD trip monitoring setpoints through the measurement of radial peaking factors.

3.5.2.2.2.2 Rod Bowing Factor

As the fuel assembly burnup increases, the gaps between fuel rods change. Decreased rod-to-rod gaps can occur, which can reduce the DNB ratio. Penalties are calculated as a function of burnup and applied to the DNBR or peak linear power as appropriate.

3.5.2.2.2.3 Inlet Flow Distribution Factor

Inlet flow maldistribution is treated in the XCOBRA-IIIC model by applying a generic inlet flow penalty to the limiting assembly and its crossflow neighbors.

3.5.2.2.2.4 Flow Mixing Factor

The effects of both pressure-driven and turbulent flow mixing between channels on the hot channel enthalpy rise are calculated by the XCOBRA-IIIC computer code. The turbulent flow

mixing is modeled empirically and is based on the reduction of the data from hot mixing tests using XCOBRA-IIIC.

The geometry of the channels surrounding the hot channel and the radial power distribution affect the lateral enthalpy transport for both the pressure-driven and turbulent flow mixing.

3.5.2.2.3 Effects of Rod Bow on DNBR

In accordance with AREVA rod bow methodology (Reference 3.5-4), the magnitude of rod bow for assemblies of the type used in Millstone Unit 2 has been estimated. Significant impact on the DNBR due to rod bow does not occur until the gap closures exceed 50 percent. The maximum design exposure for AREVA reload fuel in Millstone Unit 2 is significantly less than that at which 50 percent closure occurs; therefore, rod bow does not significantly impact the minimum DNBR (MDNBR). A further consequence of the small amount of rod bow for AREVA fuel is that total power peaking is not significantly impacted.

3.5.2.3 Void Fraction and Distribution

The XCOBRA-IIIC model calculates the local thermal and hydraulic conditions for input to the DNB correlation. While local conditions of enthalpy, quality, flow rate and pressure are associated with a code-calculated local void fraction, the void fraction is not input to the DNB correlation. The DNB correlation is approved over a local quality range, but it is not a direct function of void fraction. Therefore, there is no explicit limit set on average or local void fraction beyond that implied in the test conditions used to develop the DNB correlation.

3.5.2.4 Coolant Flow Distribution

3.5.2.4.1 Coolant Flow Distribution and Bypass Flow

The minimum primary coolant flow rate at full power conditions is given in Table 3.5-1.

Tracing the coolant flow path in Figure 3.1–1, the coolant enters the four inlet nozzles and flows into the annular plenum between the reactor vessel and core support barrel. It then flows down the annulus between the reactor vessel and core barrel and up through the flow skirt to the plenum below the core lower support structure. The skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower core plate and flows axially through the fuel assemblies. A portion of the coolant passes through the lower core plate and into the guide tubes in the fuel assemblies. The fuel assembly alignment plate is not drilled through in guide tube locations without CEAs; therefore, core bypass flow is limited in these guide tubes. After passing through the core, the coolant flows into the region outside the control element assembly shrouds. From this region, the coolant flows across the control element assembly shrouds and passes out through the outlet sleeves on the core barrel to the outlet nozzles.

The coolant which does not contact any fuel rods is termed core bypass coolant. The following are the principal core bypass routes:

a. Direct inlet to outlet coolant flow at the joint between the core support barrel sleeve and reactor vessel nozzle.

b. Coolant flow into the guide tubes in the fuel assemblies.

c. Coolant flow in the region between the core support barrel and core shroud.

d. Coolant flow from the inlet nozzle region through the alignment keyways to the vessel head region.

Table 3.5-1 gives the “best estimate” value for the core bypass flow rate as a fraction of the total primary flow rate. Taking into account the core bypass flow rate, the core flow rate, which is the effective flow rate for heat transfer, can be calculated from the total primary coolant flow rate.

3.5.2.4.2 Core Flow Distribution

The core flow distribution (CFD) analysis is performed to assess cross flow between assemblies in the core for use in subsequent MDNBR subchannel analyses. A full core model provides cross-flow boundary conditions to a full assembly model at the assembly boundaries. MDNBRs are computed from a full assembly simulation.

In the analysis, each fuel assembly in the Millstone Unit 2 core is modeled as a hydraulic channel. The calculations are performed with the XCOBRA-IIIC computer code (Reference 3.5-2). Cross flow between adjacent assemblies in the open lattice core is directly modeled. The single-phase loss coefficients are used in the CFD analyses to hydraulically characterize the assemblies in the core.

This computational procedure is designed to evaluate thermal-hydraulic conditions during boiling and non-boiling conditions. One-dimensional, two phase separated, slip flow is assumed in the XCOBRA-IIIC calculation. These assumptions are valid only if the cross flow between connecting channels is small compared to the axial velocities in the individual channels. Because small cross flow does exist, mathematical models have to be postulated for both turbulent and diversion cross-flow mixing. Models of the two-phase state are also defined in terms of void fraction, which is a function of enthalpy, flow rate, heat flux, pressure, and axial position. This computational procedure is not applicable when large blockages exist in the fuel bundles since this leads to considerable cross flow which cannot be adequately represented by the one-dimensional analysis.

Table 3.5-1 summarizes the reactor and fuel design parameters used in these CFD calculations and subsequent MDNBR analyses.

3.5.2.5 Pressure Losses and Hydraulic Loads

3.5.2.5.1 Pressure Losses

The fuel assembly irrecoverable pressure losses have been calculated using standard loss coefficient methods and results from model tests. The pressure loss across the AREVA fuel assembly was determined based on the results of Reference 3.5-5 and analyses.

3.5.2.5.2 Hydraulic Loads

3.5.2.5.2.1 Hydraulic Loads on Vessel Internal Components

The design hydraulic loads for the internal components for steady state operating conditions are listed in Table 3.5-2. These loads were derived from analysis and from reactor flow model and component test results. All hydraulic loads in Table 3.5-2 are based on the maximum expected system flow rate and a coolant temperature of 500°F. When these hydraulic loads are used in the structural analysis, they are adjusted for coolant temperature. The worst condition (i.e., coolant temperature) is not necessarily the same for each internal component; therefore, the loads are adjusted to reflect the difference in coolant temperature. This is done to ensure the design hydraulic stresses are acceptable during start-up and during power operation.

The types of loads considered in the analysis are: (1) steady-state drag and impingement loads, and (2) fluctuating loads induced by pump pressure pulsations, turbulence, and vortex shedding. All of these loads are not exerted on each internal component, but each component sees at least one of the loads. Table 3.5-2 lists the components and type of loads that are exerted on them.

3.5.2.5.2.2 Core Hydraulic Loads/Fuel Assembly Liftoff

The holddown spring force and the assembly weight force prevent the fuel assembly from lifting off the core support plate during reactor steady-state operation, based on the most adverse combination of component dimensional and material property tolerances. In addition, the holddown springs are designed to accommodate the additional load associated with a pump overspeed transient (resulting in possible temporary liftoff of the fuel assemblies), and to continue to ensure fuel assembly holddown following such occurrences. The limiting reactor steady-state conditions are the 4th pump startup conditions. These correspond to the minimum temperature and maximum pressure and coolant flow for reactor startup. Thermal expansion of the reactor vessel and fuel assembly is also considered.

3.5.2.6 Correlation and Physical Data

Reference 3.5-1 describes the correlations and physical data employed in heat transfer calculations performed by RODEX2. Reference 3.5-7 describes the correlations and physical data employed in the hydraulic calculations performed by XCOBRA-IIIC. Reference 3.5-3 describes the correlations and physical data employed in the DNB correlation.

3.5.2.7 Plant Parameters for Thermal-Hydraulic Design

The plant parameters considered include total primary coolant flow rate, vessel inlet temperature, primary pressure, and core thermal power. Two sets of thermal-hydraulic conditions are defined: nominal conditions and design conditions. Nominal plant conditions represent the best estimate for the primary coolant flow rate, pressure, and vessel inlet temperature and do not include allowances for instrument errors. Design plant conditions represent the lower limit on primary flow rate when uncertainties in system resistance and pump head are included, and represent the upper limit on vessel inlet temperature when design margins on steam generator performance are included. Furthermore, the variations which occur during steady state operation in the power, pressure, and inlet temperature due to controller deadband and instrument error are considered with the design plant parameters. During steady state operation, the possible variations in these parameters define an operating envelope. One combination of these parameters gives the MDNBR, and this combination is utilized in Chapter 14 as the initial conditions in transient and accident analysis. Table 3.5-1 lists the nominal plant parameters.

3.5.2.8 Summary of Thermal and Hydraulic Parameters

The thermal and hydraulic parameters for the reactor are listed in Table 3.5-1.

3.5.3 THERMAL AND HYDRAULIC EVALUATION

3.5.3.1 Analytical Techniques and Uncertainties

3.5.3.1.1 XCOBRA-IIIC DNBR Analyses

The thermal-hydraulic simulations employed to evaluate the MDNBR were performed in accordance with AREVA’s Nuclear Regulatory Commission (NRC) approved thermal-hydraulic methodology for mixed cores (Reference 3.5-8).

The MDNBR performance of the core during anticipated transients will be demonstrated to meet the thermal-hydraulic design criterion on DNBR through the performance of transient analysis of the limiting events. The results of this analysis are included in Chapter 14.

3.5.3.1.2 Parameter Uncertainties

Tables 14.0.7-2 through 14.0.7-5 identify parameter uncertainties included in the AREVA thermal and hydraulic and DNB methodology. Plant instrument calibration procedures and related specification requirements are designed so that these uncertainties do not increase.

3.5.3.2 Hydraulic Instability Analysis

Boiling flows may be susceptible to thermohydrodynamic instabilities. These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux or to undesired forced vibrations of core components. However, unlike in Boiling Water Reactors (BWRs), hydraulic stability is not a concern in PWR

cores. This statement, which is discussed below, is supported by the literature and the state of the art on instabilities occurring in two-phase flow systems.

Instabilities in vertical up-flow of a two-phase mixture in a heated channel can be broadly classified into several categories. Of these, the following relevant instabilities are discussed.

1. Flow Excursion

   Also called Ledinegg Instability, this is well described in Ref. 3.5-10. This instability occurs when the slope of the boiling channel pressure drop-flow rate curve (internal characteristic) becomes smaller than the slope of the loop supply pressure drop-flow rate curve (external characteristic), i.e.,

   $$\frac{d(\Delta P)}{dG}\Big|_{internal} < \frac{d(\Delta P)}{dG}\Big|_{external}$$

   where ΔP is the pressure drop and G is the mass flow rate.

   In this manner, a negative flow perturbation will be amplified as the internal pressure drop becomes larger than the external at the perturbed flow and the flow decelerates further until a stable point is reached.

   If the core is considered as a single average channel, the external pressure and flow characteristics as seen by the core exhibit

   $$\frac{d(\Delta P)}{dG}\Big|_{external} < 0$$

   due to the pump characteristics. This negative slope is stabilizing.

   On the other hand, considering flow in a single limiting bundle, the other parallel flow paths impose a flat pressure drop versus flow relation where $d(\Delta P)/dG = 0$. While this situation is less stable than the average core assumption, it is mitigated by the cross flow and mixing between this limiting bundle and the neighboring bundles. Ref. 3.5-11 shows experimentally a definite stabilizing influence of cross flow mixing.

   The internal pressure drop versus flow characteristics were shown to satisfy the Ledinegg stability criterion

$$\frac{d(\Delta P)}{dG}\Big|_{internal} > 0$$

for a wide range of conditions in the LOFT reactor (Ref. 3.5-12) which closely approximates a PWR core during nominal and worst case operating conditions.

Therefore, in conclusion, Ledinegg Instability is not a concern in PWR cores.

2. Density Wave Instability

Dynamic instabilities may occur even when the static stability criterion is satisfied (pressure drop increases when flow increases). For a density wave dynamic instability, consider an inlet flow increase perturbing the initial value. The rate of enthalpy rise and density effects will travel up the channel, and the pressure drop increase is delayed. In the case of a sinusoidal inlet flow perturbation of particular frequency, the lagging pressure drop response is such that its instantaneous value supports the growth of the initial perturbation (Ref. 3.5-13). Such unstable behavior requires the delayed portion of the total pressure drop (in the two-phase region) to be large compared with the single-phase pressure drop. The onset of this instability depends on the operating conditions and the distribution of pressure drop along the channel, as well as the external loop characteristics. A vast body of literature and several computer programs for the analysis of density waves exists mainly for BWR concerns (see for example the collection of papers in Ref. 3.5-14). Inferences from BWR experience are drawn to dismiss the possibility of density wave instabilities in a PWR core:

- Unlike a BWR, there is no riser section contributing significantly to the 2-phase pressure drop.

- For a single limiting channel with a constant pressure drop boundary condition, the cross flow in a PWR core has a stabilizing effect.

- Density wave oscillations are known to be stabilized with increasing pressure (decreasing enthalpy and density difference between the two phases). No unstable density wave oscillations could be obtained for pressures higher than 1200 psia (Ref. 3.5-15).

- BWR oscillations occur when the saturated boiling boundary is low (elevation <<4 feet). For a PWR, such boiling boundary can be achieved at nominal flow rates by more than doubling the power, which leaves a considerable stability margin even for the worst case transient.

- Considering the nuclear coupling, the void-reactivity coefficient in a PWR is reduced when the coolant is borated. Such reduction in the void-reactivity coefficient is stabilizing to this mode of oscillation.

- For a density wave coupled with an out-of-phase neutron flux oscillation mode, the large subcritical reactivity of the first flux harmonic stabilizes this mode of hydraulic-neutronic oscillation. This is due to the PWR core being small compared with typical BWR cores.

The LOFT reactor stability study also addressed the density wave oscillations and concluded that these are not likely (Ref. 3.5-12).

In conclusion, Density Wave Instability is not a concern in PWR cores.

3. Flow Pattern Transition Instability

The term “Flow Pattern Instability” is used in the literature in two connotations. The first refers to the slug flow pattern where a particular elevation in a heated channel experiences a succession of high void and low void flows as a vapor slug passes through (Ref. 3.5-12). As a vapor slug clears the channel exit, the average void content in the channel is temporarily reduced and vice versa resulting in pressure drop and flow rate oscillations. In a worst case condition in a PWR, slug flow may occur in a small number of channels near the exit. No significant oscillatory response is expected, particularly since the slug formation is limited to a short segment near the exit of the hot channels.

The more common meaning of the “Flow Pattern Transition Instability” refers to unstable transitions between bubbly and annular flow (Ref. 3.5-10). A flow rate perturbation decreasing the flow rate and increasing the void fraction will result in flow transition from bubbly-slug to annular pattern. The annular flow is characterized with lower pressure drop, which results in accelerating the flow. The increase in flow rate brings the void fraction back below the value required to support annular flow. Thus the transition back to bubbly-slug regime takes place.

Extensive work has been done on flow pattern transition (see for example Ref. 3.5-16). Most work was limited to pressures of 1000 psia and below where these transitions are more distinct. At higher pressures, Hosler (Ref. 3.5-17) notes for 1400 and 2000 psia, that the flow appears more homogeneous with no reliable observation of pattern transition.

Weisman et. al. (Ref. 3.5-18) observed no premature DNB due to bubbly-to-slug flow transition which they expected as the range of tested void fractions covers the transition range. Hosler (Ref. 3.5-17), on the other hand, noted that CHF occurred via a film dryout mechanism in established annular flow, which is far from the transition boundary to bubbly-slug pattern.

In conclusion, Flow Pattern Transition Instability is not a concern in PWR cores.

3.5.3.3 Core Hydraulics

3.5.3.3.1 Fuel Assembly Pressure Drop Coefficients

Pressure drop coefficients for the AREVA reload fuel presented are derived from pressure drop tests performed in AREVA's portable hydraulic test facility (Reference 3.5-5). The pressure drop coefficients are for the liquid phase and are referenced to the bare rod flow area.

The introduction of the AREVA Standard CE14 HTP fuel with M5 cladding in Reload Batch EE has a negligible effect on core hydraulics. The AREVA Standard CE14 HTP fuel with M5 cladding is hydraulically equivalent to the HTP fuel with Zircaloy-4 cladding (Reload Batches Y through DD fuel).

3.5.3.3.2 Guide Tube Bypass Flow and Heating Analysis

The guide tube thermal-hydraulic design calculations are performed to demonstrate adequate cooling of the CEA fingers and to ensure that bypass flow through the guide tubes does not unduly reduce core flow.

Flow enters the guide tube through the weep holes and cap screw and exits through the top of the guide tube. In the Millstone Unit 2 core, there are 81 assemblies under CEA positions. Of these, 73 assemblies are under active CEA positions. The CEA fingers extend a short distance into the guide tube in these 73 assemblies at the all-rods-out (ARO) position which provides a substantial reduction in the guide tube bypass flow. The remaining eight assemblies were originally under the part length CEAs which have been removed. In these eight assemblies, the flow is unimpeded, since the last flow plugging devices were removed in Cycle 12. The assembly guide tubes of 91 assemblies project a short distance into close fitting sockets in the upper alignment plate. The resulting flow annulus represents a significant resistance to guide tube bypass flow in these assemblies. The remaining 45 core locations are instrument tube locations. In these locations, the peripheral guide tubes also project a short distance into close fitting sockets in the upper alignment plate. The center guide tube contains instrumentation which produces a flow annulus which in turn reduces the flow in the center guide tubes.

The guide tube model employed in the flow and heating calculations uses loss coefficients to determine the guide tube flow path hydraulic losses. The core pressure drop at rated power and flow is employed as the driving force for flow through the guide tube. The model permits calculation of the guide tube configurations described above. The guide tube thermal model includes the effects of coolant heating by gamma deposition and neutron deceleration. The effects of heating due to neutron absorption and gamma deposition in the inserted control rod are evaluated. Heat transfer through the guide tube wall to the coolant in the surrounding assembly is accounted for in the model.

Calculations were performed to assess the maximum expected guide tube bypass flow (Reference 3.5-6). At hot full power (HFP), ARO configuration was selected as that resulting in the greatest bypass flow. The total core bypass flow, including flow through the guide tubes in this

instance, was determined to be less than 4.0 percent of vessel flow. The result confirms that guide tube bypass flow does not unduly reduce core flow.

To assess the adequacy of guide tube cooling, a simulation was also performed for a single assembly with the CEA fully inserted at HFP conditions. The fully inserted CEA fingers substantially increase the hydraulic resistance in the guide tube, and also represent a significant heat source. The exit coolant temperature is well below saturation. Heat transfer through the guide tube wall provides a significant part of the cooling.

Based on the results described above, it is concluded that ample guide tube cooling is afforded by the current design, and that bypass flow remains within acceptable limits.

3.5.3.3.3 Control Element Assembly Insertion Time Analysis

A large data base of CEA insertion time measurements has been obtained at a CE plant similar to Millstone Unit 2, with fuel identical in pertinent guide tube design characteristics to the Millstone Unit 2 AREVA reload fuel. The measurements span a time period during which reload quantities of AREVA fuel resided in the core. Statistical analysis (Reference 3.5-6) of this data indicates that the CEA 90 percent insertion time is equal to or less than 2.5 seconds, which is well below the maximum acceptable 90 percent insertion time of 2.75 seconds specified in the Technical Specifications.

Over 500 CEA insertion time measurements from nine different tests were analyzed. The measurements reflect the time required to reach 90 percent insertion from interruption of power to the CEA drive mechanism. Approximately six standard deviations separate the mean of the measured CEA insertion time data from the 2.75 second maximum allowable for Millstone Unit 2.

With over 500 data points, higher order statistics may also be applied to the data to conclude that the rod drop time will be equal to or less than the greatest time measured in the tests with a probability of 99 percent at a 99 percent confidence level. The greatest rod drop time in the tests, as noted above, was 2.50 seconds. The AREVA assemblies are, therefore, expected to conform to the maximum CEA 90 percent insertion time of 2.75 seconds with a substantial margin.

3.5.3.3.4 Fuel Assembly Liftoff

The hydraulic lift force on the fuel assembly was calculated (Reference 3.5-6) using the drag coefficient for a 14 by 14 fuel assembly with bimetallic spacer grids. This value differed slightly for Reload Batches M, N, and P (Cycles 10, 11, and 12). The replacement of a bimetallic spacer with a debris resistant Inconel HTP spacer increased the drag while the thermal rounding of the leading edges of the remaining bimetallic spacers decreased the drag. The overall effect was a slight increase in drag force. The total of the buoyancy and hydraulic lift forces was calculated to be 1194 pounds. The assembly weight and spring force totals 1801 lbs, thus providing a 607 pound holddown margin. This margin, which is more than half of the worst case steady state lift force, will envelope any minor variation due to the spacer modifications. It will also provide holddown during and after a 20% pump overspeed resulting in a 44% lift force increase. For

Reload Batch R (Cycle 13) and Batch S, the fuel assembly weight increased by approximately 40 pounds and a bimetallic spacer replaced the Inconel HTP spacer, increasing the margin to liftoff. A similar analysis was performed for the Reload Batch T design. The use of HMP spacers beginning with Reload Batch Y has a negligible effect on lift. The introduction of the AREVA Standard CE14 HTP fuel with M5 cladding in Reload Batch EE has a negligible effect on lift. The AREVA Standard CE14 HTP fuel with M5 cladding is hydraulically equivalent to the HTP fuel with Zircaloy-4 cladding (Reload Batches Y through DD fuel).

The maximum shear stress of 84,062 psi in the holddown springs occurs in the cold reactor condition. This is below the design criterion of 100,000 psi. The stress at reactor operating conditions is 74,188 psi, which is below the criterion of 90,000 psi at operating temperature.

Irradiation may cause some stress relaxation of the Inconel X-750 holddown springs while causing irradiation induced growth of the fuel assemblies. The assembly growth results in higher spring deflection which offsets any radiation induced relaxation of the springs. The springs are partially shrouded in spring cups, which minimize flow-induced vibration of the springs and prevent potential fretting wear.

3.5.4 TESTS AND INSPECTIONS

3.5.4.1 Reactor Testing

Thermal-hydraulic design criteria are verified during plant startup testing. This is accomplished by measuring the primary intrinsic parameters (e.g., levels, pressures, temperatures, flows, neutron fluence and differential pressures) and calculating the non-measurable and extrinsic parameters (e.g., power level, core peaking factors). During the operating cycle, various thermal-hydraulic parameters are periodically monitored to ensure compliance with the Technical Specifications.

3.5.4.2 AREVA DNB and Hydraulic Testing

3.5.4.2.1 DNB Testing

Details of the testing supporting the HTP DNB correlation are contained in Reference 3.5-3.

3.5.4.2.2 Fuel Assembly Hydraulic Testing

Single-phase hydraulic characteristics of the AREVA Millstone Unit 2 fuel assembly were experimentally determined by hydraulic tests (Reference 3.5-5) performed in AREVA's Portable Hydraulic Test Facility (PHTF).

The pressure drop testing characterized the component loss/flow coefficients of the lower tie plate (including the inlet hardware), spacers, and the upper tie plate (including the exit hardware).

Differential pressure measurements were taken over a range of Reynolds Numbers (N_{Re}). These data were used to drive empirical relationships, which describe the single-phase pressure drops of the Millstone Unit 2 fuel assembly and its components.

These test data from Reference 3.5-5 were used to calculate the Batch M, N, and P lower tie plate, spacer, and upper tie plate pressure drop coefficients, and the bare rod friction factor. Additional test data and analyses were used to determine the Batch R lower tie plate pressure drop coefficient correlations. The loss/flow coefficients derived from these tests and calculations are all referenced to the bare rod Reynolds Number.

3.5.5 REFERENCES

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3.5-2 XN-NF-75-21(P)(A), Revision 2, “XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation,” January 1986.

3.5-3 EMF-92-153(P)(A) Rev. 1, “HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, January 2005.

3.5-4 XN-75-32(P)(A), Supplements 1, 2, 3, and 4, “Computational Procedure for Evaluating Fuel Rod Bowing,” October 1983.

3.5-5 ANF-89-018(P), “Single-Phase Hydraulic Flow Test of ANF Millstone-2 Fuel Assembly,” January 1989.

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TABLE 3.5-1 NOMINAL REACTOR AND FUEL DESIGN PARAMETERS

Design and Operating Parameters	Value
Core Rated Power	2700 MWt
Fraction of Heat Generated in Fuel	0.975
Primary System Pressure	2250 psia
Core Inlet Temperature	549°F
Reactor Coolant Flow (Minimum)	360,000 gpm [a]
Assembly Pitch	8.18 inches
Bypass Flow Fraction (Best Estimate)	0.0303
Average Linear Heat Rate	6.206 kW/ft
Total Number of Assemblies	217

a. Flow reductions to 349,200 gpm are compensated for by reductions in the F_r^T and linear heat rate limits.

Fuel Parameters

Design and Operating Parameters	Value
Fuel Rod OD	0.440 inches
Guide Tube OD	1.115 inches
Rod Array	14 by 14
Rod Pitch	0.580 inches
Number of Fuel Rods/Assembly	176
Number of Guide Tubes/Assembly	5
Active Fuel Length	136.7 inches
Total Fuel Rod Assembly Length	146.67 inches (Batch EE and beyond)
Number of Spacers	9

TABLE 3.5-2 DESIGN OPERATING HYDRAULIC LOADS ON VESSEL INTERNALS

Component	Load Description	Load Value
Core Support Barre	Radial pressure differential directed inward opposite inlet duct	40 psi
Core Support Barrel and Upper Guide Structure	Uplift load	480,000 pounds
Flow Skirt	Radial pressure differential directed inward	6.0 psi average, 10.2 psi maximum, over 40° sector
Bottom Plate	Pressure differential load directed upward	43,400 pounds
Core Support Plate	Pressure differential load directed upward	43,100 pounds
Fuel Assembly	Uplift load	1194 lbs at 120% flow
Core Shroud	Radial load directed outward	20.8 psi at bottom, 0.0 psi at top
Upper Guide Structure	Pressure differential load directed upward	148,000 pounds
Fuel Alignment Plate	Pressure differential load directed upward	89,600 pounds
Upper Guide Plate	Pressure differential load directed downward	132,000 pounds
CEA Shrouds	Lateral drag load	4,200 pounds (dual CEA) 1,100 pounds (single CEA)

TABLE 3.5-3 UNCERTAINTY SOURCES FOR DNBR CALCULATIONS (DELETED)

3.A ANALYSIS OF REACTOR VESSEL INTERNALS

3.A.1 SEISMIC ANALYSIS

3.A.1.1 Introduction

Dynamic analyses of the reactor vessel internals for both horizontal and vertical seismic excitation were conducted to provide further bases for assessing the adequacy of their seismic design. These analyses were performed in conjunction with the dynamic seismic analyses of the reactor coolant system (RCS) which is discussed in Appendix 4.A. The following paragraphs provide a discussion of the analytical procedures used for the reactor internals, including a description of the mathematical models. Significant results are listed and compared to the results obtained from application of the design loads.

3.A.1.2 Method of Analysis

3.A.1.2.1 General

The procedure used in conducting the seismic analysis of the reactor internals consisted basically of three steps. The first step involved the formulation of a mathematical model. The natural frequencies and mode shape of the model were determined during the second step. The response of the model to the seismic excitation was determined in the third step. In this analysis, the horizontal and vertical components of the seismic excitation were considered separately and the maximum responses added to obtain conservative results.

3.A.1.2.2 Mathematical Models

For the dynamic analysis of the reactor internals, equivalent multi-mass mathematical models were developed to represent the system. Since the seismic input excitation of the reactor internals was obtained in the form of acceleration time history of the reactor vessel flange, only the internals are included in the model. The coupling effect of the internals' response on the vessel flange acceleration was accounted for by including a simplified representation of the reactor internals with the model of the RCS. This is discussed in Appendix 4.A. Since the horizontal and vertical responses were treated as uncoupled, separate horizontal and vertical models were developed to more efficiently account for the structural differences in these directions. A sketch of the internals showing the relative node locations for the horizontal model is presented in Figure 3.A–1. Figures 3.A–2 and 3.A–3 show the idealized horizontal and vertical models. Since the structural details provide for no vertical load transfer between the upper guide structure (UGS) and core or core shroud, the vertical response of the UGS is independent of the rest of the internals. Consequently, the vertical model was divided into two submodels. Model I consists of the core support barrel/thermal shield (CSB/TS), lower support structure, core shroud and core mass; Model II consists of the UGS.

The mathematical models of the internals are constructed in terms of lumped masses and elastic beam elements. At appropriate locations within the internals, points (nodes) are chosen to lump the weights of the structure. Between these nodes, properties are calculated for moments of

inertia, cross-section areas, effective shear areas, and lengths. The salient details of the models are discussed below.

3.A.1.2.2.1 Hydrodynamic Effects

The dynamic analysis of reactor internals presents some special problems due to their immersion in a confined fluid. It has been shown both analytically and experimentally (Reference 3.A-1) that immersion of a body in a dense fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to that in air. The effect is more pronounced where the confining boundaries of the fluid are in close proximity to the vibrating body as is the case for the reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been to ascribe to the system additional or "hydrodynamic mass."

This "hydrodynamic mass" decreases the frequencies of the system, but is not directly involved in the inertia force effects. The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary.

Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 3.A-1 and 3.A-2. The results of these references are applied to the internals structures to obtain the total (structural plus hydrodynamic) mass matrix which was then used in the evaluation of the natural frequencies and mode shapes for the model.

3.A.1.2.2.2 Fuel Assemblies

For the horizontal model, the fuel assemblies are treated as vibrating in unison. The member properties for the beam elements representing the fuel assemblies were derived from the results of experimental tests of the fuel assembly load deflection characteristics and natural frequency.

3.A.1.2.2.3 Core Support Barrel Flanges

To obtain accurate lateral and vertical stiffnesses of the upper and lower flanges, finite element analyses of these two regions were performed. As shown in Figures 3.A–4 and 3.A–5, the flanges were modeled with quadrilateral and triangular ring elements. Asymmetric loads, equivalent to lateral shear loads and bending moments, and symmetric axial loads were applied and the resulting displacements calculated. These results were then used to derive the equivalent member properties for the flanges.

3.A.1.2.2.4 Control Element Assembly Shrouds

For the horizontal model, the control element assembly (CEA) shrouds are treated as vibrating in unison and are modeled as guided cantilever beams in parallel. To account for the decreased lateral stiffness of the UGS due to local bending of the fuel alignment plate, a short member with properties approximating the local bending stiffness of the fuel alignment plate is included at the bottom of the CEA shrouds. Since the stiffness of the UGS support plate is large compared to that of the shrouds, the CEA shrouds are assumed to be rigidly connected to the UGS support plate.

3.A.1.2.2.5 Thermal Shield Supports

For the horizontal model, the thermal shield supports are modeled as horizontal members. The member properties of the beam elements representing the positioning pins were based on the radial stiffness of the circumferential set of pins. Likewise, the properties of the beam member representing the support lugs were based on the tangential stiffness of the circumferential set of lugs. For the vertical model, the equivalent cross-section area of the bar element representing the support lugs was based on the axial bending stiffness of the circumferential set of lugs. For both the horizontal and vertical models, the stiffness of the thermal shield supports includes the effect of local deformation of the core support barrel.

3.A.1.2.2.6 Upper Guide Structure Support Plate and Lower Support Structure Grid Beams

These grid beam structures were modeled as plane grids. Displacements due to vertical (out of plane) loads applied at the beam junctions were calculated through the use of the STRUDL computer code (Reference 3.A-3). Average stiffness values based on these results yielded equivalent member cross-section areas for the vertical model.

3.A.1.2.3 Natural Frequencies and Normal Modes

The mass and beam element properties of the models were utilized in STAR, a computer program from the MRI/STARDYNE Analysis System programs (Reference 3.A-4) to obtain the natural frequencies and mode shapes. This system utilizes the “stiffness matrix” method of structural analysis. The natural frequencies and mode shapes are extracted from the system of equations.

$$[\underline{K}-W_n^2\,\underline{M}]\underline{\phi}_n = 0$$

where:

$\underline{K}$ = Model stiffness matrix
$\underline{M}$ = Model mass matrix
W_n = Natural circular frequency for the nth mode
$\underline{\phi}_n$ = Normal mode shape matrix for n^{th} mode

The mass matrix, $\underline{M}$, includes the hydrodynamic and structural masses.

The natural frequencies and mode shapes calculated for the first 3 modes for the horizontal model are presented in Figures 3.A–6 through 3.A–8. The natural frequencies calculated for the vertical model are presented in Table 3.A-1. The modal data shown is typical and is presented for illustrative purposes. The effect of additional higher modes was included in the response analyses.

3.A.1.2.4 Response Calculations

3.A.1.2.4.1 Horizontal Direction

The time history analysis technique was utilized to obtain the response of the internals for the horizontal seismic excitation. The horizontal excitation was specified as the acceleration time history of the reactor vessel flange, resulting from the operational basis earthquake (OBE) (OBE = 0.09g ground acceleration). The flange excitation resulting from the design basis earthquake (DBE) (DBE = 0.17g ground acceleration) was conservatively specified as 0.17/0.09 times that for the OBE.

The time history response analysis was performed utilizing the MRI STARDYNE System/ DYNRE 1 Computer Program. This program utilizes the “Normal Mode Method” to obtain time history response of linear elastic structure. Details of the program and the “Normal Mode Method” are presented in References 3.A-4, 3.A-5 and 3.A-6.

Input to DYNRE 1 consisted of the modal data as determined in Section 3A.2.3, the modal damping factors, and the forcing function time history. This analysis used the modal data for all modes with frequencies below 100 cps. This included the first 14 modes. Contributions from higher modes are negligible.

The modal damping factors were obtained by the method of “Mass Mode Weighting” which gives:

$$\beta_n = \frac{\Sigma M_i |\phi_{in}| \beta_i}{\Sigma M_i |\phi_{in}|}$$

where:

β_n = Modal damping factor
M_i = Structural mass of mass node i
$|\phi_i|$ = Absolute value of the mode shape as mass mode i
β_i = Damping associated with pass point i

The damping factor assigned to the nodes representing the fuel assemblies was 5 percent. This is a conservative value derived from proprietary experimental results. A value of 1 percent was used for the other nodes.

The output from the DYNRE 1 code consists of the nodal displacement, velocity, and acceleration time history relative to the base. The member bending moments and shears were obtained from the STAR code (Reference 3.A-5) and were derived from the DYNRE 1 nodal displacement vectors at the times of peak response.

3.A.1.2.4.2 Vertical Direction

The response of the reactor internals to the vertical excitation was obtained by the response spectrum technique. Because of the high natural frequencies and resulting low levels of responses for the vertical direction, the more conservative spectrum response analysis results were used instead of time history results. The response spectrum utilized was derived from the vertical acceleration time history at the reactor vessel flange. The spectrum curve is presented in Figure 3.A–9.

An acceleration level corresponding to the natural frequency of each mode was selected from the spectrum curve. The response spectrum technique uses these acceleration values to determine the inertia forces, accelerations, and displacements of each mode. The results for each mode were conservatively combined on the basis of absolute values. For the vertical models, the first seven modes were included in the results.

3.A.1.3 Results

Combined results for the horizontal and vertical dynamic seismic analyses are presented in Table 3.A-2 in terms of stresses at critical locations in the reactor internals for the DBE. Table 3.A-2 also lists the seismic stresses which result from application of the design loads specified for the DBE. A comparison shows the results of the dynamic analysis to be less severe.

3.A.1.4 Conclusion

It is concluded that the seismic loads specified for the design of the internals are adequate. All seismic loads calculated by the dynamic seismic analysis are less than the design loads specified by the DBE.

3.A.2 NORMAL OPERATING ANALYSIS

Design analyses were performed on the reactor internals for normal operating conditions to demonstrate that the mechanical design bases were satisfied. These design calculations included appropriate vibration analyses of the component assemblies. The flow induced vibration of the CSB/TS, during normal operation, was characterized as a forced response to deterministic and random pressure fluctuations in the coolant. Methods were developed for predicting the response of components to the hydraulic forcing functions.

Emphasis was placed on analysis and design of those components which were particularly critical and susceptible to vibratory excitation, such as the thermal shield. Using a top supported, as opposed to a bottom supported, thermal shield design improves stability as it eliminates a free edge in the flow path. Increasing the number of upper supports and lower jackscrews, in the specific manner chosen, provides a much stiffer structure and the use of an all-welded shield eliminates local flexibilities and relative motion at bolted joints. Analytical studies show the thermal shield to be stable on its support system when exposed to the axial annular flow encountered during normal operation. The snubber design is based upon limiting the motion of the core support barrel under conditions of hydraulically induced vibrations. The snubbers are at

the position of maximum amplitude for the fundamental lateral bending mode of the barrel, thereby restricting motion of the barrel at the most efficient position. The circumferential distribution of snubbers assures restraint regardless of the direction of response.

The random hydraulic forcing function was developed by analytical and experimental methods. An analytical expression was developed to define the turbulent pressure fluctuation for fully developed flow. This expression was modified, based upon the result of scale model testing, to account for the fact that flow in the downcomer was not fully developed. Based upon test results, an expression was developed to define the spatial dependency of the turbulent pressure fluctuations. In addition, experimentally adjusted analytical expressions were developed to define; the peak value of the pressure spectral density associated with the turbulence and; the maximum area of coherence, in terms of the boundary layer displacement, across which the random pressure fluctuations are in phase.

The natural frequencies and mode shapes of the CSB/TS system were obtained using the axisymmetric shell finite element computer program, ASHSD (Reference 3.A-7). This computer program is capable of obtaining natural frequencies and mode shapes of complex axisymmetric shells; e.g., arbitrary meridional shape, varying thickness, branches, multi-materials, orthotropic material properties, etc. To employ the ASHSD code, the CSB/TS were modeled as a series of conical shell frustrums joined at their nodal point circles. The length of each element, throughout the ASHSD model, was a fraction of the shell decay length. Since rapid changes in the stress pattern occur in regions of structural discontinuity, the nodal point circles were more closely spaced in such regions. The finite element model of the CSB/TS system included representation of the core support barrel upper and lower flanges, sections of different wall thickness, and thermal shield support lugs and jackscrews. Elements with orthotropic material properties were utilized to provide equivalent axisymmetric models of the structural stiffness and constraints to relative motion between the core support barrel and thermal shield provided by the thermal shield support lugs and jackscrews. Those modes which reflect the mass of the lower support structure, core shroud and fuel were simulated by the addition of concentrated masses at specific nodes in the core support barrel flange finite element model.

Applying Hamilton’s Variational Principle to the conical shell elements an equation of motion was formulated for each degree of freedom of the system. An inverse iteration technique was utilized in the program to obtain solutions to the characteristic equation, which was based on a diagonalized form of a consistent mass matrix and stiffness matrix developed using the finite element method. Four degrees of freedom — radial displacement, circumferential displacement, vertical displacement, and meridional rotation — were taken into account in the analysis, giving rise to coupled mode shapes and corresponding frequencies. Evaluation of the reduction of these frequencies for the system immersed in coolant was made by means of the “virtual mass” method outlined in Reference 3.A-2.

The random response analysis considers the response of the CSB/TS system to the turbulent downcomer flow during steady-state operation. The random forcing function is assumed to be a wide-band stationary random process with a pressure spectral density equal to the peak value associated with the turbulence. The rms vibration level of the CSB/TS system was obtained based upon a damped, single degree of freedom analysis assuming the rms random pressure fluctuation

to be spatially invariant. The analysis demonstrates that the anticipated rms response of the CSB⁄TS system is low. Snubber loads were derived using an analytical technique originally developed by a Combustion Engineering (CE) consultant using the random loads discussed above. Modeling the reactor vessel snubbers and core support barrel system as a single degree of freedom spring-mass system, the number and magnitude of snubber, core support barrel impacts was calculated based upon the response of the system to random excitation. The snubbers were designed, based upon this loading requirement, to meet the cyclic strength requirements specified in Section III of the ASME Boiler and Pressure Vessel Code.

The forced response of the reactor internals to deterministic loading was evaluated by classical analytical methods, using lumped mass and continuous elastic structural models. These calculated responses were used to verify the structural integrity of the reactor vessel internals to normal operating vibratory excitation. Components were design analyzed to assure that there were no adverse effects from dominant excitation frequencies, such as pump rotational and blade passing frequencies.

3.A.3 LOSS OF COOLANT ACCIDENT ANALYSIS

3.A.3.1 Discussion

A dynamic analysis (Reference 3.A-8) has been performed to determine the structural response of the reactor vessel internals to the transient loss of coolant accident (LOCA) loading. The analysis determined the shell, beam and rigid body motions of the internals using established computerized structural response analyses. The finite-element computer code, ASHSD (Reference 3.A-7) was used to calculate the time-dependent beam and shell response of the CSB/TS system to the transient LOCA loading. The finite-element computer code SAMMSOR-DYNASOR (Reference 3.A-9) was used to evaluate the core support barrel's potential for buckling when loaded by a net external radial pressure resulting from an outlet line break. The structural response of the reactor internals to vertical and transverse loads resulting from inlet and outlet breaks, was determined using the spring-mass computer code, SHOCK (Reference 3.A-10).

The time and space dependent pressure loads used in the above analysis were the result of a detailed hydraulic blowdown analysis. The pressure fluctuations were determined for each node in the hydraulic model for inlet and outlet line breaks. The pressure time histories at these nodal locations were then decomposed into the Fourier harmonics which define the circumferential pressure distribution at the nodal elevations. Where the hydraulic model nodes did not correspond to those of the structural model, the hydraulic model pressure components were interpolated to provide the required loading information.

The finite element computer code, ASHSD, was used to calculate the dynamic response of the CSB/TS to transient LOCA loading resulting from an inlet break. To employ the ASHSD code, the CSB/TS were modeled as a series of conical shell frustrums (elements) joined at their nodal point circles. Applying Hamilton's Variational Principle to the conical shell elements a damped equation of motion was formulated for each degree of freedom of the system. Four degrees of freedom — radial displacement, circumferential displacement, vertical displacement and meridional rotation — were taken into account in the analysis, giving rise to coupled modes. The

differential equations of motions were solved numerically using a step integration procedure. To ensure computational stability of the numerical solution, the integration time step was chosen such that it is small compared to the shortest period of the finite element system. The model developed for the CSB/TS system is shown in Figure 3.A–10. The length of each element, throughout the analytical model, was a fraction of the shell decay length. Since rapid changes in the stress pattern occur in regions of structural discontinuity, the nodal point circles were more closely spaced in such regions. The finite element model of the CSB/TS system included representation of the core support barrel upper and lower flanges, sections of different wall thickness, and thermal shield support lugs and jackscrews. Elements with orthotropic material properties were utilized to provide equivalent axisymmetric models of the structural stiffness and constraints to relative motion between the core support barrel and thermal shield provided by the thermal shield support lugs and jackscrews. Those modes which reflect the mass of the lower support structure, core shroud and fuel were stimulated by the addition of concentrated masses at specific nodes in the core support barrel flange finite element model.

In performing the dynamic analysis of the CSB/TS system, the transient load harmonics were applied in two successive phases to account for time-dependent boundary conditions at the snubbers. The first phase used those harmonics which excite the beam modes, whereas the second phase used those harmonics which excite the shell modes. During the first phase, the lower end of the core support barrel was unrestrained. Within a very few milliseconds, the clearances between the core support barrel and reactor vessel snubbers were closed and for the remainder of the LOCA transient, the core support barrel was restrained radially at the snubber level. Transient responses were computed throughout each loading phase.

The ASHSD code computed the nodal point displacement, resultant shell forces, shell stresses and maximum principle stresses as functions of time. The maximum principle stresses at the internal and external surfaces of the CSB/TS were determined from the bending and membrane components during each phase of transient loading. Stress intensity levels calculated from the principle stresses were combined with normal operating and seismic induced stresses for comparison with design criteria.

Accurate representation and analysis of the CSB/TS shell structures was obtained through use of the finite element code ASHSD. Accurate representation of the remainder of the internals (i.e., fuel, core shroud, CEAs, UGS, lower support structure, etc.) was obtained using the SHOCK code.

The SHOCK code determines the response of structures which are represented as lumped-mass systems and subjected to arbitrary loading functions. The code solves the differential equations of motion for each mass by a numerical step-integration procedure. The lumped mass model can represent a vertically or laterally responding system subject to arbitrary loading functions and initial conditions. Options are available for describing steady state loads, preloads, input accelerations, linear and nonlinear springs (including tension and compression only springs) gaps, and structural and viscous damping.

The reactor internals were developed in terms of a spring-mass system for both vertical and lateral directions; see Figures 3.A–11 and 3.A–12. For both models, the spring rates were generally

evaluated using strength of material techniques. However, in complex areas such as at the core support barrel flanges and UGS support flange, the stiffness was derived from finite element model analyses. The lumped mass weights were generally based upon the mass distribution of the uniform support structures, but included at appropriate nodes, local masses such as snubber blocks, fuel end fittings, thermal shield lugs, etc. The net result was a lumped-mass system having the same distribution of mass as the actual structure. To simulate the effect associated with the internals oscillating laterally in the water filled vessel, a distributed virtual mass was calculated based upon the procedure outlined in Reference 3.A-8 (which includes the annulus effect) and was added to the structural lumped-mass system, to provide an analytical model with a dynamic response quantitatively similar to the actual internals. In the case of the vertical model, the hydraulic effect is notably one of reducing the effective weight of the reactor internals and this effect was included in the structural lumped-mass system.

The SHOCK code provided excellent facility for modeling clearances, preloads and component interfaces. In the lateral model, the core support barrel, reactor vessel snubber clearance was simulated by a nonlinear spring which accounted for the increased resistance to core support barrel motion when snubbing occurred. In the vertical model, nonlinear springs in the form of compression only springs, were used extensively to simulate preload and interface conditions, such as exist between the UGS support plate and core support barrel upper flange; at the fuel hold-down spring; at the fuel, core support plate interface and at the core shroud, core support plate interface. Tension only springs were used to simulate the effect of the core shroud tie rods.

In both the vertical and lateral SHOCK models, damping was varied throughout the system to simulate structural and hydraulic frictional effects within the reactor internals. The effect of hydraulic drag in the vertical model was simulated by a force time-history applied to the fuel lower end-fitting. Vertical loads were used directly from the detailed hydraulic analysis, whereas lateral loads were obtained by integrating those harmonics which excite the beam modes to obtain the net lateral load on the CSB/TS system.

The SHOCK code calculated the vertical and lateral response of the system in terms of displacements, velocities and accelerations and internal force, moments and shears as related to each model. These quantities were sufficient to permit calculation of membrane and where appropriate bending stresses for comparison with design criteria.

The finite-element code SAMMSOR-DYNASOR was used to determine the dynamic response of the core support barrel, with initially imperfect geometry, to a net external radial pressure resulting from an outlet line break. The above analysis has the capability of determining the nonlinear dynamic response of axisymmetric shells with initial imperfections subjected to arbitrarily varying load configurations.

Since SAMMSOR-DYNASOR is a finite-element program, a model was developed, Figure 3.A–13, of the core support barrel using axisymmetric finite-elements similar to those used for the ASHSD analysis. As was for the ASHSD model, the SAMMSOR-DYNASOR finite-element lengths were considerably less than the decay length of the core support barrel. The boundary condition at the core support barrel flange was considered fixed, whereas at the core support barrel lower flange radial displacements were restrained. These boundary conditions represented

the restraint due to the expansion compensating ring and pressure vessel head at the top and the snubbers and lower support structure at the bottom. For conservatism, the stiffening effects of the fuel alignment plate, core shroud and core support plate were neglected.

Since the basic phenomenon in buckling is nonlinear instability, the initial deviation of the structure from a perfect geometry greatly affects its response. The initial imperfection was applied to the core support barrel by means of a pseudo-load so developed to provide the maximum imperfection over each of the desired number of circumferential harmonics. The actual transient loading in terms of its harmonics was applied to the initially “imperfect” geometry core support barrel and the response obtained for each of the imperfection harmonics for the combined loading harmonics.

3.A.3.2 Analysis Codes

ASHSD (Reference 3.A-7) is a structural finite-element computer code developed at the University of California, Berkeley, and supported in part by the National Science Foundation. It performs dynamic analyses of complex axisymmetric structures subjected to arbitrary dynamic loadings or base accelerations. The frequencies of free vibrations as calculated by ASHSD compare well to those calculated by the equations of Hermann-Mirshy and Flugge, References 3.A-11 and 3.A-12, respectively. The authors also make comparisons with available experimental results (Reference 3.A-13) of free vibrations of cylindrical shells. The resulting comparison is good. Comparison of the numerical solution (Reference 3.A-14) of the dynamic response of a shell to suddenly applied loads and the finite-element (ASHSD) solution of the same problem are in good agreement. The response of a shell to a moving axisymmetric pressure load was evaluated by ASHSD and analytically (Reference 3.A-15) with the results being in good agreement.

SAMMSOR-DYNASOR (Reference 3.A-9) is a finite-element computer code developed at Texas A&M University and supported in part by a NASA grant from the Manned Spacecraft Center, Houston, Texas. This code has the capability of determining the nonlinear dynamic response of axisymmetric shells subjected to arbitrary dynamic loads. Asymmetrical dynamic buckling can be investigated using this program. The program has been extensively tested, using problems the solutions to which have been reported by other researchers, in order to establish the validity of the codes. Among these are a shallow shell with axisymmetric loading as described in Reference 3.A-16. Identical results are obtained with those of Reference 3.A-17 for the analytical evaluation of blast loadings on a cylindrical shell. Calculations made by SAMMSOR-DYNASOR for the symmetric buckling of a shallow spherical cap is in good agreement with the analyses of References 3.A-18 and 3.A-19 and the experimental data of References 3.A-20 and 3.A-21.

SABOR-5 - DRASTIC, (Reference 3.A-22) is a structural finite-element computer code developed at the Aeroelastic and Structures Research Laboratory, Department of Aeronautics at the Massachusetts Institute of Technology. The work was administered by the Air Force Systems Command with technical monitoring by the Aerospace Corp. SABOR 5 - DRASTIC is the end result of combining a finite-difference solution procedure and a finite-element program to permit predicting the transient response of complex shells of revolution which are subjected to arbitrary transient loadings. Comparisons with reliable independent analytical predictions (notably finite-

difference transient response solutions submitted by AVCO) confirm the accuracy and reliability of the SABOR 5 -DRASTIC dynamic response predictions. An experiment and accompanying analysis were performed by the Aerospace Corp. (Reference 3.A-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 3.A-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.

3.A.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 3.A-24.

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

3.A.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 3.A-25 through 3.A-29.) Subsequent to the commission review and approval, weld overlays were applied to dissimilar metal welds (DMWs) at the shutdown cooling, the safety injection and the pressurizer

surge nozzles. A revised LBB analysis was performed for these nozzles (see Reference 3.A-30). 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 excluded from the design basis of the following reactor vessel and reactor internals components:

Core barrel snubbers, core barrel stabilizer blocks
Reactor vessel core support ledge
Reactor Cavity Seal Plate, Neutron Shielding

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TABLE 3.A-1 NATURAL FREQUENCIES FOR VERTICAL SEISMIC ANALYSIS MATHEMATICAL MODEL

Mode Number	Sub-Model I Frequency, cps	Sub-Model II Frequency, cps
1	21.60	72.98
2	67.75	404.09
3	124.59	-

TABLE 3.A-2 SEISMIC STRESSES IN CRITICAL REACTOR INTERNALS COMPONENTS FOR THE DESIGN BASIS EARTHQUAKE

Structural Component	Location	Stress Mode	Design Load Stress	Dynamic Analysis Stress
Core Support Barrel	Upper Section of Barrel	Tension & Bending	1,129 psi	746 psi
Lower Core Support	Beam Flange	Bending	5,278 psi	929 psi
CEA Shrouds: Single	End of Shroud	Tension & Bending	3,548 psi	1,295 psi
CEA Shrouds: Dual	End of Shroud	Tension & Bending	2,762 psi	697 psi
Upper Grid Beams	Center of Beam	Bending	1,652 psi	127 psi
Upper Guide Structure Flange	Junction of Flange & Barrel Cylinder	Tension & Bending	2,823 psi	146 psi

FIGURE 3.A–1 REPRESENTATIVE NODE LOCATIONS - HORIZONTAL MATHEMATICAL MODEL

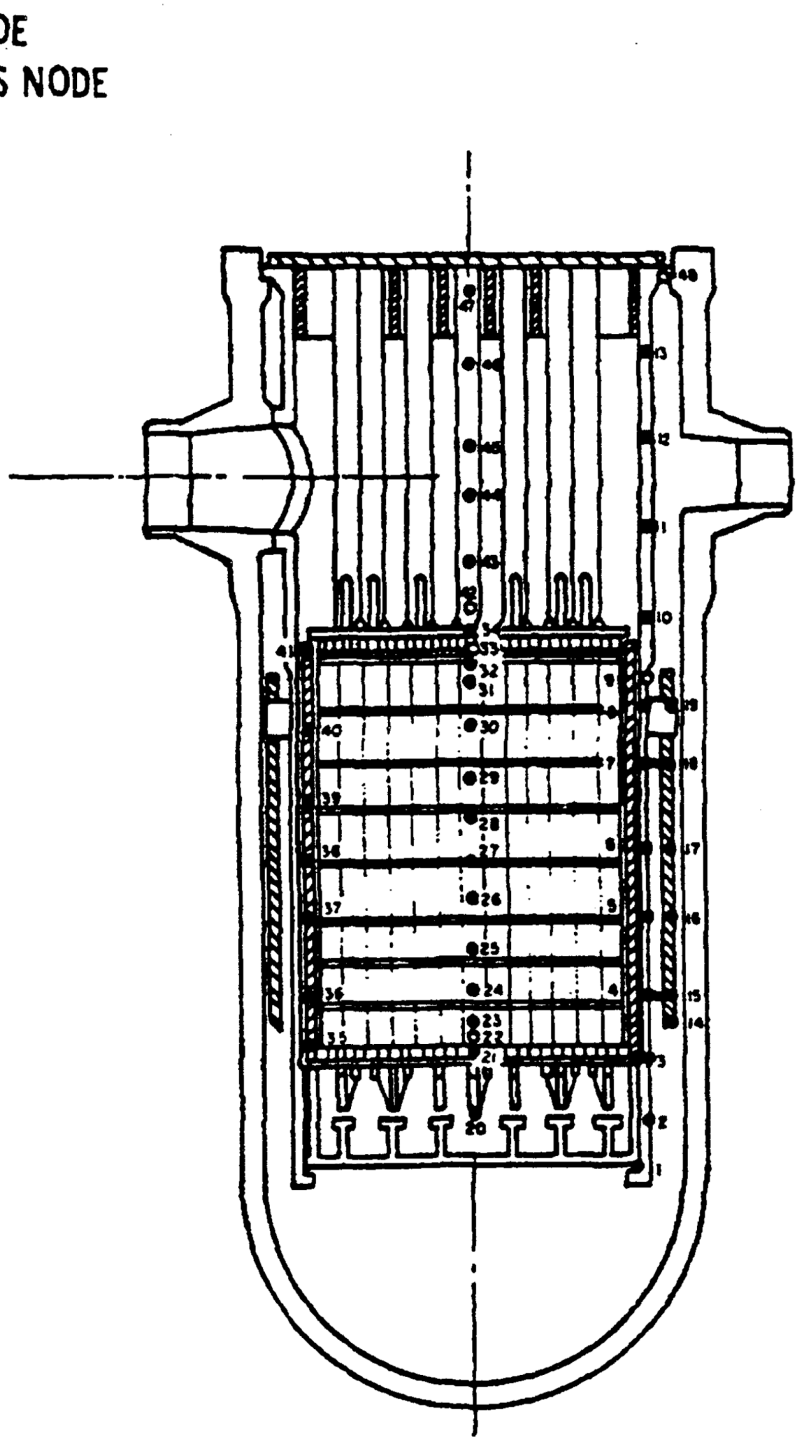


FIGURE 3.A–2 MATHEMATICAL MODEL - HORIZONTAL SEISMIC ANALYSIS

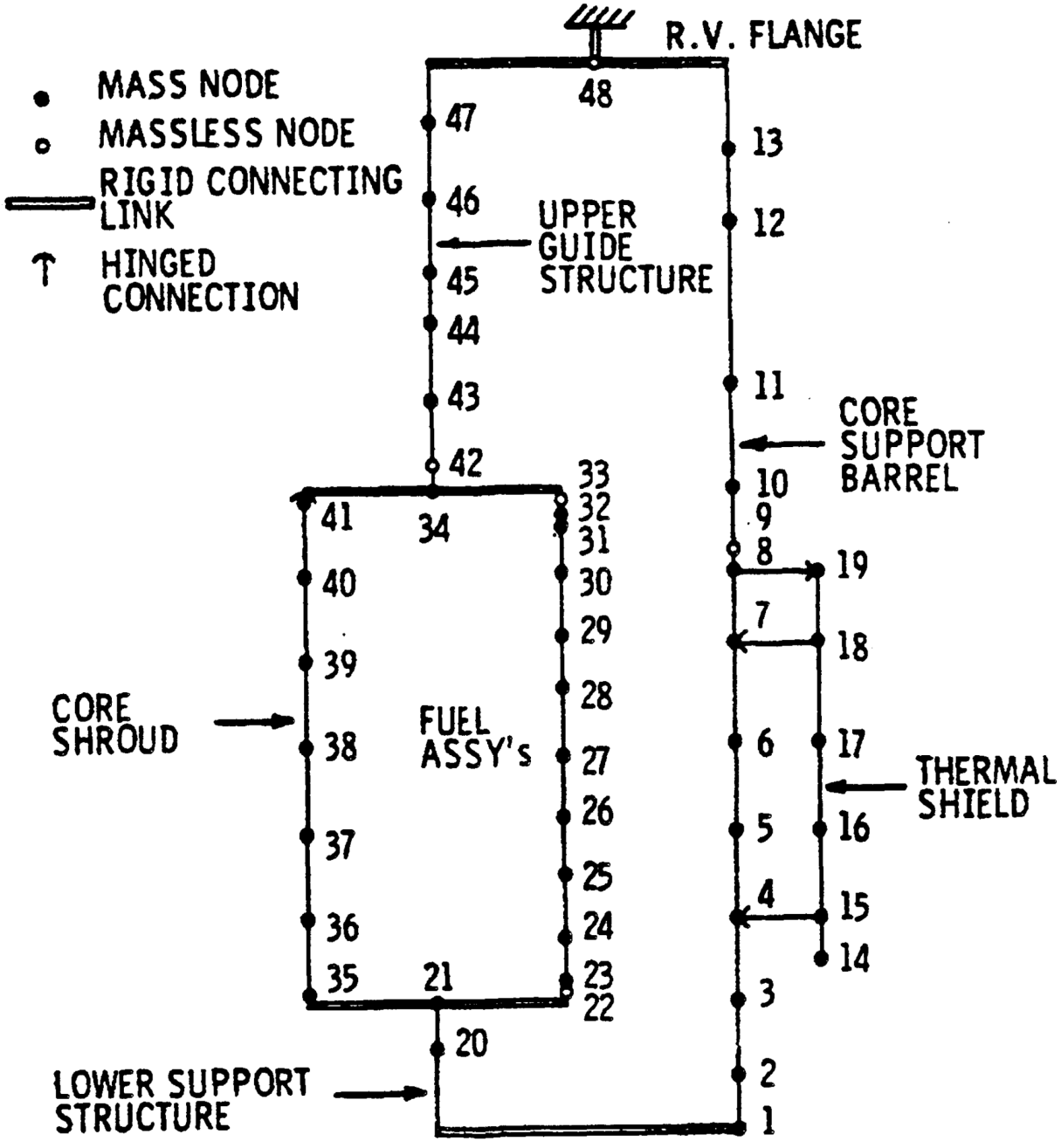


FIGURE 3.A–3 MATHEMATICAL MODEL - VERTICAL SEISMIC ANALYSIS

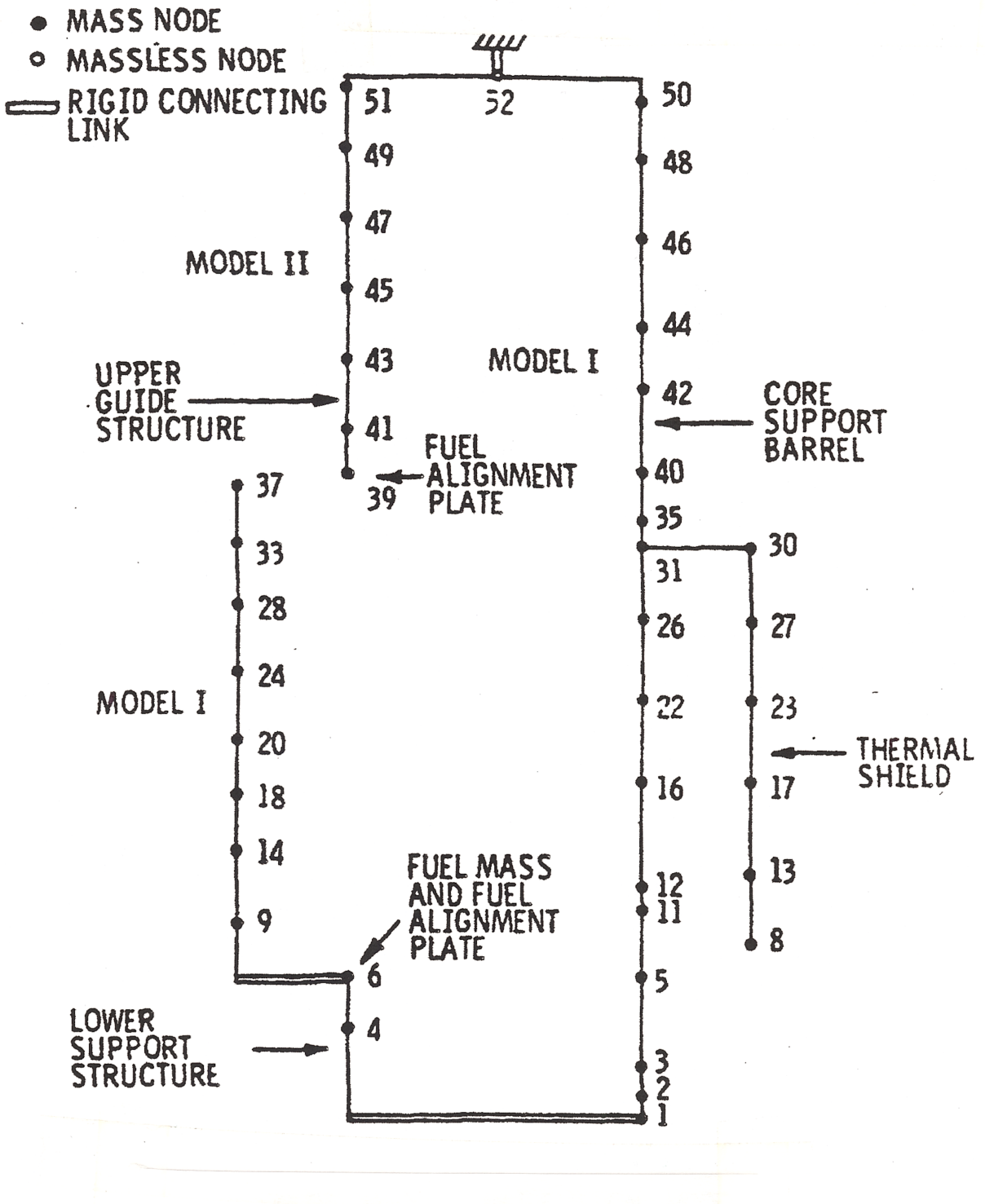


FIGURE 3.A–4 CORE SUPPORT BARREL UPPER FLANGE - FINITE ELEMENT MODEL

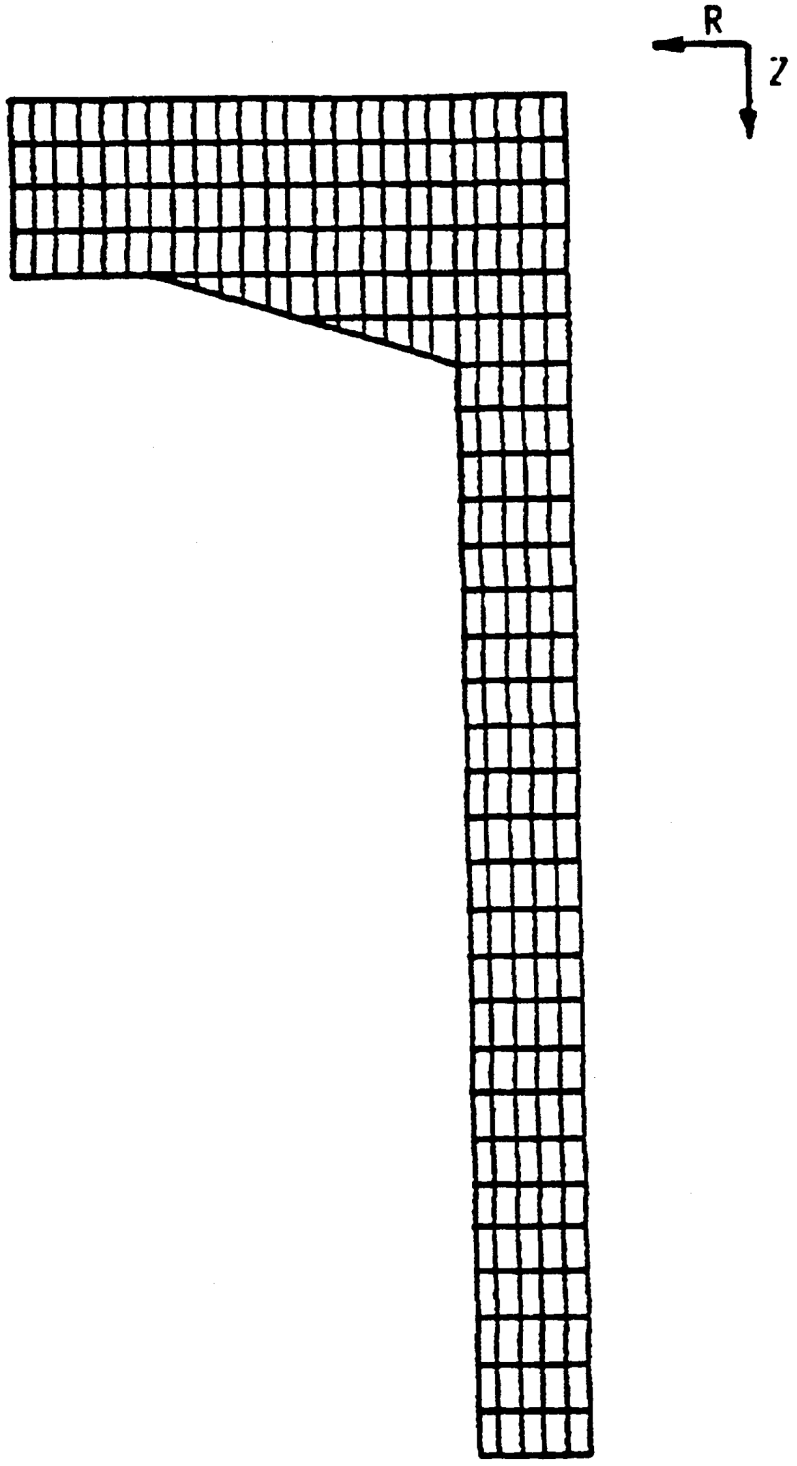


FIGURE 3.A–5 CORE SUPPORT BARREL LOWER FLANGE - FINITE ELEMENT MODEL

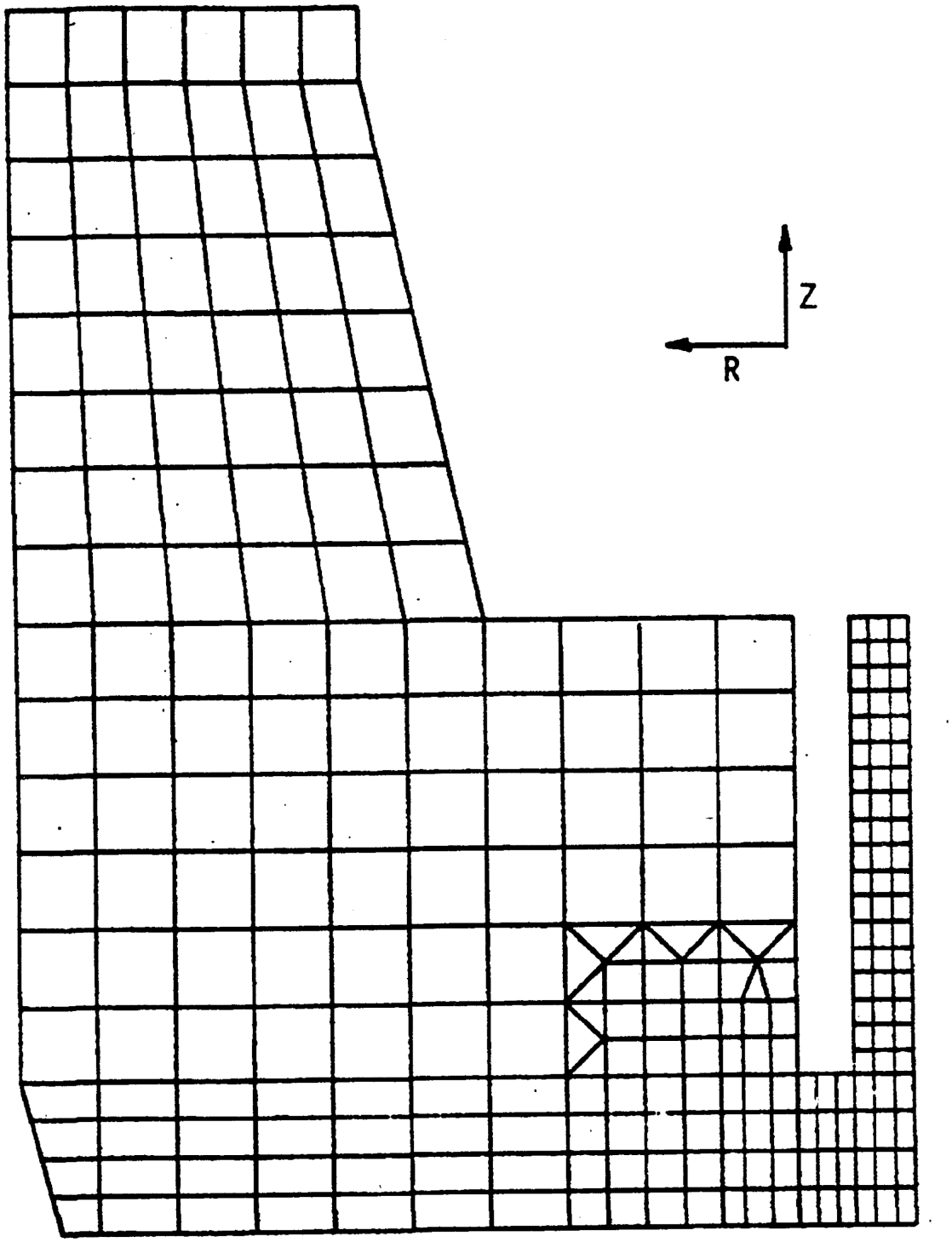


FIGURE 3.A–6 LATERAL SEISMIC MODEL - MODE 1, 3.065 CPS

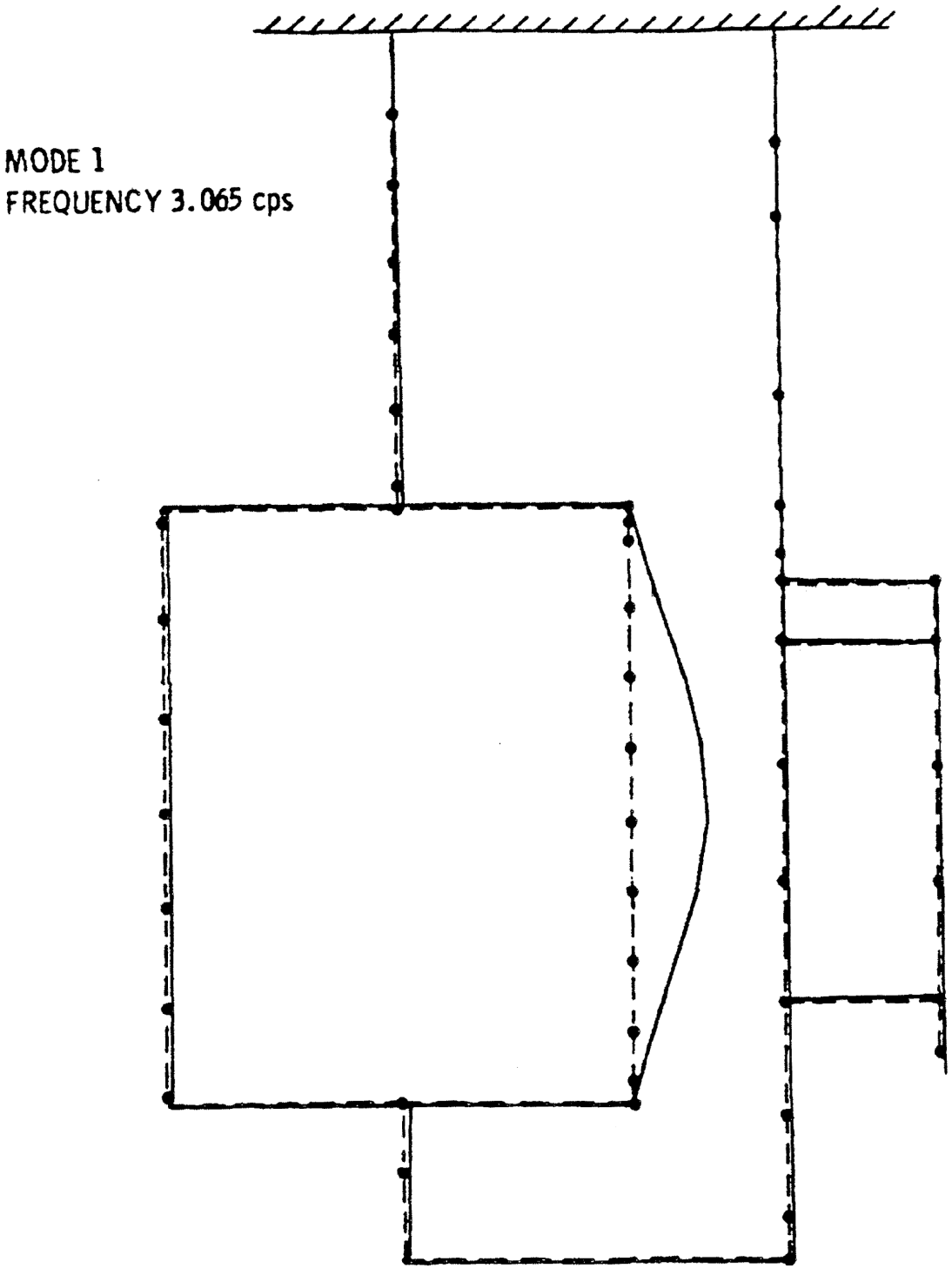


FIGURE 3.A–7 LATERAL SEISMIC MODEL - MODE 2, 5.118 CPS

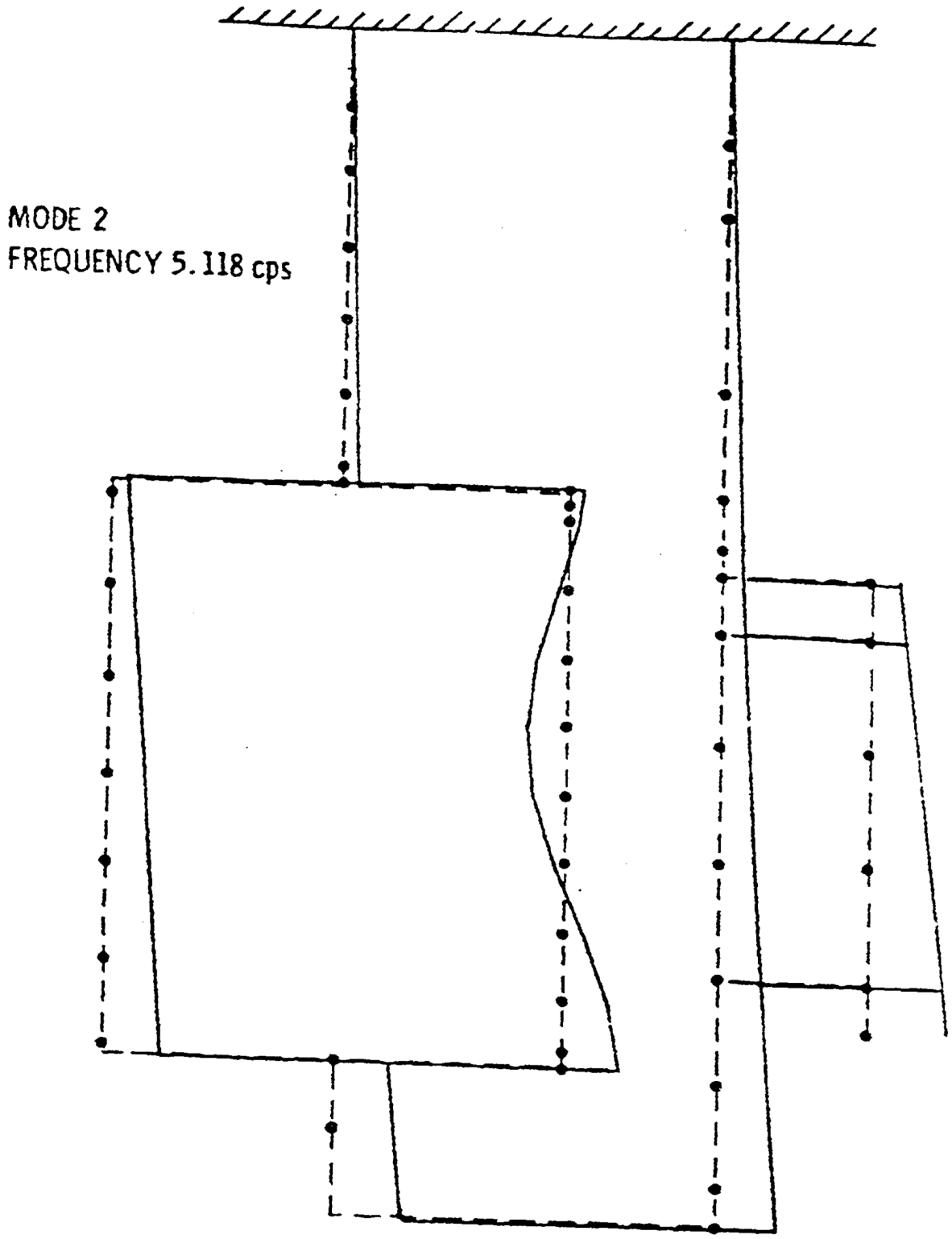
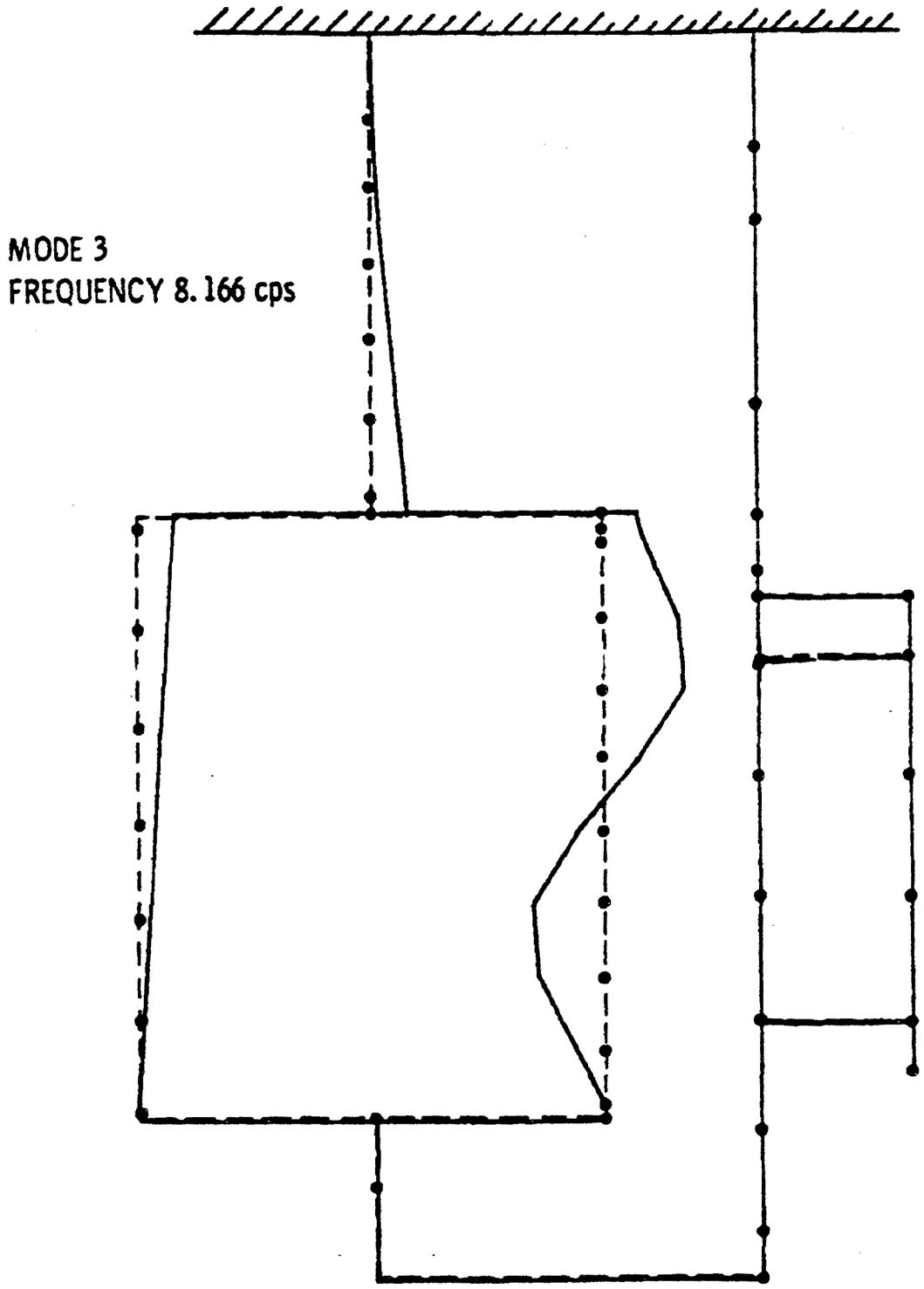


FIGURE 3.A–8 LATERAL SEISMIC MODEL - MODE 2, 5.118 CPS



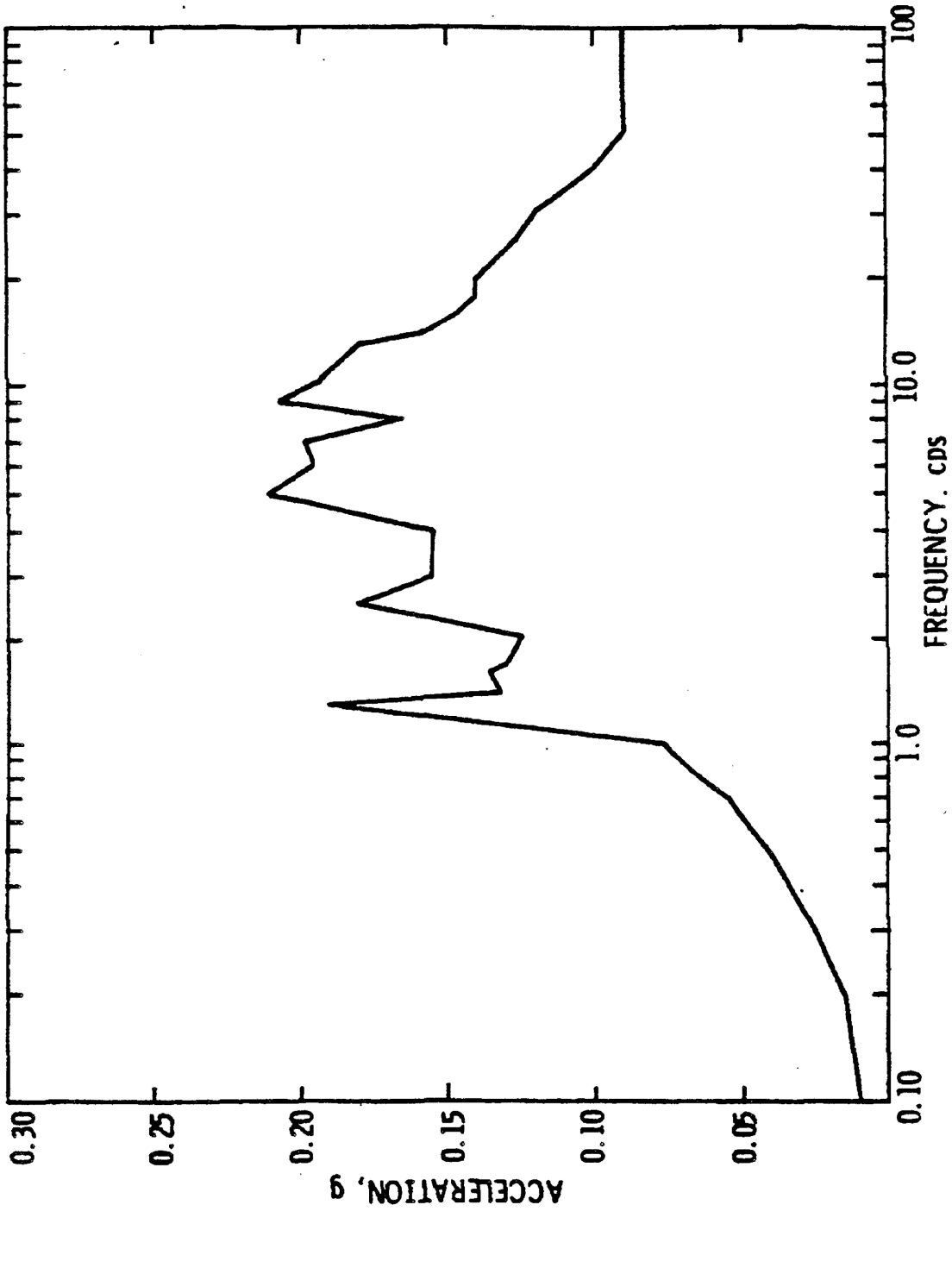


FIGURE 3.A–9 REACTOR VESSEL FLANGE VERTICAL RESPONSE SPECTRUM (1% DAMPING)

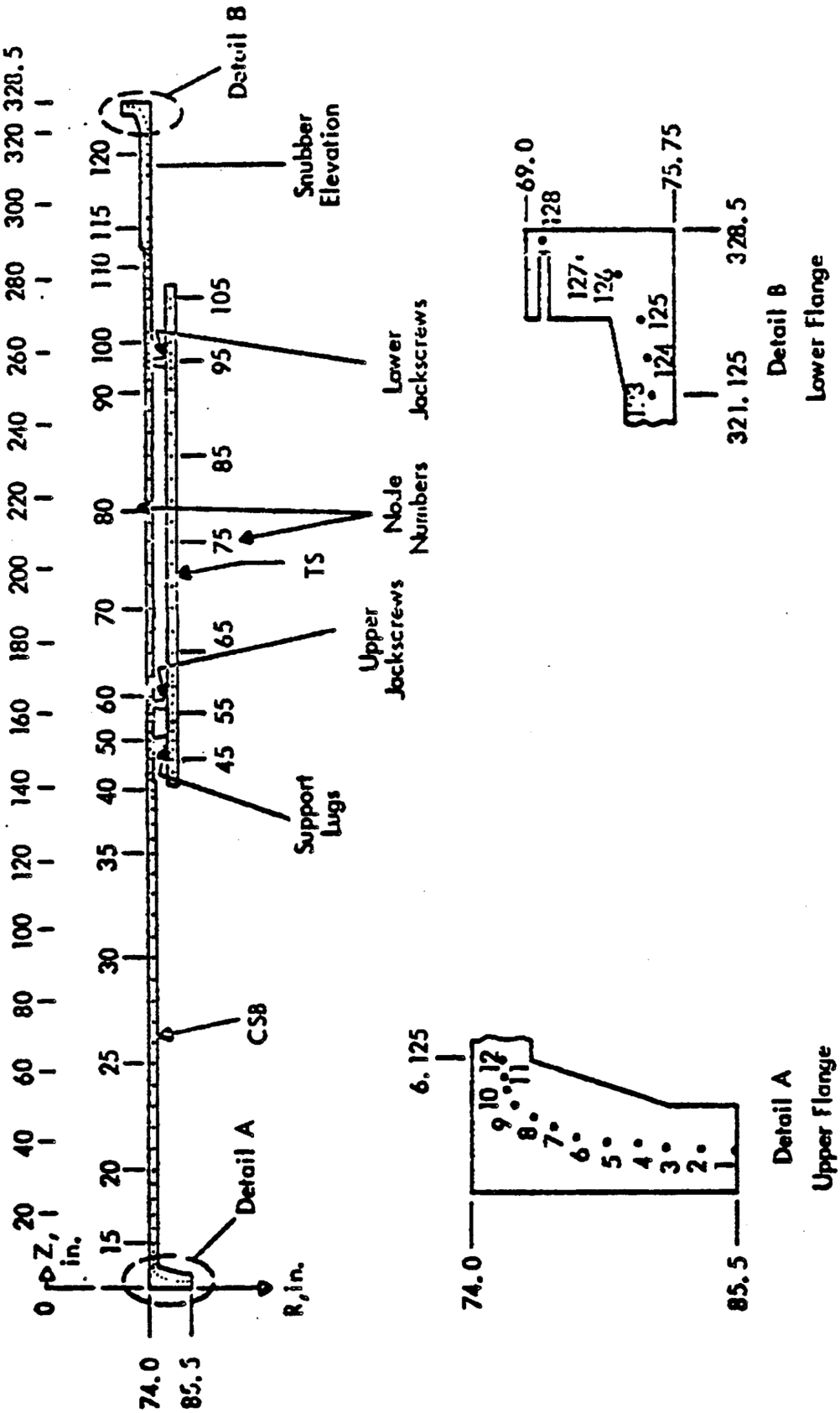


FIGURE 3.A–10 ASHSD FINITE ELEMENT MODEL OF THE CORE SUPPORT BARREL/THERMAL SHIELD SYSTEM

FIGURE 3.A–11 VERTICAL SHOCK MODEL

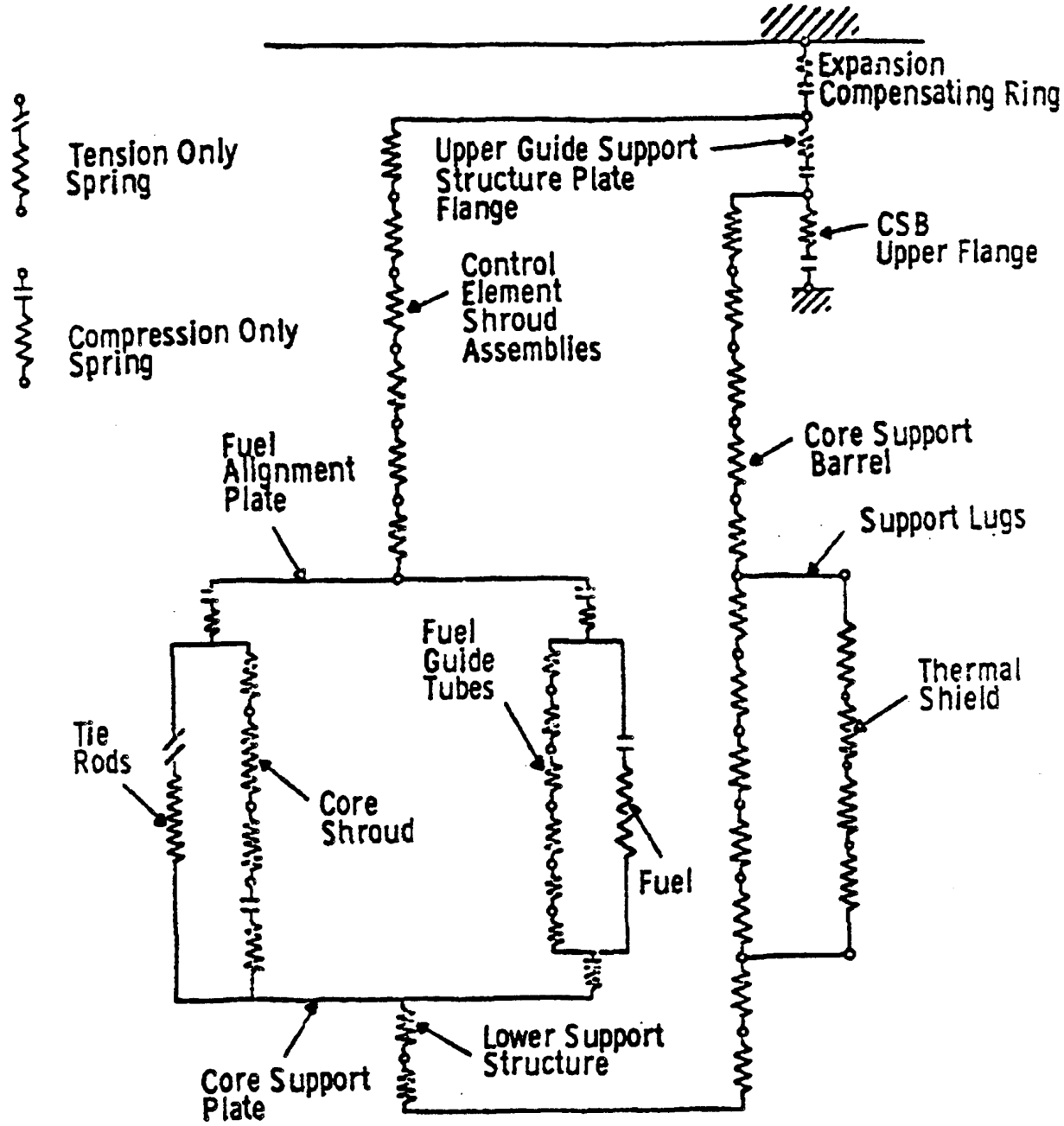


FIGURE 3.A–12 LATERAL SHOCK MODE

APRIL, 199(

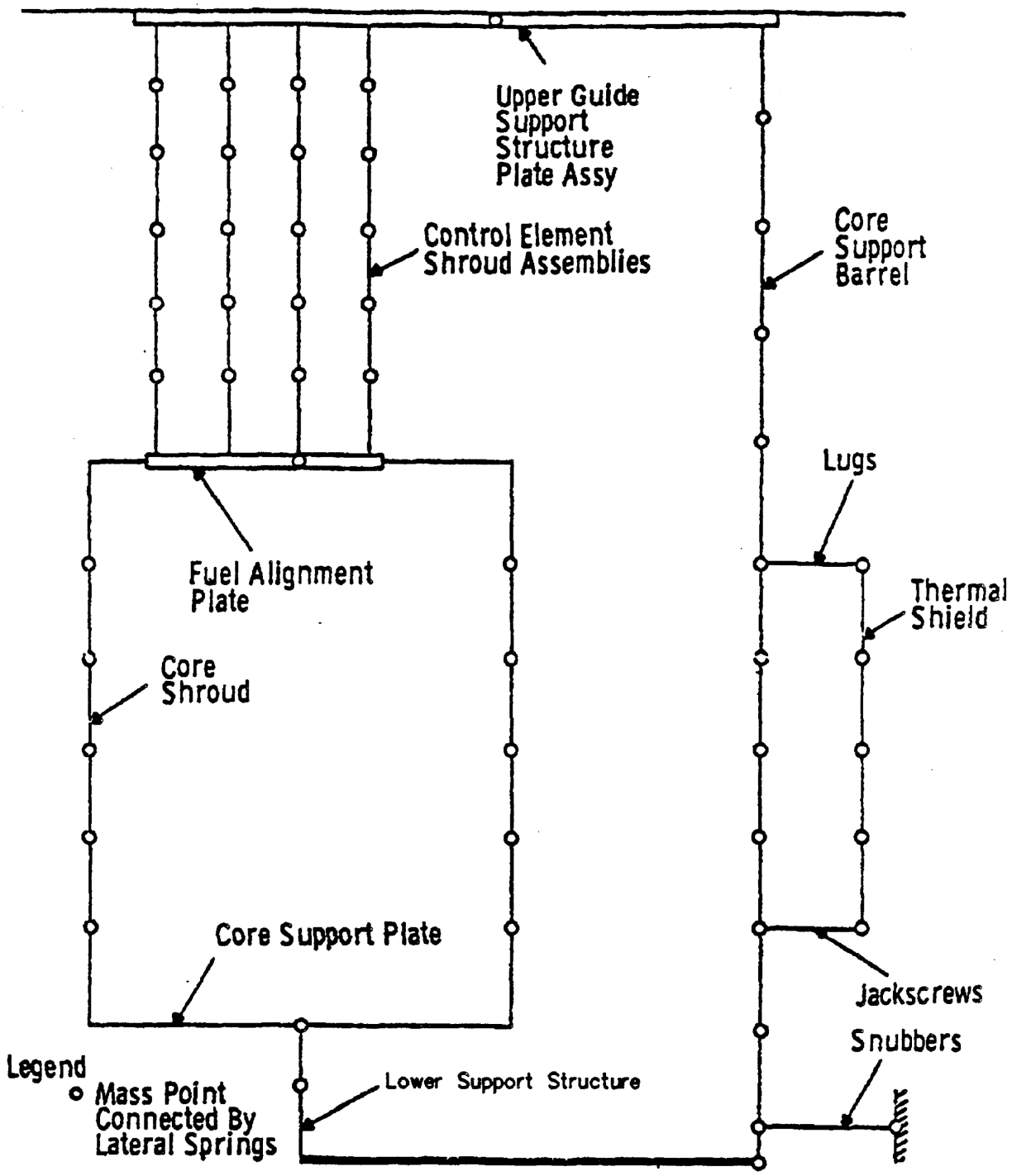
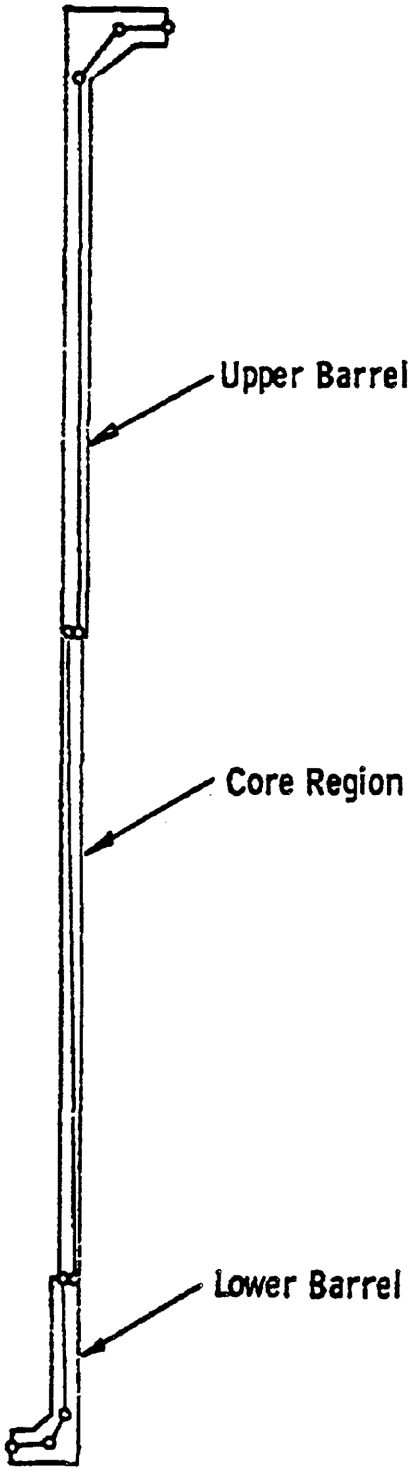


FIGURE 3.A–13 SAMMSOR DYNASOR FINITE ELEMENT MODEL OF CORE SUPPORT BARREL



Millstone Power Station Unit 2
Safety Analysis Report

Chapter 4: Reactor Coolant System

CHAPTER 4—REACTOR COOLANT SYSTEM

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CHAPTER 4 – REACTOR COOLANT SYSTEM

4.1 GENERAL SYSTEM DESCRIPTION

The function of the reactor coolant system is to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system. The reactor coolant system, which is entirely located within the containment building, consists of two heat transfer loops connected in parallel across the reactor pressure vessel. Each loop contains one steam generator, two reactor coolant pumps, connecting piping, and flow and temperature instrumentation. Coolant system pressure is maintained by a pressurizer connected to one of the loop hot legs.

A piping and instrumentation diagram of the reactor coolant system is shown in Figure 4.1–1. The legends for the piping and instrumentation diagram are given in Figures 9.1-1, 9.1-2 and 9.1-3. Elevation and plan views of the reactor coolant system are shown in Figure 4.1–2 and Figure 4.1–3, respectively. During operation, the four pumps circulate water through the reactor vessel where it serves as both coolant and moderator for the core. The heated water enters the two steam generators, transferring heat to the secondary (steam) system, and then returns to the pumps to repeat the cycle.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by the contraction or expansion of the reactor coolant. The pressurizer is located with its base at a higher elevation than the reactor coolant loop piping. This eliminates the need for a separate pressurizer drain, and ensures that the pressurizer is drained before maintenance operations.

The Reactor Coolant System (RCS) is protected against overpressure by two ASME Section III Code approved spring-loaded safety valves. In addition, two solenoid-operated power relief valves (PORVs) are provided as described in Section 4.3.7. Both the safety valves and the PORVs are connected to the top of the pressurizer. Steam discharged from the valves is cooled and condensed by water in a quench tank. In the unlikely event that the discharge exceeds the capacity of the quench tank, the tank is relieved via a rupture disc to the containment atmosphere. The rupture disc is provided as the tank code over pressure protection device. The quench tank is located at a level lower than the pressurizer. This ensures that any power-operated relief valve or pressurizer valve leakage from the pressurizer, or any discharge for these valves, drains to the quench tank.

Overpressure protection for the secondary side of the steam generators is provided by ASME Code safety valves located in the main steam line pipes upstream of the steam line isolation valves. Power-operated steam dump and bypass valves are provided to prevent opening of the secondary safety valves following a loss-of-load incident. The secondary pressure protection is described in Sections 10.3 and 4.3.2.

To maintain reactor coolant chemistry within the limits discussed in Section 4.4.3 and to control pressurizer level, a continuous but variable bleed flow from one loop upstream of the reactor coolant pump is maintained. This bleed flow is controlled by pressurizer level.

Constant coolant makeup is added by charging pumps in the chemical and volume control system. Two charging nozzles and one letdown nozzle are provided on the reactor coolant piping for these operations.

An inlet nozzle on each of the four reactor vessel inlet pipes allows injection of borated water into the reactor vessel from the safety injection system in the event emergency core cooling is needed. During a normal plant shutdown, these nozzles are also used to supply shutdown cooling flow from the low pressure safety injection pumps. An outlet nozzle on one reactor vessel outlet pipe is used to remove shutdown cooling flow.

Vent and drain connections in the reactor coolant piping are provided for draining the reactor coolant system to the radioactive waste processing system for maintenance operations. A connection is also provided on the quench tank for draining it to the radioactive waste processing system following a relief valve or safety valve discharge. Other reactor coolant loop penetrations include sampling connections (Section 9.6) and instrument connections. The nozzle identifications are tabulated in Figure 4.1–3. Normal draining of the RCS is through the chemical and volume control system.

Where required to reduce heat losses and protect personnel from high temperatures, components and piping in the reactor coolant system are insulated with a material compatible with the temperatures involved. All insulation material used on stainless steel has a soluble chloride content of less than 600 ppm to minimize the possibility of chloride-induced stress corrosion.

Electroslag welding was not used in the construction of any reactor coolant boundary component.

The major reactor coolant system components are designed for a 40 year service life. To assure that this objective can be attained, strict quality assurance standards as outlined in Sections 4.6.4and 4.6.5 were followed.

Protection provided the reactor coolant system against environmental factors such as fires, floods and missiles is described in other sections (see Chapters 1, 5, 9 and 11).

A tabulation of the RCS volumes is contained in Table 4.1–1.

TABLE 4.1–1 REACTOR COOLANT SYSTEM VOLUMES

Component	Volume (ft^3)
Reactor Vessel	4652
Steam Generators	3386
Reactor Coolant Pumps	449
Pressurizer	1500
Piping: Hot Leg	280
Piping: Cold Leg	752
Piping: Surge Line	32
Quench Tank	217

FIGURE 4.1–1 P&ID FOR REACTOR COOLANT SYSTEM & PUMP (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

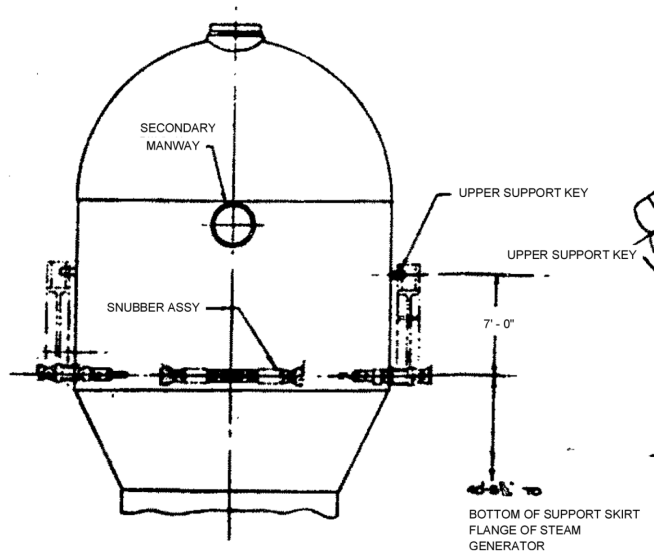
FIGURE 4.1–1 P&ID FOR REACTOR COOLANT SYSTEM & PUMP (SHEET 2)

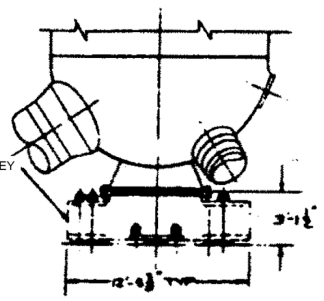
The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

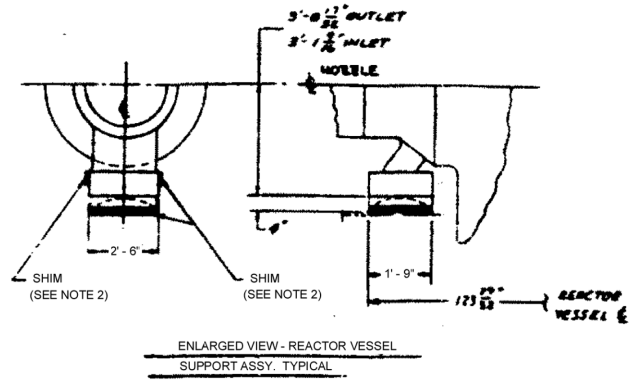
FIGURE 4.1–1 P&ID FOR REACTOR COOLANT SYSTEM & PUMP (SHEETS 3)

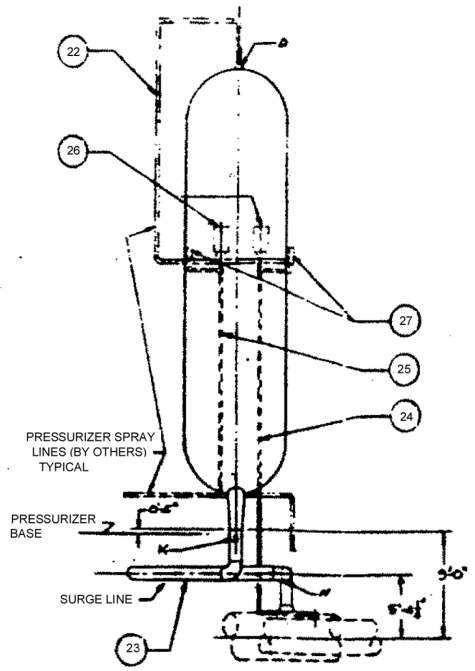
The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

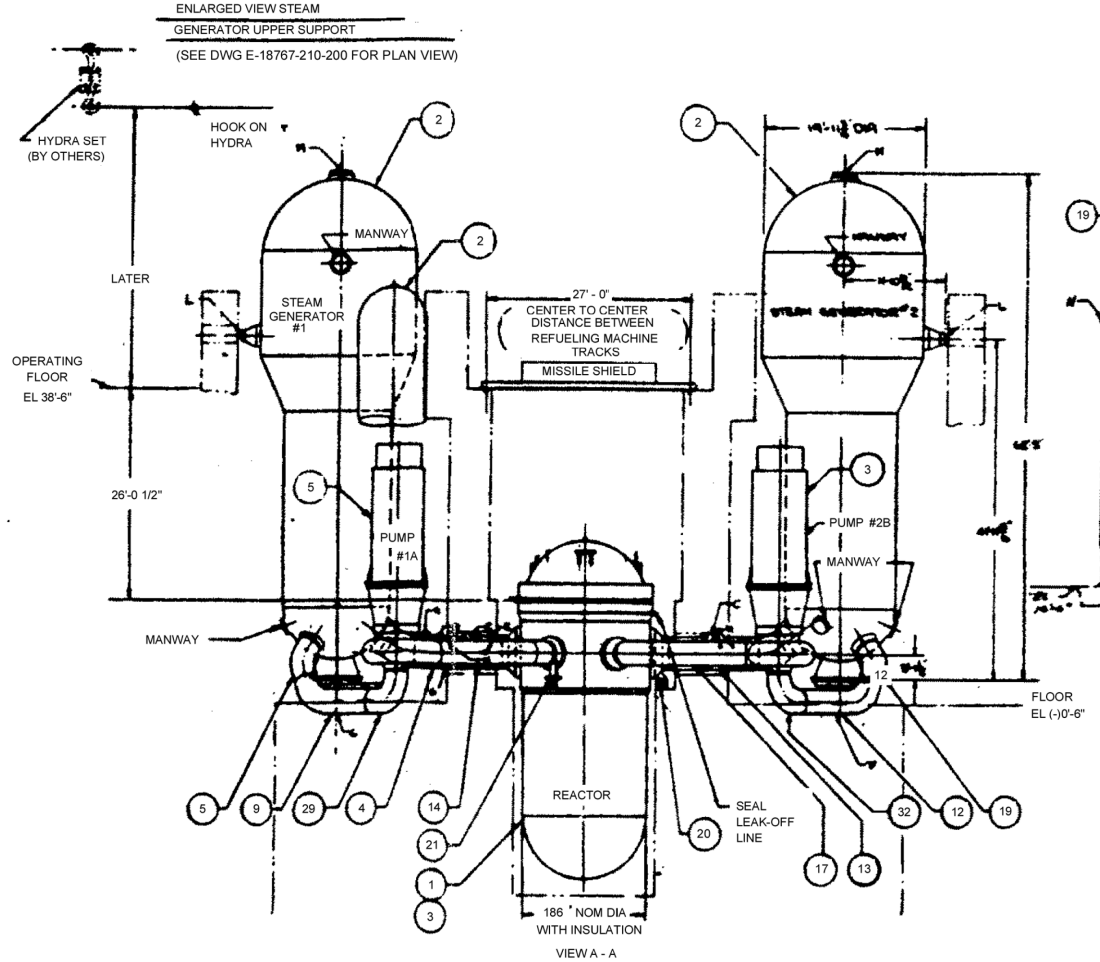
FIGURE 4.1–2 REACTOR COOLANT SYSTEM ARRANGEMENT-ELEVATION

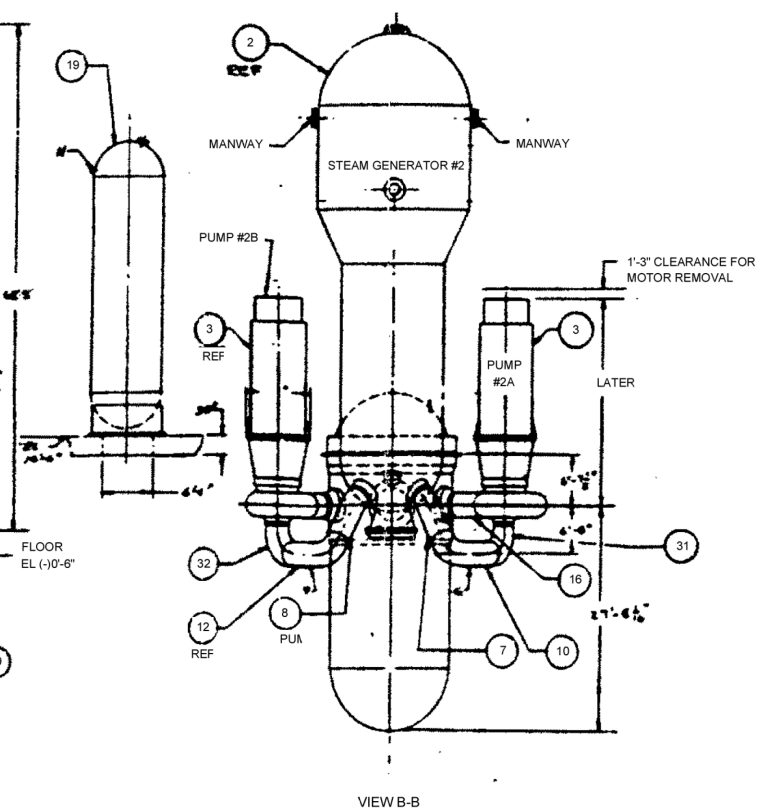






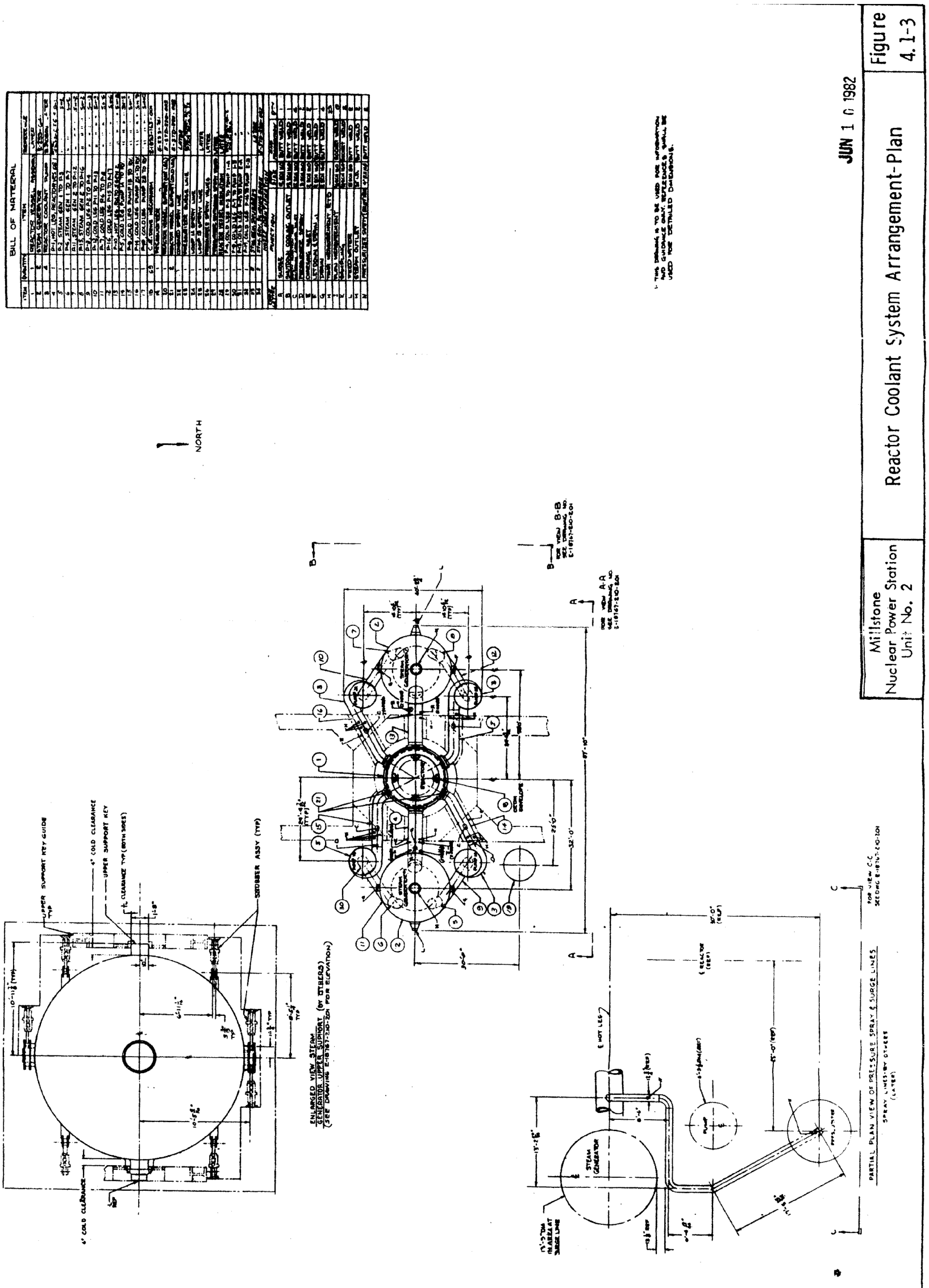






3. SEE SHEET 1 DWG. NO. E-18767-210-200
2. SHIMS TO BE MACHINED BY OTHERS TO OBTAIN REQUIRED REACTOR VESSEL LEVELNESS
1. THIS DRAWING IS TO BE USED FOR INFORMATION AND GUIDANCE ONLY. REFERENCES SHALL BE USED FOR DETAILED DIMENSIONS.

FIGURE 4.1–3 REACTOR COOLANT SYSTEM ARRANGEMENT-PLAN



4.2 DESIGN BASIS

4.2.1 DESIGN PARAMETERS

The reactor coolant system operated initially at a core power level of 2560 MWt but has since been uprated to a core power level 2700 MWt. The major systems and components which bear significantly on the acceptability of the site have been evaluated for operation at a core power level of 2700 MWt (NSSS power of 2715 MWt).

The reactor design described in Chapter 3 predicates hot leg temperature, cold leg temperature, minimum reactor coolant flow and reactor vessel pressure drop. These thermodynamic and hydrodynamic data are used in the design of the steam generator, reactor coolant pump, and reactor coolant piping as described in Section 4.3 for each of these components.

The principal design parameters for the reactor coolant system are listed in Table 4.2-1. The design parameters for each of the major components are given in Section 4.3. The reactor coolant system is designated a Class 1 system for seismic design and is designed to the criteria for load combinations and stresses which are presented in Table 4.2-2A and 4.2-2B. Seismic Analysis is discussed in Appendix 4.A.

The system design temperature and pressure are conservatively established and exceed the combined normal operating value and the change due to anticipated operating transients. They include the effects of instrument error and the response characteristics of the control system. The change due to the anticipated transients also considers the effect of reactor core thermal lag, coolant transport time, system pressure drop and the characteristics of the safety and relief valves.

The following design cyclic transients, which include conservative estimates of the operational requirements for the components discussed in Section 4.3, were used in the fatigue analyses required by the applicable codes listed in Table 4.2-3; the applicable operating condition category as designated by ASME Section III is indicated in each case.

a. Five-hundred heatup and cooldown cycles during the system's 40 year design life at a heating and cooling rate of 100°F/hr between 70°F and 532°F.

   The replacement reactor vessel head is designed for 200 steady state and transient operating cycles for plant heatup and cooldown at a rate of 100°F/hr between 70°F and 532°F.

   The replacement pressurizer is analyzed for 500 heatup cycles at a rate of 100°F/hour and 500 cooldown cycles at a rate of 200°F/hour.

   Category: Normal Condition.

b. Pressurizer spray piping is limited to 160 plant heatup and cooldown cycles. Primary manway studs of the replaced steam generators are limited to 200 heatup

and cooldown cycles at a heating and cooling rate of 100°F/hr between 70°F and 532°F.

c. Fifteen-thousand power change cycles over the range of 15 percent to 100 percent of full load with a ramp load change of five percent of full load per minute increasing and decreasing.

The replacement reactor vessel closure head is analyzed to 15,000 cycles for plant loading and unloading at 5% of full load/minute.

Category: Normal Condition.

d. Primary manway studs for the replaced steam generators are limited to 1000 cycles with a ramp load change of 5% per minute decreasing and 30% per hour increasing (plant loading/unloading).

e. Two-thousand cycles of ten percent of full load step power changes, increasing from an initial power level of 15 and 90 percent of full power and decreasing from an initial power level between 15 and 100 percent of full power.

Category: Normal Condition.

f. Ten cycles of hydrostatic testing the reactor coolant system at 3110 psig and a temperature at least 60°F above the Nil Ductility Transition Temperature (NDTT) of the component having the highest NDTT.

The replacement pressurizer is analyzed to 10 cycles of hydrostatic test at 3125 psia and 70°F - 400°F, above the minimum RT_{NDT} + 60°F.

Category: Test Condition.

g. Two-hundred cycles of leak testing at 2485 psig and at a temperature at least 60°F greater than the NDTT of the component having the highest NDTT.

Category: Test Condition; evaluated as Upset Condition.

h. Primary manway studs for the replaced steam generators are limited to 80 cycles of leak testing at 2485 psig.

i. 10^6 cycles of normal variations of ± 100 psi and ± 6 F at operating temperature and pressure.

The replacement pressurizer is analyzed for 10^6 cycles of normal variations of ± 100 psi and ± 7°F at operating temperature and pressure.

Category: Normal Condition.

j. Four-hundred reactor trips from 100 percent power.

Category: Upset Condition.

k. Primary manway studs for the replaced steam generators are limited to 80 bolt preloading cycles from unbolted state.

The manway studs for the replaced pressurizer are analyzed for 100 bolt/unbolt cycles. The vent port studs for the replaced pressurizer are analyzed for 200 bolt/unbolt cycles.

Category: Normal Condition

l. Primary manway studs for the replaced steam generators are limited to 1500 cycles of 10% of full load step power changes, increasing from an initial power level of 15% to 90% of full power and decreasing from initial power level between 15% and 100% of full power.

Category: Normal Condition

m. Primary manway studs for the replaced steam generators are limited to 200 reactor trips from 100% power.

Category: Upset Condition

In addition to the above list of normal design transients, the following abnormal transients were also considered when arriving at a satisfactory usage factor as defined in Section III of the ASME Boiler and Pressure Vessel Code; however, emergency condition transients were not used to form the basis for the code design of the components based on Paragraph N-417.10(f) of ASME III, 1968 Edition, Summer 1968 addenda.

1. Forty cycles of loss of turbine load from 100 percent power with a delayed, reactor trip.

Category: Upset Condition.

2. Forty cycles of total loss of reactor coolant flow when at 100 percent power.

Category: Upset Condition.

3. Five cycles of complete loss of secondary system pressure.

Category: Emergency Condition.

The reactor coolant system and its associated controls are designed to accommodate plant step load changes of ± ten percent of full power and ramp changes of ± five percent of full power per minute without reactor trip. The system will accept, without damage, a complete loss of load.

4.2.2 CODES ADHERED TO

The Codes adhered to and component classifications are listed in Table 4.2-3 and conform to 10 CFR Part 50, Section 50.55a. The construction permit date was December 11, 1970; thus in all instances, code dates identified below meet or exceed those required. The impact properties of all materials which form a part of the pressure boundary meet the requirements of the ASME Boiler and Pressure Vessel Code Section III, Paragraph N330, at a temperature of 40°F. The impact properties of the replacement reactor vessel closure head and replacement pressurizer meet the requirements of ASME Section III, NB 2300.

In general, code editions and addenda in effect on the date of the original purchase order to a manufacturer apply in the design, manufacture and testing of those components. The code editions and addenda which apply for the components in Tables 1.2-1 and 4.2-3 are specified in applicable FSAR subsections.

Full compliance with Safety Guide 26 was not possible since most of the components covered by Safety Guide 26 were purchased and fabrications begun prior to the March 23, 1972, issue date for this guide. Table 4.2-4 provided a comparison of those components which are not in compliance with Safety Guide 26.

The instrument air system for Millstone Unit 2 is not required for safe shutdown. Therefore, Safety Guide 26 does not apply to the design codes for this system.

The codes and standards used for the components of the diesel oil supply are as follows:

Tanks (Above Ground) - API 650

Tanks (Below Ground) - NFPA Number 30

Piping - ANSI B31.1.0 (MOD C)

Valves - ANSI B31.1.0 (MOD C)

In addition, seismic category 1 requirements and Quality Assurance Program as outlined in Appendix 1B (located in the original FSAR dated August 1972) were employed in the manufacture.

The ANS N18.2 system of quality group classifications has been utilized by the NSSS vendor for Millstone Unit 2. The AEC has voted affirmatively for the adoption of N18.2 as an ANSI standard. It is more definitive than Safety Guide 26, is supported by the major NSSS suppliers and by the utility industry, and compliance with its provisions provides a satisfactory alternative to the Safety Guide.

The following Code Cases were utilized for materials and construction of components.

ASME CODE CASES

Steam Generators

1332-4	Requirements for Steel Forgings
1359-1	Ultrasonic Examination of Forgings
1335-2	Requirements for Bolting Materials
1336	Requirements for Nickel-Chromium Iron Alloy (all product forms)
N-71-13	Component Supports
N-10	UT of Pressure Vessel Welds
N-20	Steam Generator Tubes
N-294-4	Nonwelded Components
N-474-1	Inconel 690 Material

Reactor Vessel

1335-2	Requirements for Bolting Materials
1336	Requirements for Nickel-Chromium Iron Alloy (all product forms)
1359-1	Ultrasonic Examination of Forgings
N-4-12	Material Requirements for CEDM motor housing
N-525	Material Requirements for Instrumentation Nozzles and Head Vent Nozzle

Pressurizer

N-405-1	Socket Welds
N-2142-2	Classification UNS N06052 Filler Material

ANSI B31 CODE CASES

Piping

1477-1	Use of 1970 Addenda of ANSI B31.7
70	Design Criteria for Nuclear Piping Under Abnormal Conditions
74	Weld Reinforcement for B31.0 Piping
83	Weld Reinforcement for B31.7 Piping

For the most severe loading combination, which includes the Design Basis Earthquake loads, the primary stresses in the ASME Code Section III, Class 2 and 3 components and component supports are limited to levels comparable to the emergency stress limits defined in ASME Code Section III for Class 1 components.

4.2.3 QUALITY CONTROL CLASSIFICATION

The major components of the reactor coolant system have been placed in the safety classes as defined by ANS N 18.2 “Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

4.2.4 PART-LOOP OPERATION

The maximum temperature of the hot leg and cold leg will be less than the maximum temperatures for design power at design flow. Reactor power operation with less than 4 reactor coolant pumps operating or natural circulation is not allowed. However, decay heat will be transferred to the steam generator for both cases. Current Technical Specifications restrictions prohibit other than four reactor coolant pump power operations.

The adequacy of natural circulation for decay heat removal after reactor shutdown has been verified analytically and by tests on the Palisides reactor. The core ΔT in the analysis has been shown to be lower than the normal full power ΔT; thus, the thermal and mechanical loads on the core structure are less severe than normal design conditions.

To assess the margin available in a post-coastdown situation, a study was made assuming termination of pump coastdown 100 seconds after reactor trip, with immediate flow decay to the stable natural circulation condition. It should be recognized that pump rotation will not have stopped for substantially longer than 100 seconds. With the maximum decay heat load 100 seconds after trip, the system will sustain stable natural circulation flow adequate to give a power to flow ratio of less than 0.9.

Heat removed from the core during natural circulation may be rejected either by dumping to the main condenser or to the atmosphere; the rate of heat removal may be controlled to maintain core ΔT within allowable limits. The analytical techniques are verified by tests completed on the Palisades reactor (AEC Docket Number 50-225).

TABLE 4.2-1 PRINCIPAL DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM

Design Thermal Power, (NSSS)	MWt 2715, Btu/hr 9.26 x 10^9
Design Pressure, psig	2485
Design Temperature (Except Pressurizer, 700°F), °F	650
Design Coolant Flow Rate, gpm	325,000 (1)
Cold Leg Temperature, Normal Service, °F	550 (1)
Hot Leg Temperature, Normal Service, °F	604 (1)
Normal Operating Pressure, psia	2250 (1)
System Volume, ft^3 (Without Pressurizer)	9,551
Pressurizer Water Volume, ft^3	942 at 65% level
Pressurizer Steam Volume, ft^3	590 at 65% level

(1) RCS piping and vessel design parameters. Volumetric flow rate is based on 122 x 10^6 lbm/hr mass flow rate and cold leg density. See FSAR Section 14 for principal RCS parameters used in Safety Analyses.

TABLE 4.2-2A TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS

Loading Combinations	Vessels	Primary Stress Limits Piping	Supports Working Stress
1. Design Loading + Design Earthquake (OBE)	$P_M < S_M$, $P_B + P_L \leq 1.5S_M$	$P_M < S_M$, $P_B + P_L \leq 1.5S_M$	
2. Normal Operating Loadings + Maximum Hypothetical Earthquake (DBE).	$P_M \leq S_D$	$P_M < S_D$	Within Yield
See Note b.		$P_B + P_L \leq 2.25S_m$	
See Note c.	$P_B \leq 1.5[1-(P_M/S_D)^2]S_D$ $P_B \leq 4S_D/\pi \cos[\pi P_M/2S_D]$		
3. Normal Operating Loadings + Pipe Rupture + Maximum Hypothetical Earthquake (DBE).	$P_M \leq S_L$	$P_M \leq S_L$	Deflection of supports limited to maintain supported equipment within limits shown in columns 1 and 2
See Notes a and b.	$P_B \leq 1.5[1-(P_M/S_L)^2]S_L$	$P_L + P_B \leq 3S_M$	
See Note c.	$P_B \leq 4S_L/\pi \cos[\pi P_M/2S_L]$		

Notes:

(a) This load combination is not applied to the piping run within which a pipe break is considered to have occurred.
(b) For loading combinations 2 and 3, stress limits for vessels, with the symbol P_M changed to P_L, should also be used in evaluating the effects of local loads imposed on vessels and/or piping.
(c) These stress limits are used for cylindrical structures (e.g., CRDM housings) in the vessel design.
(d) See Table 4.2-2B for replacement reactor vessel closure head loading combinations and primary stress limits.

TABLE 4.2-2A TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS (CONTINUED)

P_M = Calculated Primary Membrane Stress
P_B = Calculated Primary Bending Stress
P_L = Calculated Primary Local Membrane Stress
S_M = Tabulated Allowable Stress Limit at Temperature from ASME Boiler and Pressure Vessel Code, Section III or ANSI B31.7.
S_Y = Tabulated Yield at Temperature, ASME Boiler and Pressure Vessel Code, Section III.
S_D = Design Stress
$S_D = S_Y$ (for ferritic steels)
$S_D = = 1.2S_M$ (for austenitic steels)
$S_L = S_Y + 1/3 (S_u - S_Y)$
S_u = Tensile Strength of Material at Temperature

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lbs/ square inch

Material	S_Y (1)	S_U	S_M (1)	S_P	S_L
A-106B	25.4	60.0 (2)	17.0	25.4	36.9
SA-533B	41.4	80.0 (2)	26.7	41.4	54.3
SA-508, CL2	41.4	80.0 (2)	26.7	41.4	54.3
304 SS	17.0	54.0 (3)	15.3	18.4	29.3
316 SS	18.5	58.2 (3)	16.7	20.0	31.7

(1) From ASME Boiler and Pressure Vessel Code, Section III, 1968 ED., at 650°F.
(2) Minimum value at room temperature which is approximately the same at 650°F for ferritic materials.
(3) Estimated

TABLE 4.2-2B TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS FOR THE REPLACEMENT REACTOR VESSEL HEAD AND REPLACEMENT PRESSURIZER

Loading Combinations		ASME Code Subsection
Design	$P_m \leq S_m$	NB-3221.1
Service Level A (Normal)	$P_l \leq 1.5S_m$	NB-3221.2
Service Level B (Upset)	$P_l + P_b \leq 1.5S_m$	NB-3221.3
	$P_l + P_b + Q \leq 3.0S_m$	NB-3222.2
	$P_e \leq 3.0S_m$	NB-3222.3
	$\Sigma U_i \leq 1.0$	NB-3222.4
	$S_{TH} \leq y' S_y$	NB-3222.5
Service Level D (Faulted)	$P_m \leq$ Lesser of $2.4S_m$ and $0.7S_u$	NB-3225 Appendix F
	$P_l \leq 1.5P_m$	NB-3225 Appendix F
	$P_l + P_b \leq 1.5P_m$	NB-3225 Appendix F

Where:
P_m = General Primary Membrane Stress Intensity
P_l = Local Membrane Stress Intensity
P_b = Bending Stress Intensity
Q = Secondary Stress Intensity
P_e = Expansion Stress Intensity
U_i = Actual Service condition cycles divided by allowable cycles, based on calculated alternating stress and fatigue design curve.
S_{TH} = Thermal Stress Range
S_m = Design Stress Intensity
S_u = Tensile Strength
S_y = Yield Stress
y' = Maximum allowable range of thermal stress on an elastic basis divided by S_y.

TABLE 4.2-2B TABLE OF LOADING COMBINATIONS AND PRIMARY STRESS LIMITS FOR THE REPLACEMENT REACTOR VESSEL HEAD AND REPLACEMENT PRESSURIZER (CONTINUED)

The following typical values in ksi are selected for the replacement reactor vessel head materials at 650°F.

Material	Sy	Su	Sm	
SA508 Grade 3 Class 1	41.5	80.0	26.7	ASME Section II, Part D
SB 167 (Alloy 690)	20	80	20.0	ASME Code Case N525/Section II, Part D
SB 166 (Alloy 690)	27.5	80.0	23.3	ASME Section II, Part D
SA 182 F316LN	17.8	62.8	16.0	ASME Section II, Part D
SA 312 TP316L	15.3	61.7	13.8	ASME Section II, Part D

The following typical values in ksi are selected for the pressurizer materials at 650°F.

Material	Sy	Su	Sm	
SA508 Grade 3 Class 2	53.9	90.0	30	ASME Section II, Part D
SA 182 Grade F 316	18.5	67.0	16.7	ASME Section II, Part D

TABLE 4.2-3 REACTOR COOLANT SYSTEM CODE REQUIREMENTS

Components	Codes
Reactor Vessel (excluding replacement Reactor Vessel Closure Head and Nozzles), Original Upper Shell of Steam Generator	1. ASME, Section III, Class A, 1968 Edition, Addenda through Summer 1969
Replacement Reactor Vessel Closure Head and Nozzles, Replacement Pressurizer	1. ASME Section III, 1998 Edition through 2000 Addenda.
Replacement Lower Steam Generator	1. ASME, Section III, Class I, 1983 Edition, Addendum to Summer of 1984
Reactor Coolant Pumps	1. Draft ASME Code for Pumps and Valves for Nuclear Power, Class 1, November 1968, including March 1970 Addenda. 2. 2. ASME Section III, paragraph N153 in Summer 1969 Addenda. 3. ASME Section III, Appendix IX.
Quench Tank	ASME Section III, Class C, 1968
Pressurizer Safety Valves	1. ASME Section III, Class A, 1968 Edition, Addenda through Summer of 1970, Code Case 1344-1
Piping	1. ANSI B31.7, Class 1, 1969 Edition. 2. ASME Section III, paragraph N153 in Summer 1969 Addenda. 3. Code Case 70 to B31.7.
Secondary Safety Valves	ASME Standard Code for Pumps and Valves for Nuclear Power, Class 2, March, 1970 Draft

Note: The spare and original safety valves are used interchangeably. Refurbishment and retesting of the safety valves are performed periodically and the safety valves (spare and/or original) are then installed into the system rotationally.

TABLE 4.2-4 COMPARISON WITH SAFETY GUIDE 26

System	Components	Classification Per Safety Guide 26	Applicable Code Per Safety Guide 26	Code Used In Manufacture	Requirements Imposed In Addition to Code Requirements Used in Manufacture
Reactor Building Closed Cooling Water System	Pressure Vessels	Quality Group C	ASME Section III, Class 3	ASME Section VIII Division I	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)], Spot radiographed
	Piping (excluding Containment Penetrations)	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0 MOD B	Seismic Category I, 10% random radiography of butt welds for piping 4 inch and larger, material identification, manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Pumps	Quality Group C	ASME Section III, Class 3	Standards of the Hydraulic Institute	All pressure containing parts where hydrostatically tested at a minimum of 1.5 times the design pressure seismic I manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Valves (excluding Containment)	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0 MOD B	Seismic Category I, MT-PT Examination, material traceability on pressure retaining parts, manufactured under Quality Assurance Program of Appendix 1B [(a)]

TABLE 4.2-4 COMPARISON WITH SAFETY GUIDE 26 (CONTINUED)

System	Components	Classification Per Safety Guide 26	Applicable Code Per Safety Guide 26	Code Used In Manufacture	Requirements Imposed In Addition to Code Requirements Used in Manufacture
Safety Injection System	Shutdown Heat Exchangers	Quality Group C	ASME Section III, Class 2	ASME Section III, Class 3, TEMA R	Seismic Category I
	Refueling Water Tank	Quality Group B	ASME Section III, Class 2	ASME Section III, Class 3	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Piping HCD(C)	Quality Group B	ASME Section III	ANSI B31.1.0 MOD(C)	Seismic Category I, Material Identification per ASTM Specification
	Pumps	Quality Group B	ASME Section III, Class 2	ASME Code for Pumps and Valves for Nuclear Power, Class II	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)] (See Question 4.4)
	Valves HCD(C) Only	Quality Group B	ASME Section III, Class 2	ANSI B31.1.0 MOD(C)	Seismic Category I, Material Identification per ASTM Specification
	Piping 4 inch HCD-3 and 6 inch HCD-3	Quality Group B	ASME Section III, Class 2	ANSI B31.1.0	Seismic Category I

TABLE 4.2-4 COMPARISON WITH SAFETY GUIDE 26 (CONTINUED)

System	Components	Classification Per Safety Guide 26	Applicable Code Per Safety Guide 26	Code Used In Manufacture	Requirements Imposed In Addition to Code Requirements Used in Manufacture
Auxiliary Feedwater System	Condensate Storage Tank	Quality Group B	ASME Section III, Class 2	AWWA D100 NFPA Volume 6	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)] later modified to API 620 or equivalent based on Design Change to a pressurized tank
	Piping (excluding Containment Penetrations)	Quality Group B	ASME Section III, Class 2	ANSI B31.1.0 MOD B	Seismic Category I, 10% random radiography of butt welds for piping 4 inch and larger, material identification manufactured under Quality Assurance Program of Appendix 1B
	Pumps	Quality Group B	ASME Section III, Class 2	ASME Code for Pumps and Valves for Nuclear Power, Class II	Seismic Category I manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Valves (excluding Containment Isolation)	Quality Group B	ASME Section III, Class 3	ASNE B31.1.0 MOD B	Seismic Category I, MT-PT Examination, material traceability on pressure retaining parts, manufactured under Quality Assurance Program of Appendix 1B [(a)]

TABLE 4.2-4 COMPARISON WITH SAFETY GUIDE 26 (CONTINUED)

System	Components	Classification Per Safety Guide 26	Applicable Code Per Safety Guide 26	Code Used In Manufacture	Requirements Imposed In Addition to Code Requirements Used in Manufacture
Service Water System	Piping	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Pumps	Quality Group C, Class 3	ASME Section III	ASME Section VIII	Perform Test according to ASME PTC 8.2 1965, Seismic Category I manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Valves	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)]
Vital Chilled Water System	Piping	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Valves	Quality Group C	ASME Section III, Class 3	ANSI B31.1.0	Seismic Category I, manufactured under Quality Assurance Program of Appendix 1B [(a)]
	Condenser/ Evaporators	Quality Group C	ASME Section III, Class 3	ASME Section VIII, Division 1	Seismic Category I, manufactured under Quality Assurance Program [(a)]

(a)Appendix 1B was located in the original FSAR dated August 15, 1972.

4.3 SYSTEM COMPONENT DESIGN

4.3.1 REACTOR VESSEL

The reactor vessel (Figure 4.3–1) is supported by three pads welded to the underside of the reactor vessel nozzles. The arrangement of the vessel supports, allows radial growth of the reactor vessel due to thermal expansion while maintaining it centered and restrained from movement caused by seismic disturbances. Departure from levelness of not more than 0.002 inch per foot of flange diameter is maintained during construction to facilitate proper assembly of reactor internals. The design parameters for the reactor vessel are given in Table 4.3-1.

The vessel closure flange is a forged ring with a machined ledge on the inside surface to support the reactor internals and core. No other ring forgings are used for reactor vessel shell sections. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seal. The vessel closure contains 54 studs, 7 inches in diameter, with eight threads per inch. The stud material is ASTM A-540, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 40 ksi. Calculations show that 32 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or four adjacent broken studs will cause O-ring leakage.

Six radial nozzles on a common plane are located just below the vessel closure flange. Extra thickness in this vessel-nozzle course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around each outlet nozzle on the inside diameter of the vessel wall provides a mating surface for the internal structure which guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to reduce core bypass leakage. A fixed hemispherical head is attached to the lower end of the shell. There are no penetrations in the lower head.

The removal top closure head is hemispherical. The closure head is single piece low alloy steel forging replaced during refueling outage 16. All surfaces in contact with reactor coolant are clad with a quarter inch nominal thickness weld deposit similar to type 304 stainless steel. The nozzles in the reactor vessel head support the Control Element Drive Mechanisms (CEDMs). The CEDM, In-Core Instrumentation (ICI) and vent nozzles are constructed from Inconel alloy 690 material to minimize the susceptibility to Primary Water Stress Corrosion Cracking. Threaded housing flanges made of stainless steel are joined to the guide tubes by Gas Tungsten Arc Welding. The head flange is drilled to match the vessel flange stud bolt locations. The 54 stud bolts are fitted with spherical washers located between the closure nuts and head flange to maintain stud alignment during head flexing due to boltup. To ensure uniform loading of the closure seal, the studs are tensioned with hydraulic stud tensioners. Stud elongation is then measured to ensure proper preload on all the studs.

Flange sealing is accomplished by a double-seal arrangement utilizing two silver-plated Ni-Cr-Fe alloy, self-energized O-rings. The space between the two rings is monitored to allow detection of any inner ring leakage. The control element drive mechanism (CEDM) nozzles (Ni-Cr-Fe alloy

through the head, stainless steel flange) terminate with threaded and seal-welded flanges at the upper end. There are eight instrumentation nozzles with Grayloc connectors to maintain pressure boundary. In addition to these nozzles, there is a three-quarter inch vent connection.

The core is supported from an internally machined core support ledge.

4.3.2 STEAM GENERATOR

The nuclear steam supply system (NSSS) utilizes two steam generators (Figure 4.3–2) to transfer the heat generated in the reactor coolant system (RCS) to the secondary system and produce steam at the warranted steam pressure and quality. The design parameters for the steam generators are given in Table 4.3-2.

The steam generator is a vertical U-tube heat exchanger. The steam generator operates with the reactor coolant in the tube side and the secondary fluid in the shell side.

Reactor coolant enters the steam generator through the inlet nozzle, flows through three-quarter inch OD U-tubes, and leaves through two outlet nozzles. Vertical partition plates in the lower head separate the inlet and outlet plenums. The plenums are stainless steel clad, while the primary side of the tube sheet is Ni-Cr-Fe clad. The vertical U-tubes are Ni-Cr-Fe alloy. The tube-to-tube sheet joint is welded on the primary side.

Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring having top discharge “J” nozzles which direct the flow into the downcomer. The downcomer is an annular passage formed by the inner surface of the steam generator shell and the cylindrical shell wrapper which encloses the vertical U-tubes.

At the bottom of the downcomer, the secondary water is directed upward past the vertical U-tubes where it is boiled to produce steam in the evaporator. The heat transfer area is determined by the required heat transfer, the thermal driving force and the heat transfer coefficient. The heat transfer coefficient in the evaporator is calculated from experimental data.

Upon exiting from the vertical U-tube heat transfer surface, the steam water mixture enters centrifugal-type separators. These impart a centrifugal motion to the mixture and separate the steam from the water. The water leaves the primary separator through the bottom of the separator housing and is directed into the downcomer where it is mixed with the feedwater. Final drying of the steam from the centrifugal primary separators is accomplished by directing the steam through secondary cyclones. The moisture content of the outlet steam is no greater than 2.0 percent at design flow.

The steam generator primary side pressure loss is determined by summing the losses due to friction in the tubes, in the tube bends, entrances and exits, and the steam generator inlet and outlet plenums and nozzles.

The steam generator shell is constructed of carbon steel. Manways and handholes are provided for easy access to the steam generator internals.

The power-operated steam dump valves and steam bypass valves obviate opening of the main steam safety valves following turbine and reactor trip from full power. The steam dump and bypass system is described in Section 10.

Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve and provided by 16 spring-loaded ASME Code safety valves which discharge to atmosphere. Eight of these safety valves are

mounted on each of the main steam lines outside the containment upstream of the steam lined isolation valves. The opening pressure of the valves is set in accordance with ASME Code allowances. Parameters for the main steam safety valves are given in Table 4.3-3.

Instrumentation has been added to each safety relief valve (SRV) to provide a main control board annunciator alarm indication of valve closed/not closed, (Regulatory Guide 1.97, Rev. 2, D-18 variable).

The main control board annunciator window (C05 D17A/B) is a split window and will alarm when any of the SRV's on Steam Generator 1 or 2, respectively, are “not closed.”

Upon receiving this alarm, the operator would use the Plant Process Computer display of the Steam Generator Safety Valves or local indication, to determine which valve on Steam Generator 1 or 2 is not closed.

The steam generators are vertically mounted on bearing plates which allow lateral motion due to thermal expansion of the reactor coolant piping. Stops are provided to limit this motion in case of a coolant pipe rupture.

The top of each unit is restrained from sudden lateral movement by suitable stops and hydraulic snubbers mounted rigidly to the concrete structure.

In addition to the transients listed in Section 4.2.1, each steam generator is also designed for the following conditions such that no component is stressed beyond the allowable limit as described in ASME Code, Section III (Table 4.3-2):

a. Four-thousand cycles of transient pressure differentials of 85 psi across the primary head divider plate due to starting and stopping the primary coolant pumps (RCP).

   Category: Normal Condition.

b. Ten cycles of hydrostatic testing of the secondary side at 1235 psig, the primary side is at atmospheric pressure.

   Category: Test Condition.

c. Two-hundred cycles of leak testing of the secondary side at 985 psig, the primary side will be pressurized to 165 psig.

Category: Test Condition.

d. Fifteen-thousand cycles of adding 600 gpm of 70°F feedwater with the plant in hot standby condition.

Category: Normal Condition.

In addition to the normal design transients listed above, and those listed in Section 4.2.2, the following additional abnormal transient was also considered in arriving at a satisfactory usage factor as defined in Section III of the ASME Code:

Eight cycles of adding a maximum of 650 gpm of 70°F feedwater with the steam generator secondary side dry and at 620°F.

Category: Emergency Condition.

The unit is capable of withstanding these conditions for the prescribed numbers of cycles in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor as prescribed in ASME Code, Section III.

The steam generators are located at a higher elevation than the reactor vessel. The elevation difference creates natural circulation sufficient to remove core decay heat following coastdown of all RCPs.

The steam generators are equipped with a nitrogen addition system which has the capability of admitting N_2 to the bottom blowdown headers to mix the chemicals in the steam generators during wet layup, and also through nozzles in the transition cones for blanketing the steam generators and steam lines above the water level. A flowmeter provides nitrogen flow control and visual verification that flow is occurring. A drain line on the steam generator upper vent line allows checking that the steam generator has not been flooded. A test gauge may be installed on this line to measure steam generator overpressure.

In addition to the transients listed in this section, and those in Section 4.2.1 the following factors were considered in the design of the steam generators.

4.3.2.1 Flow Induced Vibration

The steam generator has also been designed to ensure that critical vibration frequencies will be well out of the range expected during normal operation and during abnormal conditions. The steam generator tubing, tube sleeves, and tubing supports are designed and fabricated with considerations given to both secondary side flow induced vibrations and RCP induced vibrations. In addition, the heat transfer tubing, tube sleeves, and tube supports are designed so that they will

not be structurally damaged under the loss of secondary pressure conditions that may produce a fluid velocity in the tube bundle four times design velocity.

Because the RCPs have a rotational speed of 900 rpm (less normal slip) the tube bundle design has considered the imposition of exciting frequencies of 14 to 15 cps and 70 and 75 cps. The lower frequency range is defined as a mechanical vibration resulting from the transmission of a mechanical impulse at the frequency of pump rotation. The upper frequency range is defined to be a sinusoidal pressure variation of ±6 psi in the primary piping that contains the pump. The pressure variation results from the impeller vanes interacting with the cutwater vane at the volute outlet during each revolution of the impeller.

It has been found that all tubes and tube sections that will experience forcing functions from cross flow and parallel flow have natural frequencies sufficiently different from the frequency of the forcing function that they will not experience damaging vibrations. The mechanical excitation frequency is sufficiently different from the lowest natural frequency for out-of-plane or lateral vibration in any tube span that critical vibration will not occur.

4.3.2.2 Tube Thinning

The original Combustion Engineering (CE) Steam Generators' margin of tube-wall thinning that could be tolerated without exceeding the allowable faulted stress limits under postulated condition of a design basis largest pipe break in the reactor coolant pressure boundary (RCPB) during reactor operation was 0.008 inches.

0.0012 inch excess material had been intentionally been provided in the tube wall thickness to accommodate the estimated degradation of tubes during the service lifetime. CE expected negligible tube wall thinning when operating under the specified secondary chemistry requirements.

Because of the use of volatile secondary water chemistry there has been no tube wall thinning experienced on the steam generators. The new steam generator tubes are designed to be at least structurally equivalent to the original and in compliance to Regulatory Guide 1.121. Therefore, tube wall thinning or other forms of tube degradation can be structurally accommodated with the same degree of margin as the original.

4.3.2.3 Potential Effects of Tube Ruptures

The steam generator tube rupture accident in a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would mix with water in the shell side of the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the main steam dump and bypass system. Noncondensible radioactive gases in the condenser are removed by the condenser air ejector system and discharged to the plant vent. Analysis of a steam generator tube rupture accident, assuming complete severance of a tube, is presented in Section 14.14.

4.3.2.4 Composition of Secondary Fluid

Radioactivity concentrations in the secondary side of the steam generator is dependent upon the radioactive concentration of the RCS, the primary to secondary leak rate, and the operating history of the steam generator blowdown system.

4.3.3 REACTOR COOLANT PUMPS

The reactor coolant is circulated by four single speed, vertical, single suction, centrifugal type pumps (Figure 4.3–3). The discharge nozzle is horizontal and the suction nozzle is in the bottom vertical position. The pressure containing components are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A.

The design flow for the RCP is determined from the reactor mass flow. This mass flow is converted to volumetric flow at the full power cold inlet temperature to determine the pump design flow. The maximum pressure loss at the design flow rate for the reactor vessel, steam generator, and piping is determined by adding an allowance for uncertainty to the best estimate for each pressure drop. These maximum values are used to establish the RCP design head. The RCP is designed to produce the minimum reactor design flow at the maximum expected system pressure loss.

The minimum RCS pressure at any given temperature is limited by required net positive suction head (NPSH) for the RCP during portions of plant heatup and cooldown. To ensure that the pump NPSH requirements are met under all operating conditions, an operating curve is used which gives permissible RCS pressure as a function of reactor coolant temperature. The RCP NPSH restriction on this curve is determined by using the NPSH requirement for each pump combination and correcting it for pressure and temperature instrument errors and pressure measurement location. The NPSH required versus pump flow is supplied by the pump vendor. Plant operation below this curve is prohibited. At low RCS temperatures and pressures, other considerations require the maximum pressure versus temperature curve to be above the NPSH curve.

The pump impeller is pinned and bolted to the shaft. A close clearance thermal barrier assembly is mounted above the water lubricated bearing to retard heat flow from the pump to the seal cavity which is located above the thermal barrier. The thermal barrier assembly also tends to isolate the hot fluid in the pump from the cooler fluid above and, in the event of a seal failure, serves as an additional barrier to reduce leakage from the pump. Each pump is equipped with replaceable casing wear rings. A water lubricated bearing is located in the fluid between the impeller and thermal barrier to provide shaft support. Additional shaft support is provided by bearings in the electric motor which is directly connected to the pump shaft by a rigid coupling.

The shaft seal assembly located above the thermal barrier consists of four, face type mechanical seals, three full-pressure seals mounted in tandem and a fourth low pressure backup vapor seal designed to withstand operating system pressure with the pump stopped. The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system (Figure 4.3–4). A controlled bleed off flow through the pump seals is maintained to cool the seals

and to equalize the pressure drop across each seal. The controlled bleed-off flow is collected and processed by the chemical and volume control system. Any minor leakage past the vapor seal (the last mechanical seal), is drained to the containment sump via the containment trench and collected in the radioactive waste processing system. Normal vapor seal leakage is minor, (approximately 0 to.08 GPM per RCP), and is considered to be negligible leakage to the containment atmosphere. The seals are cooled by circulating the controlled bleed off through the heat exchanger mounted integrally within the pump cover assembly. No damage would result in the event of pump operation without cooling water for up to five minutes. To reduce plant downtime and personnel exposure to radiation during seal maintenance, the seal system is contained in a cartridge which can be removed and replaced as a unit. The seal cartridge can be replaced without draining the pump casing. The seal detail is shown in Figure 4.3–5.

A motor-mounted flywheel reduces the rate of flow decay upon loss of pump power. The combined inertia of the pump motor and flywheel is 100,000 lbm-ft^2. Flow coastdown characteristics are discussed in Section 14.6.

The RCPs are typical centrifugal volumetric flow machines. The pump response following a Loss-of-Coolant Accident (LOCA) is predicted using generally accepted methods as described in Appendix 1 of CENPD-26. CENPD-26 is a proprietary report entitled “Combustion Engineering Analytical Techniques for Evaluating Loss of Coolant Accidents”. A spectrum of breaks in the RCP discharge line have been analyzed and the results follow a predictable pattern. Assuming loss of electrical power to the pump at the start of the LOCA, it is seen that the pumps initially lose speed because the volumetric flow through the pump is not sufficient to sustain the nominal speed of rotation. The volumetric flow increases during the transient, accelerating the pump to its maximum speed. The extent of the initial loss of speed varies with the break size. The larger the break size, the less the initial deceleration and the higher the maximum speed attained. The pump attains its maximum speed following a double-ended discharge break. The calculated torque imposed on the impeller follows the same trend as the speed, with the maximum value occurring following the double-ended discharge break.

The need for a disengaging device to prevent motor overspeed following a LOCA has been evaluated. In view of the fact that the maximum anticipated pump speed is well within the safe operating limits for all rotating parts, a means to disengage the motor from the pump is not necessary.

A break in the suction piping causes the reactor coolant to flow through the pump opposite to the normal direction of flow, decelerating the rotation of the pump until it is brought to rest against the anti-reverse rotation device.

The pump/motor assembly includes motor bearing oil coolers, seal chamber, controls and instruments. Cooling water is provided from the reactor building closed cooling water (RBCCW) system. The design parameters for the RCPs are given in Table 4.3-4. The RCP instrumentation is described in Section 4.3.8.5.

The RCP and motor are supported by four support lugs welded to the volute. The pump is supported by four spring assemblies employed between the support lugs and the floor below.

Movement in the horizontal plane to compensate for pipe thermal growth and contraction is permitted. Vertical movement is not restrained.

The major pump components wetted by the primary fluid are constructed of austenitic stainless steel to minimize corrosion. These materials are listed in Table 4.3-4. The mechanical seals consist of a rotating tungsten (1) carbide ring riding over a hard carbon face. The design life of this seal arrangement is at least 50,000 hours or 5.7 years. Each seal is designed to accept a pressure drop equal to full operating system pressure, but normally operates at one-third this pressure drop.

The predicted pump performance curve is shown in Figure 4.3–6. The air-cooled, self-ventilated pump motor is sized for continuous operation at the flows resulting from four-pump operation or partial pump operation with 0.76 specific gravity water. The motor service factor is sufficient to allow 500 heatup cycles during which the nominal horsepower load will decrease from 6000 to 4500 over a period of seven hours. The motors are designed to start and accelerate to rated speed under full load when 70 percent or more of rated normal voltage is applied. The motors are contained within standard drip-proof enclosures and are equipped with electrical insulation suitable for a zero to 100 percent humidity and radiation environment of 30 R/hr.

The design requirements of the RCPs include a minimum inertia for the rotating assembly of 100,000 lbm - ft^2; to achieve this total, a flywheel with an inertia of 70,000 lbm - ft^2 has been incorporated.

Each original RCP motor flywheel assembly consists of two solid discs bolted together, shrink fitted onto and keyed to the shaft above the rotor. The dimensions of each disc are:

Outside diameter, inches	75
Thickness, inches	6
Weight, each, lb	7,250

The selection of material, machining and manufacturing operations, quality control, and the rigorous acceptance criteria established to assure the integrity of the flywheel and to minimize operating stresses include the following:

A. The principal stress is 30 percent of the yield point of the flywheel material (based on the tensile tests per ASTM-A-20) at the design overspeed of 125% of the normal operating speed not considering keyway stress concentration factors. The minimum keyway fillet radius is one eight inch.

B. The bore in the flywheel was flame cut, with a minimum of one half inch of stock left on the radius for machining to final dimensions.

C. All flywheel discs have passed the following nondestructive testing:

(1) or silicon

1. Testing by steel vendor prior to machining

    a. One-hundred percent Ultrasonic Inspection per ASME Code, Section III, paragraph N-321.1

    b. One-hundred percent Magnetic Particle Examination per ASME Code, Section III, paragraph N-322.2.

2. Testing after finish machining

    a. One-hundred percent Ultrasonic Inspection of the flats and edges, performed in compliance with ASME Code, Section III, paragraph N-321.1

    b. Liquid penetrant inspection performed in compliance with ASME Code, Section III, paragraph N-322.3 on the bore and each side of each disc for eight inches radially from the bore.

D. The finish of the flywheel bore and the finish on each side of each disc for eight inches radially from the bore are held free of nicks, center punch marks, stencil marks, holes or other stress concentrations.

E. Welding was not performed on flywheel discs.

F. The keyway fillet radius on the bottom of the keyway is 0.125 inches minimum.

The flywheel material, which is pressure vessel quality, vacuum-improved steel plate, exceeds the requirements of ASTM-A-516, Grade 70 even though it was originally specified for ASTM-A-516 Grade 65. To improve the fracture toughness properties of the material, the flame cut discs, with the one-half inch allowance for machining, were heat treated as follows:

1. heated to 1650°F ±25°F and held for minimum of 3.5 hours;

2. water quenched to below 400°F;

3. tempered at 1140°F for one-half hour per inch of thickness, air cooled.

The composition of the material as certified by the steel vendor is as follows:

ASTM-E-30 Melt Number	ASTM-E-30 Slab Number	C (Weight %)	Mn (Weight %)	P (Weight %)	S (Weight %)	Si (Weight %)
B3725	3 & 4	0.21	0.97	0.008	0.025	0.23
A8176	2	0.22	1.16	0.006	0.020	0.24

The grain size of this silicone deoxidized steel is ASTM-E-112 size 7-8. The tensile properties of the flywheel material as certified by the steel vendor are as follows:

Tensile tests per ASTM-A-20

Drop weight test (DWT) – test at +40°F per ASTM-E-208.

Melt Number	Slab Number	Tensile Strength PSI	0.2% Offset Yield Strength PSI	Elongation in 2 inch (%)	Charpy V-notch at 40°F (ft-lbf)			DWT
B3725	3	76700	49000	30	104	109	91	OK
B3725	4	75600	50700	31	103	95	97	OK
A8176	2	76700	51500	29	84	73	83	OK

The Charpy values measured for the flywheel material at 40°F are substantially higher than the data compiled on SA-516 grade 70 material by the Research & Product Development Department of CE. The report titled “Longitudinal and transverse Charpy V-notch impact and dropweight test data for normalized and tempered SA-516 grade 70 material” issued on August 26, 1971, and further identified by Laboratory Number X-24053 and R&PD Project Number 420001, was prepared for the Industrial Cooperative Program of the Material Division of the Pressure Vessel Research Committee of the Welding Research Council.

This indicates that the toughness properties of these wheels are better than typical SA-516 Grade 70. Therefore, the nil-ductility transition (NDT) temperature is lower than the highest value of -10°F reported in that report.

Since the normal operating temperature of the flywheel is approximately 100°F, a substantial margin exists between the computer K_I for large hypothetical cracks, and the toughness K_{IC}. The critical crack size therefore, is greater than five inches from the bore of the wheel.

Crack growth calculations indicate that the number of starting cycles to cause a reasonably small crack to grow to critical size is orders of magnitude greater than the number of cycles expected during the life of Millstone Unit 2.

The loads that are considered for the calculation of the stresses in the flywheel are the combined primary stresses in the flywheel at normal operating speed. They include the stress due to interference fit on the shaft as well as the stress due to centrifugal force.

Normal operating speed of the flywheel is 900 rpm (less normal slip). The flywheel has a design overspeed of rated rpm plus 25 percent, which equals 1,125 rpm. The maximum tangential stresses in the flywheel at normal operating speed are 25 percent of the material yield strength. The maximum tangential stresses in the flywheel at design overspeed are 30 percent of the material yield strength.

The RCP flywheels are accessible for 100 percent in place volumetric ultrasonic examination during in-service inspection. Two (2) access panels, 180° apart, are provided on the outside of each RCP motor.

The replacement RCP motor flywheel is a one piece forging, 75 inches OD by 12 inches thick, shrink fitted onto and keyed to the motor shaft above the rotor.

The replacement RCP motors were procured as Quality Assurance items in accordance with 10 CFR 21 and 10 CFR 50 Appendix B. They are fully interchangeable with the original RCP motors.

The selection of material, machining and manufacturing operations, quality control, and the rigorous acceptance criteria established to assure the integrity of the flywheel and to minimize operating stresses include the following:

A. The principal stress does not exceed 30 percent of the yield point of the flywheel material (based on the tensile tests per ASTM-A-370) at the design overspeed of 125 percent of the normal operating speed not considering keyway stress concentration factors. The minimum keyway fillet radius is one-eighth inch.

B. If the bore in the flywheel was flame cut, a minimum of one-half inch of stock was left on the radius for machining to final dimensions.

C. All flywheel discs passed the following nondestructive testing:

Prior to machining:

a. The flywheel material was subjected to a 100 percent volumetric ultrasonic examination using procedures and acceptance criteria as specified in Paragraphs NB2532.1 and NB2532.2 of the ASME B&PV Code, Section III.

The composition of the material as certified by the steel vendors is as follows:

Chemical Composition (weight percent)

Heat Number	C	Si	Mn	P	S	Cr	Mo	Ni	V
320647	0.16	0.04	0.23	0.004	0.003	1.60	0.43	3.60	0.01
515100	0.20	0.08	0.33	0.007	0.005	1.61	0.44	3.54	0.03

The heat treatment was as follows:

1. Heated to 1,560°F to 1,580°F and held for a minimum of 7.5 hours.

2. Water quenched.

3. Tempered at 1,110°F to 1,185°F for 12 hours, air cooled.

Tensile tests were performed on the flywheel material per ASTM A370.

The NDT temperature of the flywheel material, as obtained from DWT performed in accordance with the specification ASTM E 208 was no higher than -30°F.

The Charpy V-notch (C_v) upper-shelf energy level in the "weak" direction of the flywheel material was at least 50 ft-lbs. A minimum of three C_v specimens were tested from each forging in accordance with ASTM A 370.

The mechanical properties of the flywheel material as certified by the steel vendors are as follows:

Mechanical Properties

Heat Number	Tensile Strength (psi)	0.2% Offset Yield Strength (psi)	Elongation in 2 inches (%)	Charpy V-notch at 20°F Energy (ft-lb_f)			Drop Weight Test at -30°F
320647	110,000	93,700	21.7	130	126	92.9	No break
515100	112,000	96,000	22.4	140	130	122	No break

Fracture Toughness

The minimum static fracture toughness of the material at the normal operating temperature of the flywheel was equivalent to a critical stress intensity factor, K_{IC}, of at least $150\sqrt{}$ksi in. Compliance was demonstrated by either of the following:

a. Testing of the actual material to establish the K_{IC} value at the normal operating temperature.

b. Determining that the normal operating temperature is at least 100°F above the RT_{NDT}.

In-service inspection includes 100 percent ultrasonic examination of the flywheel of one (1) of the four (4) RCPs during each inspection interval. The acceptance criteria is in accordance with ASME - Boiler and Pressure Vessel Code, Section III, for Class I vessels.

4.3.4 REACTOR COOLANT PIPING

The RCS piping consists of two loops which connect the steam generators to the reactor vessel. Each loop can be considered to consist of 42 inch ID “hot leg” piping connecting the reactor vessel outlets to the steam generator inlets and 30 inch ID piping connecting the steam generator outlets to the RCPs and the coolant pumps to the reactor vessel inlets. The two 30 inch piping segments are referred to as the “pump suction leg” and the “cold leg” respectively. A 12 inch schedule 160 surge line connects one loop hot leg to the pressurizer. Design parameters for the reactor coolant piping are given in Table 4.3-5.

The reactor coolant piping is designed and fabricated in accordance with the rules and procedures of ANSI B31.7, Class I. The anticipated transients listed in Section 4.2.1 form the basis for the required fatigue analysis to ensure an adequate usage factor.

The reactor coolant piping is fabricated from SA 516 Gr 70 carbon steel mill clad internally with roll bonded type 304L stainless steel. A minimum clad thickness of one-eighth inch is maintained. The 12 inch surge line is fabricated from ASTM A351 Gr CF8M alloy steel.

Thermal sleeves are installed in the surge line nozzle, charging nozzles and shutdown cooling inlet nozzle to reduce thermal shock effects from auxiliary system. Clad sections of piping are fitted, where necessary, with safe ends for field welding to stainless steel components.

In response to industry experience, half nozzle replacements have been performed on selected instruments and sampling nozzles as a mitigation technique against pressurized water stress corrosion cracking (PWSCC).

The piping is shop fabricated and shop welded into subassemblies to the greatest extent practicable to minimize the amount of field welding. Fabrication of piping and subassemblies is done by shop personnel experienced in making large heavy wall welds. Welding procedures and operations meet the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. All welds are 100 percent radiographed and liquid-penetrant or magnetic-particle tested and all reactor coolant piping penetrations are attached in accordance with the requirements of ANSI B31.7. Cleanliness standards consistent with nuclear service are maintained during fabrication and erection. There are no dissimilar metal field welds.

4.3.5 PRESSURIZER

The pressurizer maintains RCS operating pressure and compensates for changes in coolant volume during load changes. Table 4.3-6 gives design parameters for the pressurizer. The pressurizer is shown in Figure 4.3–7.

Pressure is maintained by controlling the temperature of the saturated liquid volume in the pressurizer. At full load nominal conditions, slightly more than one-half the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. A number of the pressurizer heaters are operated continuously to offset the heat losses and the continuous minimum spray,

thereby maintaining the steam and water in thermal equilibrium at the saturation temperature corresponding to the desired system pressure.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 4.3–8. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level causing the reactor system pressure to decrease. The pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value and the available charging pumps in the chemical and volume control system (CVCS) are automatically started to add coolant to the system and restore pressurizer level.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program (Figure 4.3–8). The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer (in-surge), compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the RCP discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a power dependent function. A high level error signal produced by an in-surge causes the letdown control valves to open, releasing coolant to the CVCS and restoring the pressurizer to the prescribed level.

Small pressure and coolant volume variations are accommodated by the steam volume which absorbs flow into the pressurizer and by the water volume which allows flow out of the pressurizer. The total volume of the pressurizer is determined by consideration of the following factors:

a. Sufficient water volume is necessary to prevent draining the pressurizer as the result of a reactor trip or a loss-of-load accident. In order to preclude the initiation of safety injection and of automatic injection of concentrated boric acid by the charging pumps, the pressurizer is designed so that the minimum pressure observed during such transients is above the setpoint of the safety injection actuation signal (SIAS);

b. The heaters should not be uncovered by the out-surge following load decreases; ten percent step decrease, and five percent per minute ramp decrease;

c. The steam volume should be sufficient to yield acceptable pressure response to normal system volume changes during load change transients;

d. The water volume should be minimized to reduce the energy release and resultant containment pressure during a LOCA;

e. The steam volume should be sufficient to accept the reactor coolant in-surge resulting from loss-of-load without the water level reaching the safety and power-operated relief valve (PORV) nozzles;

f. During load following transients, the total coolant volume change and associated charging and letdown flows should be kept as small as practical and be compatible with the capacities of the volume control tank, charging pumps, and letdown control valves in the CVCS.

To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 4.3–9. High or low water level error signals result in the control actions shown in Figure 4.3–10 and described above.

The pressurizer heaters are single unit, direct immersion heaters which protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown. Approximately 20 percent of the heaters are connected to proportional controllers which adjust the heat input as required to account for steady state losses and to maintain the desired steam pressure in the pressurizer. These heaters are separated into two banks (approximately 160 kW each) and are provided with diverse vital power.

The remaining heaters are connected to on-off controllers. These heaters, called backup heaters, are normally deenergized but are turned on by a low pressurizer pressure signal or high level error signal. This latter feature is provided since load increases result in an in surge of relatively cold coolant into the pressurizer, decreasing the temperature of the water volume. The action of the CVCS restoring the level results in a pressure undershoot below the desired operating pressure. To minimize the pressure undershoot, the backup heaters are energized earlier in the transient, contributing more heat to the water before the low pressure setting is reached. An interlock will prevent operation of the backup heaters if the high level error signal occurs concurrent with a high pressurizer pressure signal. A low-low pressurizer level signal deenergizes all heaters to prevent heater burnout.

The pressurizer spray is supplied from each of the RCP discharges on one loop to the pressurizer spray nozzle. Automatic spray control valves control the amount of spray as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the PORVs during normal load following transients. A small continuous flow is maintained through the spray lines at all times to keep the spray lines and the surge line warm, reducing thermal shock during plant transients. This continuous flow also aids in keeping the chemistry and boric acid concentration of the pressurizer water equal to that of the coolant in

the heat transfer loops. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the RCPs are shut down.

In the event of an abnormal transient which causes a sustained increase in pressurizer pressure, at a rate exceeding the control capacity of the spray, a high pressure trip level will be reached. This signal trips the reactor and opens the two PORVs. The steam discharged by the relief valves is piped to the quench tank where it is condensed. In accordance with Section III of the ASME Boiler and Pressure Vessel Code, the RCS is protected from overpressure by two spring-loaded safety valves. The discharge from the safety valves is also piped to the quench tank. See Section 4.3.7 for the safety valve design parameters.

The pressurizer is supported by a cylindrical skirt welded to the lower head. Since the pressurizer surge line has sufficient flexibility, no provisions are made for horizontal movement and the skirt is bolted rigidly to the floor.

The pressurizer assembly was replaced in 2006 with a new pressurizer assembly fabricated from materials that are less susceptible to primary water stress corrosion cracking. The replacement pressurizer is fabricated and installed to the same design criteria as the original pressurizer with some improvements. The cylindrical shell sections, upper and lower heads including the large bore nozzles of the replacement pressurizer are forged components, thereby minimizing the welds and weld inspections. Safe ends made of stainless steel are provided as required on the large bore nozzles to facilitate field welds to the connecting piping. The interior surface of the pressurizer is clad with weld deposited stainless steel. The heater sleeves, instrument nozzles and the vent/pass nozzle are fabricated from stainless steel instead of the originally used Inconel Alloy 600 material. The nozzles are attached to the pressurizer by j-groove welds to the clad buildup. The total number of heaters is reduced from 120 to 60. The heat output of each heater is roughly twice that of the replaced heater, thus maintaining the same amount of total heat output.

A six and one-half inch inside diameter vent port was added on the upper head of the replacement pressurizer. This vent port is a substitute for the removal of the pressurizer manway during routine refueling outages with RCS nozzle dams installed. The vent port is sized to avoid a RCS pressurization that would exceed the design pressure of the RCS nozzle dams following a postulated loss of shutdown cooling with the reactor vessel head on and reactor coolant water levels one foot below the reactor vessel flange or lower. For non-routine outages where nozzle dams are installed and reactor coolant water levels are higher than one foot below the reactor vessel flange and the reactor vessel head is on, the pressurizer manway will need to be removed in order to avoid a RCS pressurization that would exceed the nozzle dam design pressure following a postulated loss of shutdown cooling. In addition, removal of either the vent port or the pressurizer manway provides an adequate RCS vent path for low temperature overpressure protection in Mode 5 and Mode 6 when the head is on the reactor vessel.

4.3.6 QUENCH TANK

The quench tank is designed to prevent the discharge of the pressurizer relief or safety valves from being discharged to the containment. The quench tank is shown in Figure 4.3–11. The steam discharged into the quench tank from the pressurizer is discharged under water by a sparger to

enhance condensation. The normal quench tank water volume of 135 cubic feet is sufficient to condense the steam released from the pressurizer safety and relief valves. The quench tank is sized to accommodate the steam released as a result of a loss-of-load accident followed immediately by an uncontrolled rod withdrawal accident with no coolant letdown or pressurizer spray.

The water temperature rise in the quench tank is limited to 281°F, assuming a maximum initial water temperature of 120°F. The gas volume in the tank is sufficient to limit the maximum tank pressure after the above steam release to 35 psia. The contents of the quench tank are cooled by recirculation through the primary drain tank and quench tank cooler. The temperature of the water in the quench tank is indicated on the main control console. A high temperature alarm is also provided. A high quench tank temperature alerts the operator to cool the tank contents.

A measurement channel provides a quench tank pressure indication on the main control console and actuates a high pressure alarm. High quench tank pressure indicates that the tank has received a discharge from the safety or relief valves, or from the HPSI test line relief valve. The operator would then take action to restore the tank to normal operating conditions.

Quench tank level indication and high and low level alarms are also provided on the main control console.

The quench tank can condense the steam discharged during a loss-of-load accident as described in Section 14.2 without exceeding the rupture disc set point, which is rated for 96 psig at 72°F and 89 psig at 350°F, assuming normal closing of the safety valves at the end of the accident. It is not designed to accept a continuous uncontrolled safety valve discharge. The rupture disc on the quench tank provides code overpressure protection of the tank. The rupture disc vents to the containment. The quench tank parameters are given in Table 4.3-7.

The tank normally contains demineralized water under a nitrogen overpressure. The sparger, spray header, nozzles and rupture disc fittings are stainless steel. The tank is designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class C.

The quench tank is located at a level lower than the pressurizer. This ensures that any PORV or pressurizer safety valve leakage from the pressurizer, or any discharge from these valves, drains to the quench tank.

4.3.7 VALVES

The design parameters for the pressurizer spray valves (RC-100E, RC-100F) are given in Table 4.3-8. The PORV isolation valve (RC-403, RC-405) parameters are given in Table 4.3-9. The position of each valve on loss of actuating signal (failure position) is selected to ensure safe operation. System redundancy is considered when specifying the failure position of any given valve. Valve position indication is provided at the main control panel.

Manually operated valves in the RCS have backseats to limit stem leakage when in the open position. Globe valves are installed with flow entering the valve under the seat. This arrangement will reduce stem leakage during normal operation or when closed.

The two PORVs, designated RC-402 and RC-404 on Figure 4.1–1, relieve sufficient pressure to avoid opening of the RCS safety valves. The relief valves are actuated by the high RCS pressure trip signal. Parameters for these valves are given in Table 4.3-10.

The valves are solenoid-operated power relief valves. The two half capacity valves are located in parallel pipes which are connected to the pressurizer relief valve nozzle in the inlet side and to the relief line piping to the quench tank on the outlet side. A motor-actuated isolation valve is provided upstream of each of the relief valves to permit isolating the valve for maintenance or in case of valve leakage.

The capacity of the PORVs is sufficient to pass the maximum steam surge associated with a continuous control rod withdrawal accident starting from low power. Assuming that a reactor trip is effected on a high pressure signal, the capacity of the PORVs is sufficient so that the safety valves do not open. The relief valve capacity is also large enough so that the safety valves do not open during a loss-of-load accident from full power. This assumes normal operation of the pressurizer spray system, and reactor trip on high pressure. The PORVs also function to provide low temperature overpressurization protection (LTOP) to the RCS. This is accomplished by manually selecting a reduced valve setpoint as described in Section 7.4.8.

Two safety valves, designated RC-200 and RC-201 on Figure 4.1–1 are located on the pressurizer to provide overpressure protection for the RCS. They are totally enclosed, backpressure compensated, spring-loaded safety valves meeting ASME Code requirements. Parameters for these valves are given in Table 4.3-11.

Both safety valves (RC-200, 201) are equipped with acoustic valve position monitors. These monitors will provide operators with an indication of the status of the safety valves. The PORVs have been upgraded to monitor their position from the control room by the position indication lights instead of the acoustic monitoring method previously used. The valve monitoring system conforms with NUREG-0578 C lessons learned from TMI Task Force Report.

The safety valves pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design (2,735 psig), following a complete loss of turbine generator load without simultaneous reactor trip. Reactor trip occurs on a high RCS pressure signal. To determine the maximum steam flow, the only other pressure relieving system assumed operational is the steam system safety valves. Conservative values for all system parameters, delay times, and core moderator coefficient are assumed. This analysis is given in Section 14.2, Loss-of-Load External Electrical Load and/or Turbine Trip.

The mounting of pressure relieving devices (safety valves and relief valves) within the RCPB and on the main steam lines outside of the containment is in accordance with the applicable provisions of ASME Boiler and Pressure Vessel Code Section III.

The pressurizer safety and relief valves are connected to nozzles on the top of the pressurizer vessel. The loads which the nozzles experience during normal plant operation and when the valves are relieving are included in the specification for the pressurizer.

All overpressure relief valves and their connected piping (i.e., headers, header connections and discharge piping) are designed to withstand the following conditions without exceeding the applicable codes primary stress allowable: maximum loads due to valve discharge thrust, internal pressure, dead weight and earthquake applied simultaneously. When more than one relief valve is attached to a piping system, the loads due to all relief valves discharging simultaneously are applied to the system along with the above mentioned primary loads. In addition, the loads from the most critical combination of valves discharging are applied. The local stresses in the main steam line outside the containment at the connection of the relief valves were computed as specified in “Welding Research Council Bulletin 107” and contained below the allowable primary stress level.

The pressurizer safety and relief valve discharge piping system was modified by deleting the loop seals upstream of the Pressurizer Safety Valves (PSVs) and Power-Operated Relief Valves (PORVs). The piping for both the PSVs and PORVs was again modified during the replacement of the pressurizer. The PORVs were also replaced with upgraded PORVs and credited as an acceptable pressurizer steam space vent path during post accident conditions. The new piping configuration was evaluated to address NUREG-0737, Action Item II.D.1 (relief valve and safety valve testing).

The thermal-hydraulic analysis was performed to calculate the transient fluid time-history forcing functions acting on the pipe segments due to each safety or relief valve discharge. These forcing functions are combined in a common entry mode to maximize the forces acting on the discharge piping. The potential case of water discharge through relief valves during low temperature modes of operation was also considered.

The structural reanalysis of the PSV and PORV discharge piping system included the normal plant loading, plant transients described in Section 4.2.1, seismic loading, and the fluid transient forcing functions. Seismic analysis of the piping system was performed by the modal analysis response spectra method. Dynamic response of the piping system to the PSV and PORV discharge loads was performed by the time-history modal superposition method.

Pipe primary and secondary stress intensities and fatigue usage factors were found to be within the Code allowable values. The pipe supports of the PSV and PORV discharge piping system were modified to accommodate the load and displacements resulting from the reanalysis.

The main steam relief valve system is designed so that the blowdown force is transmitted directly to the structure by mechanical/structural devices, and not through the piping. These relief valves are provided with discharging stacks to direct steam blowdown to atmosphere. Stacks are designed so that backpressure does not result in a valve reaction force.

Pumps and valves within the RCPB are classified as either active or inactive components. Active components are those whose operability is relied upon to perform a safety function, as well as

reactor shutdown function, during the transients or events considered in the respective operating condition categories. Inactive components are those whose operability (e.g., valve opening or closure, pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition categories. Thus, certain pumps and valves (classified as active components) within the RCPB are required not only to serve as pressure-retaining components (as in the case of passive components such as vessel and piping) but also to operate reliably to perform a safety function such as safe shutdown of the reactor and mitigation of the consequences of a pipe break accident under the loading combinations considered in RCPB as either active or inactive. The only pumps within the RCPB are the RCPs which are inactive under faulted conditions.

Active components outside of the RCPB have been designed to function as required when subjected to the loadings and the maximum pressure and temperature occurring under normal and accident conditions in the areas in which the components are located. Anticipated temperature and pressure transients which would have an effect on operability of components were specified as design requirements.

A complete list of the active pumps and valves located within the RCPB was provided previously as part of Amendment 14, and is found as Table 4.3-12.

The following is a list of active pumps and valves located outside the RCPB whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the RCPB.

Pumps

Component	Quantity	Materials Code	Test Code	Normal Position	Post LOCA Position
High Pressure Safety Injection	3	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, Nov. 1968	ASI Standard 610, ASME Power Test Code PTC- 8.2 and Standards of the Hydraulic Institute	Off	On
Low Pressure Safety Injection	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, Nov. 1968	ASI Standard 610 Standard of the Hydraulic Institute, and ASME Power Test Code PTC-8.2	Off	On

Pumps

Component	Quantity	Materials Code	Test Code	Normal Position	Post LOCA Position
Containment Spray Pumps	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968	Standard of the Hydraulic Institute	Off	On

Valves

Component	Quantity	Materials Code	Test Code	Normal Position	Post LOCA Position
High Pressure Safety Injection Pump Suction (Check)	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open
High Pressure Safety Injection Pump Discharge (Check)	4	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open
Low Pressure Safety Injection Pump Discharge (Check)	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open
Containment Isolation Valves	-	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968	-	As Required	As Required
Containment Spray (Motor Operated)	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open
Containment Sump Discharge (Check)	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open

Valves

Component	Quantity	Materials Code	Test Code	Normal Position	Post LOCA Position
Refueling Water Storage Tank Discharge (Check)	2	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Open during Injection Mode. Closed during Recirculation Mode.
Shutdown Cooling Heat Exchanger Bypass (2-SI-306) (Air Operated)	1	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Open	Open
Shutdown Cooling Heat Exchanger Flow Control (2-SI-657) (Air Operated)	1	Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968		Closed	Closed

For inactive valves and pumps other than the RCPs, the rules of the Pump and Valve Code (March 1970 Draft) for design conditions are applied in evaluating the loadings produced by the Emergency and Faulted Conditions.

For active components, additional requirements are imposed on the design to assure operability during the faulted operating conditions. As appropriate, these additional requirements consist of simulated tests and/or supplementary calculations which demonstrate that the active component will perform its required function during the specified conditions. Where calculations are employed, the primary stresses produced by the faulted conditions are limited to values less than the Emergency Condition limits of Subparagraph HB-3224.1 in all regions of the active component where deformations may impair the required function.

For the RCPs, as appropriate, the stress criteria for Vessels, as discussed above, are applied in evaluating the Emergency and Faulted Conditions (elastic analysis). These limits are consistent with the “Design by Analysis” method of Article 4, ASME Section III Code which is applied in the design calculation. For design conditions other than those explicitly addressed by Section III of the ASME Code, and where design calculations are used to evaluate stresses and deformations in pumps, the methods and criteria applied are in accordance with Article 4, ASME Section III Code. The calculations will include the effects of gross and local structural discontinuities and the loadings produced by geometrical eccentricities. Current state-of-the-art analytical methods including finite element techniques, are employed in the calculations. Experimental techniques were not employed.

4.3.8 INSTRUMENTATION APPLICATION

The measurement channels necessary for operational control and protection of the RCS are described below. A brief explanation of the purpose of each measurement channel is made and a summary of resulting action is given. A detailed description of critical instrument channel actions may be found in Chapter 7.

Four independent measurement channels are provided for each parameter which initiates protective system action. Two independent signals are required to initiate protective action, thereby, preventing spurious actions resulting from the failure of one measurement channel. This arrangement results in a high degree of protective measurement channel reliability in terms of initiating action when required and avoiding unnecessary action from spurious signals. Two independent measurement channels are provided for parameters which are critical to operational control. These control channels are separate from the protective measurement channels. To avoid control conflicts, control action is derived from only one channel at any time, while the second channel serves as a backup. This allows continued operation of the facility if one channel fails and permits maintenance on the failed channel during operation. This arrangement results in increased availability.

4.3.8.1 Temperature

4.3.8.1.1 Hot Leg Temperature

Each of the two hot legs contains five narrow range channels to measure coolant temperature leaving the reactor vessel. Four of these channels are used to furnish a hot leg temperature signal to the reactor protective system. The fifth hot leg temperature measurement channel provides a signal to the average temperature computer which is a part of the reactor regulating system (RRS). The average temperature of the loop is recorded on a two-channel recorder in the main control room. The second channel on each loop’s recorder records an average temperature reference signal received from the RRS.

A high temperature alarm is provided on this channel to alert the operator to a high temperature condition. The temperature from this measurement channel is indicated in the main control room in addition to being recorded. The other hot leg temperature channels are also displayed in the main control room.

4.3.8.1.2 Cold Leg Temperature

Each of the four cold legs contains three temperature measurement channels. The cold leg Resistance Temperature Detectors (RTD) are located downstream of the RCPs. Two channels from each cold leg (four per heat transfer loop) are used to furnish a cold leg coolant temperature signal to the reactor protective system (RPS). All eight of these cold leg temperature measurements are indicated in the main control room. Two of the remaining four cold leg temperature measurement channels, one each from opposite loops, are narrow range temperature measurements that input to the RRS for calculation of average temperature and the Feedwater regulating system for dynamic compensation in single element control. These loops also provide

control room indication of narrow range temperature. The remaining two cold leg measurement channels, one each from opposite loops, are wide range temperature measurements that provide input to the subcooled margin monitor (Inadequate Core Cooling ICC) and Low Temperature Over Pressure (LTOP) channels. These loops also provide control indication and recording of wide range temperature.

4.3.8.1.3 Surge Line Temperature

This measurement channel provides an indication of surge line temperature in the main control room. A low surge line temperature condition activates an alarm in the main control room. The low-temperature alarm during normal operation is an indication that the continuous spray rate has decreased.

4.3.8.1.4 Pressurizer Vapor Phase Temperature

The pressurizer vapor phase RTD is located on the upper dome of the pressurizer. This channel provides a wide range temperature indication of the temperature of the steam phase in the pressurizer.

4.3.8.1.5 Pressurizer Water Phase Temperature

The pressurizer water phase RTD is located at an elevation below the top of the pressurizer heaters. This channel provides a wide range temperature indication in the main control room and is used during plant heatup and cooldown.

4.3.8.1.6 Spray Line Temperature

An RTD in each spray line provides a temperature indication and a low temperature alarm in the control room for each spray line. A low temperature alarm during normal operation is an indication the continuous spray rate has decreased.

4.3.8.1.7 Relief and Safety Valve Discharge Temperature

Temperatures in the pressurizer safety valve and PORV discharge lines are measured and indicated in the main control room. A high temperature in one of these lines is an indication that the associated valve may be leaking. High temperature alarms are provided to alert the operator to this condition.

4.3.8.1.8 Quench Tank Temperatures

The temperature of the water in the quench tank is indicated in the main control room. A high temperature alarm is also provided. A high quench tank temperature alerts the operator to the requirement for cooling of the tank contents.

4.3.8.1.9 Reactor Vessel Flange Seal Leakage Temperature

This RTD is located in the reactor vessel flange leak-off line. The channel is displayed in the main control room and actuates a high temperature alarm. A high temperature is indicative of excessive leakage past the first reactor vessel flange seal.

4.3.8.1.10 RCS High Point Vents Leakage Temperature

Thermocouple installed on the downstream side of solenoid valve train of the reactor vessel head vent system is utilized to monitor leakage past the system solenoid valves. Under normal operating conditions, the thermocouple will measure the ambient temperature in the piping downstream of the solenoid valve train. The output of each thermocouple is continuously recorded by the plant computer. Note that monitoring of the leakage past the PORVs in the pressurizer steam space vent path is described in Section 4.3.8.1.7, above.

Any leakage through the system valves will cause an increased temperature in the downstream piping which will be detected by the thermocouple. At a predetermined setpoint, an alarm will be actuated, identifying a high temperature reading on the appropriate thermocouple. Once a high temperature alarm is received, further actions will be governed by the Technical Specifications for reactor coolant system leakage.

4.3.8.2 Pressure

4.3.8.2.1 Pressurizer Pressure

Four independent narrow range pressure channels are provided for initiation of protective systems action. The pressure transmitters are connected to the upper portion of the pressurizer via the upper level measurement nozzles and measure pressurizer vapor pressure. All four channels are indicated in the main control room and actuate separate high, low, or low low pressure alarms in the control room.

The protection actions these pressure signals initiate are:

1. Reactor trip on high primary system pressure. The reactor trip signals are also used to open the PORVs;

2. Safety injection system actuation on low low primary system pressure;

3. Reactor trip on a low primary system pressure. The set point is a function of the coolant temperatures in the hot and cold legs. The variable set point has high and low limits alarmed in the control room and is not allowed to decrease below 1865 psia.

4.3.8.2.2 Pressurizer Pressure

Two independent pressure channels provide narrow range pressure signals for controlling the pressurizer heaters and spray valves. The output of one of these channels is manually selected to perform the control function. During normal operation, a small group of heaters are proportionally controlled to offset heat losses. If the pressure falls below a low pressure set point, all of the heaters are energized. If the pressure increased above the high pressure set point, the spray valves are proportionally opened to increase the spray flow rate as pressure rises. An interlock will prevent operation of the backup heaters in the event of a high level error signal concurrent with a high pressure condition. These two channels are also used to provide pressurizer pressure signals to the RRS. The two channels are continuously recorded in the main control room and are provided with high and low pressure alarms.

4.3.8.2.3 Pressurizer Pressure

Two low-range pressure measurement channels provide a control room indication of RCS pressure during plant startup and shutdown in the main control room. They also provide independent pressure signals to the shutdown cooling suction isolation valves (refer to Section 9.3.4.1) which prevent these valves from opening above a selected set point. If the shutdown cooling suction valves are open when the pressure exceeds a selected set point (280 psia), an annunciator receives a signal to alarm from these pressure channels. The channels also provide signals to actuate the PORVs for LTOP. These two instrument channels are independent and redundant.

4.3.8.2.4 Quench Tank Pressure

This measurement channel provides a quench tank pressure indication in the main control room and actuates a high pressure alarm. High quench tank pressure indicates that the tank has received a discharge from the safety or relief valves, or from the HPSI test line relief valve. The operator will then take action to restore the tank to normal operating conditions.

4.3.8.3 Level

4.3.8.3.1 Pressurizer Level

Two pressurizer level channels are used to provide two independent level signals for control of the pressurizer liquid level. These signals are used to deenergize the pressurizer heaters on low low pressurizer level to prevent heater burn out, provide input to one channel in the two-channel recorder in the control room, and actuate high and low pressurizer level alarms in the main control room. The second channel on the level recorder records the programmed pressurizer level setpoint computed by the RRS as a function of the average reactor coolant temperature. The level transmitters are compensated for the steam and water densities existing in the pressurizer during normal operation.

The liquid level in the pressurizer is programmed to vary as a function of average reactor coolant temperature. This level set point is computed by the RRS and furnished to controls associated

with these measurement channels. One of the two measurement channels is manually selected to furnish an actual liquid level signal to the controls. If these two signals differ, the level control adjusts the CVCS charging or letdown flow rates to make the difference zero. Each of the level channels is indicated in the main control room and is equipped with a high and low level alarm in the control room.

4.3.8.3.2 Pressurizer Level

One wide range pressurizer channel is provided for main control room indication of pressurizer level during plant startup and shutdown.

4.3.8.3.3 Quench Tank Level

A quench tank level channel indicates quench tank level in the main control room. The transmitter also activates high and low level alarms in the main control room.

4.3.8.4 Reactor Coolant Loop Flow

For independent differential pressure measurement channels are provided in each heat transfer loop to measure the pressure drop across the steam generators. Four pressure taps are located in each hot leg piping section just before the elbow entering the steam generator and four pressure nozzles are located in the steam generator outlet plenum. Four differential pressure transmitters are connected between the four hot leg nozzles and the steam generator nozzles, resulting in four steam generator differential pressures.

The outputs of the transmitters are sent to four analog summing devices in the low total flow trip logic. Each summing device receives two differential pressure signals with the summation of these signals representing the total core flow at all times, even during coast down transients.

The summing devices provide four independent total flow signals. The four signals are indicated separately in the main control room and activate separate low-flow alarms. In the RPS, they are compared with the low-flow reactor trip set point, selected by the operator to correspond with the number of operating RCPs. If two channels indicate flow which is less than the flow set point, the reactor is tripped.

4.3.8.5 Reactor Coolant Pump Instrumentation

The RCPs and motors are equipped with the instrumentation necessary for proper operation and to warn of incipient failures. (See Figure 4.3–4.) A description of the major channels follows:

4.3.8.5.1 Pump Seal Temperatures

The reactor coolant temperature in the lower seal cavity may be indicated in the main control room through its selection on a multiposition switch. The switch also permits display of all of the remaining temperature measurement channels for each pump. The pump seal temperature is alarmed to alert the operators to a high-temperature condition. A high temperature condition is an

indication that the integral heat exchanger is not performing satisfactorily or is a backup indication that the component cooling water flow has decreased or supply temperature has increased.

4.3.8.5.2 Motor Stator Temperatures

Each RCP motor is provided with six RTD's embedded in the stator windings. During initial pump testing the highest reading RTD was selected for this temperature measurement control. The signal output of this RTD may be selected for indication in the main control room by a multiposition switch. Should stator temperature exceed a predetermined limit, a high temperature alarm will be sounded in the control room. High temperature is detrimental to motor winding insulation life, and may be caused by high ambient temperature, reduction in the cooling air flow to the stator or inadequate time delay between successive starts of the motor.

4.3.8.5.3 Motor Thrust Bearing Temperatures

Temperatures of the motor upward thrust bearing, and downward thrust bearing may be indicated in the main control room through selection on a multiposition switch. A high temperature alarm is provided for each pump in the main control room which is annunciated if any one of the bearing temperatures in the pump exceeds a safe value.

4.3.8.5.4 Pump Controlled Bleed-Off Temperature

The temperature of the controlled bleed-off flow may be displayed in the main control room through its selection on a multiposition switch. An alarm signal is provided should the controlled bleed off temperature exceed a high limit. A high temperature condition is an indication that the integral heat exchanger is not operating properly.

4.3.8.5.5 Antireverse Device Bearing Temperature

This measurement channel provides a status on the operating temperature of the antireverse device bearing. Display in the main control room is available through selection on a multiposition switch. A high temperature alarm is provided to alert plant operators to an abnormal condition.

4.3.8.5.6 Upper and Lower Guide Bearing Temperature

The upper and lower guide bearing temperatures may be displayed in the main control room at the option of the operators. An alarm signal is provided if the high temperature limit is exceeded.

4.3.8.5.7 Lube Oil Cooler Inlet and Outlet Temperature

The inlet and outlet temperatures of the lube oil cooler are available for display in the main control room and are alarmed to alert plant operators to high temperature conditions.

4.3.8.5.8 Lower Bearing Oil Temperature

The lower bearing oil temperature may be displayed in the main control room at the option of the plant operators.

4.3.8.5.9 Pump Seal Pressures

The middle, upper, and controlled bleed-off pump seal cavities in each pump are provided with pressure sensors which generate a signal proportional to the pressure within the cavity. The pressure in any seal may be selected for display in the main control room through its selection on a three-position switch. A high and low pressure alarm is annunciated for the middle and upper seal measurement channels. Abnormally, high pressure in the upper and middle seal cavities indicates a failed or failing lower or middle seal. A low pressure condition in the middle seal cavity indicates a failed or failing upper seal. A recorder in the control room records seal pressures in order to recognize pump seal degradation.

4.3.8.5.10 Motor Oil Lift Pressure

This pressure measurement channel provides a signal to the RCP control circuit to prevent motor start with insufficient motor thrust bearing oil pressure.

4.3.8.5.11 Lube Oil Filter Pressure Differential

There are three lube oil filters, each with differential pressure indication capability. These measurement channels provide alarms for high differential pressure across any of the filters, indicating clogging.

4.3.8.5.12 Pump Controlled Bleed-Off Flow

Flow instruments are used to measure the controlled bleed off flow from the pump upper seal cavity to the chemical and volume control system. These instruments provide a remote indication of the flow rate in the East and West electrical penetration rooms, and annunciate high and low flow alarms in the control room. There is also an input to the plant computer and a recorder in the control room which records flow.

4.3.8.5.13 Lube Oil and Antireverse Device Lube Oil Flow Switch

These measurement channels provide alarm signals when a low flow condition exists in the thrust bearing lube oil loop or in the antireverse device lube oil flow.

4.3.8.5.14 Motor Oil Reservoir Level

Float-type sensors or differential pressure transmitters are used to produce signals proportional to the oil levels in the upper and lower motor oil reservoirs. These signals are used to provide an indication on the main control panel of the oil level in each oil reservoir. Either oil reservoir level

may be selected for display in the control room through use of a two-position switch. Each signal also annunciates a high or low level alarm in the control room.

4.3.8.5.15 Vibration Instrumentation

Motor vibration is sensed by two velomitor probes attached to the upper motor frame. Excessive motor vibration will cause an alarm on the plant process computer.

Pump shaft orbit is monitored by two proximity probes mounted 90 degrees to each other in a horizontal plane. A separate key phasor probe provides a single pulse per revolution as a reference for determining angular changes in shaft orbit. Outputs from the probes are also recorded.

4.3.8.5.16 Reverse Rotation Switch

Reverse rotation of an RCP is sensed by a reverse rotation switch. This switch actuates an alarm in the control room. Reverse rotation indicates failure of the mechanical anti-reverse device.

4.3.8.5.17 (Deleted)

4.3.8.5.18 RCP Underspeed Reactor Trip

An RCP underspeed trip has been added which uses a speed sensor on the RCP motor. The sensor generates a frequency signal proportional to speed which is in turn converted to a voltage. This voltage signal is maintained by the RPS, with one pump signal being monitored by only one channel in the RPS. When any two pump speeds drop below 830 RPM a reactor trip will occur. This trip is designed to protect against simultaneous loss of all pumps and the resulting loss of coolant flow, with a faster reaction or trip time, than is provided by the steam generator differential pressure signal.

4.3.9 REACTOR COOLANT VENTING SYSTEM

The RCS venting system provides the capability for removing noncondensible gases collected in the system in order to allow for satisfactory long term core cooling.

The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent noncondensible gases will provide additional assurance that the requirements of 10 CFR 50.44 will be met. For events beyond the design basis, this venting capability will substantially increase the ability to deal with large quantities of noncondensible gas.

Reactor Vessel Head Vent System

The reactor coolant head vent system is equipped with two (2) solenoid-operated globe valves in series in each piping train. Each valve has remote-manual control capability from the control room with open and closed position indication. Provision of two (2) solenoid operated globe valves in series for each vent train minimizes the probability of a vent path failing to close, once

opened. The power source for each valve train is an independent redundant DC emergency bus, energized from separate redundant battery systems. In addition, the valves also receive power from redundant independent AC emergency buses. All valves fail closed upon loss of power supply to the actuator. The solenoid vent valves are qualified to IEEE-344-1975.

A manual isolation valve is installed downstream of the solenoid operated vent valves. This valve is normally open and is intended to isolate leakage that may develop through the solenoid-operated valves. Also, a manual isolation valve is located just upstream of the two solenoid valves on each train of the RCS head vent piping, capable of isolating the affected train. Piping upstream of the valve is designed to reactor coolant system pressure rating, providing an at-power isolation capability.

The discharge sparger for the reactor coolant head vent system is located in the vicinity of the containment air coolers where the vented gases will be cooled, mixed with additional containment air, and discharged into the lower elevation of the containment. Uniform mixing of the containment post accident atmosphere is provided by the post-accident recirculation system. The RCS vent system piping has been analyzed in accordance with ASME, Section III, of the Boiler and Pressure Vessel Code, for the Class 1 portion of the system. The balance of the system has been analyzed pursuant to ANSI B31.1 Power Piping Code.

Pressurizer Steam Space Vent System

Venting of the noncondensible gases can also be performed from the pressurizer. The pressurizer venting system consists of two flow paths with redundant isolation valves in each flow path. Each flow path has a power operated relief valve and a block valve. The block valve in each train is normally open. During the pressurizer replacement project the two (2) power operated relief valves were upgraded so they can be credited to perform the venting function post accident. The PORV’s are EEQ qualified and receive power from battery backed independent vital DC buses. The noncondensibles are vented via the PORV discharge piping into the quench tank and then to the waste gas header as necessary. If the pressure reaches in excess of the rupture disc relief pressure, the noncondensibles would be released into the containment. The PORV inlet and discharge piping has been analyzed in accordance with ASME, Section III, of the Boiler and Pressure Vessel Code, Class 1 and Class 2 and 3 requirements respectively.

The reactor vessel and pressurizer vents were designed to utilize existing penetrations within each vessel. The system can pass in excess of the gas volume equivalent to one-half the RCS volume in one (1) hour. Although the RCS vents are larger than the size corresponding to the definition of LOCA (10 CFR 50, Appendix A), consequences of ruptures of the vents are bounded by the results of current small break loss of coolant accident (SBLOCA) analyses.

Since the vent lines are sized larger than the size corresponding to the definition of LOCA, the system is equipped with two (2) solenoid-operated globe valves in series in each piping train. Each valve has remote-manual control capability from the control room with open and closed position indication. A manual isolation valve is installed downstream of the solenoid operated vent valves on each of the reactor and pressurizer vent systems. This valve is normally open and is intended to isolate leakage that may develop through the solenoid operated valves. Also, a manual

isolation valve is located just upstream of the two solenoid valves on each train of the RCS head vent piping, capable of isolating the affected train. Piping upstream of the valve is designed to reactor coolant system pressure rating, providing an at-power isolation capability.

The design of the venting system minimizes the probability of a vent path failing to close, once opened. This has been accomplished by providing two (2) solenoid operated globe valves in series for each vent train. The power source for each valve train is an independent redundant DC emergency bus, energized from separate redundant battery systems. In addition, the valves also receive power from redundant independent AC emergency buses. All valves fail closed upon loss of power supply to the actuator.

The discharge sparger for the RCS vent system is located in the vicinity of the containment air coolers where the vented gases will be cooled, mixed with additional containment air, and discharged into the lower elevation of the containment. Uniform mixing of the containment post-accident atmosphere is provided by the post-accident recirculation system.

The RCS vent system piping has been analyzed in accordance with ASME, Section III, of the Boiler and Pressure Vessel Code, for the Class 1 portion of the system. The balance of the system has been analyzed pursuant to ANSI B31.1 Power Piping Code. The solenoid vent valves are qualified to IEEE-344-1975.

4.3.10 PERMANENT REACTOR CAVITY SEAL

The permanent reactor cavity seal is designed to contain refueling water in the refueling cavity for fuel shuffle during outages. The reactor cavity seal consists of seal membrane sub-assembly and support structure sub-assembly. The seal membrane sub-assembly consists of a stainless steel membrane, inner and outer legs attached to the reactor vessel seal ledge and the embedment ring by circumferential fillet welds all around the reactor vessel. The support structure assembly functions as the load bearing assembly consisting of radial members around the annulus, resting on the reactor vessel flange and the embedment ring. The reactor cavity seal is shown in Figure 4.3–12.

The reactor cavity seal has multiple openings, which provide access to the ex-core nuclear instrumentation as well as ventilation for the reactor cavity cooling air. These openings have hinged cover plates, which are closed for flooding the transfer canal during refueling. O-rings are installed to provide watertight seal to prevent leakage into the reactor cavity. Prior to reactor operation, the hinged cover plates will be laid back to allow for ventilation of cooling air from the reactor cavity.

The cavity seal is designed to withstand the pressure loads from the refueling water as well as the motions of the reactor vessel and containment building due to seismic displacements. It can also accommodate axial and radial growth from the normal and transient thermal conditions of the reactor vessel. The cavity seal is also designed and tested to withstand loads imposed by a dropped fuel assembly. The cavity seal is designed for 80 heat up/cool down cycles and 50 cycles of maximum allowed water static head.

Leak before break analysis is applied as the basis for design of the reactor cavity seal and neutron shielding. Compartmental pressurization due to reactor coolant loop rupture is excluded for the design of reactor cavity seal and the neutron shielding in accordance with Reference 3.A-29.

The support structure is fabricated from 300 series stainless steel conforming to ASME Section II Part A. The seal membrane is allowed to deflect such that it can rest on the support structure during flooding. The cavity seal membrane, although not required, is designed and analyzed to the guidelines of ASME B&PV Code Section III, Appendix XIII and ASME Section II, Parts A and C, 1995 Edition, through 1996 Addenda.

TABLE 4.3-1 REACTOR VESSEL PARAMETERS

Design Pressure, psig	2485
Design Temperature, °F	650
Nozzles	
Inlet (4 each), inches	30
Outlet (2 each), ID inches	42
CEDM (69), ID, inches	2.718
Instrumentation (8), nominal, inches	4 5/8
Vent (1), nominal, inches	0.75
Dimension	
Inside Diameter, nominal, inches	172
Overall Height, Including CEDM Nozzles, inches	503.75
Height, Vessel Without Head, inches	408-9/16
Wall Thickness, minimum, inches	8-5/8
Upper Head Thickness, minimum inches	7-3/8
Lower Head Thickness, minimum inches	4-3/8
Cladding Thickness, Replacement Closure Head, inches nominal	0.25
Cladding Thickness, Replacement Closure Head, inches minimum after machining	1/8
Cladding Thickness, Bottom Head, minimum	3/16
Cladding Thickness, Remainder of vessel, minimum, inches	1/8
Material	
Shell/Bottom Head	SA-533-65 Grade B, Class 1 Steel
Closure Head	SA 508 Grade 3 Class 1 Carbon Steel
Forgings	A-508-64, Class 2
Cladding	Stainless Steel (1) and NiCrFe Alloy
CEDM Nozzles	Ni-Cr-Fe Alloy welded to SA-182, F316LN
Instrumentation Nozzles	Ni-Cr-Fe Alloy welded to SA-182, F316LN

(1)Weld deposited austenitic stainless steel with a composition approximately equivalent to SA-240, type 304 in contact with coolant.

TABLE 4.3-1 REACTOR VESSEL PARAMETERS (CONTINUED)

Dry Weights	
Closure Head lb.	147,900
Vessel, without flow skirt, lb.	682,000
Studs, Nuts and Washers, lb.	38,900
Total lb.	878,900
Volumes	
Bottom of Core, ft^3	1113.11
Center of Core, ft^3	1680.93
Top of Core, ft^3	2248.74
Full Vessel, ft^3	4651.27

TABLE 4.3-2 STEAM GENERATOR PARAMETERS

Number	2
Type	Vertical U-Tube
Number of Tubes (Design/Actual)	8523
Tube Outside Diameter, inches	0.750
Heat Transfer Rate, each, Btu/hr	4.63×10^9
Nozzles and Manways	
Primary Inlet Nozzle (1 each), ID, inches	42
Primary Outlet Nozzle (2 each), ID, inches	30
Steam Nozzle (1 each), ID, inches	34
Feedwater Nozzle (1 each), nominal, inches	18
Instrument Taps (12 each), nominal, inches	1
Primary Manways (2 each), ID, inches	18
Secondary Manways (2 each), ID, inches	16
Secondary Handhole (4 each), ID, inches	8
Bottom Blowdown (2 each), nominal, inches	4
Nitrogen Addition (1 each), nominal, inches	1
Wet Layup (1 each), nominal, inches	2
Primary Side Design	
Design Pressure, psig	2485
Design Temperature, °F	650
Design Thermal Power (NSSS), MWt	2715
Coolant Flow (Each), lb/hr	74×10^6
Normal Operating Pressure, psia	2250
Coolant Volume, each, ft^3	1693
Secondary Side Design	
Design Pressure, psig	1000
Design Temperature, °F	550
Normal Operating Steam Pressure, psia	880, Full Load
Normal Operating Steam Temperature, °F	530, Full Load
Blowdown Flow, Design, Maximum, Each, lb/hr	112,000

TABLE 4.3-2 STEAM GENERATOR PARAMETERS (CONTINUED)

Steam Flow, Each, lb/hr	5.9×10^6
Steam Moisture Content, Maximum, percent	2.0
Feedwater Temperature, °F	435
Number of Steam Primary Separators, each Steam Generator	170
Number of Secondary Separators, each Steam Generator	170
Dimensions	
Overall Height, Including Support Skirt, inches	749
Upper Shell Outside Diameter, inches	239.75
Lower Shell Outside Diameter, inches	166
Weights	
Dry, lb.	1,070,400
Flooded, lb.	1,666,600
Operating, lb.	1,283,000

TABLE 4.3-3 MAIN STEAM SAFETY VALVE PARAMETERS

Main Steam Piping Design Pressure, psig	1,100 (NOTE)
Main Steam Piping Design Temperature, °F	600
Fluid	Saturated Steam

Valve Number	Set Pressure psia	Capacity lb/hr (each valve)
1, 2	1000	794,060
3, 4	1005	794,060
5, 6	1015	794,060
7, 8	1025	794,060
9, 10	1035	794,060
11, 12	1045	794,060
13, 14, 15, 16	1050	794,060

Total Capacity (16 valves) lb/hr 12,704,960

Material

Body	ASTM A105 Gr II
Disc	ASTM 565 Gr 616 or ASME SB-637 UNS N07750 Type 3
Trim	ASTM A451 Gr CPF8

NOTE: ASME Section III Code requires pressure rise to be limited to no more than 10% above piping design pressure. Operability of the safety valves ensures that the main steam system pressure will be limited to within 110% of its design pressure. Design pressure for main steam safety valves - 1035 psig.

TABLE 4.3-4 TECH PUB REVIEW PKG FOR T4.3-4REACTOR COOLANT PUMP PARAMETERS

Synchronous Speed, rpm	900
Type	Vertical, Limited Leakage, Centrifugal
Shaft Seals	Type N-9000 Assembly
Stationary Face Ring	Carbon-Morganite CNFJ
Rotating Face Ring	Tungsten or Silicon Carbide
Design Pressure, psig	2,485
Design Temperature, °F	650
Normal Operating Pressure, psig	2,235
Maximum Operating Temperature, °F	549
Design Flow, gpm	81,200
Total Dynamic Head, ft, Minimum	243
Maximum Flow (one-pump Operation), gpm	120,000
Dry Weight, lb.	168,050
Flooded Weight, lb.	175,050
Reactor Coolant Volume, ft^3 per pump	112
Shaft Material	ASTM A-182 Type F-304
Casing Material	ASME SA-351 Gr CF8M
Casing Wear Ring Material	ASTM A-351 Gr CF8
Hydrostatic Bearing	
Bearing Material	ASTM A-351 Gr CF8
Journal Material	ASTM A-351 Gr CF8
Motor	
Voltage, volts (source)	6,900
Voltage, volts (rated)	6,600
Frequency, Hz	60
Phase	3
Horsepower (rated)/rpm	6,500/887
Synchronous Speed, rpm	900

TABLE 4.3-4 TECH PUB REVIEW PKG FOR T4.3-4REACTOR COOLANT PUMP PARAMETERS (CONTINUED)

Instrumentation	**Quantity** (per pump)
Seal Temperature	1
Pump Casing Differential Temperature	1
Seal Pressure	3
Controlled Bleed off Flow	1
Controlled Bleed off Temperature	1
Motor Oil Level	2
Motor Bearing Temperature	5
Motor Stator Temperature *	6
Reverse Rotation Flow	1
Vibration Monitoring System ***	5
Oil Lift Pressure	1
Lubrication Oil Temperature	3

Total Seal Assembly Leakage (Normal and Stand-by Operation)

Three Pressure Seals Operating, gpm **

Two Pressure Seals Operating, gpm **

One Pressure Seal Operating, gpm **

* NMAC, Main Coolant Pump Seal Maintenance Guidelines, Final Report-December 1993 can be used to calculate these volumes more accurately at different conditions.

** Six (6) resistance temperature detectors (RTDs) were imbedded into the stator (2 per phase) during fabrication. During first full load plant operation, all 6 RTDs were monitored. The RTD reading the highest temperature was chosen for use. The other five RTDs remain as spares.

*** Vibration Monitoring System consists of two (2) velomitors on the motor casing, and two (2) proximity probes and one (1) key phasor on the pump shaft.

TABLE 4.3-5 REACTOR COOLANT PIPING PARAMETERS

Number of loops	2
Flow per loop, lb/hr	61×10^6
Pipe Size	
Reactor outlet, ID, inches	42
Reactor inlet, ID, inches	30
Surge line, nominal, inches	12
Design Pressure, psig	2485
Design Temperature, °F	650
Velocity, Hot leg, ft/sec	40.4
Velocity, Cold leg, ft/sec	36.3

TABLE 4.3-6 PRESSURIZER PARAMETERS

Design Pressure, psig	2,485
Design Pressure, °F	700
Normal Operating Pressure, psia	2250
Normal Operating Temperature, °F	653
Internal Free Volume, ft^3	1500
Normal Operating Water Volume, Full Power, ft^3	800
Normal Steam Volume, Full Power, ft^3	700
Installed Heater Capacity, kW	1600
(Note: Total Heater Capacity may be less due to Heater unavailability)	
Spray Flow, Maximum, gpm	375
Spray Flow, Continuous, gpm	1.5
Nozzles	
Surge Line (1) nominal, inches	12
Safety Valves (2) nominal, inches	4
Relief Valve (1) nominal, inches	4
Spray (1) nominal, inches	4
Heater Sleeve (60) ID, inches	0.905
Manual Vent (1) nominal inches	
Manway nominal, inches	16
Alternate Vent Port nominal, inches	6.5
Instrument Nozzles	
Level (4) nominal, inches	1
Temperature (2) nominal, inches	1
Pressure (2) nominal, inches	1
Materials	
Vessel	SA-508 Grade 3, Class 2
Cladding - Cylinder Shell, Upper and Lower Head	Type 308 Stainless Steel (1)

(1)Weld deposited austenitic stainless steel in contact with coolant.

TABLE 4.3-6 PRESSURIZER PARAMETERS (CONTINUED)

Dimensions		
Overall Length, inches (bottom of support skirt to tip of relief nozzle)		434.06
Outside Diameter, inches	106.56	
Inside Diameter, inches (with cladding)	96.16	
Cladding Thickness, inches (minimum)	1/8	
Dry Weight, Including Heaters, lb.	202,731	
Flooded Weight, Including Heaters, lb.	297,433	

TABLE 4.3-7 QUENCH TANK PARAMETER

Design Pressure, psig	100 INT/15 EXT
Design Temperature, °F	350
Normal Operating Pressure, psig	3
Normal Operating Temperature, °F	120
Internal Volume, ft^3	217
Normal Water Volume, ft^3	135
Normal Gas Volume, ft^3	82
Blanket Gas	Nitrogen
Nozzles	
Pressurizer discharge (1), inch, nominal	10
Demineralized water (1), inch, nominal	2
Rupture Disc (1), inch, nominal	18
Drain (1), inch, nominal	3
Temp. Instrument (1), inch, nominal	1
Level Instrument (1), inch, nominal	1
Pressure and Level Instrument (1) inch, nominal	1
Vent (1), inch., nominal	1.5
Vessel Material	ASTM-A-240 TP 304
Dimensions	
Overall Length, inches	145.5
Outside Diameter, inches	60
Dry Weight, lb.	4600
Flooded Weight, lb.	18,120

TABLE 4.3-8 PRESSURIZER SPRAY (RC-100E, RC-100F) VALVE PARAMETERS

Service - Pressurizer Spray Control	
Design Temperature, °F	650
Design Pressure, psig	2485
Flow, gpm	440
Pressure Drop, psi	8.5 - 40
Failure Position	Failed Closed
Manufacturer	Fisher Controls Co.
Design Code	Pump and Valve Code, Nov. 1968 Draft, Class I
Seismic Class	I
Materials	Body 316 SST ASTM A351-CF8M

TABLE 4.3-9 POWER-OPERATED RELIEF VALVE ISOLATION VALVE PARAMETERS (RC-403, RC-405)

Service - Pressurizer Power-Operated Relief Valve Isolation	
Design Temperature, °F	675
Design Pressure, psig	2,485
Actuator	Electric Motor
Failure Position	As Is
ANSI Class	2,500 lb
Manufacturer	Velan Valve Company
Design Code	Pump and Valve Code Nov. 1968 Draft, Class I
Seismic Class	I
Materials	Body ASTM A182 Grade F316

TABLE 4.3-10 PRESSURIZER POWER-OPERATED RELIEF VALVE PARAMETERS (RC-402, RC-404)

Design Pressure, psig	2485
Design Temperature, °F	675
Fluid	Saturated Steam 0.1% (wt) Boric Acid
Number	2
Capacity, lb/hr (minimum) each	153,000
Type	Solenoid Operated
Set Pressure, psig	2,385 and 400 psig (low temperature overpressurization)
Failure Position	Closed
Design Code	ASME Section III, 1977 Edition through Winter 1979 Addenda
Materials	Body 316L SS, SA182

TABLE 4.3-11 PRESSURIZER SAFETY VALVE PARAMETERS (RC-200, RC-201)

Design Pressure	2,485
Design Temperature, °F	675
Fluid	Saturated Steam, 0.1% (wt) Boric Acid
Set Pressure	
RC-200, psig	2,485
RC-201, psig	2,485
Capacity, lb/hr, at set pressure	
RC-200	294,000
RC-201	294,000
Type	Spring loaded-balances bellows, enclosed bonnet
Accumulation, %	3
Back pressure Compensation	Yes
Blowdown, %	12
Design Code	ASME Section III, Class A, 1968 Edition, Addenda through Summer of 1970, Code Case 1344-1
Materials	Body 316 SST, ASTM A 182

TABLE 4.3-12 ACTIVE AND INACTIVE VALVES IN THE REACTOR COOLANT SYSTEM BOUNDARY

Line	Valve Type/ Number	Classification Active - A Inactive - I	Normal Position	Post LOCA Position
Shutdown Cooling	Motor / 2	A	Closed	Closed
Charging [a]	Air / 2	A	Open / Closed	Open / Closed
	Check / 2	A	Open / Closed	Open / Closed
Letdown	Air / 2	A	Open	Closed
	Manual / 1	I	Open	Open
Auxiliary Spray	Air / 1	I	Closed	Closed
	Check / 1	I	Closed	Closed
Pressurizer Spraya	Air / 2	I	Open / Closed	Closed
	Manual / 6	I	Open	Open
Pressurizer Relief	Motor / 2	I	Open	Open
	Solenoid / 2	I	Closed	Closed
Pressurizer Safety	2	I	Closed	Closed
Safety Injection Tank	Motor / 4	I	Open	Open
	Check / 4	A	Closed	Open
Leakage Control (SIS)	Air / 4	I	Closed	Closed
Safety Injection	Check / 8	A	Closed	Open
LPSI Header	Check / 4	A	Closed	Open
	Motor / 4	A	Closed	Open
HPSI Header	Check / 8	A	Closed	Open
	Motor / 8	A	Throttled Open	Open
Drain: Reactor Coolant Loop	Manual / 13	I	Closed	Closed
	Air / 1	I	Open	Open
Charging	Manual / 4	I	Closed	Closed
Pressurizer Spray	Manual / 4	I	Closed	Closed
Letdown	Manual / 4	I	Closed	Closed
Safety Injection	Manual / 14	I	Closed	Closed
Auxiliary Spray	Manual / 4	I	Closed	Closed

TABLE 4.3-12 ACTIVE AND INACTIVE VALVES IN THE REACTOR COOLANT SYSTEM BOUNDARY (CONTINUED)

Line	Valve Type/ Number	Classification Active - A Inactive - I	Normal Position	Post LOCA Position
Shutdown Cooling	Manual / 2	I	Closed	Closed
Vent / Test: Reactor Vessel	Manual / 2	I	Closed	Closed
Vent / Test: Pressurizer	Manual / 2	I	Closed	Closed
Vent / Test: Pressurizer Spray	Manual / 3	I	Closed	Closed
Vent / Test: Letdown	Manual / 3	I	Closed	Closed
Vent / Test: Charging	Manual / 4	I	Closed	Closed
Vent / Test: Safety Injection	Manual / 10	I	Closed	Closed
Vent / Test: Auxiliary Spray	Manual / 2	I	Closed	Closed
Sampling	Manual / 3	I	Open	Open
Shutdown Cooling Relief	1	I	Closed	Closed

a. Valves may be open or shut during normal operation or post-incident.

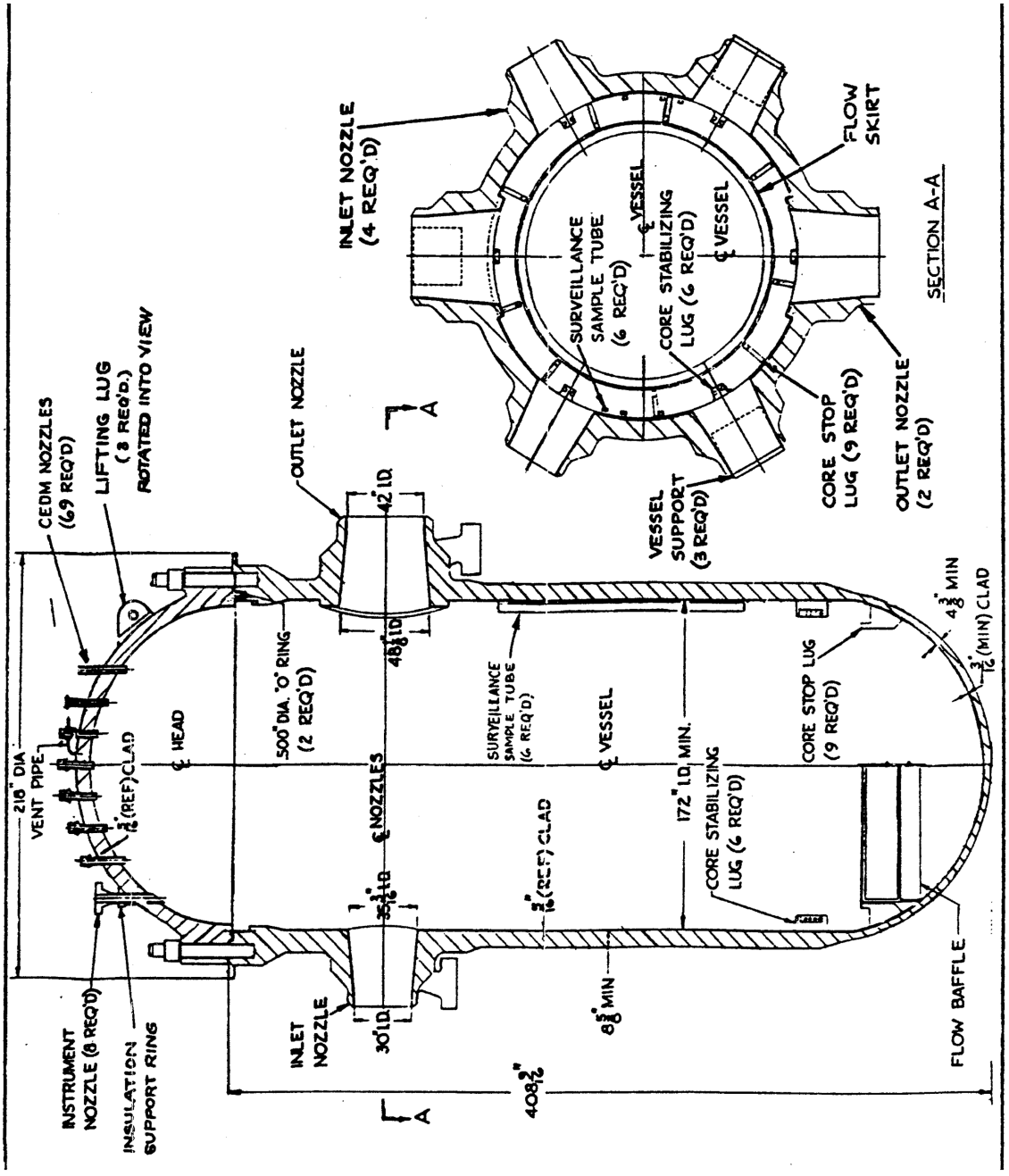


FIGURE 4.3–1 REACTOR VESSEL

FIGURE 4.3–2 STEAM GENERATOR

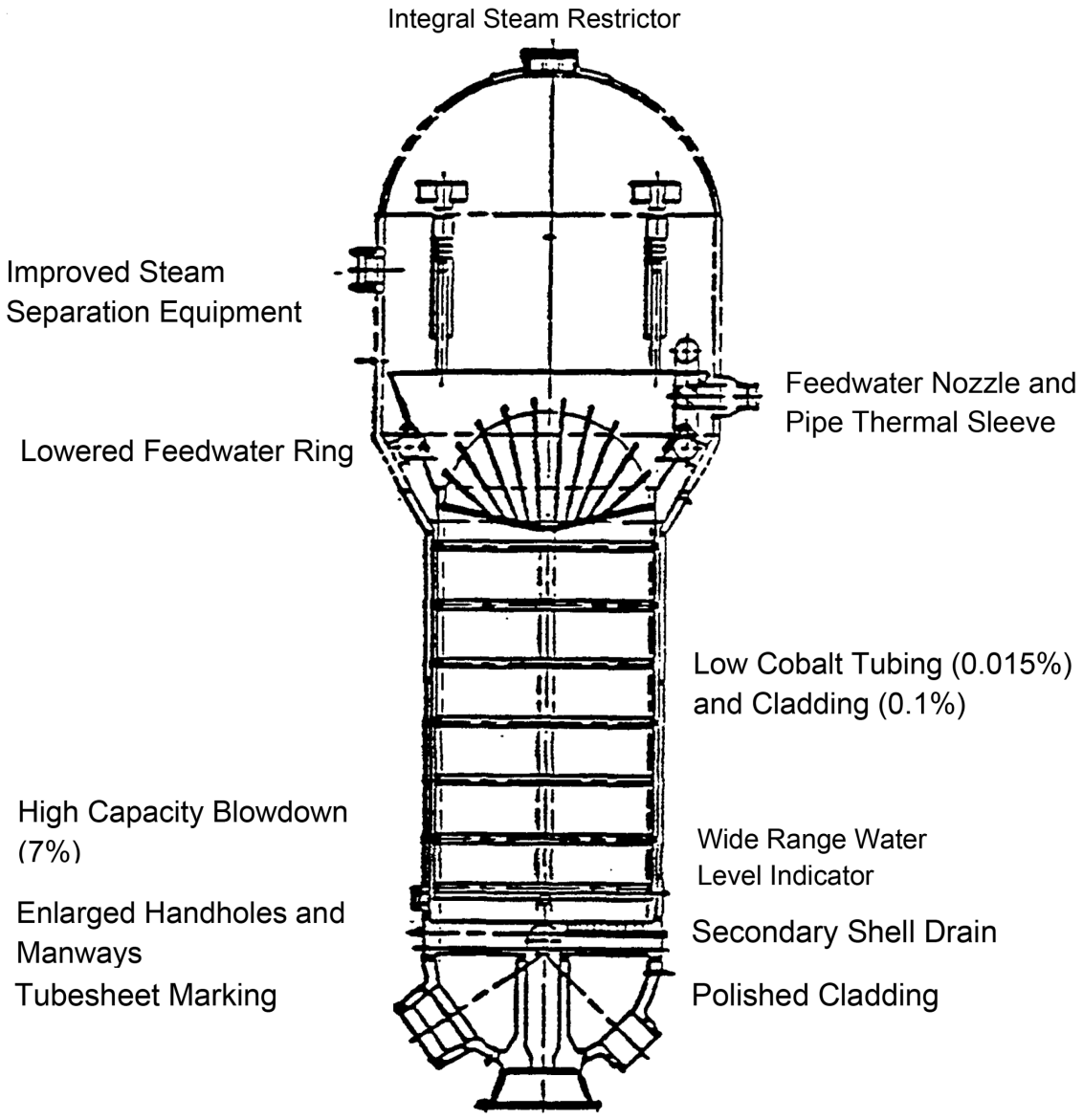


FIGURE 4.3–3 REACTOR COOLANT PUMP

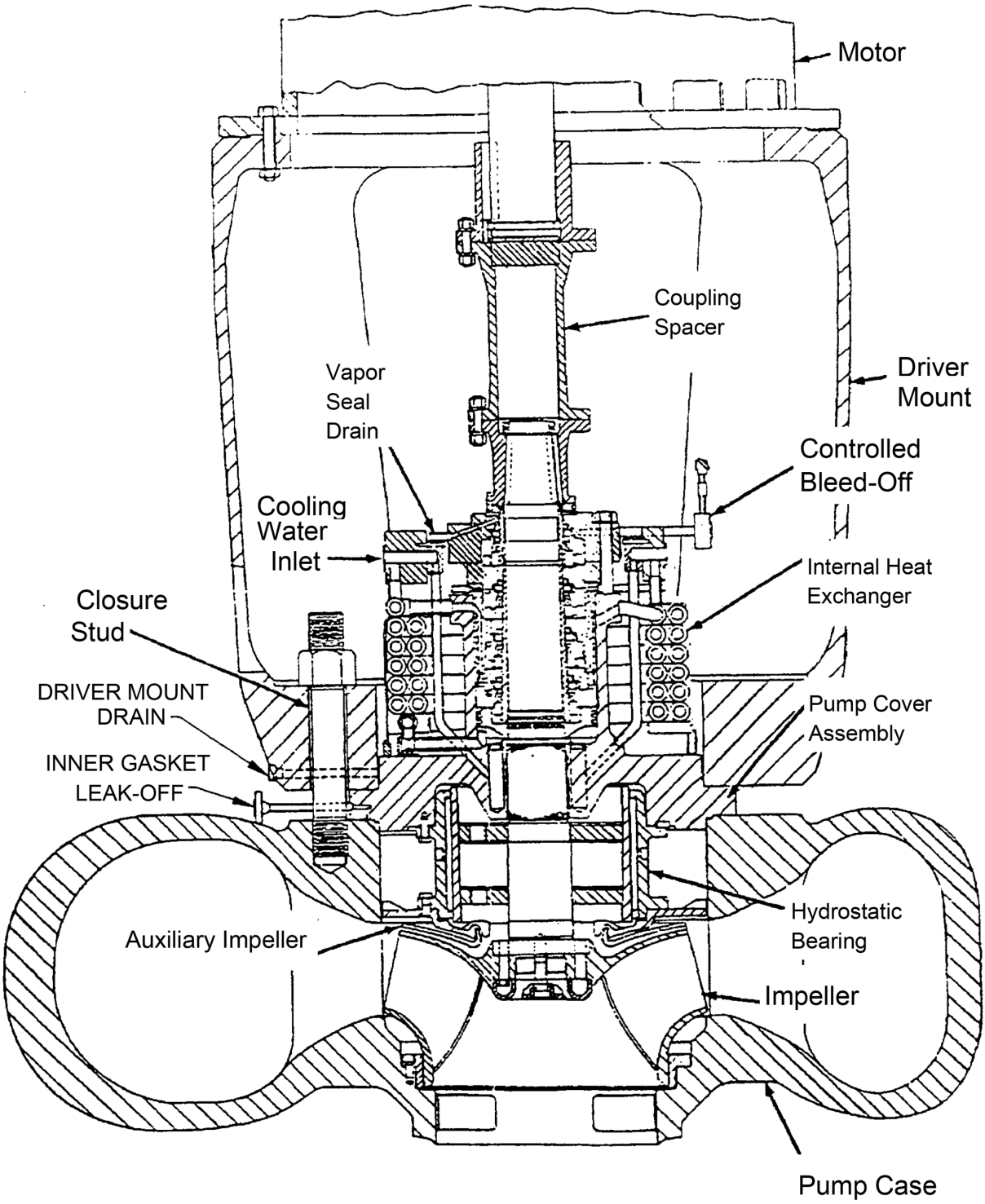
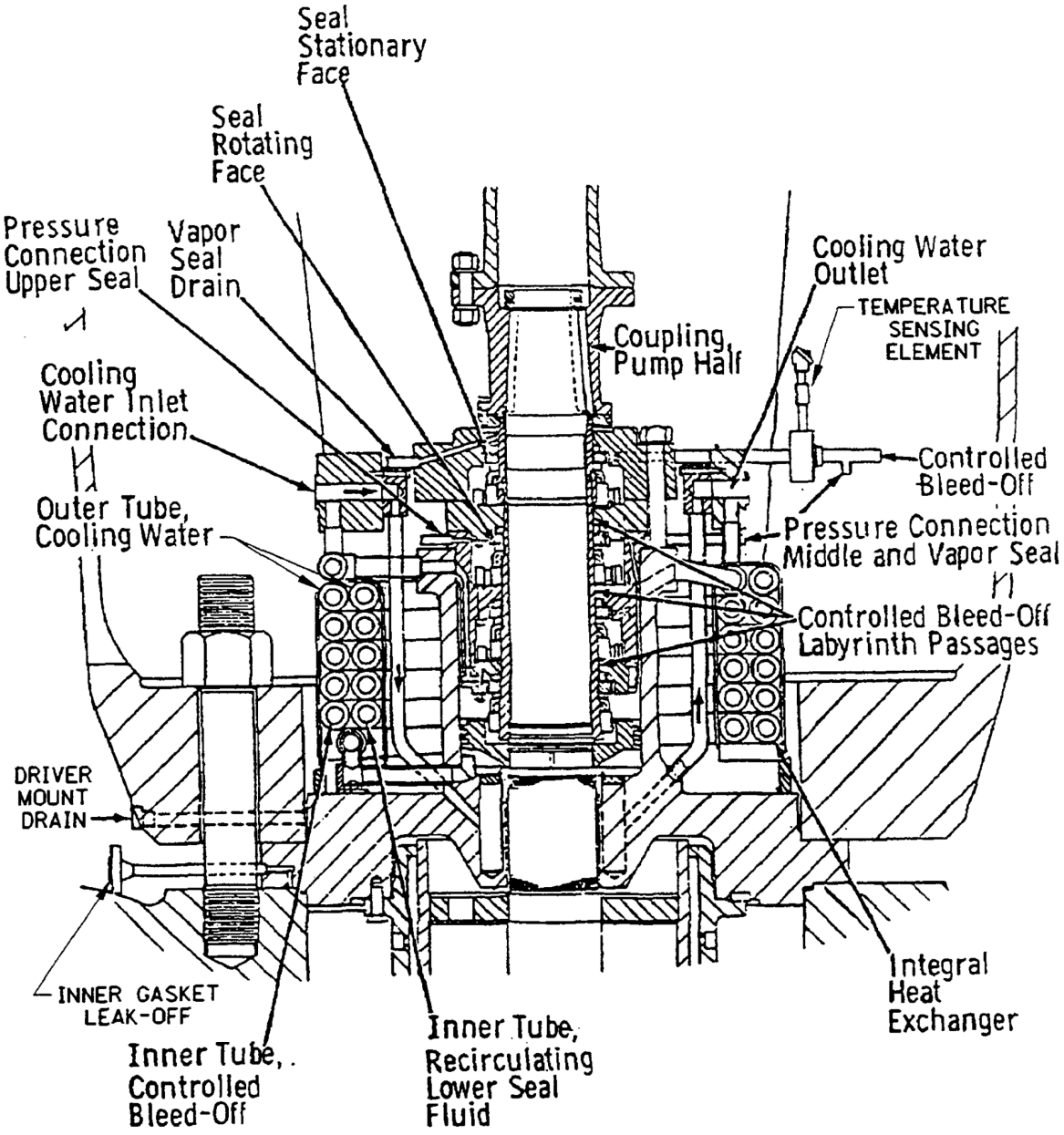


FIGURE 4.3–4 P&ID REACTOR COOLANT PUMP

The figures indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 4.3–5 REACTOR COOLANT PUMP SEAL AREA



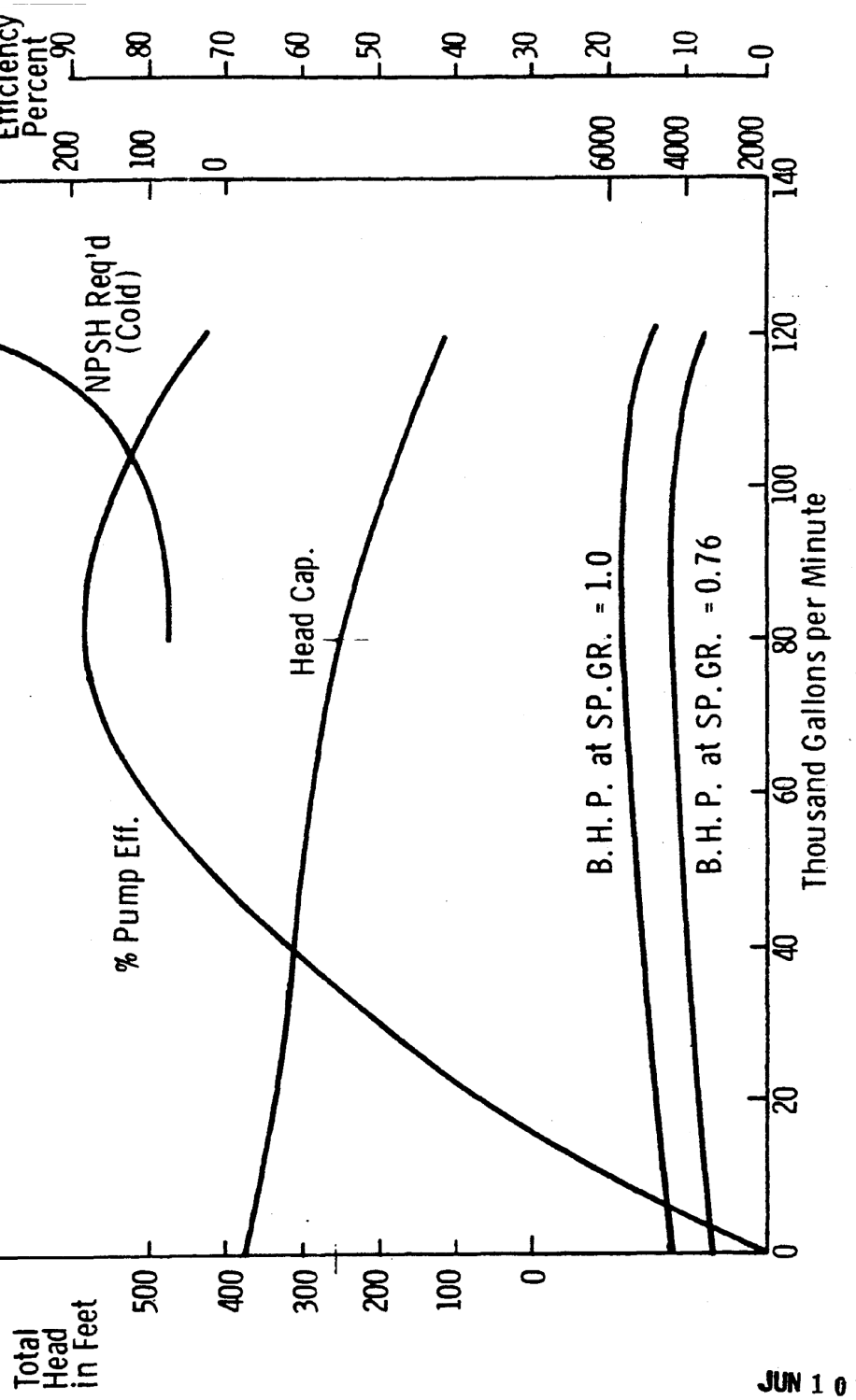


FIGURE 4.3–6 REACTOR COOLANT PUMP PREDICTED PERFORMANCE

FIGURE 4.3–7 PRESSURIZER

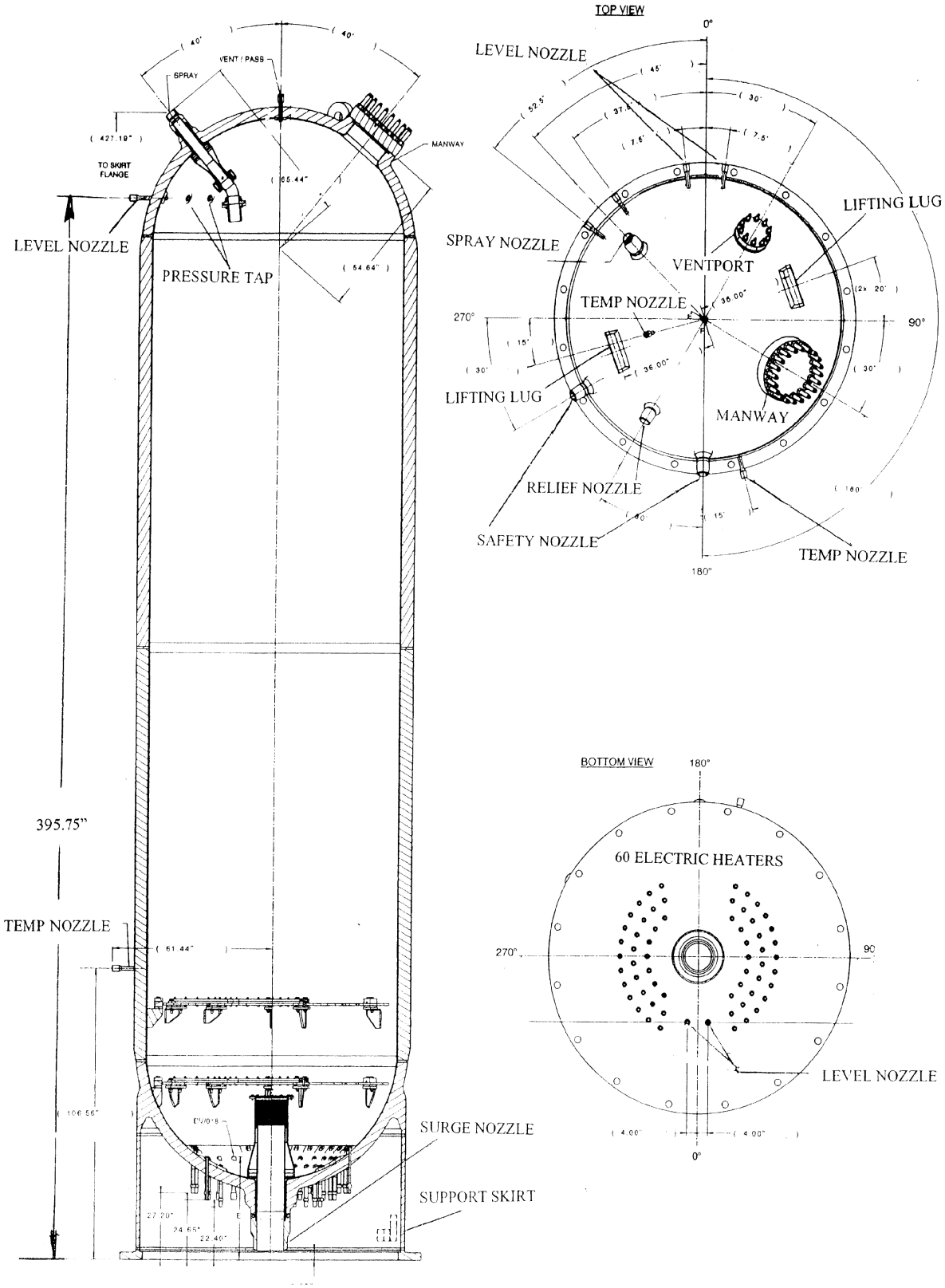
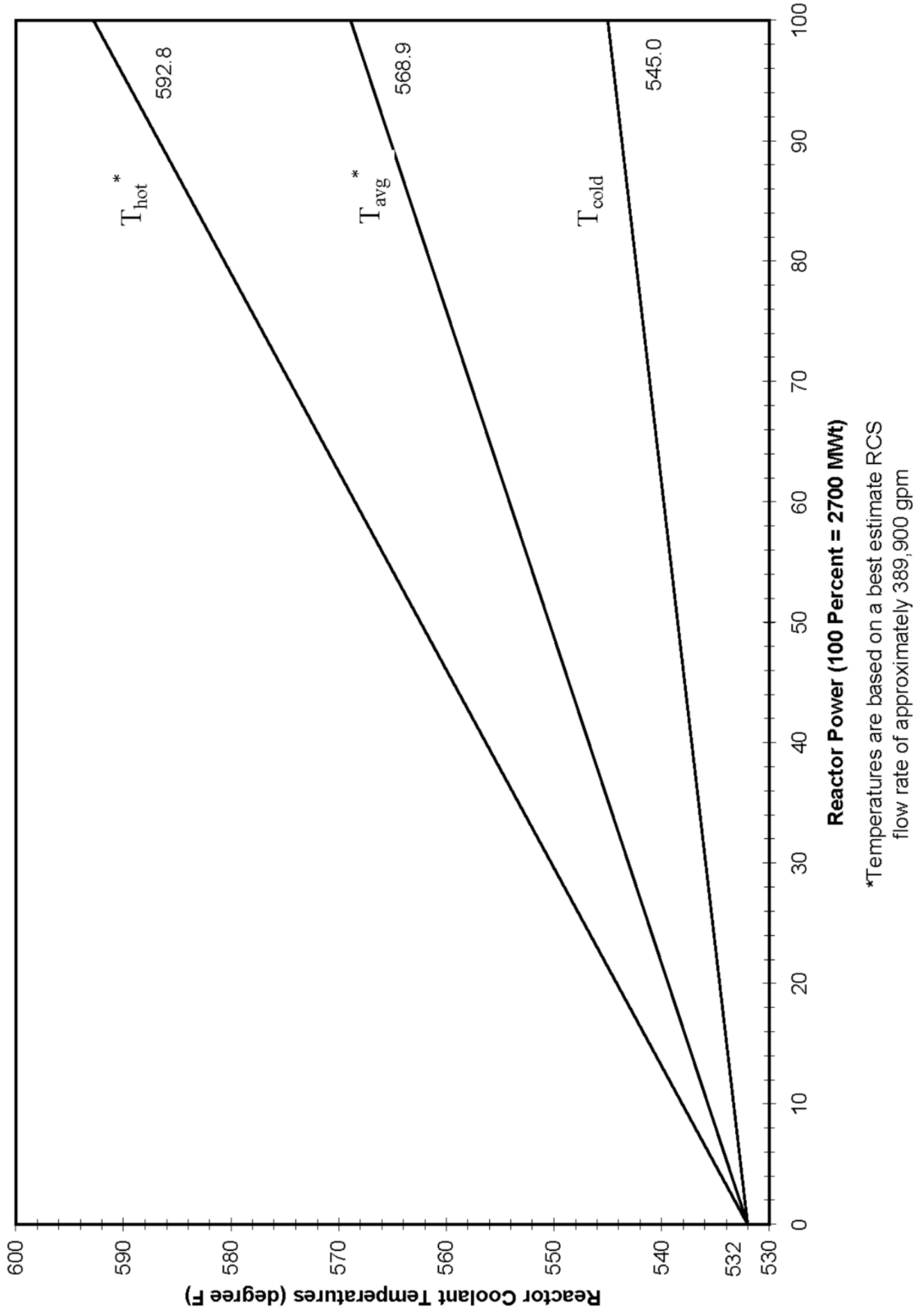


FIGURE 4.3–8 TEMPERATURE CONTROL PROGRAM



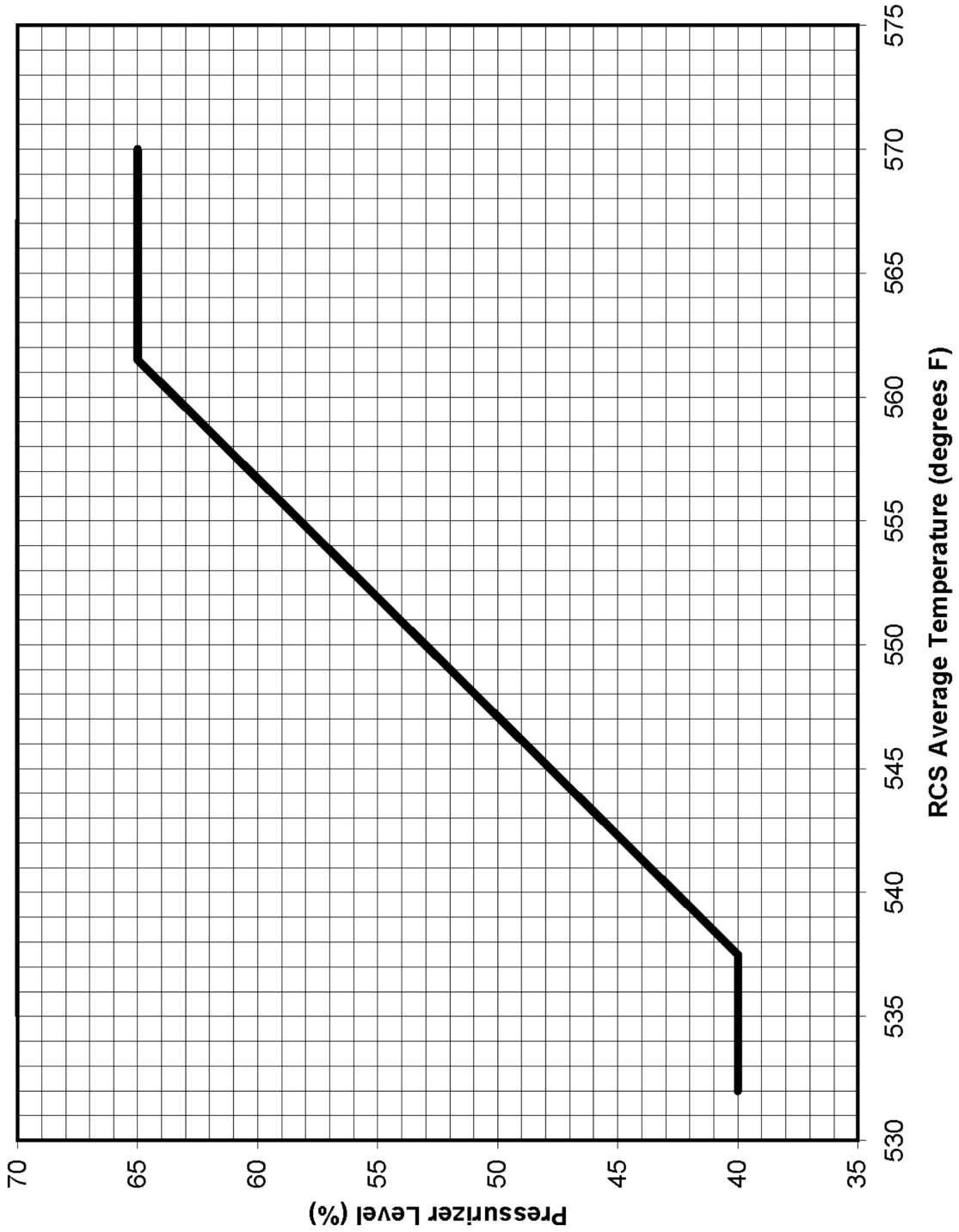
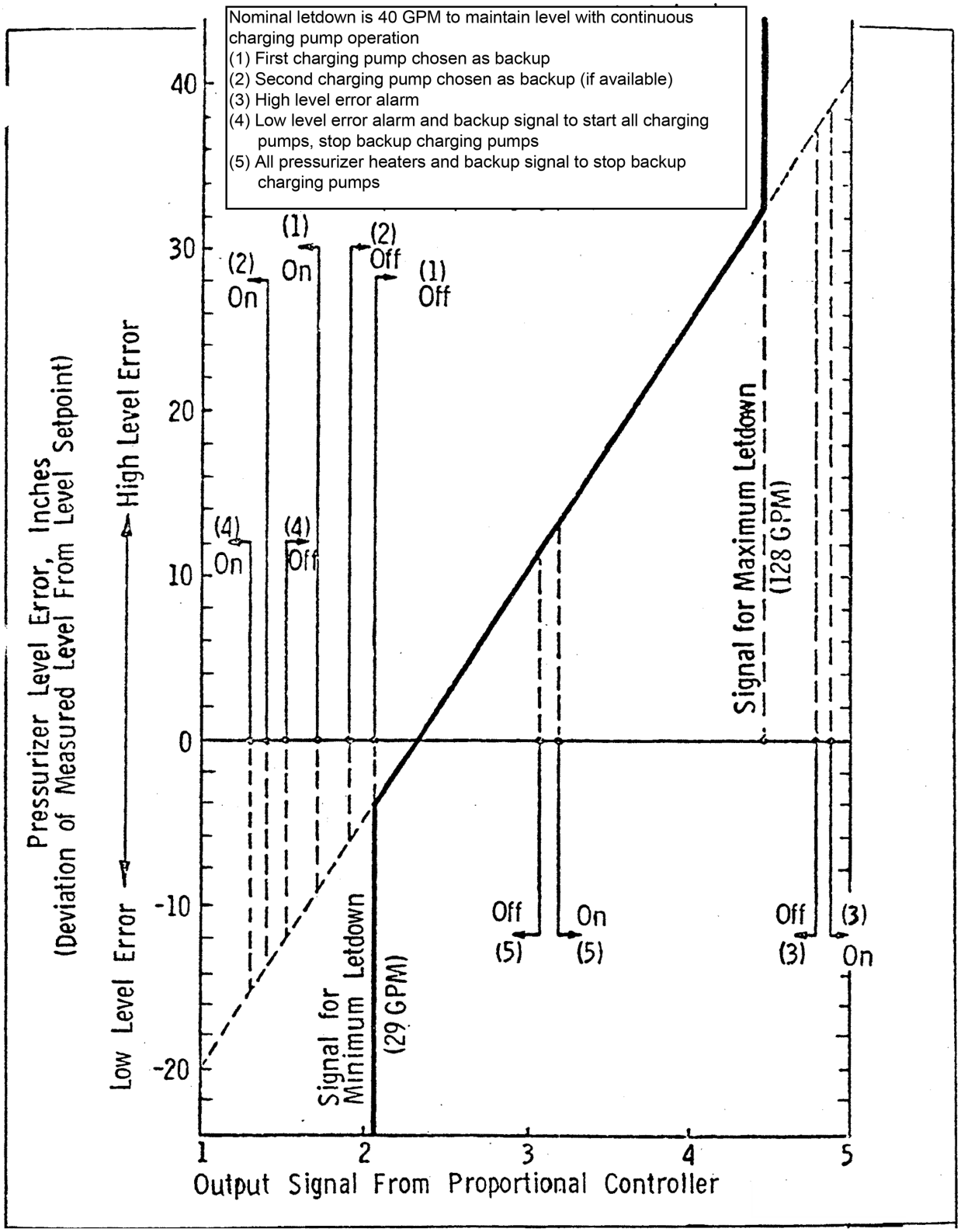


FIGURE 4.3–9 PRESSURIZER LEVEL SETPOINT PROGRAM

FIGURE 4.3–10 PRESSURIZER LEVEL CONTROL PROGRAM



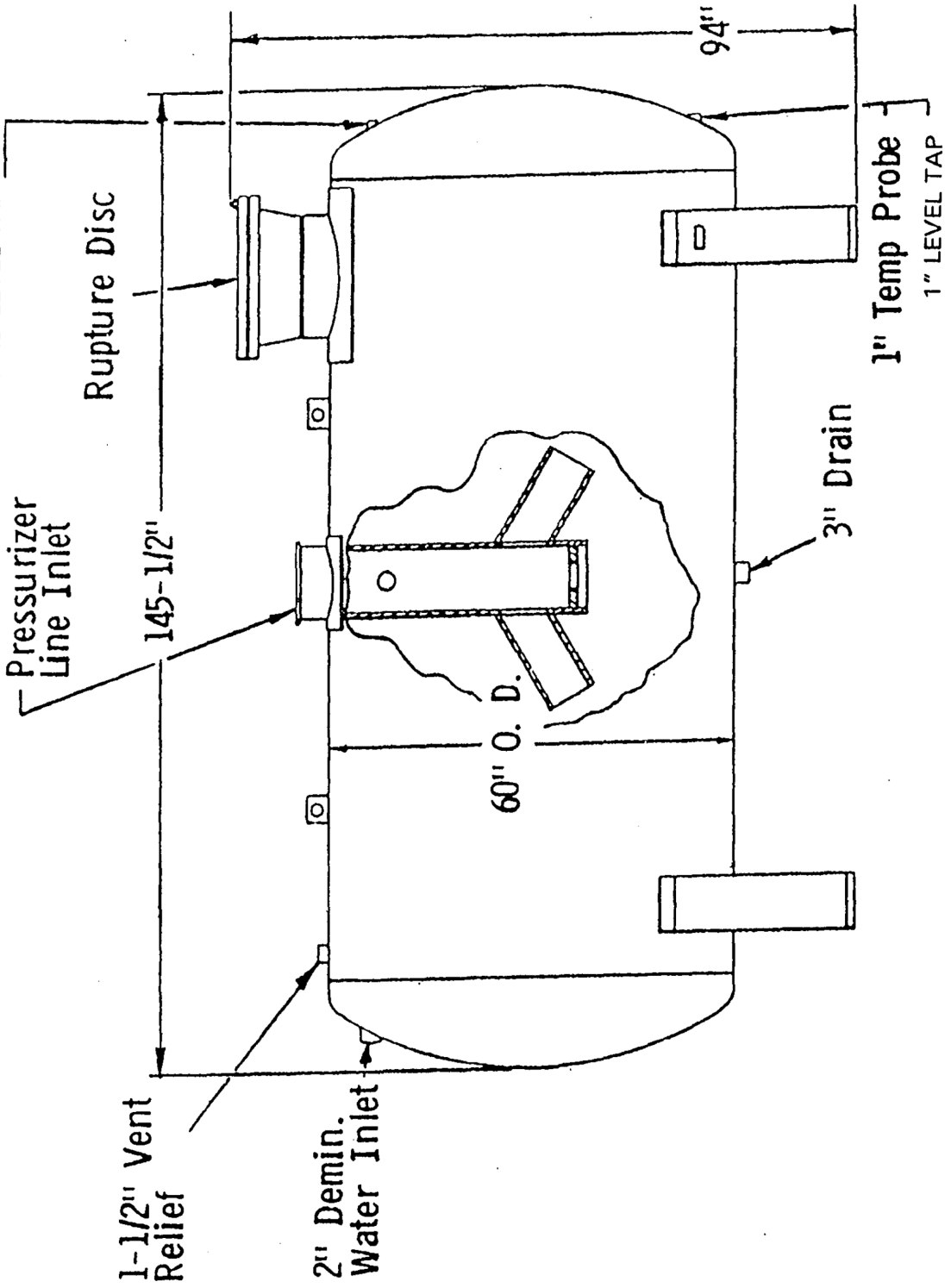
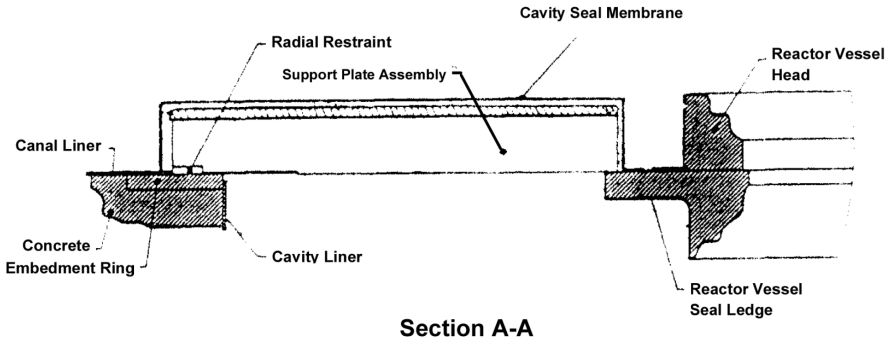
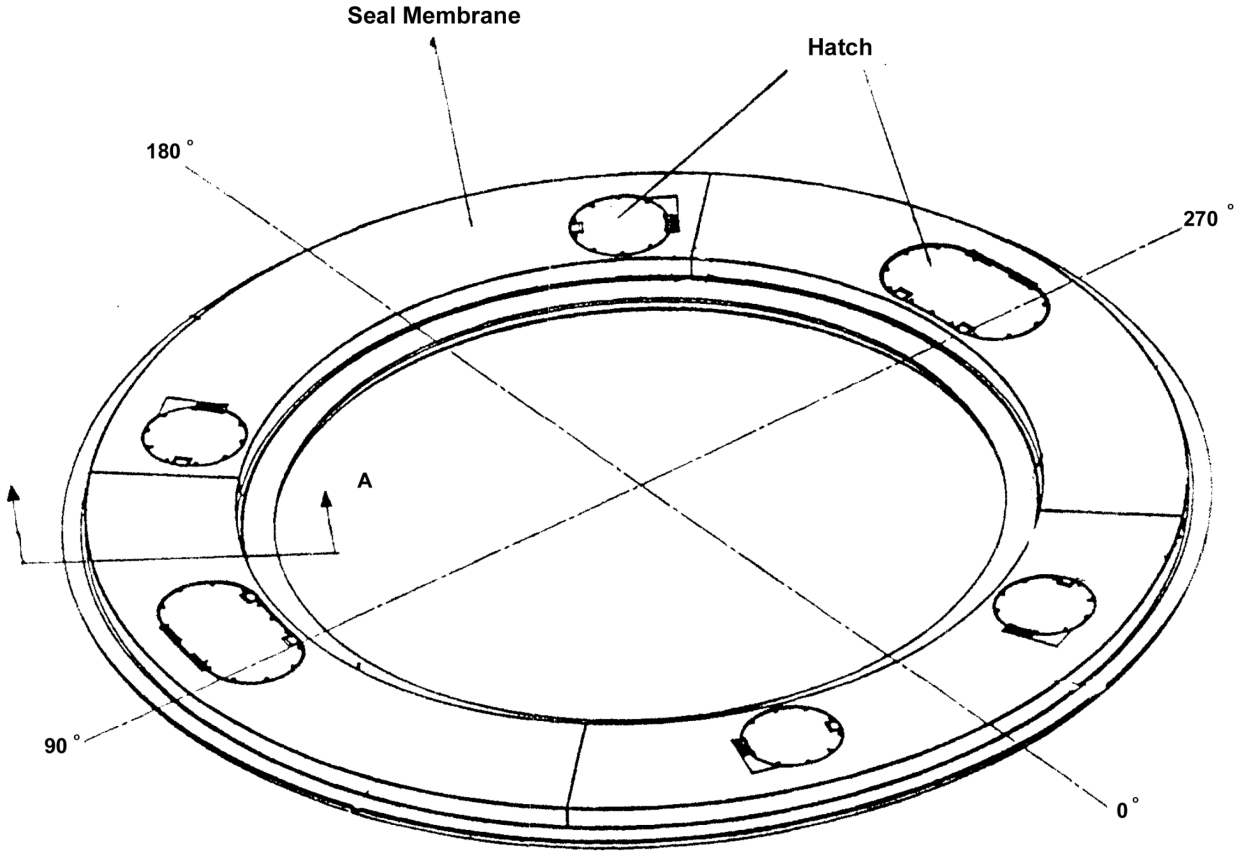


FIGURE 4.3–11 QUENCH TANK

FIGURE 4.3–12 PERMANENT REACTOR CAVITY SEAL PLATE



Section A-A



4.4 MATERIALS COMPATIBILITY

4.4.1 MATERIALS EXPOSED TO COOLANT

The materials exposed to the reactor coolant have shown satisfactory performance in operating reactor plants. A listing of materials is given in Table 4.4-1.

4.4.2 INSULATION

Piping and equipment are insulated with granular-type, reflective-type and NUKON fiberglas-type material compatible with the temperature and functions involved. All insulation material used on stainless steel has a low (< 600 ppm) soluble chloride content to minimize the possibility of chloride induced stress corrosion. Removable metal reflective-type thermal insulation is provided on weld areas of the reactor coolant system subject to inservice inspection. The replacement pressurizer is entirely covered with reflective metallic insulation, taking into consideration the containment sump clogging concerns as detailed in NRC Regulatory Guide 1.82. The chemical makeup of the insulation conforms to NRC Regulatory Guide 1.36. Nonremovable metal reflective-type thermal insulation is provided on the reactor cavity wall. The primary channel heads of the two steam generators are covered with the NUKON fiberglas-type material.

Removable blanket insulation is provided on the CEDM, instrumentation nozzle areas of the replacement reactor vessel head. Interconnected rigid panel insulation covers the lower portion of the head and the head flange for easy removal.

The thickness of insulation is such that the exterior surface temperature is not higher than approximately 50°F above the maximum containment ambient (120°F). All insulation support attachments are attached prior to final stress relief.

The possibility of leakage of reactor coolant onto the reactor vessel head or other part of the reactor coolant pressure boundary causing corrosion of the pressure boundary has been investigated by Combustion Engineering (CE).

Detailed laboratory examinations have shown that:

a. Reactor coolant (containing boric acid) alone, at temperatures greater than about 250°F, does not result in significant corrosion of low alloy steels. Therefore, under normal operating conditions, corrosion of the pressure boundary is not a concern. Below this temperature, boric acid solutions can result in significant corrosion. This corrosion is controlled with an aggressive preventative maintenance program and procedures to evaluate all unidentified reactor coolant leakage.

b. Boric acid solutions dripped through the calcium silicate insulation to be used on this plant do not initiate attack.

The results of these tests have therefore shown that reactor coolant system leakage onto surfaces of the reactor coolant pressure boundary will not affect the integrity of the pressure boundary when calcium silicate insulation, NUKON fiberglas-type insulation or metal reflective-type insulation is used.

4.4.3 COOLANT CHEMISTRY

Control and variation of the reactor coolant chemistry is a function of the chemical and volume control system. Sampling lines are provided from the reactor coolant piping to provide a means for taking periodic sample of the coolant for chemical analysis. Table 4.4-2 contains the Reactor Coolant Chemistry parameters and limits listing. The water chemistry is maintained as follows:

At temperatures below 250°F, no upper limit on dissolved O_2 is specified.

1. Hydrazine should only be added during subcritical heatup at 1.5 times the measured oxygen concentration.

2. Consistent with concentration of additives.

All wetted surfaces in the reactor coolant system are compatible with the above water chemistry.

TABLE 4.4-1 MATERIALS EXPOSED TO COOLANT

Reactor	
Replacement Closure Head Cladding	Weld Deposit Type 309 SS (layer 1) Type 308 SS (subsequent layers)
Vessel Cladding	Weld Deposited Type 308 SS *
Vessel Internals	304 SS and Ni-Cr-Fe Alloy
Fuel Cladding	Zircaloy-4
Control Element Drive Mechanisms	Ni-Cr-Fe
Piping Base Metal	SA 516 Gr 70 Carbon Steel **
Piping Cladding	Austenitic Stainless Steel Type 304 L
Steam Generator	
Bottom Head Cladding	Weld Deposited Type 308 SS *
Tube Sheet Cladding	Weld Deposited Ni-Cr-Fe Alloy
Tubes	Ni-Cr-Fe Alloy
Pumps	
Casing	Austenitic Stainless Steel, Type 316
Internals	Austenitic Stainless Steel, Type 316 and Type 304
Pressurizer Cladding	
- Cylinder Shell, Upper and Lower Head	Weld Deposited Type 308 and 309 SS *
Heater Sheath	SA 213 TP 316
Heater Sleeve	SA 182 Grade F316

* Weld Deposited Austenitic Stainless Steel in contact with coolant.

** Piping Base Material is exposed on instrument nozzles that have been subjected to a half nozzle replacement.

TABLE 4.4-2 REACTOR COOLANT CHEMISTRY

PARAMETER	REACTOR COOLANT LIMITS
Suspended Solids, ppm maximum	0.35 prior to reactor startup
pH at 25°F	Determined by the concentration of boric acid and lithium present. Consistent with the Primary Chemistry Control Program. (a)
Chloride, ppm Cl^-, maximum	0.15
Fluoride, ppm F^-, maximum	0.10
Hydrogen as H_2, cc (STP)/Kg H_2O	25-50
Dissolved O_2, ppm maximum	0.1 (b) (c) (d)
Lithium as Li^7, ppm	Consistent with the Primary Chemistry Control Program (a)
Boron, ppm	0-2620 (e)
Conductivity, μS/cm at 25°C	Relative to Lithium and Boron concentration

(a) During power operation lithium is coordinated with boron to maintain a $pH_{(t)}$ of ≥ 7.0, but ≤ 7.4, consistent with the Primary Chemistry Control Program. Lithium is added to the RCS during plant startup, but prior to reactor criticality, and is in specification per the Primary Chemistry Control Program within 72 hours after criticality. Lithium may be removed from the reactor coolant immediately before, or during, shutdown periods to aid in the cleanup of corrosion products. By evaluation, a maximum lithium concentration of 4.5 ppm is permissible with a target lithium concentration of 4.3 ppm for 100% power operations.

(b) The temperature at which the Oxygen limit applies is > 250°F.

(c) The at power operation residual Oxygen concentration control value is ≤ 0.005 ppm.

(d) During plant startup, Hydrazine may be used to control dissolved Oxygen concentration at ≤ 0.1 ppm.

(e) RCS boron concentration is maintained as necessary to ensure core reactivity or shutdown margin requirements are met. Although the RCS and related auxiliary systems containing reactor coolant are designed for a maximum concentration of 2620 ppm boron, it should be noted the design basis for the TSP baskets in the containment sump assumes the RCS, SITs, and RWST are at a maximum boron concentration of 2400 ppm.

4.5 SYSTEM DESIGN EVALUATION

4.5.1 PREVENTION OF BRITTLE FRACTURE

To protect against non-ductile failure, the requirements of 10 CFR 50 Appendix G have been implemented and the requirements of 10 CFR 50.61 have been satisfied.

10 CFR 50 Appendix G provides fracture toughness requirements which ensure sufficient margins of safety against non-ductile failure during normal operation, including anticipated operational occurrences and inservice hydrostatic tests. The requirements of this Appendix are implemented by Figure 4.5–4 which provides maximum heatup and cooldown rates for the reactor coolant system and maximum reactor coolant system pressure (as indicated by pressurizer pressure) as a function of reactor vessel inlet temperature. Additional discussions addressing the development of these limits are provided in Sections 4.5.1.2 and 4.5.1.3.

10 CFR 50.61 provides additional fracture toughness requirements to protect against non-ductile failure of the reactor vessel during pressurized thermal shock (PTS) events. Compliance with these requirements is demonstrated by ensuring that the end-of-life reference temperature (RT_{PTS}) for the reactor vessel beltline materials stays below the established limits using the prescribed methods. Additional discussion addressing 10 CFR 50.61 is provided in Section 4.5.1.4.

4.5.1.1 Initial Nil-Ductility Transition Reference Temperature.

The original design requirements for carbon and low alloy materials which form the pressure boundary of the reactor coolant system (RCS) have impact properties which meet the requirements of paragraph N-330 of the Summer 1969 ASME Boiler and Pressure Vessel Code, Section III, at 40°F or less.

To address changes in regulations and demonstrate compliance with 10 CFR 50 Appendix G and 10 CFR 50.61 requirements, the original design requirements of N-330 were supplemented and the materials initial Nil-Ductility Transition Reference Temperature (RT_{NDT}) were subsequently established.

The impact properties for the replacement reactor vessel closure head meet the requirements of Article NB 2331 of the ASME B&PV Code, Section III, 1998 Edition through 2000 Addenda. Reference Temperature RT_{NDT} of - 40°F was established in accordance with SA 508 Supplementary Requirement (S10) and NB 2300.

The replacement reactor vessel closure head and the replacement pressurizer were evaluated for protection against non-ductile failure in accordance with the methodology presented in ASME Section III, Appendix G for Class 1 components. The maximum stress intensity factors for the transients meet the fracture toughness requirements of ASME Section III Appendix G for a postulated defect of 1/10th thickness of the reactor vessel head and 1/4th thickness of the pressurizer.

In the case of the replacement steam generators, the materials were required to satisfy the requirements of ASME, Section III, 1983 Edition through the Summer 1984 Addenda, Article NB-2331, and establish a maximum RT_{NDT} at a temperature of 0°F.

4.5.1.2 Nil-Ductility Transition Reference Temperature Shift

The flux of fast neutrons at the reactor vessel wall is governed by the reactor core load design, arrangement of the reactor internals and average power levels, among other factors. The integration of this flux over time, called fluence, is monitored by dosimetry materials included in the surveillance capsules located near the vessel inner wall. In the time since original plant startup the thermal shield has been removed and the core load design has changed. In addition, small crack-arresting holes have been drilled in the core barrel, affecting the flux for one vessel plate.

Based on results from surveillance capsule dosimetry retrieved through cycle 14, the known configuration history, neutron flux modeling calculations described below, and further assuming future reactor operation with the low leakage design at 2700 Mwt and 90 percent plant capacity factor, the maximum fluence at the end of the period of extended operation of 60 years will be approximately 3.83×10^{19} n/cm^2 (E > 1Mev). These fluence estimates are fully described in Reference 4.5-2. The following summary describes the key methods of the referenced analysis.

Discussion of Fluence Calculations

The prediction of neutron fluence at various locations was based on an analysis of neutron transport for given configurations and source developed to model the reactor. To further refine the model accuracy, the results of the fluence analysis were correlated with measurements of actual fluence based on dosimetry retrieved from surveillance capsules. Estimations of future fluence are then based on a combination of the currently measured fluence and the extrapolation of calculated cycle 14 flux though the end of licensed plant life. Thus, it is assumed there will be no significant change to reactor configuration or core load design.

The three dimensional discrete ordinates transport computer code DORT was used in the surveillance capsule W-83 analysis to model neutron transport within the reactor. The code results describe the space and energy dependent neutron flux present in the reactor.

The reactor was modeled as a 1/8th segment of the core, the reactor internals, core barrel, thermal shield (through cycle 5), explicit representations of the surveillance capsules at 6° and 14°, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, the primary biological cladding and shield wall.

Note that a variation of this model, in which material composition of the surveillance capsules was redefined as water, was utilized to determine the maximum neutron exposure at the pressure vessel wall in octants of the core that do not contain surveillance capsules.

The actual activation or fission measured for the retrieved capsule dosimetry was compared to the calculated activation. The ratio of measured to calculated activation, M/C, was determined to fall well within the criteria specified in Regulatory Guide 1.190.

Shift Prediction

The shift prediction methodology for the determination of the adjusted reference temperature (ART) is done in accordance with Revision 2 to Regulatory Guide 1.99 (dated May 1988). The ART, for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

The shift in RT_{NDT} is based on fluence predictions, described above, for the time period corresponding to 54 EFPY. Table 4.6-14 provides the results of the calculation.

4.5.1.3 Operational Limits

All components in the RCS are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon a rate of 100°F/hr and for cyclic operation.

The maximum allowable RCS pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III of the ASME Code including Appendix G, Protection Against Nonductile Failure and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements.

FSAR Figures 4.5–4 and 4.5–5 provide the RCS pressure-temperature limitations during plant heatup and cooldown. Figures 4.5–4 and 4.5–5 are valid for the period up to and including fifty four years of full power integrated neutron flux, as determined using the rationale of Section 4.5.1.2, and was developed using the rules of Appendix G, “Protection Against Nonductile Failure” of the ASME Boiler and Pressure Vessel Code, Section XI, 2002 Addenda. Using these rules the belt line material of the reactor vessel is established as the controlling component section throughout plant life. This established an upper boundary on RCS pressure as a function of RCS temperature and allowable heatup and cooldown rates to ensure prevention of nonductile failure.

The limitations for normal heatup and cooldown rates and the applicable temperature ranges are summarized in Table 4.5-2.

The RCS pressure-temperature limits provided by Figures 4.5–4 and 4.5–5 have been corrected to indicated pressurizer pressure versus indicated cold leg temperature.

Indicated cold leg temperature is the best available indication of the reactor vessel downcomer temperature and will normally be monitored as RCS cold leg temperature when reactor coolant pumps are operating or natural circulation is occurring. In the instances where the shutdown cooling (SDC) system is operating without RCP’s, the SDC system return temperature will be

used. Appropriate corrections which account for dynamic flow losses, static elevation differences and instrumentation uncertainties have been applied.

Also shown is an allowable region for shutdown cooling system operation. This region is established based upon the design pressure-temperature ratings of components of the shutdown cooling system and the normal operation of this system as described in Chapter 6 (Section 6.3) and Chapter 9 (Section 9.3).

The reactor vessel beltline material consists of six plates. The NDT temperatures (T_{NDT}) of each plate was established by drop weight test (DWT). Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs absorbed energy and 35 mils lateral expansion. (Data points were based on average of three specimens.)

Similar testing was not performed on all remaining material in the RCS. However, sufficient impact testing was performed to meet appropriate design code requirements and a conservative RT_{NDT} of 50°F has been established for longitudinal direction.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the RT_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ Mev) fluxes of the reactor vessel are described in Section 4.5.1.2 of the FSAR.

Since the neutron spectra and flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured reference transition temperature shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation.

The actual shift in RT_{NDT} will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.6.3 and shown in Figure 4.6–1 of the FSAR. To compensate for any increase in the RT_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown.

In addition to the requirements provided by ASME Section XI, Appendix G and 10 CFR 50, Appendix G, the following items were also considered in the development of the RCS pressure-temperature limits provided by Figures 4.5–4 and 4.5–5.

Lowest Service Temperature - As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 50°F. ASME III, Art. NB-2332(b) requires a lowest service temperature of RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature a pressure of 20 percent of the system hydrostatic test pressure cannot be exceeded.

a. Maximum Pressure for Shutdown Cooling – This pressure is established by considering the design pressure of the shutdown cooling system, shutoff head of the Low Pressure Safety Injection (LPSI) pumps, elevation head from the pressurizer to the LPSI pumps, margin to SDC safety valve setpoints and the design temperature of the shutdown cooling system.

4.5.1.4 Pressurized Thermal Shock

In accordance with 10 CFR 50.61, reactor pressure vessel belt line materials have been reviewed to establish a reference temperature for pressurized thermal shock (RT_{PTS}). This review evaluated core loading patterns and the actual or best estimate of copper and nickel in the vessel material. In addition, the reactor vessel material composition and properties were compared to those of surveillance capsule materials from which actual tests and measurements were taken. A summary of this review is as follows:

a. Copper/Nickel Content

   Best estimate copper/nickel content values for the Reactor Vessel Beltline Plates and welds are given in Table 4.6-13.

   - Plates; Full chemistry results available for beltline plates.
   - Welds; Where chemistry results were unavailable, nickel content was conservatively estimated using data available for the type of wire used.

b. Core Configuration

   The neutron fluence values given in Table 4.6-13 have been used and have been calculated as described in Section 4.5.1.2. These end-of-life fluence values represent the most recent surveillance capsule (capsule W-83) evaluation.

Calculated RT_{PTS} values have been obtained using the above assumptions. Table 4.6-13 provides the results of the calculations. This table will be updated whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in the RT_{PTS} projected values, as required by 10 CFR 50.61(b)(1). The values that were calculated do not exceed the RT_{PTS} screening criteria of 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld materials at 54 effective full power years.

4.5.2 SEISMIC DESIGN

The nuclear steam supply system (NSSS) is designed to withstand the loads imposed by the maximum hypothetical accident and the maximum seismic disturbance without loss of functions required for reactor shutdown and emergency core cooling. The method of combining stresses produced by these simultaneous conditions is described in Section 4.2.1.

A seismic analysis of the RCS is given in Appendix 4.A.

The RCS components are considered Class 1 for seismic design. Loadings which result from earthquake conditions are categorized and, in combinations with other specified loadings, are evaluated in accordance with the rules of ASME Boiler and Pressure Vessel Code, Section III.

a. Operating Basis Earthquake (OBE) - The OBE condition is categorized as an upset condition. In evaluations using normal and upset conditions, loadings resulting from the OBE shall be considered to occur during normal operation at full power. 200 cycles of the OBE have been specified in the system design. The procedure used to account for the number of earthquake cycles during one seismic event includes consideration of the number of significant motion peaks expected to occur during the event. The number of significant motion peaks during one seismic event would be expected to be equivalent in severity to no more than 40 full load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire event. Based upon this consideration and the assumption that seismic events equivalent to 5 OBEs will occur during the life of the plant, Category I systems, components and equipment are designed for a total of 200 full load cycles.

b. Design Basis Earthquake (DBE) - Two faulted conditions, which include loadings resulting from the DBE, are defined.

    - Loadings resulting from the combined effects of the DBE and normal operation at full power.

    - Loading resulting from the combined effects of the DBE, normal operation at full power and pipe rupture conditions.

4.5.2.1 Piping

The primary stress limits applied in evaluating the emergency and faulted conditions for the RCS piping are specified as follows:

The RCS piping is designed in accordance with the requirements for Class I piping of ANSI B31.7, Code for Nuclear Power Piping. The primary stress limits of ANSI B31.7, Case 70, Design Criteria for Nuclear Power Piping Under Abnormal Conditions, are applied in evaluating the emergency and faulted conditions, except in the case of Faulted Condition (1) noted above in Section 4.5.2.b. In this case, the primary stress limits for emergency conditions (Case 70) are applied.

4.5.2.2 Vessels

The primary stress limits applied in evaluating the emergency and faulted conditions for vessels in the RCS are specified as follows:

The reactor vessel, steam generators and pressurizer were designed in accordance with the rules for Class A vessels of the ASME Boiler and Pressure Vessel Code Section III. The replacement steam generator subassemblies are designed in accordance with the rules of Class I vessels, ASME Section III. The primary stress limits of Section III, Paragraph N-417-10, Stress Limitations for Emergency Conditions, are applied in evaluating the emergency conditions and in evaluating Faulted Condition (1) noted above in Section 4.5.2.b.

With respect to Faulted Condition (2) in Section 4.5.2.b, the primary stress limits of Section III, N-417-11, Stress Limitations for Faulted Conditions, modified as follows, are applied.

a. In lieu of the value suggested in N-417.11b, the yield strength value to be used in applying the limit analysis procedure will be equal to tabulated yield strength plus one-third of the difference between the tensile strength and the tabulated yield strength, with values taken at temperature.

b. In the piping run within which a pipe break is considered to have occurred, the stress criteria for this condition need not be applied to the relevant nozzles or to the nozzle-vessel region within the limits of reinforcement given in N-454(a), except in the case of nozzles integral with component support assemblies. In the case of nozzles integral with component support assemblies, the criteria is applicable in all regions which sustain support reactions.

The replacement reactor vessel closure head is designed in accordance with ASME Section III, Class 1 vessels, 1998 Edition through 2000, Addenda. The primary stress limits of Section III, NB 3225 Appendix F, F-1331.1 (a, b and c) are applied in evaluating the faulted conditions (Table 4.2-2B). The load combination for the faulted condition (Service Level D) is defined in Section 4.5.2.b. DBE and LOCA loads are combined using the Square Root Sum of Squares (SRSS) method for evaluating the faulted condition. The emergency conditions (Service Level C) are bounded by the design conditions and therefore, the replacement reactor vessel head is evaluated using the primary stress limits of NB 3221.

The pressurizer was replaced to the design requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NB, 1998 Edition through 2000 Addenda. The primary stress limits for the emergency and faulted conditions are applied in accordance with subsection NB 3000 of the design code.

4.5.2.3 Pumps and Valves

The primary stress limits applied in evaluating the emergency and faulted conditions for the pumps and valves in the RCS are specified as follows:

The RCS pumps and valves are designed in accordance with the rules of ASME Code for Pumps and Valves for Nuclear Power - March 1970, Draft. Supplementary to these rules, the primary stress limits used for Vessel, as discussed under Item b. above, are applied in evaluating the emergency and faulted conditions for the reactor coolant pumps (RCP). In the case of valves, the

rules of the Pump and Valve Code are considered adequate to assure that, as regards primary stresses in the pressure boundary, the piping, not the valves, will be the limiting element. Therefore, no supplementary criteria which limit the primary stresses during abnormal conditions are necessary.

4.5.3 OVERPRESSURE PROTECTION

4.5.3.1 Overpressure Protection During Normal Operation

The RCS is structurally designed for operation at 2485 psig and 650°F (pressurizer 700°F). Operation of the system 2235 psig nominal and 600°F will result in material stresses 90 percent of design values. Detailed structural analyses have been performed by the component vendors and reviewed independently by Combustion Engineering (CE) for all portions of the system. Welding materials used have physical properties superior to the materials which they join. Inspection procedures and tests specified and independently reviewed by CE were carried out to assure that pressure-containing components have the maximum integrity obtainable with present code-approved inspection techniques.

The RCS is protected against overpressure by two ASME Code approved safety valves which limit system pressure to a maximum of 110 percent of design. In addition, two solenoid-operated relief valves are provided as described in Section 4.3.7.

4.5.3.2 Low Temperature Overpressurization Protection

The RCS low temperature overpressurization protection (LTOP) system, along with administrative procedures, provides protection against exceeding the ASME Section III, Appendix G (Protection Against Brittle Fracture) requirements during cold plant conditions; i.e., temperature < 275°F. The LTOP system consists of two redundant relief trains each with one power operated relief valve (PORV) with a setpoint of ≤ 400 psig and associated relief piping as described in Section 7.4.8.

4.5.4 REACTOR VESSEL THERMAL SHOCK

An analysis of the thermal stresses produced in the reactor vessel wall due to the operation of the safety injection system has been performed. The analysis has been reported in a CE report “Thermal Shock Analysis on Reactor Vessels due to Emergency Core Cooling System (ECCS) Operation,” A-68-9-1, and was submitted for the record on Docket Number. 50-309, Maine Yankee Atomic Power Station. The results show that there will be no failure of the reactor vessel due to brittle fracture.

Work has also been performed to refine the surface heat transfer coefficient. The temperature quench data obtained during the heat treatment of several heavy section steel plates was reviewed. With this background, CE planned and conducted additional quench tests to develop experimental heat transfer coefficients.

These tests were performed on a plate approximately 2 foot by 2 foot by one-half foot thick and instrumented with 11 thermocouples. The plate was heated to 550°F and lowered quickly into an agitated (turbulent) water bath at 80°F, nearly duplicating the temperature conditions which would be present in the reactor during ECCS operation. The temperature of all thermocouples was recorded throughout the cooldown of the plate.

Subsequently, the data was compared to a heat transfer computer model of the plate to obtain an effective heat transfer coefficient. A detailed report covering this work, entitled "Experimental Determination of Limiting Heat Transfer Coefficients during Quenching of Thick Steel Plates in Water" (A-68-10-2, December 13, 1968), was submitted to the AEC (now NRC) and made part of the public record. The report concludes that an effective heat transfer coefficient of 300 Btu/hr-ft^2-F provides a realistic upper limit for thick steel plates quenched in highly agitated room temperature water.

The stress near the tip of axial and circumferential vessel cracks of various depth has been determined by the finite element method. This work was reported by a CE report, "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel during Emergency Core Cooling," A-70-19-2, January 1970, and is part of the public record.

These reports substantiate the analytical conclusion that a vessel failure will not occur due to ECCS operation. An acute crack, even if formed, will not propagate.

4.5.5 LEAK DETECTION

Methods are provided to alert the operator of the presence of leakage from the RCS in a timely manner to allow detection and isolation of the leak to ensure the leakage does not exceed acceptable limits. Detection of leaks from the RCS can be accomplished by one or a combination of the below listed means.

Leak Detection Within the Containment

Leaks within the containment may be indicated by:

a. Increased pressure and temperature in the containment;

b. Monitoring the normal containment sump level;

c. An increase in airborne activity as measured by the containment air radiation monitor system. The sensitivity and response time of the particulate and gaseous detectors are dependent on many factors. Although the airborne activity detectors may very well give an early warning of an RCS leak, any correlation of these radiation monitor readings and the RCS leakage rate will be weak. These monitors are best used for trending purposes and a trigger to check other indications for leakage source;

d. Monitoring the reactor building closed cooling water (RBCCW) temperature to and from the containment air recirculation (CARS) and cooling units.

Reactor Coolant System

Leakage from the RCS is indicated by the level in the pressurizer or in the primary drain tank and/or high RCS makeup flow from the chemical and volume control system.

Relief and Safety Valves Located on the Reactor Coolant System

Piping from the relief and safety valves located on the pressurizer is provided with temperature sensors with readout in the main control room. Any temperature increase will indicate relief or safety valve leakage. In addition, an increase in pressurizer quench tank level, pressure, or temperature will also indicate leakage.

Reactor Vessel Head Closure

The space between the double O-ring seal is monitored to detect an increase in pressure, which indicates a leak past the inner O-ring. A leak indicator in containment indicates pressure. Upon high temperature a control room alarm is sounded.

Leakage Through Steam Generator Tubes or Tube Sheet

An increase in radioactivity indicated by radiation monitors for the gases from the condenser air ejectors or steam generator blowdown system monitors will indicate leakage through steam generator tubes to the secondary side. The N-16 radiation monitors will indicate primary to secondary leakage when the reactor is at power.

4.5.6 PREVENTION OF STAINLESS STEEL SENSITIZATION

Sensitization of stainless steel occurs when unstabilized Type 300 Series stainless material is held in the temperature range of 900-1400°F for sufficient time to form a continuous network of chromium carbide precipitates. Sensitization occurs after approximately 100 hours at 900°F, as compared to one hour at 1400°F. Stabilized Type 300 Series stainless material avoids continuity of chromium carbide precipitates in the grain boundaries by careful control of metal chemistry.

There are no furnace sensitized stainless steels in the reactor coolant pressure boundary (RCPB). Sensitization is precluded from the NSSS through materials selection and control to all welding and heat treating procedures.

Major portions of the RCPB in CE's nuclear plants are shown in preceding tables in this section to be formed by carbon steels and a high nickel base alloy. None of these materials is susceptible to furnace sensitization (a continuous network of iron-chromium grain boundary carbides) in the sense of unstabilized Type 300 Series stainless steels. All internal carbon steel surfaces are weld-deposit or roll-bond clad with Inconel or stainless steel, to preclude excessive corrosion.

Internal surfaces of the reactor vessel, pressurizer and steam generator primary side are overlaid with Type 308, 309 or 308L weld deposited metal. Weld metal composition is carefully controlled to overcome interface dilution and promote an austenoferritic duplex structure. Therefore, during the stress relief heat treatment (1150 ±25°F) required by the ASME Code for the pressure vessel,

a continuous network of chromium carbide precipitates is not formed in the Type 308 or 308L weld overlay even though this material has been subjected to a furnace heat treatment. The delta ferrite acts as a carbon sink and prevents continuity of carbide precipitates.

Extensive testing has confirmed that, properly formulated (a duplex structure), Type 308 weld deposited metal does not form a continuous carbide network within grain boundaries even following a typical vessel post weld heat treatment (viz, 1150°F for 20 hours). Hence, the material is immune to intergrannular corrosion.

The replacement reactor vessel head is overlaid inside with stainless steel weld material. The first layer is type 309 stainless steel and all subsequent layers including the final layer that is in contact with the reactor coolant is type 308L stainless steel.

Susceptibility to underclad / reheat cracking was minimized by controlling the welding heat input and chemical composition of the RV Head forging to maintain the Delta G function less than 0.1 in accordance with the correlations developed by the vendor (Mitsubishi Heavy Industries). In addition, in order to comply with the intent of Reg Guide 1.43, examinations were performed on three representative penetration holes in the RV Head “J” groove preparations prior to depositing the buttering to specifically check for underclad / reheat cracking on the susceptible area up to 1⁄2 inch below the fusion line between the base material and cladding.

The bimetallic weld on the CEDM and instrumentation nozzles for the replacement head joins the stainless steel threaded connector or flange to the alloy 690 penetration tube. Additionally, the materials for welding to Alloy 690 have lower susceptibility to PWSCC than the original alloy 82⁄182 welds.

The J-groove weld design for the guide tubes is modified to control the amount of overwelding to a minimum and thus minimize residual weld stresses. The weld surface finish was also improved to facilitate surface examinations.

All other Type 300 Series stainless steel used either is not subjected to a furnace sensitization heat treatment or, as is the case of cladding on the primary piping, is of Type 304L (low carbon, 0.03% max.) composition and is not susceptible to the formation of continuous chromium carbide grain boundary networks.

The replacement pressurizer is overlaid inside with stainless steel weld material. Susceptibility to underclad/reheat cracking was precluded by performance testing of the cladding weld procedure specifications in accordance with NRC Regulatory Guide 1.43.

The RCS pump casing is CF8M (Cast 316), which again is a duplex material. The casting is solution annealed after welding; hence, this component will not have a sensitized structure.

Nitrogen-enhanced stainless steel was not used in the fabrication of any RCPB component.

Because carbon steel piping is used in the RCS, no safe ends are required on the reactor vessel, or steam generator primary nozzles. Where small diameter solid stainless pipes are employed (or in the instance of welding the coolant pump casing to carbon steel), an Inconel-182 weld deposit is

built up on the nozzle prior to vessel post-weld heat treatment. Thereafter, an annealed stainless steel safe end is shop welded to the Inconel-182 buildup using alloy 82/182 weld metal.

In joining small diameter annealed solid stainless steel piping, as is used in the pressurizer surge line, charging pump lines and safety injection systems, some carbide precipitation will occur as a result of welding. However, the precipitation that occurs in the weld heat affected zone does not sensitize the material in the context of forming continuous grain boundary carbide precipitates. Samples from typical such welds pass the industry accepted standard for intergrannular corrosion susceptibility (i.e., Strauss Test – ASTM-A393). Metallographic examination of such welds reveal that only discontinuous grain boundary precipitates are present.

The following four welding processes are used to weld stainless steel in CE's NSSSs. Welding processes are performed in accordance with written procedures, as provided in the Quality Assurance Program. Nitrogen is not used as a purge gas in the welding process in lieu of argon or helium gas.

- Shielded Metal Arc (SMA)
- Gas Tungsten Arc (GTA)
- Gas Metal Arc (GMA)
- Submerged Arc (SA)

a. Shielded metal arc (SMA) is a process wherein coalescence is produced by heating with an electric arc drawn between a flux covered metal electrode and the work.

b. In the gas tungsten arc (GTA), coalescence is produced by heating with an electric arc drawn between a tungsten electrode and the work. Filler metal, if required, is added by feeding a bare metal rod or wire into the weld pool. Shielding of the weld is obtained from an inert gas mixture.

c. With gas metal arc (GMA), coalescence is effected by heating with an arc drawn between a continuous feed wire electrode and the work. Shielding of the weld is obtained from an externally supplied inert mixture.

d. Submerged arc (SA) produces coalescence by heating with an arc or arcs drawn between a bare metal (filler) electrode or electrodes and the work. The arc and weld are shielded by a blanket of granular fusible flux.

Table 4.5-1 lists the nozzles on the steam generator, reactor vessel, pressurizer and piping. The table also indicates the size of the nozzle, base material of the nozzle and, where applicable, the material of the nozzle safe end.

The procedures used in welding nozzles within CE manufacturing facilities are generally as follows: (1) For nozzles with stainless steel safe ends, the safe ends are not attached until after final stress relief, and (2) the stainless steel safe end is welded to Inconel buttering on the alloy

steel and the weld made using Inconel weld wire. With this procedure furnace sensitizing of stainless steel is precluded.

During manufacture of the core structures, various parts of the core structure are tested for sensitization using the Strauss Test (ASTM A393). Test specimens consist of: (1) mockups of various welded joints, and (2) monitoring specimens included in any heat treatment of various components. None of the specimens tested in conjunction with fabrication of reactor vessel internals for previous CE plants have failed the Strauss Test. The replacement reactor vessel head and replacement pressurizer weld materials and welding are controlled in accordance with Regulatory Guide 1.44, May 1973, “Control of the use of Sensitized Steel”, to preclude sensitization of austenitic stainless steels.

The typical weld heat input with the above processes as used by CE to joint Type 300 Series stainless steel varies from 6000 joules per inch GTA to 96,000 joules per inch SA. To avoid weld heat affected zone sensitization, CE limits the interpass temperature on multipass welds in stainless steels to 350°F maximum. The replacement pressurizer uses only low carbon stainless steel materials. The CE interpass temperature limits do not apply to the replacement pressurizer. AREVA limited the interpass temperature for low carbon stainless steel materials to 250°C (482°F), which is sufficient to prevent sensitization. The combination of normal heat input using the above welding procedures and control of interpass temperature assures minimum carbide precipitation in the weld heat affected zone. Samples from large welds have been examined in the laboratory and none has failed the Strauss Test.

In field welding operations, Bechtel uses welding procedures that limit heat input to the weld areas, and thus preclude the possibility of sensitization of austenitic stainless steels. Most of the welding employed is of the manual SMA process; a minor amount of GMA welding is also used. Neither one of these processes would be classified as an excessively high heat input welding procedure.

Further precautions employed to preclude field sensitization of austenitic stainless steels consist of:

a. Preheat and interpass temperatures are limited to 350°F maximum.

b. Controlled welding sequence is used to minimize heat input.

c. The practice of block welding is prohibited.

d. Postweld heat treatment is prohibited on equipment and/or parts that are completely or partially fabricated of austenitic stainless steel. During the fabrication of the replacement pressurizer post weld heat treatment was performed on the safety/relief nozzles and spray nozzle safe ends. Nozzle welds were post heat-treated using a Post Weld Heat Treatment (PWHT) procedure specially qualified for the heat of material used for the safe ends in accordance with Regulatory Guide 1.44 to preclude sensitization.

e. Application of heat to correct weld distortions resulting in dimensional deviations in equipment and/or parts fabricated of austenitic stainless steel is prohibited.

In preparing for, and engineering, the field welding requirements, close liaison is maintained between Bechtel and CE. Detailed welding parameters prepared by CE are submitted to CE for review and mutual concurrence and approval before they are adopted for use. Bechtel quality assurance procedures for field welding are discussed in Appendix 1B. Appendix 1B was located in the original FSAR dated August 15, 1972.

The delta ferrite content of all austenitic stainless steel weld metals used to fabricate CE's RCS components is controlled to 5-18% in the as-deposited condition. Delta ferrite content is confirmed from chemical analysis and the Schaeffler or McKay Diagrams. In addition, a calibrated ferrite measuring instrument (Seven Gauge or similar) is used. The ferrite requirement is met for each heat and/or lot of filler metal used in fabrication. The ferrite content in the weld materials and welding for the replacement reactor vessel head and replacement pressurizer are in accordance with Reg Guide 1.31, "Control of Ferrite in Stainless Steel weld metal."

Where field welding of austenitic stainless steels is required in the RCS only the inert GTA and manual SMA processes are used. The welding procedures were qualified in accordance with Section IX of the ASME Boiler & Pressure Vessel Code. Tensile test specimens were taken to exhibit the tensile strength of the welded joints and bend tests were taken to indicate the ductility of the weld joints. Filler material compositions are in accordance with the ASME SFA/AWS filler material specification and are selected on the basis of the austenitic stainless steels to be welded. In addition to these requirements the filler materials must be capable of depositing 8 to 25% ferrite. This is verified for each heat or lot of filler material by plotting the heat analysis for bare wire or the analysis of an all weld metal deposit for covered electrodes on the Schaeffler, DeLong or equivalent diagram to determine its ferrite content and the acceptability of the filler materials. The austenitic stainless steel materials including cladding are analyzed for delta ferrite in accordance with NB-2433. The FN shall be 5FN to 15FN. The FN for undiluted ER309L deposit shall be in the range of 5FN - 22FN. By control of the welding processes, the filler materials and the welding parameters (by specifying a maximum interpass temperature of 350°F), the welds should contain sufficient delta-ferrite in the austenitic matrix to avoid hot cracking in the austenitic stainless steel welds.

4.5.7 REFERENCES

4.5-1 W. G. Counsil (NU) letter to J. R. Miller (NRC), "Millstone Nuclear Power Station, Unit No. 2, Proposed Revisions to Technical Specifications, Pressure-Temperature Curves" (January 4, 1984), Attachment 2 - CE Report TR-N-MCM-008, "Evaluation of Irradiated Capsule W-97" (April, 1982).

4.5-2 S. E. Scace (DNC) to U.S. NRC, "Millstone Nuclear Power Station, Unit No.2, Submittal of Third Reactor Vessel Surveillance Capsule Report," (February 2003), Enclosure - WCAP-16012 Revision 0, "Analysis of Capsule W-83 from the Dominion Nuclear Connecticut Millstone Unit 2 Reactor Vessel Radiation Surveillance Program," (February 2003).

TABLE 4.5-1 REACTOR COOLANT SYSTEM COMPONENT NOZZLES, NOZZLE SIZES AND NOZZLE MATERIALS

Steam Generator

Component	(Number)	Size	Material
Primary Inlet	(1)	42 inch ID	Carbon steel safe ends clad with stainless steel
Primary Outlet	(2)	30inch ID	Carbon steel safe ends clad with stainless steel
Pressure Taps	(4)	1inch Schedule 160	Inconel B-166
Steam Outlet	(1)	36 inch ID	Carbon steel
Feedwater	(1)	18 inch Schedule 80	Carbon steel
Bottom Blowdown	(2)	4 inch ID	Carbon steel
Liquid Level	(12)	1 inch Schedule 80/ Schedule 160	Carbon steel/Inconel B-166
Nitrogen Addition	(1)	1 inch Schedule 80	Carbon steel
Wet Layup	(1)	2 inch Schedule 160	Carbon steel

Reactor Vessel and Head

Component	(Number)	Size	Material
Primary Outlet	(2)	42 inch ID	Carbon steel clad with stainless steel
Primary Inlet	(4)	30 inch ID	Carbon steel clad with stainless steel
CEDM/HJTC	(69)	2.718 inch ID	Ni-Cr-Fe and stainless steel
Instrumentation	(8)	4.625 inch ID	Ni-Cr-Fe and stainless steel
Vent	(1)	0.75 inch Schedule 80	Inconel SB-167

TABLE 4.5-1 REACTOR COOLANT SYSTEM COMPONENT NOZZLES, NOZZLE SIZES AND NOZZLE MATERIALS (CONTINUED)

Pipe

Component	(Number)	Size	Material
Surge Nozzle	(1)	12 inch Schedule 160	A-105 Gr II with A-351 Gr CF8M safe end
Pressure	(8)	0.75 inch Schedule 160	Inconel B-166 with A-182 Type 316 safe end
RTD	(25)	1 inch nominal	Inconel B-166
Shutdown Cooling	(1)	12 inch Schedule 140	A-105 Gr II with A-351 Gr CF8M safe end
Spray	(2)	3 inch Schedule 160	A-105 Gr II with A-182 Type 316 SS safe end
Safety Injection	(4)	12 inch Schedule 140	A-182 F1 with A-351 Gr CF8M safe end
Charging Inlet	(2)	2 inch Schedule 160	A-105 Gr II with A-182 Type 316 SS safe end
Sampling	(2)	0.75 inch Schedule 160	Inconel B-166 with A-182 Type 316 SS safe end
Drain/Letdown	(5)	2 inch Schedule 160	A-105 Gr II with A-182 Type 316 SS safe end
Pump	(8)	30 inch ID	A-516 Gr 70 clad with A-240, Type 304L SS with A-351, CF8M safe end

TABLE 4.5-2 REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITS

Cooldown		Heatup [a]	
Indicated Cold Leg Temperature	**Limit**	**Indicated Cold Leg Temperature**	**Limit**
≤ 220°F	≤ 50°F/hour	≤ 200°F	≤ 60°F/hour
> 220°F	≤ 100°F/hour	200°F < T ≤ 275°F	≤ 80°F/hour
		> 275°F	≤ 100°F/hour

a. These limitations apply to hydrostatic and leak test conditions.

FIGURE 4.5–1 REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS FOR 7 FULL POWER YEARS

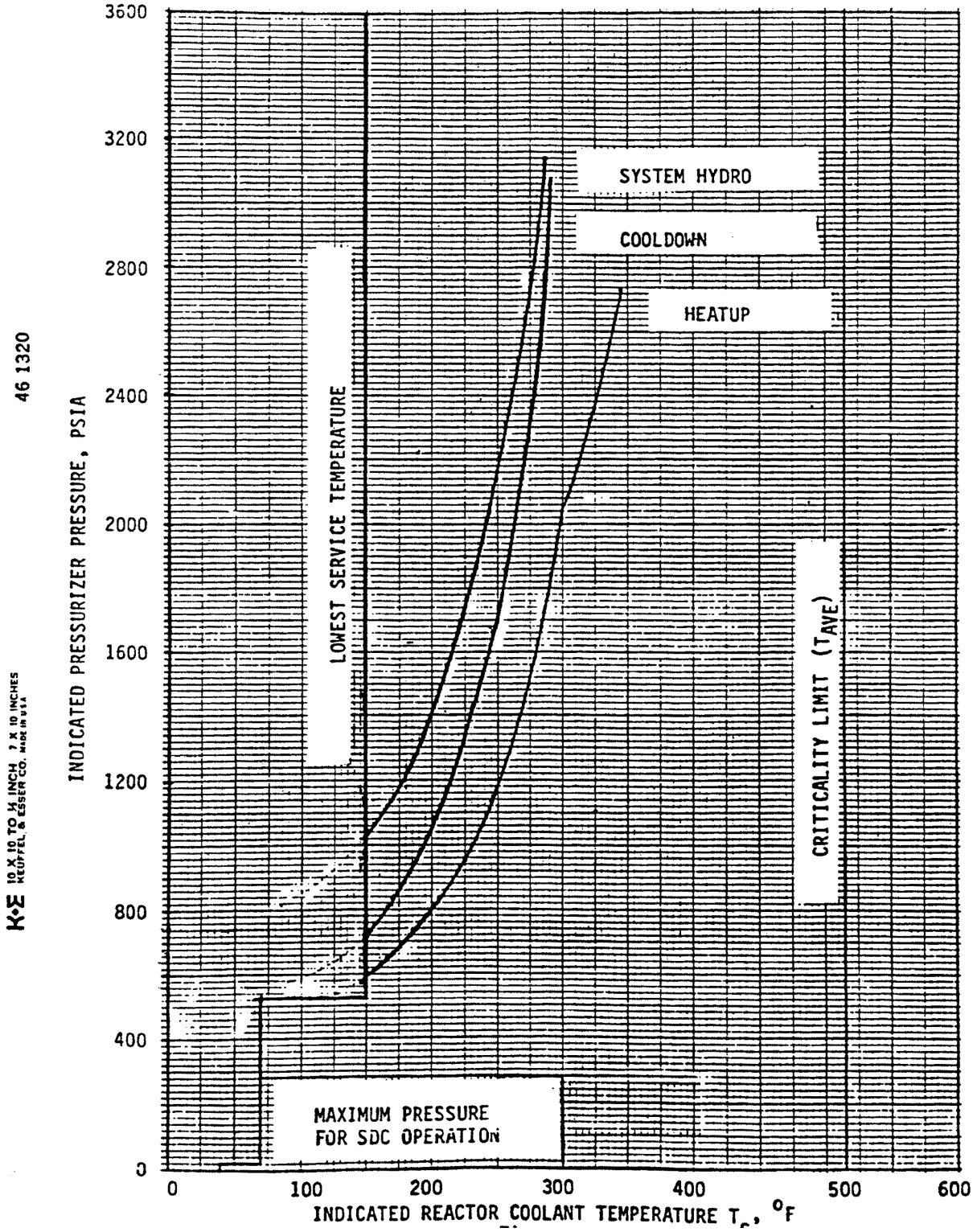


FIGURE 4.5–2 REACTOR COOLANT SYSTEM PRESSURE - TEMP LIMITATIONS DURING PLANT HEATUP/COOLDOWN AFTER 7 YEARS INTEGRATED NEUTRON FLUX

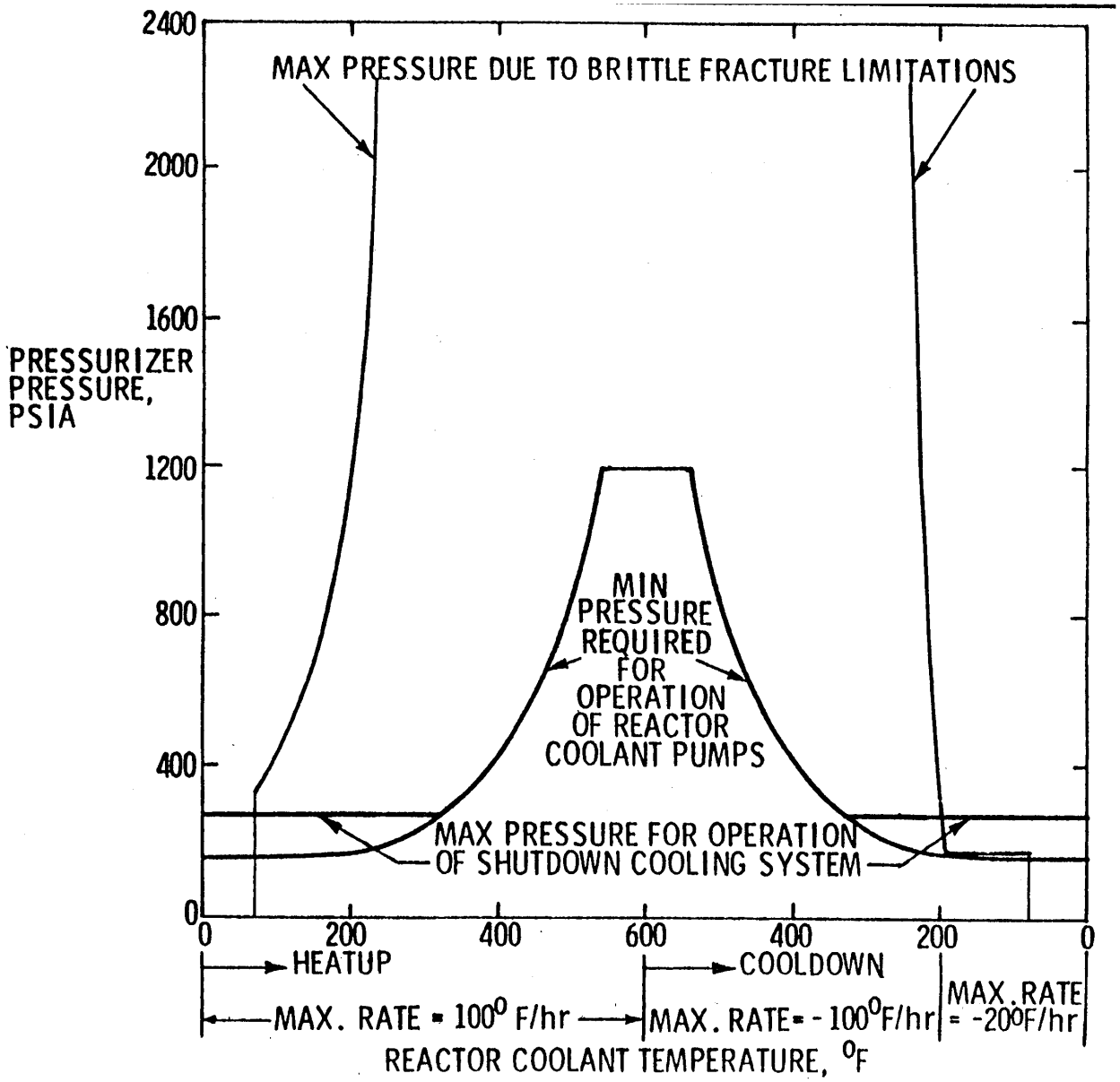


FIGURE 4.5–3 REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS FOR 0 TO 2 YEARS OF FULL POWER

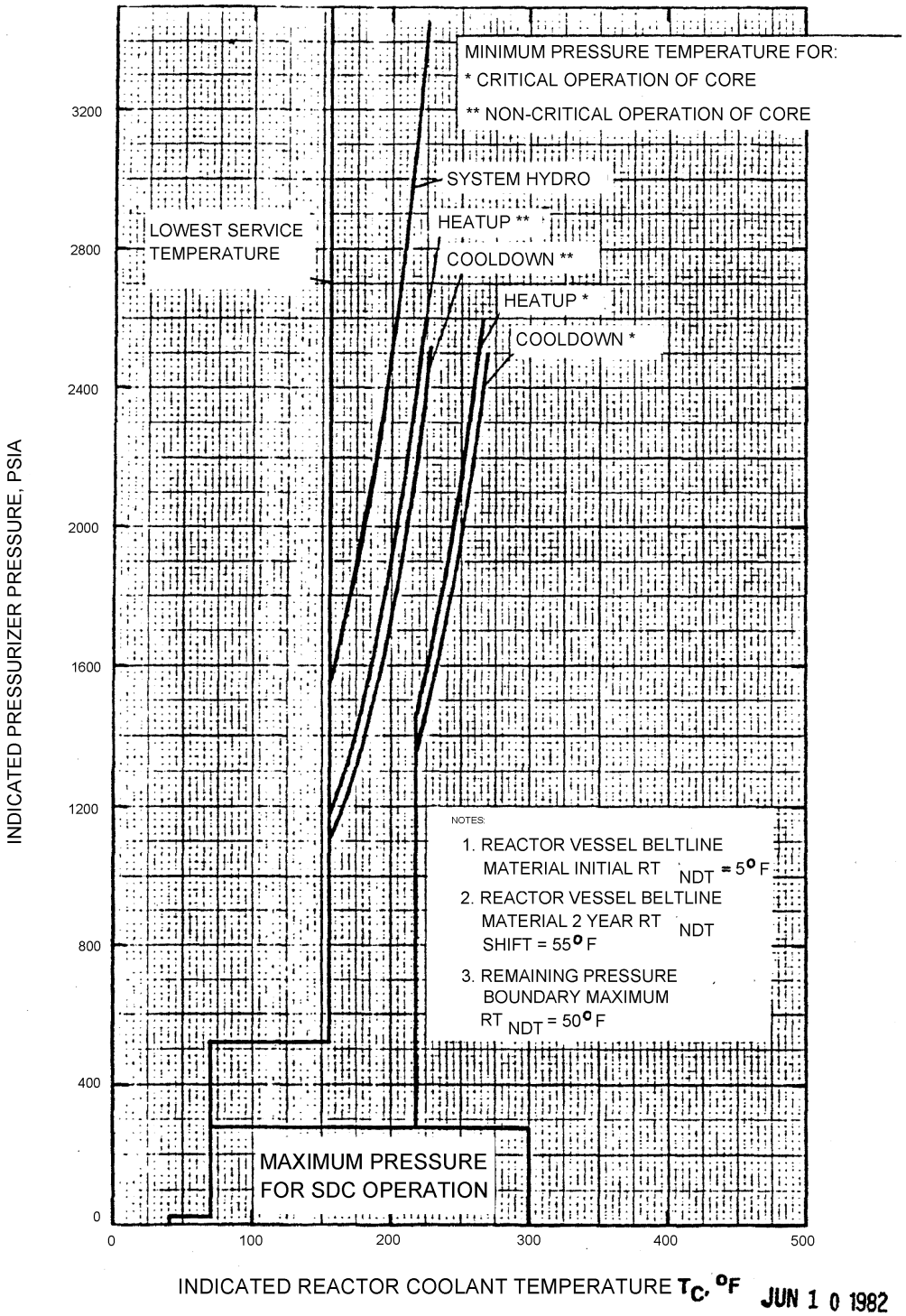


FIGURE 4.5–4 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS FOR 54 EFPY

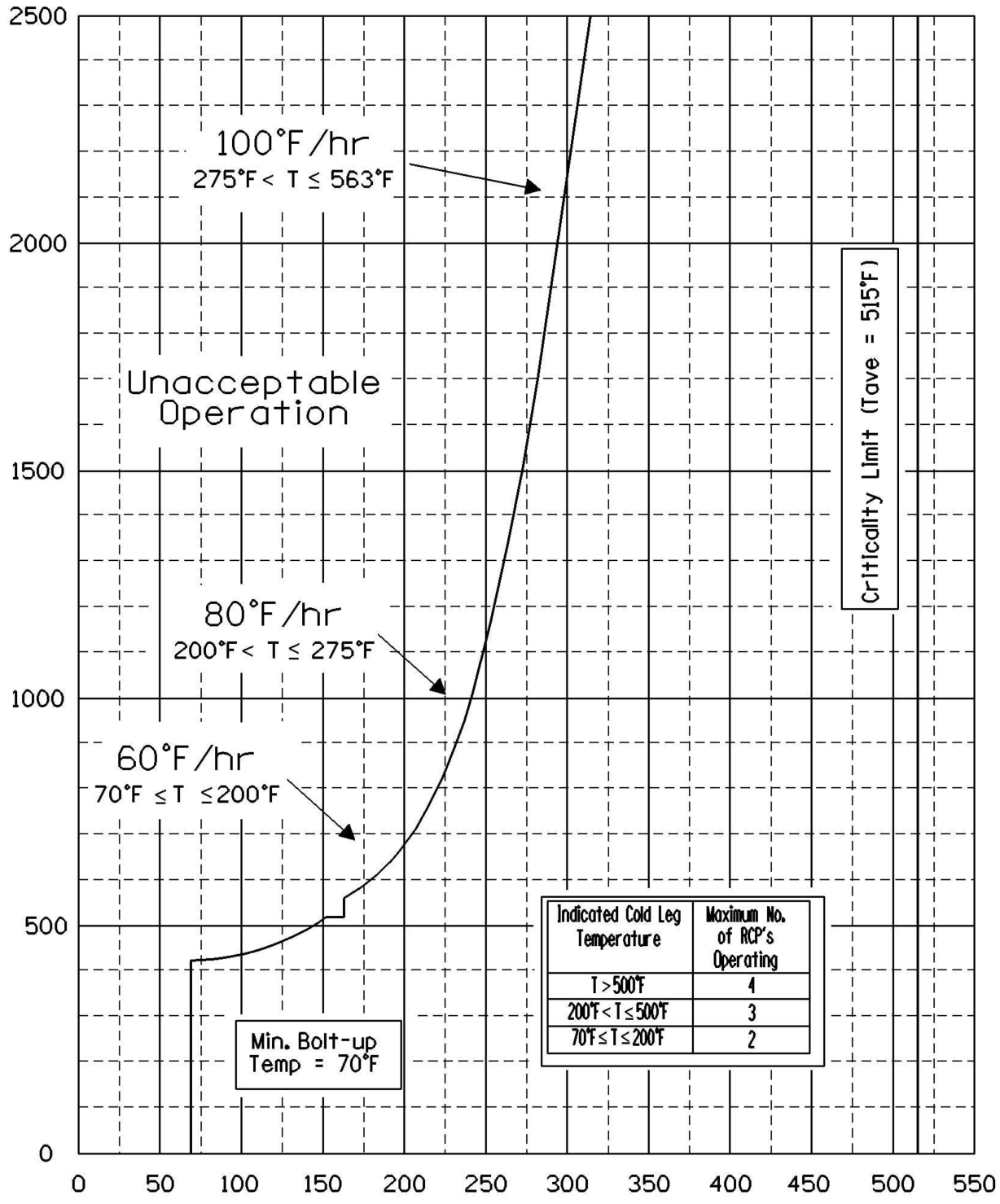
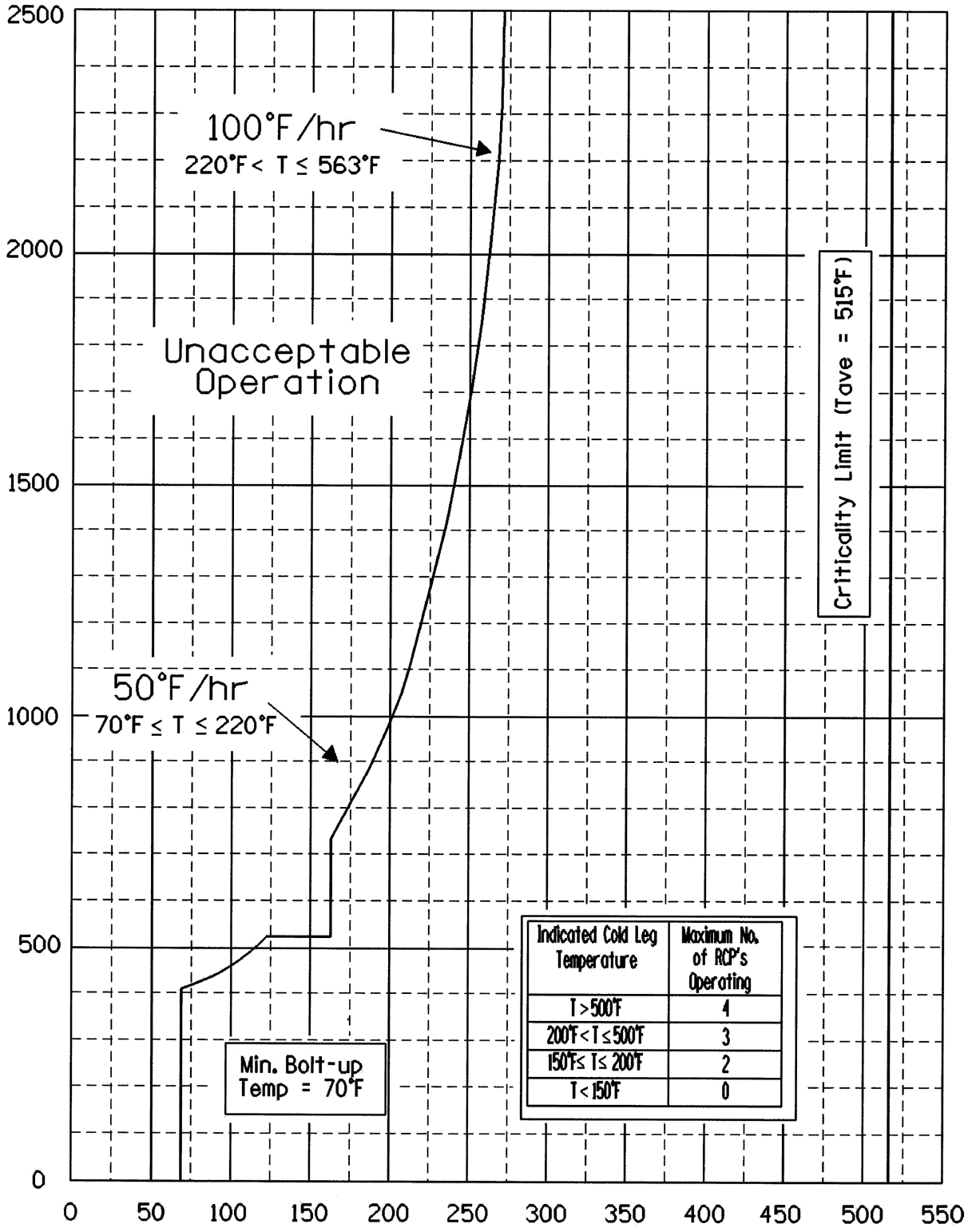


FIGURE 4.5–5 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS FOR 54 EFPY



4.6 TESTS AND INSPECTIONS

4.6.1 GENERAL

Shop inspection and tests of all major components are performed at the vendors' plants prior to shipment. An inspection at the site is performed to assure that no damage has occurred in transit. Testing of the reactor coolant system (RCS) are performed at the site upon completion of the plant construction. These tests will include hydrostatic tests of all fluid systems. A complete visual inspection of all welds and joints are performed prior to the installation of the insulation. All field welds are radiographically and dye penetrant inspected in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

In addition to the code-required examinations on the replacement reactor vessel head, base line examinations were performed prior to the component being place in service. These baseline examinations include a full volumetric examination of the CEDM nozzle base material, a minimum of 2 inches from the high point of the J-groove weld down to the threaded portion. Additionally eddy current examination was performed on the wetted surface of at least 28 peripheral CEDM nozzles in the bimetallic weld area.

A hot flow test of the reactor coolant loop up to zero power operating pressure and temperature without the core installed will be made. The system will be checked for vibration and cleanliness. Auxiliary systems will be checked for performance (see Chapter 13).

4.6.2 NIL DUCTILITY TRANSITION REFERENCE TEMPERATURE

The carbon and low alloy steels which form the reactor coolant pressure boundary are required to satisfy the requirements of 10 CFR 50, Appendix G which utilizes the nil ductility transition reference temperature (RT_{NDT}) as the basis for establishing operational limitations for the reactor coolant system.

The original materials associated with the reactor coolant pressure boundary were ordered and tested to the requirements of ASME Code, Section III, Paragraph N-330. The impact properties of these materials were required to meet the acceptance criteria noted in Paragraph N-330 at +40°F. These tests determined the nil ductility transition temperature (NDTT) for these materials. As the original design requirements were insufficient to directly establish the initial RT_{NDT} of the pressure boundary materials and comply with the requirements of 10 CFR 50, Appendix G, the RT_{NDT} was either estimated using the procedures of MTEB 5-2, performance of supplemental testing of surplus material in accordance with NB-2331 of Section III to the ASME Code or developed from generic impact testing.

In the case of the reactor vessel, the beltline materials will experience a shift (an increase) in nil ductility transition reference temperature due to neutron irradiation. This increase is calculated in accordance with regulatory positions and is described in Section 4.5.1.3.

To assure that the change in the fracture toughness properties behave in the expected fashion, a reactor vessel material surveillance program is conducted. A description of the surveillance program is provided in Section 4.6.3.

The design material toughness test requirements were as follows:

Reactor Vessel

Carbon and low alloy steel materials which form a part of the pressure boundary shall meet the requirements of the ASME Code, Section III, Paragraph N-330 at a temperature of +40°F. It shall be an objective that the materials meet this requirement at -10°F. Charpy tests shall be performed and the results used to plot a transition curve of impact values vs. temperature extending from fully brittle to fully ductile behavior. The actual NDT temperature of inlet and outlet nozzles, vessel and head flanges, and shell and lower head materials shall be determined by drop weight tests per ASTM E-208.

Replacement Reactor Vessel Closure Head

The impact properties of carbon and low alloy steel materials including weld filler metals for the replacement head shall meet the requirements of ASME Section III, Subsection NB 2331, 1998 Edition through 2000 Addenda. Charpy V-notch transition curves were established in accordance with SA 508 supplementary requirements (S3) for temperatures showing, upper shelf energy, lower shelf energy and transition. The actual T_{NDT} shall be determined by drop weight test in accordance with ASTM E208. Reference Temperature, RT_{NDT} shall be determined in accordance with NB 2331.

Steam Generator and Pressurizer

It shall be an objective that impact properties of all ferritic steel materials which form a part of the pressure boundary shall meet the requirements of the ASME code Section III, at a temperature of +10°F; alternate higher temperature levels up to 40°F may be used only if the material fails at +10°F. Such higher temperature levels, if applicable, shall be determined and documented. This objective is applicable to the pressurizer and the original steam generators, of which the original steam drums are still in use. For the steam generator replacement subassemblies, the maximum allowable RT_{NDT} as defined in paragraph NB-2311(a) of ASME Section III Code is 0°F. For the replacement pressurizer ferritic steel materials, the RT_{NDT} shall be performed at a temperature of 10 degrees F or less. The actual T_{NDT} is determined by drop weight test in accordance with NB 2331 to ASTM-208-91.

Reactor Coolant Piping

Materials used to fabricate the pipe fittings shall be specified, examined and tested to satisfy as a minimum the requirements of Chapter I-III of the American Nuclear Society (ANS) Code for Pressure Piping B31.7, Class 1.

Impact properties of carbon steel materials, including welds, shall have a minimum V-notch value of 20 ft-lb (average of three specimens) or 15 ft-lb (any individual specimen) at 40°F. It shall be a design objective that the materials meet this requirement at 10°F. Weld procedure qualifications and weld metal certifications records may serve to demonstrate impact properties of welds.

The initial toughness test data for these components are provided in Tables 4.6-1, 4.6-2, 4.6-3 and 4.6-4. Table 4.6-1 also provides RT_{NDT} values.

Toughness test data are summed as follows:

a. The maximum NDT temperature for the reactor vessel as obtained from drop weight tests is +10°F. Drop weight tests were conducted only for material used in the reactor vessel.

b. The maximum temperature corresponding to the 50 ft-lb value of the C_v fracture energy for the reactor vessel is +65°F.

   Refer to the tables presented above the Charpy V-notch data at 10°F for the steam generators, pressurizer, and reactor coolant piping.

c. The minimum upper-shelf C_v energy value for the strong direction of the material used in the reactor vessel is 103 ft-lb.

   The upper shelf-C_v energy was not determined for the material used in fabricating the steam generator, pressurizers, or reactor coolant piping. The data was not obtained for the weak direction in the material used to fabricate the reactor vessels, thus branch technical position MTEB5.2 is used to establish transverse (WR) properties.

Due to regulatory changes during the construction of the facility, it was necessary to establish the RT_{NDT} of the reactor coolant pressure boundary materials to comply with the requirements of 10 CFR 50 Appendix G. In most instances, this has been accomplished by utilizing the guidance of MTEB 5.2 to estimate the RT_{NDT} of the material based upon the available data. An initial RT_{NDT} of 50°F has been established for the reactor coolant pressure boundary materials (excluding the reactor vessel beltline) based on MTEB 5-2. However, in the case of the reactor vessel beltline materials, the RT_{NDT} was not determined through a combination of methods including utilizing the guidance of MTEB 5-2, testing surplus materials to the requirements of NB-2300, and utilizing generic data. The initial RT_{NDT} values for the beltline materials are provided in Table 4.6-13. In addition, the primary side of the steam generator has been replaced and the materials RT_{NDT} values have been determined from testing in accordance with NB-2300.

4.6.3 SURVEILLANCE PROGRAM

The surveillance program is implemented to monitor the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials in accordance with the requirements of 10 CFR 50, Appendix H. Changes in the impact properties of the material are evaluated by the comparison of pre-irradiation and post-irradiation Charpy impact test specimens. Changes in mechanical properties are evaluated by the comparison of pre-irradiation and post-irradiation data from tensile test specimens.

Three metallurgically different materials representative of the pressure vessel are investigated. These are base metal, weld metal, and weld HAZ material. In addition to the materials from the reactor vessel, materials from a standard heat of A533B, made available through the Heavy Section Steel Technology (HSST) Program, are also used. This reference material is fully processed and heat treated and is used for Charpy impact specimens so that a comparison may be made between the irradiations in various operating power reactors and in experimental reactors. A complete record of the chemical analysis, fabrication history and mechanical properties of all surveillance test materials is maintained.

The exposure locations and a summary of the specimens at each location is presented in Table 4.6-8. The pre-irradiation NDT temperature of each plate in the intermediate and lower vessel shell courses is determined from the drop weight tests and correlated with Charpy impact tests.

Base metal test specimens are fabricated from sections of the shell plate in either the intermediate or the lower shell course which exhibits the highest unirradiated NDT temperature. All base material test specimens are cut from the same shell plate. This material is heat treated to a condition of the base metal in the completed reactor vessel.

Weld metal and HAZ material are produced by welding together two plate sections from the intermediate or lower shell course of the reactor vessel. All HAZ test materials are also fabricated from the plate which exhibits the highest unirradiated NDT temperature.

The material used for weld metal and HAZ test specimens was adjacent to the test material used for ASME Code, Section III tests and was at least one plate thickness from any water-quenched edge. The procedures used for making the shell girth welds in the reactor vessel was followed in the preparation of the weld metal and HAZ test materials. The procedures for inspection of the reactor vessel welds was followed for inspection of the welds in the test materials. The welded plate was heat treated to a condition which is representative of the final heat treated condition of the completed reactor vessel.

Additional information from the baseline surveillance program includes the chemical composition of the surveillance plate and weldment (made from two separate heats of weld wire), which are given in Table 4.6-5. The baseline mechanical properties of the base metal (WR and RW), weld metal, heat affected zone (HAZ), and standard reference material (SRM), are shown in Tables 4.6-6 and 4.6-7. The Charpy V-notch impact energy and lateral expansion data as a function of test temperature are shown in Figures 4.6–5 through 4.6–14.

The test specimens are contained in six irradiation capsule assemblies. The axial position of the capsules bisected by the midplate of the core. The circumferential locations include the peak flux regions. The reactor vessel surveillance program was designed in accordance with ASTM E185 (no edition specified). The program complies with ASTM E185-73 and 10 CFR 50, Appendix H.

The location of the surveillance capsule assemblies is shown in Figure 4.6–1. A typical surveillance capsule assembly is shown in Figure 4.6–2. A typical Charpy impact compartment assembly is shown in Figure 4.6–3. A typical tensile monitor compartment assembly is shown in Figure 4.6–4.

Fission threshold detectors (U-238) were inserted into each surveillance capsule to measure the fact neutron flux. Threshold detectors of Ni, Ti, Fe, S, and Cu with known Co content have been selected for this application to monitor the fast neutron exposure. Cobalt is included to monitor the thermal neutron exposure.

The selection of threshold detectors is based on the recommendations of ASTM E-261. “Method for Measuring Neutron Flux by Radioactive Techniques.” Activation of the specimen material will also be analyzed to determine the amount of exposure.

The maximum temperature of the encapsulated specimens will be monitored by including in the surveillance capsules small pieces of low-melting-point eutectic alloys or pure metals individually sealed in quartz tubes.

The temperature monitors will provide an indication of the highest temperature to which the surveillance specimens were exposed but not the time-temperature history or the variance between the time-temperature history of different specimens. These factors, however, will affect the accuracy of the estimated vessel material NDT temperature to only a small extent.

Test specimens removed from the surveillance capsules are tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens will be compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material will then be made. This assessment of the NDT temperature shift is based on the temperature shift in the average Charpy curves, the average curves being considered representative of the material.

The periodic analysis of the surveillance samples permit the monitoring of the neutron radiation effects upon the vessel materials. If, with due allowance for uncertainties in NDT temperature determination, the measured NDT temperature shift turns out to be greater than predicted, then appropriate limitations would be imposed on permissible operating pressure-temperature combinations and transients to ensure that the existing reactor vessel stresses are low enough to preclude brittle fracture failure.

The original six surveillance capsules were inserted into their designated holders during the final reactor assembly operation. Each capsule remains in the reactor for the tentative schedule listed in Table 4.6-9. Table 4.6-9 shows the target fluence for each of the capsules.

The fluence values in Table 4.6-9 are accurate within +30 percent.

4.6.4 NONDESTRUCTIVE TESTS

Prior to and during fabrication of the reactor vessel, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code were performed on all welds, forgings and plates as follows:

All full penetration pressure retaining welds were 100 percent radiographed to the standards of paragraph N-624 of Section III of the ASME Boiler and Pressure Vessel Code. Other pressure retaining welds such as those used for the attachment of mechanism housings, vents and instrument housings or the replacement reactor vessel head and J groove welds were inspected to additional inspection criteria listed in Table 4.6-11.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75 percent of screen height, was based on interpretation of indications causing complete loss-of-back reflection. Rejection under shear wave inspection was based on indications, exceeding the amplitude of the indication from a calibration notch whose depth is three percent of the forging thickness not exceeding 2⁄8 inch with a length of 1 inch.

All forgings were also subjected to magnetic-particle examination or liquid-penetrant testing depending upon the material. Rejection was based on Section III of the ASME Code, paragraph N626.3 for magnetic-particle and paragraph N627.3 for dye penetrant testing.

Plates were ultrasonically tested using longitudinal ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME Code, with calibration so that the first back reflection is at least 50 percent of screen height, was based on defects causing complete loss of back reflection. Any defect which showed a total loss of back reflection which could not be contained within a circle whose diameter is the greater of three inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above which caused a complete loss-of-back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above.

Nondestructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as frequent calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict adherence to the nondestructive testing requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The detection of flaws in irregular geometries was facilitated because most nondestructive testing of the materials was completed while the material was in its simplest form. Nondestructive inspection during fabrication was scheduled so that full penetration welds were capable of being radiographed to the extent required by Section III of the ASME Boiler and Pressure Vessel Code.

The new replacement reactor vessel closure head is a single piece forging. The head was replaced during refueling outage 16. Prior to and during fabrication of the replacement reactor vessel head, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code 1998 Edition through 2000 Addenda were performed on the forging and welds.

Ultrasonic and magnetic particle examinations were performed on the head forging in accordance with ASME Section III, NB 2000 and NB 5000 respectively. Head cladding was ultrasonically examined for both bond and defects using a calibration block typical of the cladding and base material. Any indications that produce amplitude equal to or greater than the amplitude received from the three-eighths inch flat bottom hole, regardless of length, were unacceptable (see Table 4.6-11 for additional owners inspection requirements).

Pressure retaining welds were 100% radiographed and liquid penetrant examined in accordance with NB 5000. Canopy seal welds were liquid penetrant examined. J-groove welds were liquid penetrant examined at half thickness and again at final surface. No indications were allowed for the final PT of the J welds on the CEDM and instrument and vent nozzles and attachments of mechanism housings. Hydrostatic tests at the shop were conducted to 3107 psig. Visual examinations, magnetic particle and liquid penetrant tests were performed to reveal any surface discontinuities.

In addition to the pre-service examinations required by Section XI of the ASME Boiler and Pressure Vessel Code, augmented ultrasonic and eddy current examination were performed on the inconel nozzle bore material to meet the examination requirements of NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles”, August 3, 2001 and NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity” March 18, 2002.

Each of the vessel studs received one ultrasonic test and one magnetic-particle inspection during the manufacturing process.

The ultrasonic test was a radial longitudinal beam inspection, and a discontinuity which causes an indication with a height which exceeded 20 percent of the height of the adjusted first back reflection was cause for rejection. Any discontinuity which prevents the production of a first back reflection of 50 percent of the screen height was also cause for rejection.

The magnetic-particle inspection was performed on the finished studs. Linear axially aligned defects whose lengths are greater than one inch long and linear nonaxial defects were unacceptable.

The vessel studs are elongated by the stud tensioners during each installation of the vessel head. The amount of elongation for the desired preload was specified by the vendor to fall within a predetermined acceptance range. Maintenance procedures for vessel head installation are in full compliance with the vendor specified range.

The replacement pressurizer assembly consists of upper and lower heads and two shells attached by circumferential seam welds. The shells and the heads are forged components. The large bore

nozzles (safety, relief, spray and surge) are integral with the upper and lower head eliminating the nozzle to shell welds. The attachment weld between the lower head prolongation and the support skirt is greater than one “t” limit of the ASME Boiler and Pressure Vessel Code Section XI, Subsection IWB jurisdictional boundary. Therefore ASME Section XI, Subsection IWF jurisdiction applies to the attachment weld.

Non destructive examinations were performed on forgings and welds in accordance with ASME Section III, Boiler and Pressure Vessel Code 1998 Edition through 2000 Addenda. Ultrasonic and magnetic particle examinations were performed on the forged components in accordance with ASME Section III, NB 2000 and 5000. The heater sleeves, vent and instrument nozzles are J-groove weld attachments to the reinforced weld buildup of the replacement pressurizer. The partial penetration welds were examined by liquid penetrant tests in accordance with NB-5245. The electrical heater sheaths were completely liquid penetrant inspected in accordance with NB-5000. In addition to the code requirements, additional inspection criteria listed in Table 4.6-11 for the pressurizer were required for acceptability.

Hydrostatic test of the replacement pressurizer was conducted at 3125 psia.

Prior to and during fabrication of the components of the RCS, nondestructive testing based upon the requirements of Section III of the ASME Boiler and Pressure Vessel Code is used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C were fabricated to the standards of Subsection C, Article 21 of Section III of the ASME Code.

Table 4.6-10 summarizes the components inspection program during fabrication and construction.

In addition to the inspections conducted during fabrication and construction of the pressurizer shown in Table 4.6-10, pre-service inspections were performed on several components of the replacement pressurizer. UT inspections were performed on the circumferential welds of the pressurizer shell and heads. Magnetic particle inspection of the skirt to vessel was performed. UT and PT inspections were performed on the safety, relief, spray and surge nozzles.

Periodic tests and examinations of the RCS are conducted after startup on a regular basis.

For preoperational and in-service structure surveillance of the RCS, refer to the Technical Specifications; tests for RCS integrity after the system is closed following normal opening, modification or repair are specified in Technical Specifications.

4.6.5 ADDITIONAL TESTS

During design and fabrication of the reactor vessel, additional operations beyond the requirements of the ASME Boiler and Pressure Vessel Code, Section III were performed by the vendor. Table 4.6-11 summarizes the additional tests by components.

During the design of the reactor vessel, detailed calculations were performed to assure that the final product would have adequate design margins. A detailed fatigue analysis of the vessel for all

design conditions has been performed. In those areas which are not amenable to calculation, stress concentrations have been obtained through the use of photo-elastic models. In addition, Combustion Engineering (CE) has performed test programs for the determination and verification of analytical solutions to thermal stress problems. Also fracture mechanics and brittle fracture evaluations have been performed.

The thermal and structural analyses of CEDM and ICI head adapters of the replacement reactor vessel closure head were performed using finite element models. Structural responses from loads including branch line pipe break, faulted conditions (combination of line break and seismic), normal operating mechanical and acoustic excitation were evaluated. ASME Appendix G evaluations were performed assuming meridional and circumferential flaws at critical locations of the reactor vessel closure head. Detailed fatigue analysis using finite element models of bimetallic welds at the interface between alloy 690 and stainless steel for the ICI nozzles were performed.

All material used in the reactor vessel was carefully selected and precaution were taken by the vessel fabricator to ensure that all material specifications were adhered to. To assure compliance, the quality control staff of CE reviewed the mill test reports and the fabricator's testing procedures.

All welding methods, materials, techniques, and inspections comply with Sections III and IX of the ASME Boiler and Pressure Vessel Code. Before fabrication was begun, detailed qualified welding procedures, including methods of joint preparation, together with certified procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welders and welds by subjection to a complete and thorough testing program in order to ensure maximum quality of welded joints.

During the manufacture of the reactor vessel, in addition to the areas covered by the ASME Boiler and Pressure Vessel Code, Section III, quality control by the vendor included:

a. preparation of detailed purchase specifications which included cooling rates for test samples;

b. requiring vacuum degassing for all ferritic plates and forgings;

c. specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication;

d. use of written instructions and manufacturing procedures which enable continual review based on past and current manufacturing experiences;

e. performance of chemical analysis of welding electrodes, welding wire, and materials for automatic welding, thereby providing continuous control over welding materials;

f. the determination of NDT temperature through use of drop weight testing methods as well as Charpy impact tests;

g. and test programs on fabrication of plates up to 15 inches thick to provide information about material properties as thickness increases.

RW wave ultrasonic testing was performed on 100 percent of all plate material.

Cladding for the reactor vessel is a continuous integral surface of corrosion-resistant material, 5⁄16 inch nominal thickness. The detailed procedure used, i.e., type of weld rod, welding position, speed of welding, nondestructive testing requirements, etc., was in compliance with the ASME Boiler and Pressure Vessel Code. The cladding is ultrasonically inspected for lack of bond at intervals not to exceed 12 inches WR to the direction of welding. Unbonded areas equal to or in excess of calibration require additional scanning of the surrounding material until the boundary of the discontinuity is established. An area of unbounded clad in excess of acceptance standards is repaired.

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with Section III, paragraph N-618 of the ASME Boiler and Pressure Vessel Code.

Surveillance of the quality control program was also carried out during the manufacture of the vessel by the Windsor Quality Control Section of CE and by Northeast Nuclear Energy Company (NNECO) with an independent consultant. This work included independent review of radiographs, magnetic-particle tests, ultrasonic tests, and dye penetrant tests conducted during the manufacture of the vessel. A review of material certifications and vendor manufacturing and testing procedures was also conducted. Manufacturers' records such as heat-treat logs, personnel qualification files and deviation files were also included in this review.

The nominal cladding thickness of the replacement reactor vessel closure head following machining (i.e., grinding) is quarter inch.

The cladding was ultrasonically examined such that each pass of the scanner overlap a minimum 10 percent of the transducer dimension perpendicular to the direction of the scan. Magnetic particle and liquid penetrant tests were conducted following hydrostatic test of the reactor vessel head to find any surface discontinuities. The magnetic particle and liquid penetrant tests were conducted in accordance with ASME Section III subsection NB 5000.

The replacement head was fabricated and inspected by Mitsubishi Heavy Industries at their facility. Representatives from Dominion Nuclear, CT (DNC) were present to witness the weld inspections and hydrostatic tests at various hold points. DNC also reviewed material certifications, test results conducted during fabrication.

The replacement pressurizer has a minimum stainless steel cladding of 1/5 inch. The cladding was ultrasonically examined such that each pass of the scanner overlap a minimum 10 percent of the

transducer dimension perpendicular to the direction of the scan. Magnetic particle and liquid penetrant tests were conducted on 100% of the accessible surface to detect surface discontinuities were conducted in accordance with ASME Section III subsection NB 5000. In addition to the weld inspections, DNC personnel witnessed the hydrostatic test at various hold points at the fabricator's shop for the replacement pressurizer.

4.6.6 IN-SERVICE INSPECTION

A preoperational inspection was performed in compliance with Section XI of the ASME Boiler and Pressure Vessel Code, In-Service Inspection of Nuclear Reactor Coolant Systems, 1971, where possible. CE and Southwest Research Institute were retained to assist in the development of the in-service inspection program. Access was provided, where possible, to permit inspection of the areas listed in the code. In-service inspections are performed in accordance with Section XI.

A large portion of the insulation for the reactor vessel has been placed on the reactor cavity wall to permit inspection of the vessel outer surface. Essentially all vessel internals can be removed so that a complete visual internal inspection is possible and a volumetric internal inspection of the vessel is also possible. Pads have been welded to the outer surface of the reactor vessel to facilitate prompt location of welds for inspection purposes.

Access openings are provided on the permanent reactor cavity seal and the neutron shielding around the reactor vessel to facilitate inspection and maintenance of the neutron detector wells during refueling.

Biological shielding around the primary piping in the area of the reactor pressure vessel has been designed to afford access to the circumferential and RW welds, as well as the transition piece-to-nozzle welds.

All primary piping, as well as major components, excluding the reactor pressure vessel, have been provided with easily removable insulation in the areas of all welds and adjacent base metal requiring examination as defined by Section XI. Removable blanket insulation is provided on the CEDM, vent and instrumentation nozzles areas of the replacement reactor vessel closure head.

Plant arrangement and piping has been designed to assure that adequate access exists for either direct personnel access or for remote handling equipment to perform the examinations required by Section XI. Service connections, e.g., air, water, electricity, have been located adjacent or in close proximity to each inspection area. Consideration has been given to the type of examination and equipment requirements.

Access holes have been provided in the support skirt of each steam generator to provide a means of examining the tube sheet support stay cylinder weld.

TABLE 4.6-1 RT_{NDT} DETERMINATION FOR REACTOR VESSEL BASE METAL MILLSTONE UNIT NUMBER 2

Piece Number (Reference Drawing E-233-426-1)	**CODE Number**	**HEAT Number**	**VESSEL LOCATION**	**DROP WEIGHT NDTT**	**T50 (°F)**	**T35 (°F)**	**T_{CV} (°F)**	**RT_{NDT} (°F)**	**UPPER SHELF C_V ENERGY FOR LONGITUDINAL DIRECTION**
203-02	C-500	4P2989 5P3286-4310V1	Vessel Flange	+10F	5	50	70	10	149
204-02	C-511	C-5823-3A	Bottom Head Dome	-70	61	68	88	28	115
204-03A	C-510-1	C-5823-3B	Bottom Head Peel	-60	-20	-2	18	-42	150
204-03C	C-510-2	C-5892-3A		-10	-22	-20	0	-60	135
204-03B	C-510-3	C-5892-3B		-40	20	20	40	-20	140
204-03D	C-510-1	C-5823-3B		-60	20	-2	18	-42	150
204-03E	C-510-2	C-5892-3A		-10	-22	-20	0	-60	135
204-03F	C-510-3	C-5892-3B		-40	20	20	40	-20	140
205-02C	C-503-1	9-7395-1-1	Inlet Nozzles	-20	45	10	65	5	111
205-02B	C-503-2	9-7401-1-2		+10	30	12	50	10	120
205-02A	C-503-3	9-7454-1-3		-20	12	12	32	-28	108
205-02D	C-503-4	9-7458-1-4		-60	30	20	50	-10	123
205-03A	C-508-3	AV2999-9G-1283	Inlet Nozzle Extension	-20F	30	2	50	-10	146
205-03B	C-508-1	AV2999-9G-1281		-	30	2	50	-10	
205-03C	C-508-2	AV2999-9G-1282			30	2	50	-10	
205-03D	C-508-4	AV2999-9G-1262			30	2	50	-10	
205-06A	C-502-1	9-7356-001	Outlet Nozzles	-110	-20	-30	0	-60	132
205-06B	C-502-2	9-7375-002		-130	16	-10	36	-24	103
205-07A	C-509-1	AV3816-9G-1239	Outlet Nozzle Exten.	-20	25	15	45	-15	141
205-07B	C-509-2	AV3816-9G-1391			25	15	45	-15	
215-01C	C-504-1	C-5804-2	Upper Shell	+10	62	53	82	22	118
215-01A	C-504-2	C-5809-2		+10	62	48	82	22	131

TABLE 4.6-1 RT_{NDT} DETERMINATION FOR REACTOR VESSEL BASE METAL MILLSTONE UNIT NUMBER 2 (CONTINUED)

Piece Number (Reference Drawing E-233-426-1)	CODE Number	HEAT Number	VESSEL LOCATION	DROP WEIGHT NDTT	T50 (°F)	T35 (°F)	T_{CV} (°F)	RT_{NDT} (°F)	UPPER SHELF C_V ENERGY FOR LONGITUDINAL DIRECTION
215-01B	C-504-3	C-5809-1		-10	55	52	75	15	125
215-02A	C-505-1	C-5843-1	Intermediate Shell	-20	-	-	-	8.1	118
215-02B	C-505-2	C-5843-2		-10	-	-	-	17.5	123
215-02C	C-505-3	C-5843-3		-10	-	-	-	5	115
215-03A	C-506-1	C-5667-1 1/4T	Lower Shell	+10	-	-	-	7	112
215-03C	C-506-2	C-5667-2 1/4T	Lower Shell	-40F	-	-	-	-33.7	135
215-03B	C-506-3	A-5518-1 1/4T		-30	-	-	-	-19.2	136
A1		03W62-1-1	Replacement Closure Head	- 40	- 44	- 70	-	- 40	148
B1		03W62-1-1	Replacement Closure Head	- 40	- 18	- 22	-	- 40	139

T50 (°F) corresponds to the temperature at which 50 ft-lb energy is absorbed.

T35 (°F) corresponds to the temperature at which 35 mils lateral expansion is exhibited.

T_{CV} (°F) reflects the greater of T50 and T35 and incorporates guidance from MTEB 5-2 where necessary based upon available data.

TABLE 4.6-2 CHARPY V-NOTCH AND DROP WEIGHT TEST VALUES - PRESSURIZER MILLSTONE UNIT NUMBER 2

Description of Part	Heat Number	Drop Wt/ RT_{NDT} °F[1]	Charpy V-notch Test Temp °F	Charpy V-notch Test Orientation	Impact Energy (ft-lbs)	Lateral Expansion (mils)
Upper Head	T4987	-8 (+19)	+52	0°	147-147-147	90-87-87
				180°	160-172-157	87-94-90
Upper Shell	T5086	+1 (+28)	+61	0°	142-162-155	90-98-94
				180°	144-128-125	90-87-90
Lower Shell	T5085	-8 (+19)	+52	0°	137-140-125	94-90-87
				180°	127-111-113	87-79-79
Lower Head	T4986	-17 (+10)	+43	0°	177-169-153	94-90-87
				180°	169-169-138	87-90-87
Support Skirt	4-1575	-	+10	Base Ring	179-165-177	87-83-87
				Skirt A	134-131-134	79-78-78
				Skirt B	131-171-131	76-89-76
Manway / Vent Covers	11746	-	+59	Vent	116-119-115	79-83-83
				Manway	119-159-127	85-94-89
Lifting Lug	11764	-	+59	Lug 1	130-167-133	91-94-93
				Lug 2	136-144-174	93-94-94
Manway Stud	N9952	-	+10	-	37.6-37.6-36.9	29.3-28.9-28.1
Manway Nut	81025	-	+10	-	50.1-48-7-49.4	41.0-39.5-42.2
Ventport Stud	N9879	-	+10	-	36.9-36.1-36.1	27.0-26.6-26.2
Ventport Nut	81025	-	+10	-	50.1-48.7-49.4	41.0-39-5-42.2

Notes:

(1) Per the replacement pressurizer vendor, the crack started welds for drop weight testing specimens were deposited in accordance with the 1981 edition of ASTM Standard Test Method E 208 instead of the specified 1991 edition. A conservative shift in RT_{NDT} of 27°F above the values reported by the fabricator CMTRs has been determined to bound the resulting uncertainty in RT_{NDT} values. The adjusted values are shown in parentheses and were used in relevant ASME Code and analyses.

TABLE 4.6-3 CHARPY V-NOTCH AND DROP WEIGHT TEST VALUES - STEAM GENERATOR

PART Number	HEAT	LOCATION/ MATERIAL	Steam Generator Unit Number	RT_{NDT} (°F) (1)	CHARPY V-NOTCH IMPACT TEST RESULTS (2)		CVN TEST TEMP °F	NOTES
5056021	W70029-1	Tubesheet / SA508 Class 3	2	-70	126.9, 118.8, 104.4	146.5, 125.0, 94.0	-10 °F	
5056021	W70022-1		1	-60	130.9, 146.5, 152.1	144.6, 134.9, 130.9	0°	
5056020-2	88D108-1-1	Primary Head / SA508, Class 3	2		124.6, 147.5, 142.3 (La)	136.4, 142.3, 132.7 (La)	40	3
	88C115				195.4, 190.3, 147.5 (La)		40	
				-30	114.3, 95.8, 112 (ME)	104, 120, 119 (Me)	30	
				0	173, 181, 151 (ME)		60	
5056020-1	88D101-1-1		1		129, 111, 136 (La)	122, 125, 123 (La)	40	3
	88C110				157, 131, 123 (La)		40	
				-30	90, 98, 98 (ME)		30	
				-40	98.8, 90, 107 (ME)		20	
				0	142, 134, 164 (ME)		60	
5056024-1	727957	Stay Cylinder / SA508, Class 3	1		98.1, 95.8, 97.3	112, 98.1, 102	+40	
					120.2, 113, 116	126, 90.7, 101.7	+40	
				-35	99.5, 112, 110	95.8, 72.2, 116	+25	
				-35	129, 135.7, 127.6	123.9, 128.3, 131	+25	

TABLE 4.6-3 CHARPY V-NOTCH AND DROP WEIGHT TEST VALUES - STEAM GENERATOR (CONTINUED)

PART Number	HEAT	LOCATION/ MATERIAL	Steam Generator Unit Number	RT_{NDT} (°F) (1)	CHARPY V-NOTCH IMPACT TEST RESULTS (2)		CVN TEST TEMP °F	NOTES
5056024-2	727957		2	-27	106.9, 118, 112.8	104.7, 109.1, 113.6	+33	
				-35	104.7, 104.7, 109.1		+25	
				-27		115, 118, 120.9	+33	
					106, 110, 120	99.5, 104.7, 118	+40	
					122.4, 107.6, 112.8	126, 102, 109	+40	
5056025-1	724015	Safe End-Inlet / SA508, Class 1	1	-8°F	130.5, 127.6, 118 (TANG)		+40	4
				-8°F	99.6, 80.4, 110.6		+52	
5056025-2	724015		2	-8°F	103.2, 118, 109.1 (TANG)		+40	
				-8°F	103.3, 89.9, 97.36		+52	
5056026-1, 2, 3, 4	724015	Safe End-Outlet / SA508, Class 1	1 & 2	-8°F	115.8, 105.5, 109.1 (TANG)	99.5, 95.88, 88.5	+40	4
							+52	
5069670-2	87633-2	Manway	1	-22°F	109, 121, 124 (Top)		+37.4	
5069670-		SA533, Grade B, Class 1		-49		108, 108, 91 (Bottom)	+10	
5069670-5	87659-2	Manway SA 533, Grade B, Class 1	1 + 2	-49	107.7, 97.5, 115 (Top)		+10	

TABLE 4.6-3 CHARPY V-NOTCH AND DROP WEIGHT TEST VALUES - STEAM GENERATOR (CONTINUED)

PART Number	HEAT	LOCATION/ MATERIAL	Steam Generator Unit Number	RT_{NDT} (°F) (1)	CHARPY V-NOTCH IMPACT TEST RESULTS (2)		CVN TEST TEMP °F	NOTES
5069679-1	87659-2			-40		137, 125.5, 119 (Bottom)	+20	
5069670-3	87660-2		2	-40	143, 103, 107 (Top)		+10	
				-31		106, 165, 140 (Bottom)	+30	
5062982-1, 2	37314	Vessel Support A533, Grade B, Class 1	1 + 2	-40	181, 178, 140 (Top)		+20	
	(66961)			-31		184, 200, 187 (Bottom)	+30	
5062980-1, 2, 3, 4, 5, 6	37410		1 + 2	-22	177, 190, 151 (Top)		+40	
	(66922)			-22		178, 160, 182 (Bottom)	+40	
5056024-1	727957	Inlet Nozzle / SA508, Class 3	1		124, 113, 115	123, 129, 125	+40	
				-45	71.5, 67.8, 81.8	73, 84.8, 70	-15	
5056024-2	727957		2		134, 152, 136	178, 205, 179	+40	
				-62	58, 89, 87	88, 84, 60	-3	
5056023-1	727957	Outlet Nozzle / SA508, Class 3	1		110, 111, 118		+40	
				-36		59, 74.5, 73	+42	

TABLE 4.6-3 CHARPY V-NOTCH AND DROP WEIGHT TEST VALUES - STEAM GENERATOR (CONTINUED)

PART Number	HEAT	LOCATION/ MATERIAL	Steam Generator Unit Number	RT_{NDT} (°F) (1)	CHARPY V-NOTCH IMPACT TEST RESULTS (2)	CVN TEST TEMP °F	NOTES
5056023-2	727957		2		116, 125, 116	+40	
				-18	93, 112, 82.6	+42	
5056023-3	727957		1		118, 119, 135	+40	
				-45	75, 67, 86	+15	
5056023-4	727957		2		112, 122, 118	+40	
				-35	81.1, 85.56, 76.7	+25	

Notes:

(1) RT_{NDT} determined from drop WEIGHT and Charpy V-notch test results.

(2) Charpy-v-notch impact test results are listed in sets of three as tested. Specimen location and orientations were varied and removed for testing in accordance with ASME Section III requirements. Specimen location and orientation have been listed where possible.

(3) Specimen location identified as La and Me were orientated in longitudinal and meridional direction, respectively.

(4) Specimen location identified as TANG were oriented in tangential direction.

TABLE 4.6-4 CHARPY V-NOTCH VALUES - PIPING

Assembly Number	PIECE Number	CODE Number	DESCRIPTION	MATERIAL	CHARPY V-NOTCH VALUES - FT-LB			Temperature °F
503-01	502-20-1	C-4601-1	Pipe Segment	SA-516 Grade 70	50	53	49	+10
Hot Leg	502-20-2	C-4601-2	Pipe Segment	Grade 70	37	34	38	+10
	502-02-1	C-4605-1	Ell Segment	Grade 70	55	41	61	+10
	502-02-2	C-4605-1	Ell Segment	Grade 70	55	41	61	+10
	506-02	C-4613-1	Nozzle Forging	A 105 Grade 2	43			+10
	512-03	C-4611	Nozzle Forging	Grade 2	43	36	32	+10
503-02	502-02-3	C-4605	Ell Segment	SA 516 Grade 70	55	41	61	+10
Hot Leg	502-02-4	C-4605	Ell Segment	Grade 70	55	41	61	+10
	502-20-3	C-4601-3	Pipe Segment	Grade 70	41	47	43	+10
	502-20-4	C-4601-4	Pipe Segment	Grade 70	45	47	51	+10
	506-08	C-4612-1	Nozzle Forging	A 105 Grade 2	24	30	28	+10
504-01	502-12-1	C-4604-1	Pipe Segment	SA 516 Grade 70	58	61	63	+10
Cold Leg	502-12-2	C-4604-2	Pipe Segment	Grade 70	36	36	36	+10
Pump Discharge	502-08-1	C-4608	Ell Segment	Grade 70	34	30	39	+10
	502-08-2	C-4608	Ell Segment	Grade 70	34	30	39	+10
	507-02-1	C-4615-1	Nozzle Forging	SA 105 Grade 2	43	36	32	+10
	508-02-1	C-4610-1	Nozzle Forging	SA 182 Grade F1	98	95	104	+10
504-03	502-10-9	C-4609-2	Ell Segment	SA 516 Grade 70	64	41	66	+10

TABLE 4.6-4 CHARPY V-NOTCH VALUES - PIPING (CONTINUED)

Assembly Number	PIECE Number	CODE Number	DESCRIPTION	MATERIAL	CHARPY V-NOTCH VALUES - FT-LB			Temperature °F
Cold Leg Pump Discharge	502-10-10	C-4609-2	Ell Segment	Grade 70	64	41	66	+10
	502-18-1	C-4602-1	Pipe Segment	Grade 70	58	67	62	+10
	502-18-2	C-4602-2	Pipe Segment	Grade 70	58	67	62	+10
	508-02-2	C-4610-2	Nozzle Forging	SA 182 Grade F1	95	94	110	+10
504-04	502-12-3	C-4604-3	Pipe Segment	SA 516 Grade 70	59	66	60	+10
Cold Leg Pump Discharge	502-12-4	C-4604-4	Pipe Segment	Grade 70	59	66	60	+10
	502-08-3	C-4608	Ell Segment	Grade 70	34	30	39	+10
	502-08-4	C-4608	Ell Segment	Grade 70	34	30	39	+10
	507-02-2	C-4615-2	Nozzle Forging	SA 105 Grade 2	43	36	32	+10
	508-02-3	C-4610-3	Nozzle Forging	SA 182 Grade F1	106	120	114	+10
	507-07-2	C-4616-2R	Nozzle Forging	SA 105 Grade 2	43	36	32	+10
504-05	502-10-11	C-4609-2	Ell Segment	SA 516 Grade 70	64	41	66	+10
Cold Leg Pump Discharge	502-10-12	C-4609-2	Ell Segment	Grade 70	64	41	66	+10
	502-18-3	C-4602-3	Pipe Segment	Grade 70	58	61	57	+10
	502-18-4	C-4602-4	Pipe Segment	Grade 70	58	61	57	+10
	508-02-4	C-4610-4	Nozzle Forging	SA 182 Grade F1	109	76	92	+10
	507-07-1	C-4616-1	Nozzle Forging	SA 105 Grade 2	43	36	32	+10
503-05-1	502-04-1	C-4606-1	Ell Segment	SA 516 Grade 70	58	51	50	+10
Cold Leg Pump Suction	502-04-2	C-4606-1	Ell Segment	Grade 70	58	51	50	+10

TABLE 4.6-4 CHARPY V-NOTCH VALUES - PIPING (CONTINUED)

Assembly Number	PIECE Number	CODE Number	DESCRIPTION	MATERIAL	CHARPY V-NOTCH VALUES - FT-LB			Temperature °F
503-05-2	502-04-3	C-4606-1	Ell Segment	SA 516 Grade 70	58	51	50	+10
Cold Leg Pump Suction	502-04-4	C-4606-1	Ell Segment	Grade 70	58	51	50	+10
503-05-3	502-04-5	C-4606-2	Ell Segment	SA 516 Grade 70	44	49	53	+10
Cold Leg Pump Suction	502-04-6	C-4606-2	Ell Segment	Grade 70	44	49	53	+10
503-03-1	502-06-1	C-4607	Ell Segment	SA 516 Grade 70	47	55	52	+10
503-05-4	502-04-7	C-4606-2	Ell Segment	SA 516 Grade 70	44	49	53	+10
Cold Leg Pump Suction	502-04-8	C-4606-2	Ell Segment	Grade 70	44	49	53	+10
Cold Leg Pump Suction	502-06-2	C-4607	Ell Segment	Grade 70	47	55	52	+10
	502-16-1	C-4603-1	Pipe Segment	SA 516	63	70	64	+10
	502-16-2	C-4603-2	Pipe Segment	Grade 70	63	70	64	+10
503-03-2	502-06-3	C-4607	Ell Segment	SA 516 Grade 70	47	55	52	+10
Cold Leg Pump Suction	502-06-4	C-4607	Ell Segment	Grade 70	47	55	52	+10
	502-16-1	C-4603-1	Pipe Segment	Grade 70	63	70	64	+10
	502-16-2	C-4603-2	Pipe Segment	Grade 70	63	70	64	+10
503-03-3	502-06-5	C-4607	Ell Segment	SA 516 Grade 70	47	55	52	+10
Cold Leg Pump Suction	502-06-6	C-4607	Ell Segment	Grade 70	47	55	52	+10
	502-16-1	C-4603-1	Pipe Segment	Grade 70	63	70	64	+10
	502-16-2	C-4603-2	Pipe Segment	Grade 70	63	70	64	+10

TABLE 4.6-4 CHARPY V-NOTCH VALUES - PIPING (CONTINUED)

Assembly Number	PIECE Number	CODE Number	DESCRIPTION	MATERIAL	CHARPY V-NOTCH VALUES - FT-LB			Temperature °F
503-03-4	502-06-7	C-4607	Ell Segment	SA 516 Grade 70	47	55	52	+10
Cold Leg Pump Suction	502-06-8	C-4607	Ell Segment	Grade 70	47	55	52	+10
	502-16-1	C-4603-1	Pipe Segment	Grade 70	63	70	64	+10
	502-16-2	C-4603-2	Pipe Segment	Grade 70	63	70	64	
503-07-1	502-14-1	C-4604-5	Pipe Segment	SA 516 Grade 70	23	21	20	+10
Cold Leg Pump Suction	502-14-2	C-4604-6	Pipe Segment	Grade 70	52	57	54	+10
	502-10-1	C-4609-1	Pipe Segment	Grade 70	50	43	35	+10
	502-10-2	C-4609-1	Pipe Segment	Grade 70	50	43	35	+10
	507-10-4	C-4614-4R	Nozzle Forging	SA 105 Grade 2	38	42	43	+10
503-07-2	502-14-1	C-4604-5	Pipe Segment	SA 516 Grade 70	23	21	20	+10
Cold Leg Pump Suction	502-14-2	C-4604-6	Pipe Segment	Grade 70	52	57	54	+10
	502-10-3	C-4609-1	Ell Segment	Grade 70	50	43	35	+10
	502-10-4	C-4609-1	Ell Segment	Grade 70	50	43	35	+10
	507-10-1	C-4614-3R	Nozzle Forging	SA 105 Grade 2	38	42	43	+10
505-10-1	502-14-3	C-4604-7	Pipe Segment	SA 516 Grade 70	52	57	54	+10
Cold Leg Pump Suction	502-14-4	C-4604-8	Pipe Segment	Grade 70	58	61	63	+10
	502-10-5	C-4609-1	Ell Segment	Grade 70	50	43	35	+10
	502-10-6	C-4609-1	Ell Segment	Grade 70	50	43	35	+10
	507-10-2	C-4614-2	Nozzle Forging	SA 105 Grade 2	43	36	32	+10
505-10-2	502-14-3	C-4604-7	Pipe Segment	SA 516 Grade 70	52	57	54	+10

TABLE 4.6-4 CHARPY V-NOTCH VALUES - PIPING (CONTINUED)

Assembly Number	PIECE Number	CODE Number	DESCRIPTION	MATERIAL	CHARPY V-NOTCH VALUES - FT-LB			Temperature °F
Cold Leg Pump Suction	502-14-4	C-4604-8	Pipe Segment	Grade 70	58	61	63	+10
	502-10-7	C-4609-2	Ell Segment	Grade 70	64	41	66	+10
	502-10-8	C-4609-2	Ell Segment	Grade 70	64	41	66	+10
	507-10-3	C-4614-1	Nozzle Forging	SA 105 Grade 2	43	36	32	+10

TABLE 4.6-5 PLATE AND WELD METAL CHEMICAL ANALYSIS

Element	Weight Percent		
	Plate C-506-1	1/4 T-ID Weld C-506-2/C-506-3	1/4 T-OD Weld C-506-2/C-506-3
Si	0.12	0.17	0.15
S	0.014	0.013	0.013
P	0.006	0.015	0.016
Mn	1.26	1.13	1.13
C	0.21	0.12	0.12
Cr	0.10	0.04	0.05
Ni	0.61	0.06	0.06
Mo	0.62	0.54	0.53
V	0.004	0.006	0.007
Cb	< 0.01	< 0.01	< 0.01
B	0.0006	0.0003	0.0003
Co	0.011	0.009	0.009
Cu	0.14	0.30	0.21
Al	0.020	< 0.001	< 0.01
W	< 0.01	0.01	< 0.01
Ti	< 0.01	< 0.01	< 0.01
As	0.011	0.011	0.012
Sn	0.009	0.004	0.003
Zr	0.002	0.002	0.002
N_2	0.009	0.008	0.009

TABLE 4.6-6 BELTLINE MECHANICAL TEST PROPERTIES - REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Code Number	Orientation [a]	NDTT (°F)	30 ft-lb Fix (°F)	50 ft-lb Fix (°F)	35 Mils Lat. Exp. Fix (°F)	RT_{NDT} (°F)	Upper Shelf (ft-lb)	RT Yield (ksi)
Base Metal Plate	C-506-1	Transverse (WR)	-10	22	66	54	6	106.5	67
Base Metal Plate	C-506-1	Longitudinal (RW)	-10	36	84	65	24 [b]	124.5	67
Weld Metal	C-506-2 / C-506-3	Transverse (WR)	-60	-26	5	2	-55	129.5	76
HAZ Metal	C-506-1	Transverse (WR)	-30	-44	35	34	-25	123.0	68

a. With respect to the plates' major rolling direction for base metal; with respect to the welding direction for weld and HAZ.

b. Not valid per 10 CFR 50, Appendix G.

TABLE 4.6-7 TENSILE TEST PROPERTIES - REACTOR VESSEL SURVEILLANCE MATERIALS

Material	Plate Code Number	Orientation [a]	Test Temp (F)	Yield Strength (ksi)	Tensile Strength (ksi)	Elongation TE(%)/UE(%)	R.A. (%)
Base Metal Plate	C-506-1	Transverse (WR)		67	88	27/11	68
			250	64	81	25/10	67
			550	58	84	23/09	62
Base Metal	C-506-1	Longitudinal (RW)	71	67	86	29/12	71
			250	61	79	26/10	70
			550	56	83	26/10	69
Weld Metal	C-506-2/C-506-3	Longitudinal (RW)	71	76	86	28/11	74
			250	74	81	26/09	71
			550	67	85	25/10	65
HAZ Metal	C-506-1	Transverse (WR)	71	68	88	24/09	70
			250	60	80	25/08	72
			550	62	83	21/07	68

a. With respect to the plates' major rolling direction for base metal; with respect to the welding direction for weld and HAZ.

TABLE 4.6-8 SUMMARY OF SPECIMENS PROVIDED FOR EACH EXPOSURE LOCATION

Capsule Location on Vessel Wall	Base Metal Impact L [a]	Base Metal Impact T [b]	Base Metal Tensile	Weld Metal Impact	Weld Metal Tensile	HAZ Impact	HAZ Tensile	Reference Impact [c]	Total Impact	Specimens Tensile
83°	12	12	3	12	3	12	3	-	48	9
97°	12	12	3	12	3	12	3	-	48	9
104°	12	-	3	12	3	12	3	12	48	9
263°	12	-	3	12	3	12	3	12	48	9
277°	12	12	3	12	3	12	3	-	48	9
284°	12	12	3	12	3	12	3	-	48	9
	72	48	18	72	18	72	18	24	288	54

(a) L = Longitudinal

(b) T = Transverse

(c) Reference material correlation monitors

TABLE 4.6-9 CAPSULE REMOVAL SCHEDULE

CAPSULE	LOCATION	LEAD FACTOR [a]	REMOVAL TIME (EFPY) [b]	FLUENCE (n/cm^2, >1.0MeV) [(a)]
W-97	97°	1.40	3.0	3.24×10^{18} [c]
W-104	104°	0.95	10.0	9.49×10^{18} (c)
W-97 [d]	97°		10.0	--
W-83	83°	1.31	15.3	1.74×10^{19} (c)
W-277	277°	1.31	EOL [e]	See Note (d)
W-263	263°	1.31	Standby	--
W-284	284°	0.97	Standby	--

a. Updated in Capsule W-83 dosimetry analysis

b. Effective Full Power Years (EFPY) from plant startup

c. Plant specific evaluation

d. Flux Monitor

e. EOL is defined as the end-of-license period corresponding to the original 40 year license. Capsule W-277 is projected to receive 1.31 times the reactor vessel peak EOL surface fluence of 2.4×10^{19} n/cm^2 ($E > 1.0$ MeV). Capsule W-277 will receive the vessel peak EOL surface fluence at 23.2 EFPY. It will be removed before it receives twice the peak vessel surface fluence of 4.80×10^{19} n/cm^2 ($E > 1.0$ MeV).

TABLE 4.6-10 INSPECTION OF REACTOR COOLANT SYSTEM COMPONENTS DURING FABRICATION AND CONSTRUCTION

1. Reactor Vessel	
Forgings	
Flange	UT, MT
Studs	UT, MT
Cladding	UT, PT
Nozzles	UT, MT
Plates	UT, MT
Cladding	UT, PT
Welds	
Main Seams	RT, MT
CRD Head Nozzle Connection	UT, PT, ET
Instrumentation and Vent Nozzles	UT, PT
Main Nozzles to Shell	RT, MT
Cladding	UT, PT
Nozzle Safe Ends	RT, MT
Vessel Support Buildup	UT, MT
All Welds - After Hydrostatic Test	MT, PT [1]
2. Steam Generator	
Tube Sheet	
Forging	UT, MT
Cladding	UT, PT
Primary Head	
Forging	UT, MT
Cladding	UT, PT
Secondary Shell and Head	
Plates	UT, MT
Tubes	UT, ET
Nozzles (Forgings)	UT, MT
Studs	MT
Welds	
Shell, Longitudinal	RT, MT
Shell, Circumferential	RT, MT

1. Liquid penetrant tests of J-welds only.

TABLE 4.6-10 INSPECTION OF REACTOR COOLANT SYSTEM COMPONENTS DURING FABRICATION AND CONSTRUCTION (CONTINUED)

Cladding	UT, PT
Nozzles to Shell	RT, MT
Tube-to-Tube Sheet	PT
Instrument Connections	MT
Temporary Attachments After Removal	MT
All Welds - After Hydrostatic Test	MT
Nozzle Safe Ends	RT, (MT or PT)
Level Nozzles	MT
3. Replacement Pressurizer	
Heads,	
Forging	RT, MT
Cladding	UT, PT, MT
Shell,	
Forging	RT, MT
Cladding	UT, PT, MT
Heaters,	
Sheath	UT, PT
Nozzles (Integral to Heads)	UT, MT
Studs (Manway, Ventport)	UT, PT
Welds	
Shell to Shell and Shell to Heads Circumferential	RT, MT, PT, UT
Cladding	UT, PT, MT
Nozzle Safe Ends	RT, PT
Instrument Connections	PT
Support Skirt	RT, PT, UT
Temporary Attachments After Removal	MT
Heads to Shell and Shell to Shell Welds	PT
Heater Assembly	RT, PT
4. Pumps	
Castings	RT, PT
Forgings	UT, PT
Welds	
Circumferential	RT, PT
Instrument Connections	PT
All Welds After Hydrostatic Test	PT

TABLE 4.6-10 INSPECTION OF REACTOR COOLANT SYSTEM COMPONENTS DURING FABRICATION AND CONSTRUCTION (CONTINUED)

5. Piping	
Fittings	RT, PT
Pipe	RT, PT
Nozzles	RT, PT
Welds	UT, PT
Circumferential	RT, PT, MT
Nozzles to Run Pipe	RT, PT
Instrument Connections	PT
Cladding	UT, PT

Legend:

RT -Radiographic
PT - Dye Penetrant
ET - Eddy Current
UT - Ultrasonic
MT - Magnetic Particle

TABLE 4.6-11 REACTOR COOLANT SYSTEM INSPECTION C-E REQUIREMENTS

Reactor Vessel

<test>	C-E Requirements	Code Requirements
Ultrasonic Testing (UT)	1. UT of Weld Clad for bond.	1. None

Replacement Reactor Vessel Head [a]

<test>	C-E Requirements	Code Requirements
Ultrasonic Testing (UT))	1. Indications that produce an amplitude greater than the amplitude received from the one-eighth inch flat bottom hole and less than the amplitude from the three-eighth inch flat bottom hole are to be characterized as to size, length, and depth below the surface. Indications exceeding the following are unacceptable.	ASME Section V, Article 23, SA 578

Depth from Clad Surface	Length of Indication
Up to 0.02 inches	0.375 inches
0.02 inches - 0.06 inches	1.0 inches
0.06 inches - 0.1 inches	3.0 inches
Over 0.10 inches	6.0 inches

TABLE 4.6-12 REACTOR COOLANT SYSTEM INSPECTION C-E REQUIREMENTS (CONTINUED)

Replacement Reactor Vessel Head (continued)

<test>	C-E Requirements	Code Requirements
Liquid Penetrant Test (PT)	1. The final PT of J-weld and the half-inch area adjacent to the weld for CEDM adapters, instrument tube connections shall allow no indications of defects.	ASME Section III, NB 5245

Steam Generator

<test>	C-E Requirements	Code Requirements
Ultrasonic Test (UT)	1. UT for Defects in Tube Sheet Clad UT of Weld Clad for bond.	1. 1. None
	2. UT of Weld Clad for bond	2. None

Replacement Pressurizer [a]

<test>	C-E Requirements	Code Requirements
Ultrasonic Testing (UT)	1. Indications that produce an amplitude greater than the amplitude received from the one-eighth inch flat bottom hole and less than the amplitude from the three-eighth inch flat bottom hole are to be characterized as to their length, and depth below the surface. Indications exceeding the following are unacceptable.	ASME Section V, Article 23, SA 578

Depth from Clad Surface	Length of Indication
Up to 0.03 inches	0.375 inches
0.03 inch - 0.06 inch	1.0 inches
0.06 inch - 0.125 inch	3.0 inches
Over 0.125 inch	6.0 inches

(a) This is a Dominion requirement and not a CE requirement. The reactor vessel head and replacement pressurizer were replaced in accordance with Dominion Purchase Specifications.

TABLE 4.6-13 RT_{PTS} VALUES AT 54 EFPY

Vessel Location	Component Identification	Chemical Content: Cu (%)	Chemical Content: Ni (%)	Chemistry Factor	Initial RT_{NDT} (°F)	Margin Term (°F)	Neutron Fluence (10^{19}n/ cm^2)	RT_{PTS} (°F)
Intermediate	C-505-1	0.13	0.61	91.3	8.1	34	3.83	165.1
Shell	C-505-2	0.13	0.62	91.5	17.5	34	3.83	174.7
Course Plates	C-505-3	0.13	0.62	91.5	5	34	3.83	162.2
Lower	C-506-1	0.15	0.60	110.0	7	34	3.78	188.8
Shell	C-506-2	0.15	0.61	110.0	-33.7	34	3.78	163.5
Course Plates	C-506-3	0.14	0.66	101.5	-19.2	34	3.78	183.6
Intermediate Shell Axial Welds	2-203 A (Heat A8746)	0.15	0.13	77.7	-56	66	2.83	114.6
	2-203 B/C (Heat A8746)	0.15	0.13	77.7	-56	66	3.53	114.6
Lower Shell Axial Welds	3-203 A (Heat A8746)	0.15	0.13	77.7	-56	66	3.83	114.6
	3-203 B/C (Heat A8746)	0.15	0.13	77.7	-56	66	2.50	114.6
Intermediate-to- Lower Shell Girth Weld	9-203 (Heat 10137)	0.22	0.04	100.0	-56.3	56	3.78	134.1
	9-203 (Heat 90136)	0.27	0.07	124.3	-56.3	56	3.78	166.7

TABLE 4.6-14 ADJUSTED REFERENCE TEMPERATURES (ART) PROJECTIONS

Vessel Location	Component Identification	Neutron Fluence [(a)], 10^{19} n/cm^2 (E>1 MeV) 54 EFPY		ART Projections [(b)] 54 EFPY	
		1/4T	3/4T	1/4T	3/4T
Intermediate	C-505-1	2.283	0.811	153.8	128.0
Shell	C-505-2	2.283	0.811	163.4	137.6
Course Plates	C-505-3	2.283	0.811	150.9	125.1
Lower	C-506-1	2.253	0.809	175.2	144.1
Shell	C-506-2	2.253	0.809	134.5	103.4
Course Plates	C-506-3	2.253	0.809	138.6	110.0
Intermediate Shell Axial Welds	2-203 A (Heat A8746)	2.283	3.811	104.6	82.6
	2-203 B (Heat A8746)	1.508	0.536	96.0	73.7
	2-203 C (Heat A8746)	1.508	0.536	96.0	73.7
Lower Shell Axial Welds	3-203 A (Heat A8746)	2.283	0.811	104.6	82.6
	3-203 B (Heat A8746)	1.490	0.529	95.8	73.4
	3-203 C (Heat A8746)	1.490	0.529	95.8	73.4
Intermediate to Lower Shell Girth Weld	9-203 (Heat 10137)	2.253	0.800	121.7	93.5
	9-203 (Heat 90136)	2.253	0.800	151.3	116.2

(a) Maximum neutron fluence for vessel location.

(b) Adjusted reference temperature projections based on best-estimate copper and nickel content, chemistry factor, initial RT_{NDT} and margin given in Table 4.6-13.

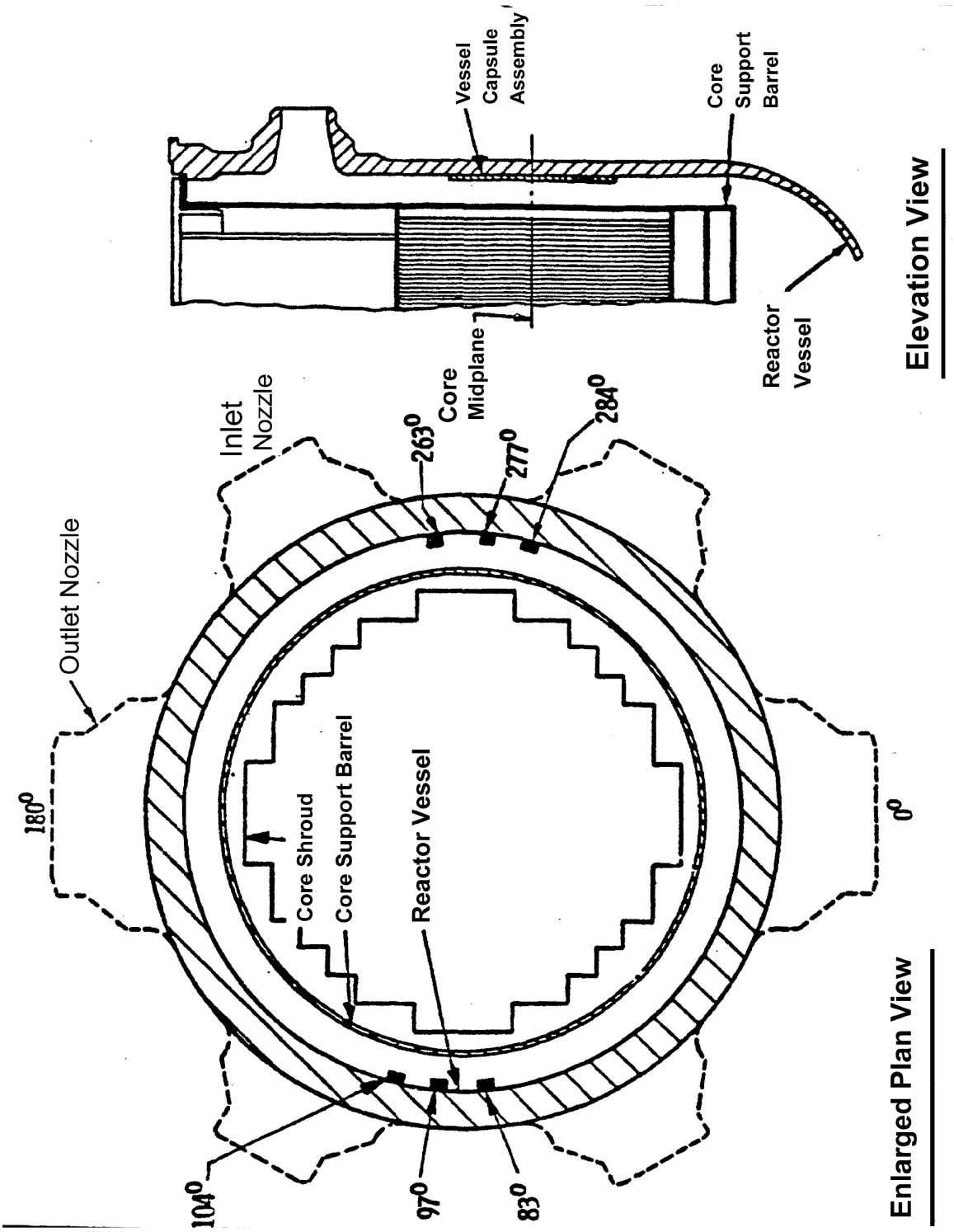


FIGURE 4.6–1 LOCATION OF SURVEILLANCE CAPSULE ASSEMBLIES

FIGURE 4.6–2 TYPICAL SURVEILLANCE CAPSULE ASSEMBLY

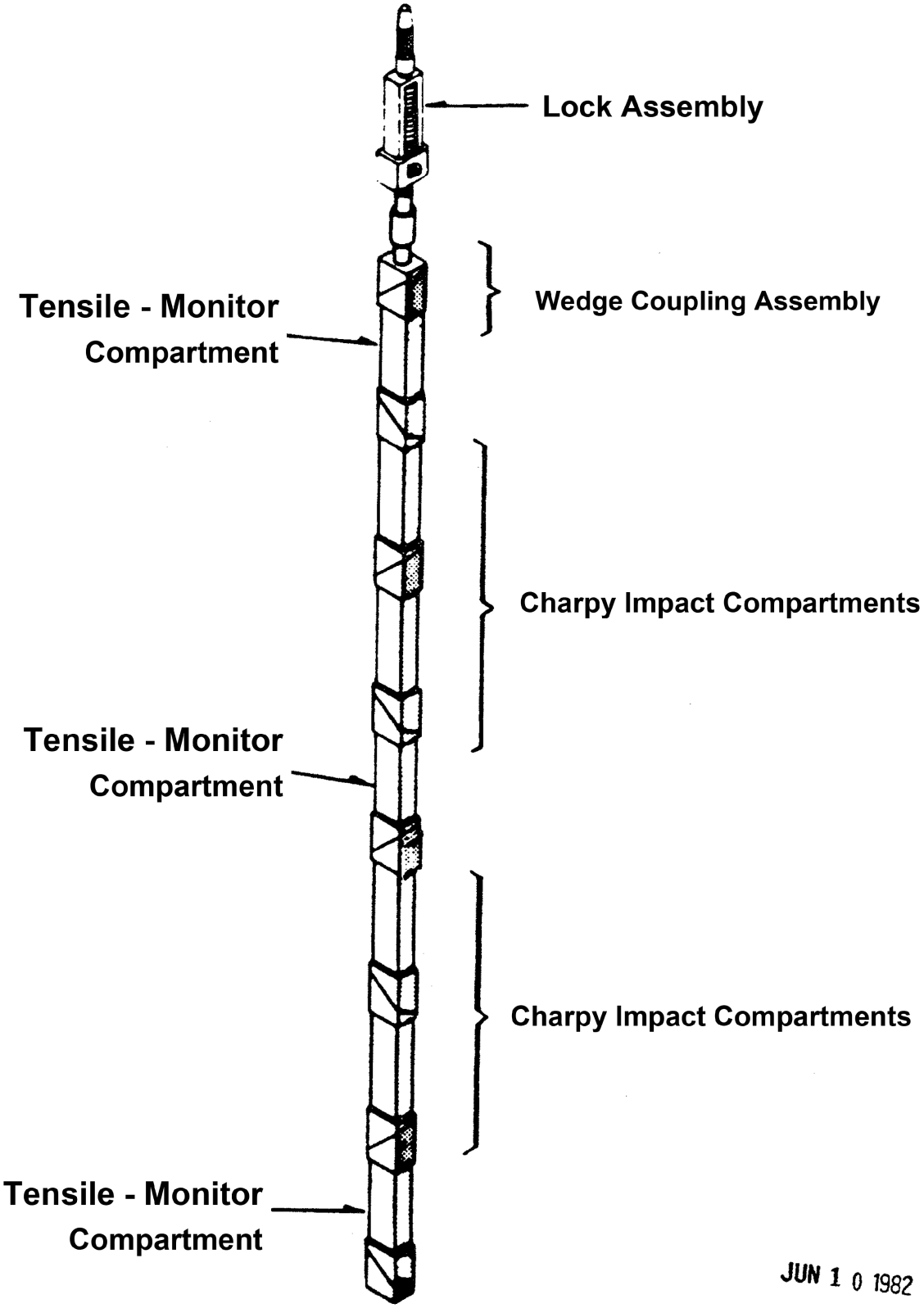


FIGURE 4.6–3 TYPICAL CHARPY IMPACT COMPARTMENT ASSEMBLY

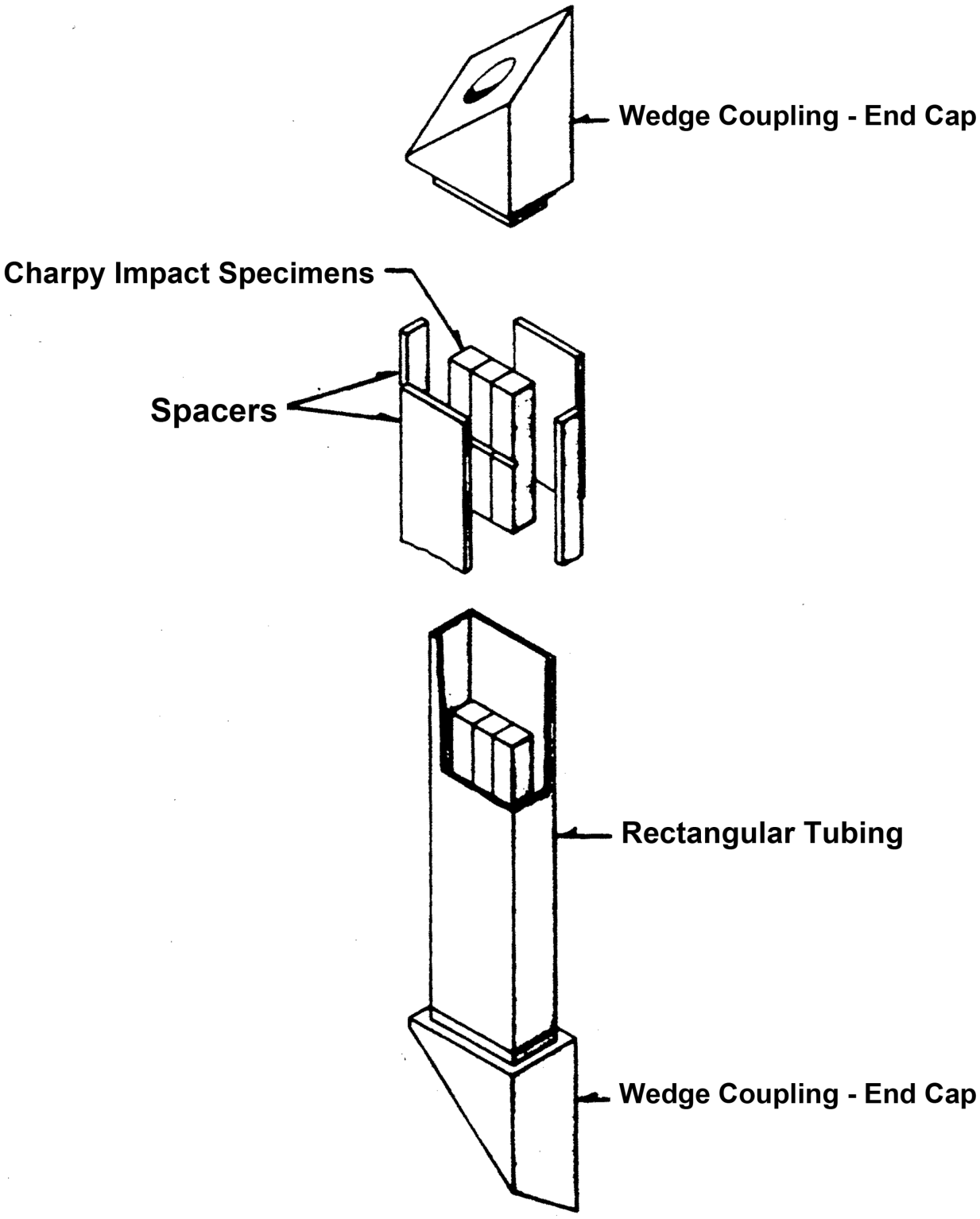


FIGURE 4.6–4 TYPICAL TENSILE-MONITOR COMPARTMENT ASSEMBLY

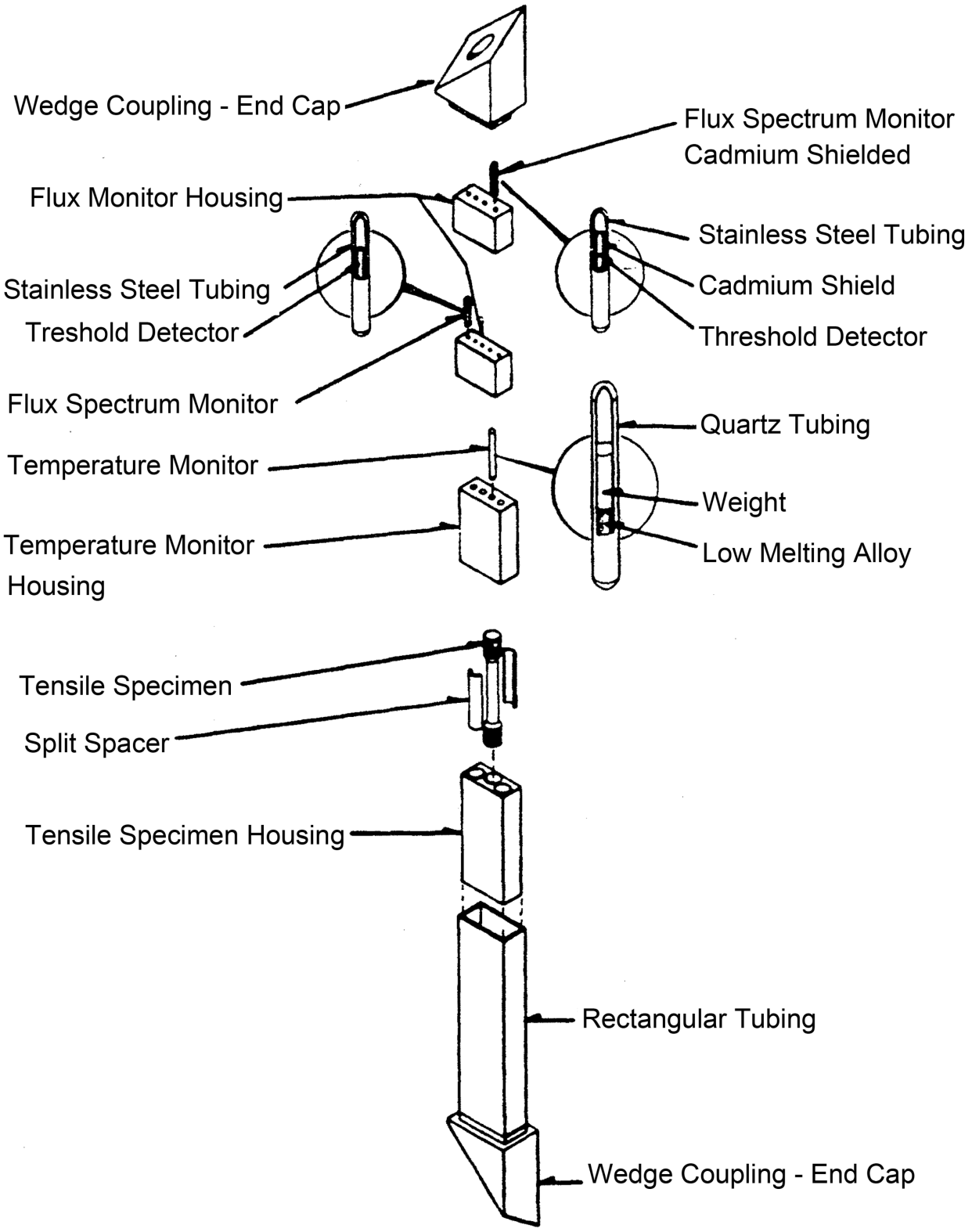


FIGURE 4.6–5 BASE METAL - WR (TRANSVERSE) PLATE C-506-1 IMPACT ENERGY VS TEMPERATURE

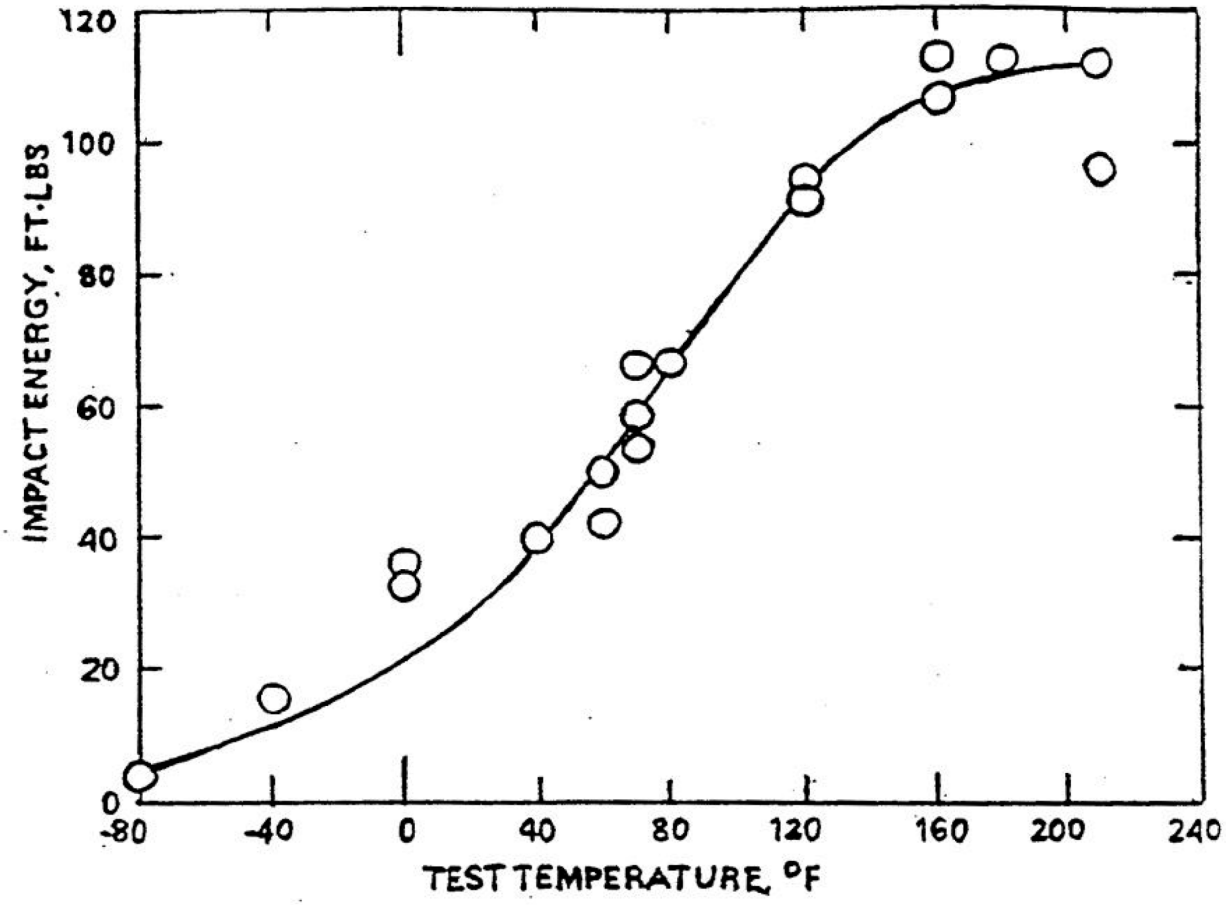


FIGURE 4.6–6 BASE METAL - WR (TRANSVERSE) PLATE C-506-1 LATERAL EXPANSION VERSUS TEMPERATURE

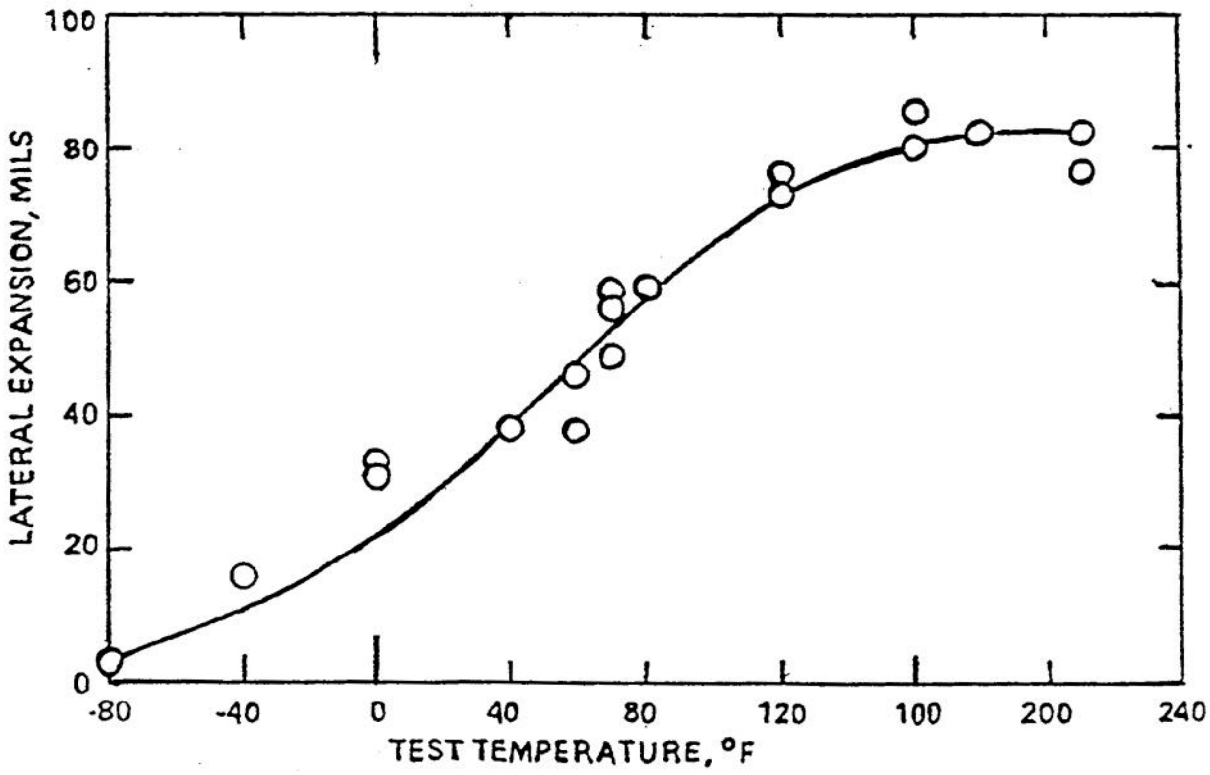


FIGURE 4.6–7 BASE METAL - RW (LONGITUDINAL) PLATE C-506-1 IMPACT ENERGY VERSUS TEMPERATURE

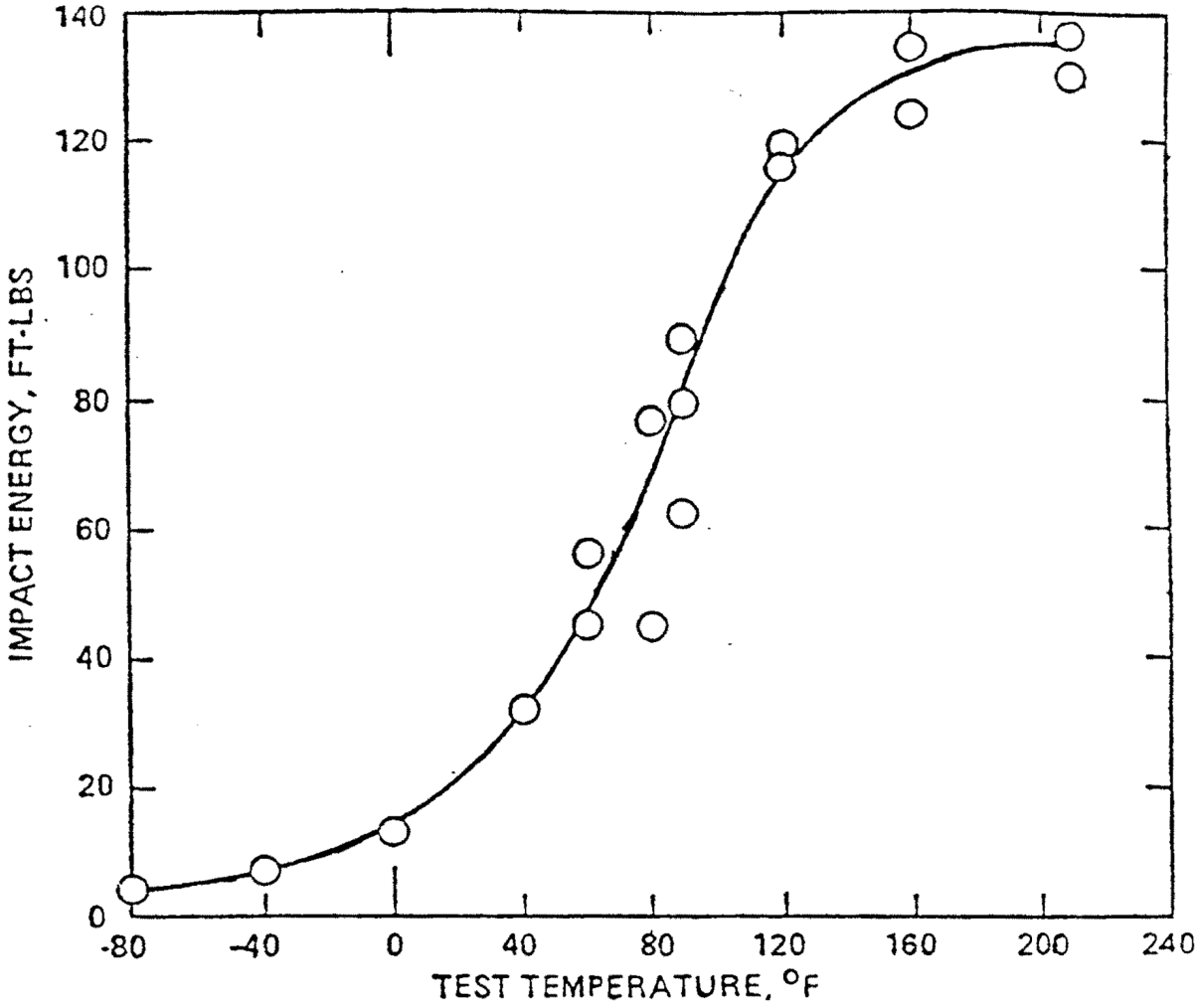


FIGURE 4.6–8 BASE METAL - RW (LONGITUDINAL) PLATE C-506-1 LATERAL EXPANSION VS TEMPERATURE

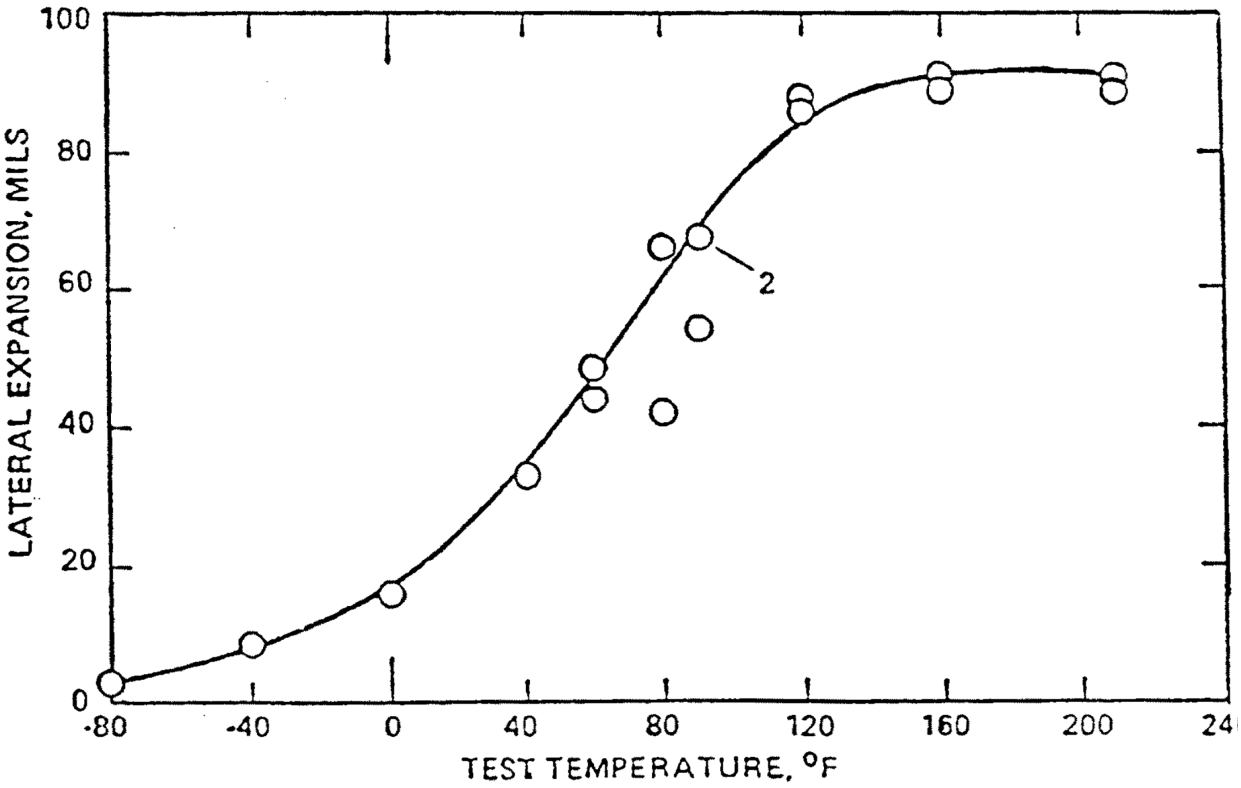


FIGURE 4.6–9 WELD METAL PLATE C-506-2/C-506-3 IMPACT ENERGY VS TEMPERATURE

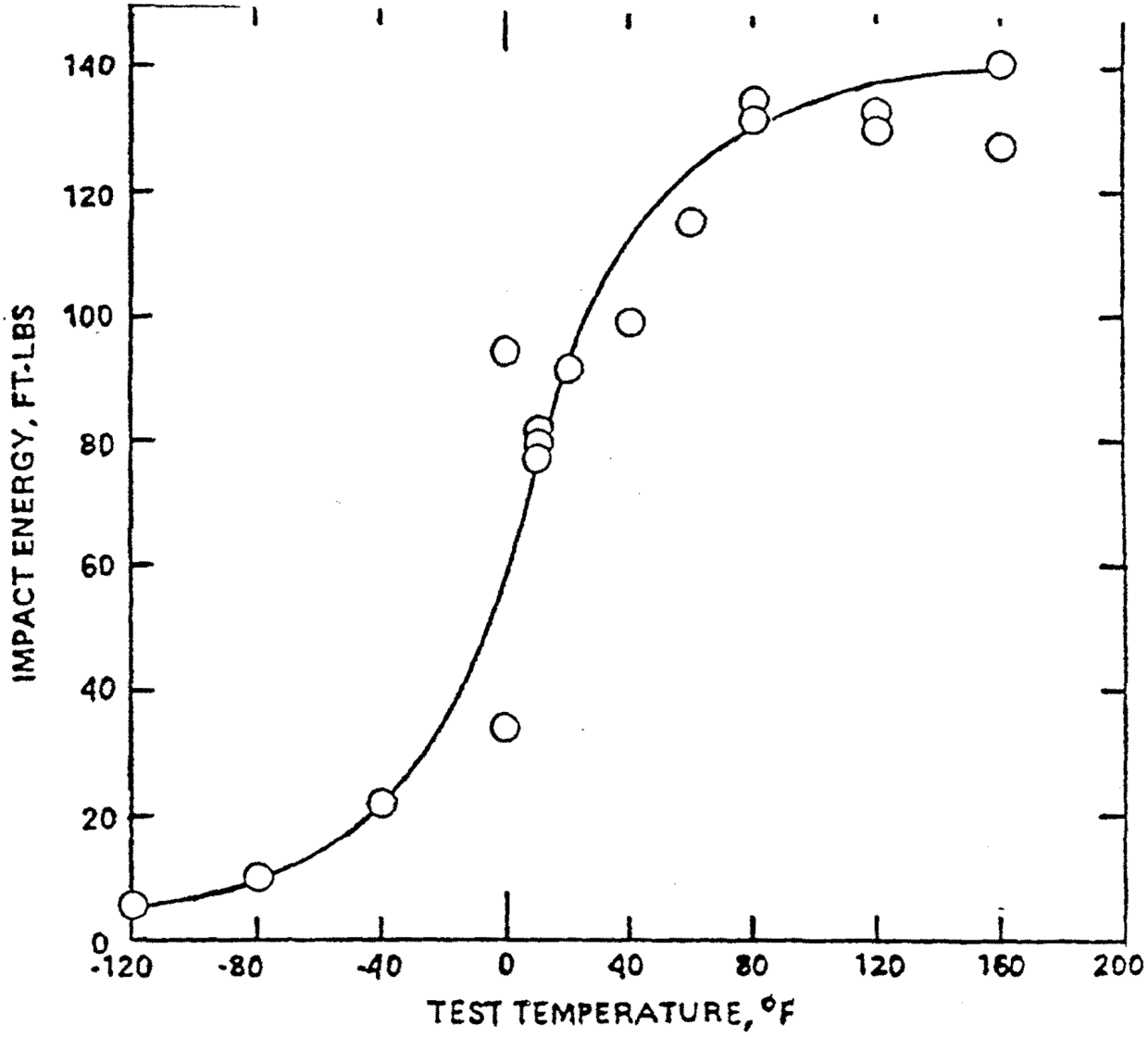


FIGURE 4.6–10 WELD METAL, PLATE C-506-2/C-506-3 LATERAL EXPANSION VS TEMPERATURE

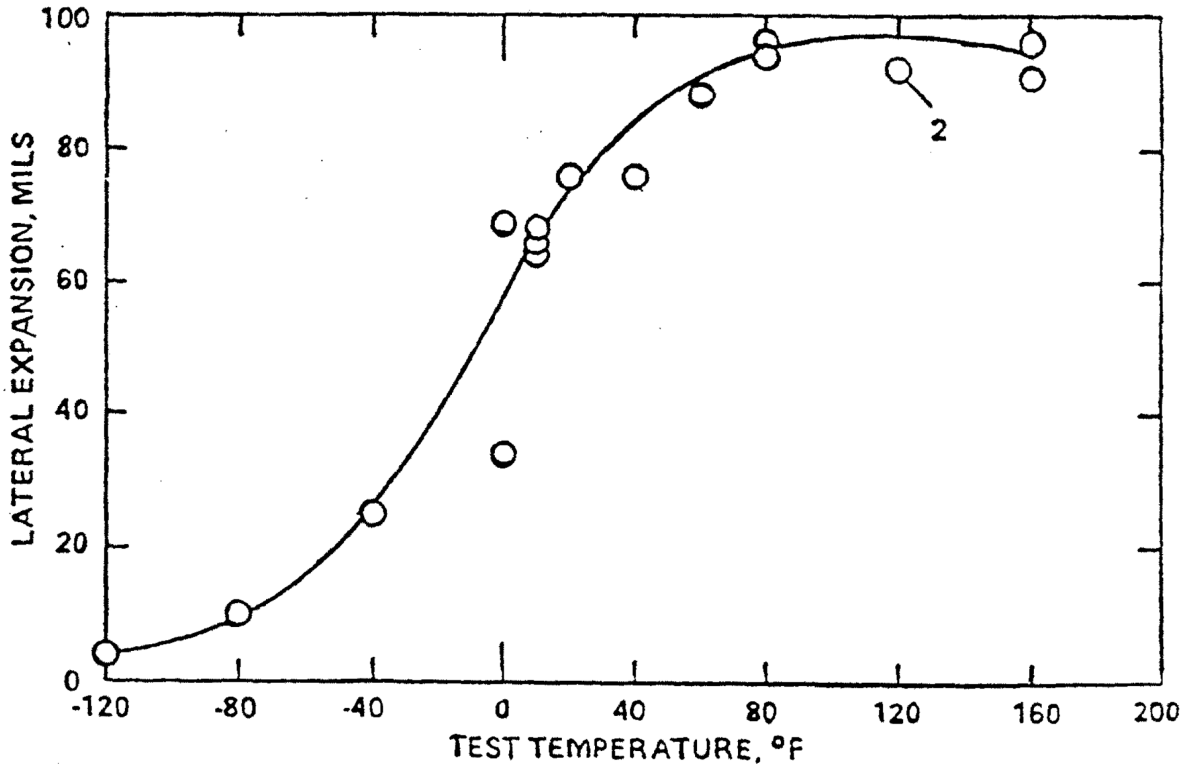


FIGURE 4.6–11 HAZ METAL, PLATE C-506-1 IMPACT ENERGY VERSUS TEMPERATURE

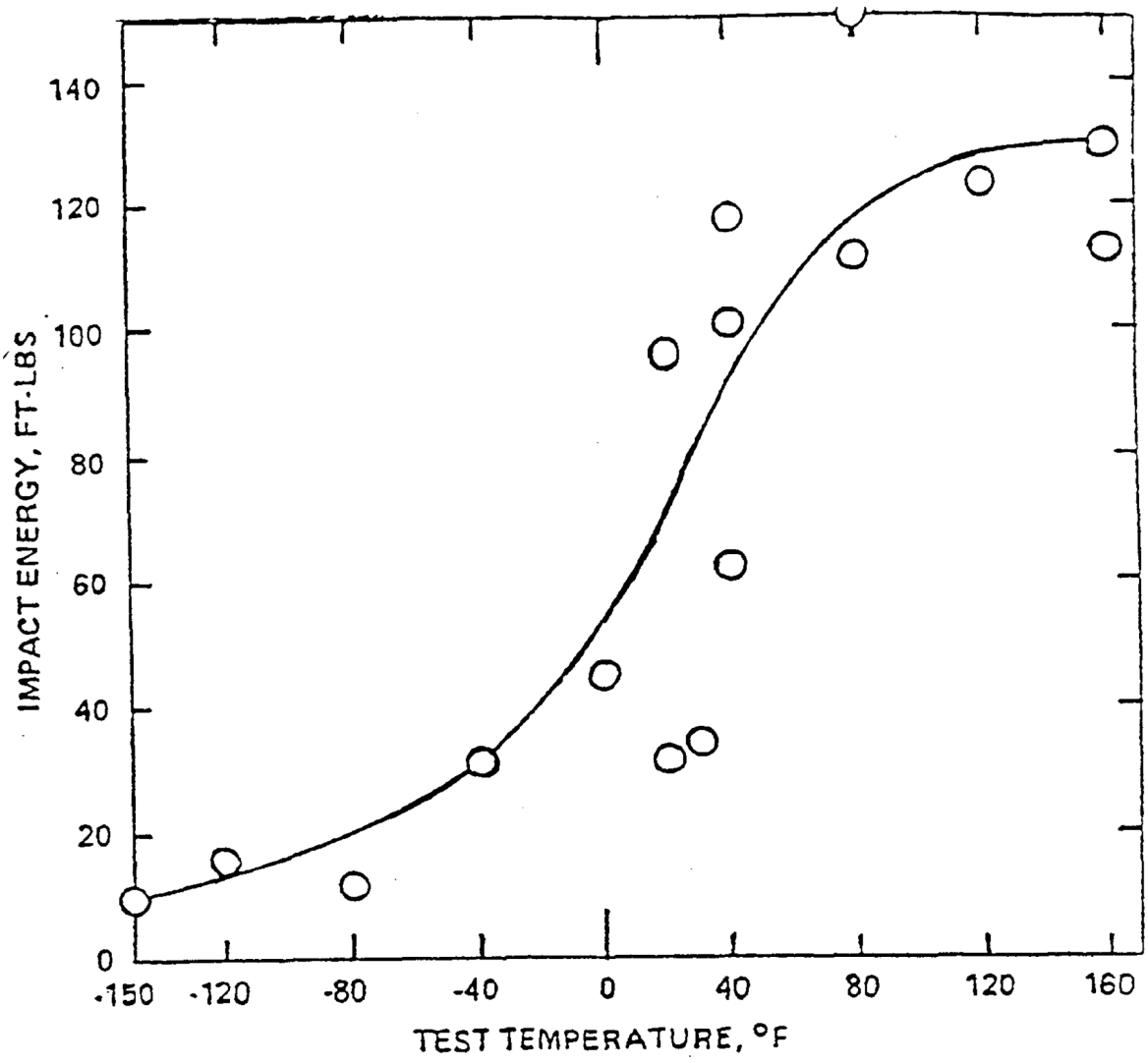


FIGURE 4.6–12 HAZ METAL, PLATE C-506-1 LATERAL EXPANSION VERSUS TEMPERATURE

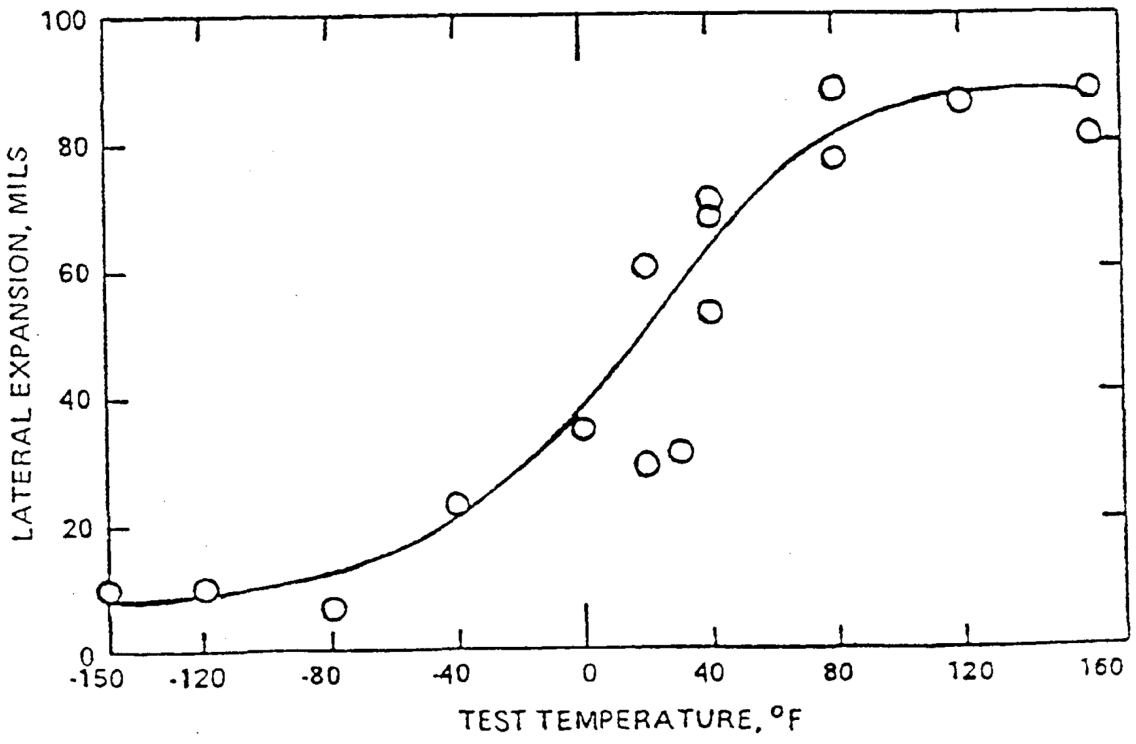


FIGURE 4.6–13 SRM (HSST PLATE 01MY - LONGITUDINAL) IMPACT ENERGY VERSUS TEMPERATURE

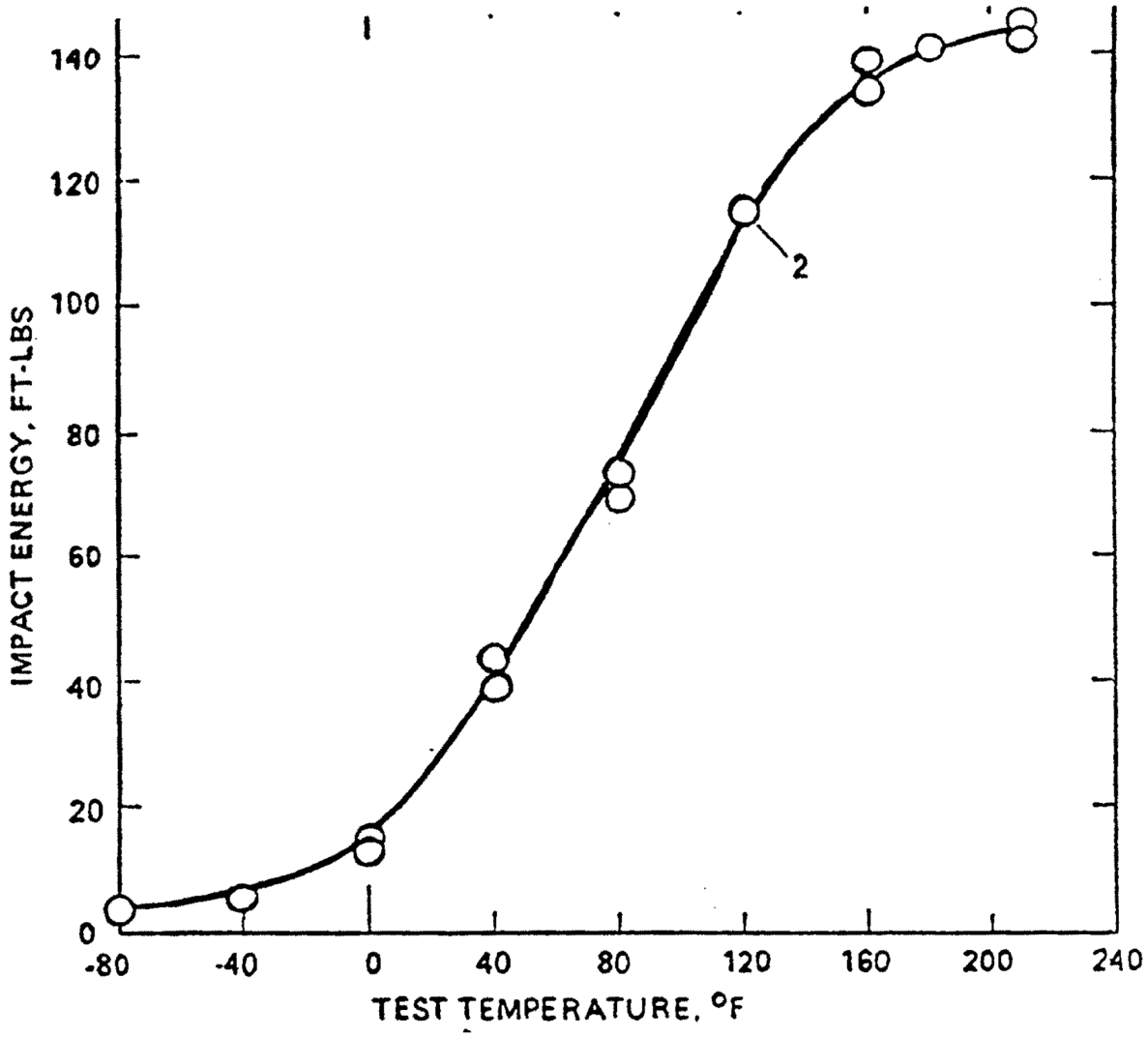
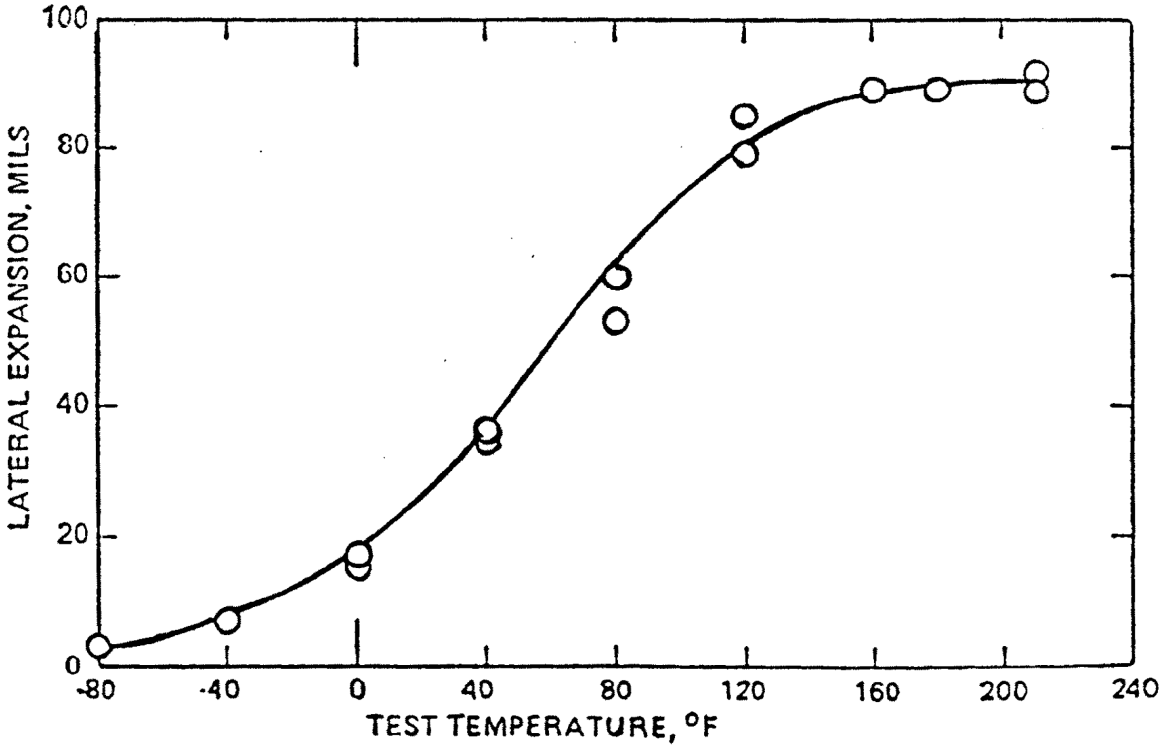


FIGURE 4.6–14 SRM (HSST PLATE 01MY - LONGITUDINAL) LATERAL EXPANSION VERSUS TEMPERATURE



4.A SEISMIC ANALYSIS OF REACTOR COOLANT SYSTEM

4.A.1 INTRODUCTION

The purpose of this appendix is to describe the methods employed and present the results obtained by dynamic seismic analyses of the reactor coolant system components. These analyses were performed to confirm the adequacy of the seismic loadings specified for the design of the components and the supports of the reactor coolant system which includes the reactor vessel, the steam generators, the reactor coolant pumps, the pressurizer, and the interconnecting reactor coolant piping.

Dynamic seismic analysis of the reactor vessel internals was performed separately and is discussed in Appendix 3.A.

4.A.2 METHOD OF ANALYSIS

4.A.2.1 General

The seismic analysis of the reactor coolant system (RCS) components was performed using either normal mode or direct integration theory in conjunction with time history and response spectrum techniques, as appropriate.

Time history techniques were employed in the analysis of the reactor vessel, the two steam generators, the four reactor coolant pumps and the interconnecting reactor coolant piping. In the analysis of these components, a single composite mathematical model, which included integral representations of each of the components and connecting piping, was employed to account for the interacting effects of dynamic coupling. The analysis of these dynamically coupled multi-supported components utilized different time dependent input excitations applied simultaneously to each support.

The analyses of the pressurizer and the surge line piping employed separate, uncoupled, mathematical models and utilized response spectrum techniques.

The input data, time histories and response spectra, applied in the analyses were provided by the analysis of the containment structure internal support structure described in Section 5.8.

The RCS components were analyzed using either modal or proportional (Rayleigh) methods of damping. Except for analysis of the surge line piping, all modal analyses used a constant damping factor of 1% of critical damping for all active modes. In the analysis of the surge line piping, a damping factor of 0.5% of critical damping was used for each mode. When proportional damping was used, Alpha and Beta were conservatively selected to provide less than 1% of critical damping at the significant frequencies of response of the major components.

4.A.2.2 Mathematical Models

In the descriptions of the mathematical models which follow, the spatial orientations are defined by the set of orthogonal axes where Y is in the vertical direction, and X and Z are in the horizontal plane, in the directions indicated on the appropriate figure. The mathematical representation of the section properties of the structural elements employs a 12 by 12 stiffness matrix for the three-dimensional space frame models, and employs a 6 x 6 stiffness matrix for the two dimensional plane frame model. Elbows in piping runs include the in-plane/out-of-plane bending flexibility factors as specified in the ASME Code, Section III.

4.A.2.2.1 Reactor Coolant System - Coupled Components

A schematic diagram of the composite mathematical model, designated MS-2, used in the original plant design analyses of the dynamically coupled components of the reactor coolant system is presented in Figure 4.A–1 and 4.A–1A. This model, Figure 4.A–1, includes 19 mass points with a total 47 dynamic degrees of freedom. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristic of the components, as follows: the reactor vessel, with internals, is represented by 5 mass points with a total of 13 dynamic degrees of freedom; each of the two steam generators are represented by 3 mass points with a total of 7 dynamic degrees of freedom; and each of the four reactor coolant pumps are represented by 2 mass points with a total of 5 dynamic degrees of freedom. The relatively small mass of the interconnecting reactor coolant piping is lumped proportionately with the masses of the adjoining components.

<component>	**Number of Mass Points**	**Number of Dynamic Degrees of Freedom**
Reactor Vessel and Internals	5	13
Steam Generators (2)	6	14
R.C. Pumps (4)	8	20
Total	19	47

The mathematical model, Figure 4.A–1, as defined, provides a complete three dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The total mass of the entire coupled system is dynamically active in each of the three coordinate directions.

In addition to Model MS-2 described above, a second model of the coupled components, designated Model RV14, was formulated for the original plant design to incorporate a more detailed representation of the reactor vessel assembly. With the exception of the representation of the reactor vessel assembly, Model RV14 is identical to Model MS-2, Figure 4.A–1. A schematic diagram of the representation of the reactor vessel assembly incorporated into Model RV14 is presented in Figure 4.A–2. This more detailed representation of the reactor vessel assembly

consists of 15 mass points with a total of 33 dynamic degrees of freedom and includes a 10 mass point, 22 dynamic degrees of freedom representation of the reactor vessel internals. The representation of the reactor vessel internals was formulated in conjunction with the analysis of the reactor vessel internals discussed in Appendix 3.A, and was designed to simulate the dynamic characteristics of the models used in that analysis. The coupled Model RV14 was used to generate time histories of absolute accelerations at the reactor vessel flange used as forcing functions in the analysis of the reactor vessel internals.

Coupled model RV14, modified to delete the reactor internals representation of the thermal shield and to reflect current design steam generator properties, was also used in analyses to determine the effects of the replacement steam generators on RCS responses. This modified model representation consists of 27 mass points with 63 dynamic degrees of freedom.

<component>	**Number of Mass Points**	**Number of Dynamic Degrees of Freedom**
Reactor Vessel	5	11
Reactor Internals	8	18
Steam Generators(2)	6	14
Reactor Coolant Pumps (4)	8	20
Total	27	63

4.A.2.2.2 Pressurizer

The mathematical model employed in the analysis of the pressurizer is shown schematically in Figure 4.A–3. This lumped parameter, planer model provides a multi-mass representation of the axially symmetric pressurizer and includes 5 mass points with a total of 6 dynamic degrees of freedom.

The replacement pressurizer was analyzed using BWSPAN for both operating basis and safe shutdown earthquake using the response spectrum method. The seismic excitation is applied at the support skirt elevation. The significant change to the analysis from the original pressurizer analysis is that the analysis is performed at 65% water level. The seismic directional responses are combined by the absolute sum of each horizontal and vertical response of the spectrum analysis.

4.A.2.2.3 Surge Line

The lumped parameter, multi-mass mathematical model employed in the analysis of the surge line is shown schematically in Figure 4.A–4. The surge line is modeled as a three dimensional piping run with end points anchored at the attachments to the pressurizer and the reactor vessel outlet piping. In the definition of the mathematical model, 10 mass points with a total of 27 dynamic degrees of freedom were selected to provide a complete three-dimensional representation of the dynamic response of the surge line. All supports and restraints defined for the surge line assembly

are included in the mathematical model. The total mass of the surge line is dynamically active in each of the three coordinate directions.

4.A.2.3 Calculations

4.A.2.3.1 General

As applied in the analysis, the simultaneous equations of motion for linear structural systems with viscous damping can be written, Reference 4.A-1:

$$M\ddot{X} + C\dot{X} + KX = -M\ddot{Y} - K_{ms}X_s$$

where:

M = diagonal matrix of lumped masses

C = square symmetric damping matrix

K = square symmetric stiffness matrix which defines the mass point force-displacement relationship.

$\ddot{Y}$ = column matrix with elements equal to the absolute acceleration of the datum support in the coordinate direction of the related dynamic degree of freedom of the structural system.

K_{ms} = rectangular matrix of stiffness coefficients which defines the mass point force, non-datum support displacement relationship.

X_s = column matrix of displacements relative to the datum at non-datum supports.

X = column matrix of mass point displacements relative to the datum.

$\dot{X}$ = column matrix of mass point velocities relative to the datum.

$\ddot{X}$ = column matrix of mass point accelerations relative to the datum.

In this form, the equations define the dynamic response of a multi-mass structural system subjected to time dependent support motion. In the analysis of systems with multiple supports, such as the coupled components of the reactor coolant system, the equations provide for different time-dependent input motions at each of the supports. In this case, one of the supports of the system is designated the reference, or datum, from which the motions of all other points of the structural system are measured. The reactor vessel support was designated as the datum in the analyses of the coupled components of the reactor coolant system.

Normal mode theory, as described in References 4.A-1 and 4.A-2, was employed to reduce the equations of motion to a system of independent equations in terms of the normal modes for the modal superposition time-history and spectrum analyses of the reactor coolant system components. For direct integration time-history analyses, the equations of motion were solved

using numerical integration methods with an integration time step of 0.001 second. In the analyses, the dynamic response of the components was determined for seismic input excitations in each of the three global coordinate directions: X (east-west), Y (vertical) and Z (north-south). The dynamic responses to vertical seismic excitation were found for both the case of initial support displacement upward and the case of initial support displacement downward. These responses were combined to determine the most severe combinations produced by the effects of seismic excitations in each of the horizontal directions applied simultaneously with either seismic excitation in the vertical direction.

4.A.2.3.2 Frequency Analysis

An eigenvalue analysis was performed utilizing the ICES STRUDL II computer code, Reference 4.A-3, to calculate the mode shapes and natural frequencies of the original plant design composite mathematical models. Modifications to the standard ICES STRUDL II program have been implemented by Combustion Engineering to include a Jacobi diagonalization procedure in the eigenvalue analysis and to provide appropriate influence coefficients and stiffness matrices for use in the response and reaction calculations.

The natural frequencies and dominant degrees of freedom calculated are shown in Table 4.A-1 for all modes used in the analysis of the reactor coolant system Model MS-2, the surge line and the pressurizer.

An eigenvalue analysis was also performed utilizing the ANSYS computer code, Reference 4.A-5, to calculate the natural frequencies and mode shapes of modified model RV14 which was used in the analysis to evaluate the effects of the replacement steam generators. A comparison of response frequencies for corresponding mode shapes identified no significant differences from those in Table 4.A-1 (see also Section 4.A.5, Effects of Replacement Steam Generators).

4.A.2.3.3 Mass Point Response Analysis

The original plant design time history mass point responses to seismic excitation were computed using TMCALC, a C-E code. This code performs a numerical integration of the equations of motion for singly or multiply supported dynamic systems utilizing normal mode theory, Reference 4.A-2, and Newmark’s Beta-Method with Beta equal to one-sixth, Reference 4.A-4. For the multiply supported system, the separate time histories of each support were imposed on the system simultaneously. The results are time history responses of the mass points. The analysis of the reactor coolant system utilized modal data for all frequencies through 40 cps.

The mass point responses resulting from the spectrum analysis were found utilizing SHAKE, a C-E computer code. This code performs a normal mode response spectrum analysis resulting in the modal inertials loads found using the response spectrum for the pressurizer support. The mass point responses of the surge line were found using an envelope of the support spectra of the interconnected major components.

The transient analysis capability of the ANSYS computer code, Reference 4.A-5, was used to compute the responses of the coupled components of the reactor coolant system with existing replacement steam generators and with the reactor thermal shield removed.

4.A.2.3.4 Seismic Reaction Analysis

The original plant design dynamically induced loads at all system design points due to the time history support excitations and mass point responses were calculated utilizing FORCE, a C-E computer code. This code performs a complete loads analysis of the deformed structure at each incremental time step by computing internal and external system reactions (forces and moments) by superposition of the reactions due to the mass point displacements and the non-datum support displacements as follows:

$$R(t) = C_m X_m(t) + C_s X_s(t)$$

where:

$R(t)$ = the matrix of all components of the reactions at the system design points.

C_m = the matrix of mass points displacement influence coefficients.

$X_m(t)$ = the column matrix of time history mass point displacements relative to the datum at each time step.

C_s = the matrix of support displacement influence coefficients.

$X_s(t)$ = the column matrix of time history support displacements relative to the datum at non-datum supports at each time step.

The support and mass point displacements due to horizontal and vertical seismic excitations are added algebraically at each time step. The maximum component of each reaction for the entire time domain, and its associated time of occurrence, are selected.

The maximum reactions for the pressurizer and surge line resulting from the response spectrum analysis were found by applying the modal inertial loads for each mode, to the structural model using the STRUDL computer code. The design point reactions due to each modal loading were conservatively combined by summing the absolute values of the modal reactions. For the replacement pressurizer, the design point reactions due to each modal loading were combined using square root sum of the squares. The surge line analysis included consideration of the relative end displacements. The reactions found by statically imposing the maximum relative displacements of the two ends of the surge line were conservatively included by absolute summation with the inertial response from the spectrum analysis.

The system design loads for the coupled components of the reactor coolant system with existing replacement steam generators and with the reactor internals thermal shield removed were calculated using the ANSYS computer code.

4.A.3 RESULTS

The reactions (forces and moments) at all design points in the system, obtained from the dynamic seismic analysis, were compared with the seismic loads in each component design specification. The results of this comparison are summarized in Table 4.A-2 for the points of maximum calculated load.

The maximum seismic loads calculated by the time history techniques are the result of a search and comparison over the entire time domain of each individual component of load due to the simultaneous application of the horizontal and either vertical excitation. The maximum calculated components of load shown in Table 4.A-2 for each design location do not in general occur at the same time, nor for the same combination of horizontal and vertical excitation, and therefore result in a conservative case.

Except for the replacement pressurizer, the maximum seismic loads calculated by the response spectrum techniques are the result of combining the modal reactions due to the horizontal and the vertical excitation on an absolute sum basis.

The results shown are for the Operational Basis Earthquake. For conservative determination of results due to the Design Basis Earthquake, both the calculated results and specification values have been multiplied by a factor of 2.0, rather than 0.17/0.09 = 1.89, the ratio of DBE to OBE maximum ground accelerations.

4.A.4 EFFECTS OF THERMAL SHIELD REMOVAL

An engineering evaluation was made to assess the effects of the thermal shield removal on the dynamic response characteristics of the reactor vessel and the reactor coolant system. The main effect of the thermal shield removal was a reduction in the weight of the reactor vessel. This reduction was approximately two percent (2%). Since the stiffness of the connection between the reactor vessel and internals did not change, it was concluded that the dynamic response characteristics of the reactor vessel and, therefore, the reactor coolant system would not change significantly with the removal of the thermal shield. Reactor vessel flange motions, originally used for a more detailed evaluation of the internal structures in Appendix 3.A, also remain unchanged.

4.A.5 EFFECTS OF REPLACEMENT STEAM GENERATORS

The most significant effects of the replacement steam generators were increases in weights and center of gravity elevations of approximately 2.7% and 13.2 inches, respectively, for these components. Modified model RV14, which includes these changes in addition to those due to removal of the reactor internals thermal shield, was used in a reanalysis of the dynamically coupled components of the RCS. The changes in steam generator properties resulted in a reduction of approximately 4.5% in the fundamental frequencies in which the steam generator response is predominant. Corresponding changes in loads which are attributed to the replacement steam generators are generally insignificant with no design governing load increases of more than

7%. The loads from this reanalysis, incorporating the effects of both thermal shield removal and steam generator replacement, are included in Table 4.A-2.

Reactor vessel flange motions were compared with those originally used in the detailed evaluation of the internal structures in Appendix 3.A. This comparison concluded that use of the original flange motions is conservative for reactor internals design.

4.A.6 CONCLUSION

It is concluded that the seismic loadings specified for the design of the reactor coolant system components and supports are adequate. All seismic loads calculated by the dynamic seismic analysis are less than the corresponding loads in the component design specification.

4.A.7 REFERENCES

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4.A-2 Hurty, W. C., and Rubinstein, M. F., “Dynamics of Structures,” Chapter 8, Prentice Hall, Inc., Englewood Cliffs, New Jersey, 1964.

4.A-3 ICES STRUDL II Engineering Users Manual, R68-91, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts.

4.A-4 Newmark, N. M., “A Method of Computation for Structural Dynamics”, Volume 3, Journal of Engineering Mechanics Division, A.S.C.E., July, 1959.

4.A-5 DeSalvo, G. P., and Swanson, J. A., “ANSYS - Engineering Analysis System,” Swanson Analysis Systems, Inc., Elizabeth, PA., 1972.

TABLE 4.A-1 NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

Mode Number	Frequency (cps)	Dominant Degrees of Freedom		
		Names	Directions	Locations
1	3.07	M66	Z	Pump 1B
2	3.08	M43	Z	Pump 2B
3	3.21	M61	Z	Pump 1A
4	3.23	M52	Z	Pump 2A
5	3.32	RI1	Z	Reactor Internals
6	3.32	RI1	X	Reactor Internals
7	5.11	M61	X,Y	Pump 1A
8	5.14	M52	X,Y	Pump 2A
9	5.21	M66	X,Y	Pump 1B
10	5.23	M43	X,Y	Pump 2B
11	10.51	SG5A, SG5B	X	Steam Generators 1 & 2
12	10.52	SG5A, SG5B	X	Steam Generators 1 & 2
13	10.60	M61	X	Pump 1A
14	10.74	M52	X	Pump 2A
15	10.98	M66	X	Pump 1B
16	11.11	M43	X	Pump 2B
17	12.14	RI2	Z	Reactor Internals
18	12.16	RI2	X	Reactor Internals
19	20.40	SG9A	Z	Steam Generator 1
20	20.42	SG9B	Z	Steam Generator 2
21	23.65	RI1	Y	Reactor Internals
22	27.26	SG9, 10, A & B	X	Steam Generators 1 & 2
23	27.79	SG5A, SG5B	Y	Steam Generators 1 & 2
24	27.82	SG5A, SG5B	Y	Steam Generators 1 & 2
25	28.99	M65, M42	Z	Pumps 1B & 2B
26	29.91	M65, M42	Z	Pumps 1B & 2B
27	30.37	M42	Z	Pump 2B
28	31.43	M60	Z	Pump 1A
29	31.92	M51	Z	Pump 2A
30	37.51	V1, V3, V4	X, Z	Reactor Vessel
31	39.34	SG5B	Z	Steam Generator 2

TABLE 4.A-1 NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

Mode Number	Frequency (cps)	Dominant Degrees of Freedom		
		Names	Directions	Locations
32	39.41	SG5A	Z	Steam Generator 1
1	13.78	-	X, Z	Replacement Pressurizer
2	44.65	-	X, Z	Replacement Pressurizer
3	48.99	-	X	Replacement Pressurizer
4	62.35	-	X, Z	Replacement Pressurizer
5	75.86	-	X, Z	Replacement Pressurizer
6	75.97	-	Y	Replacement Pressurizer
7	99.45	-	X, Z	Replacement Pressurizer
1	5.75	7, 8, H3	Y	Surge Line
2	11.29	5, 7, H1	X	Surge Line
3	15.98	5, H1	Y	Surge Line
4	21.79	8, 9, H3	Z	Surge Line
5	25.81	9, H3	Y	Surge Line
6	32.12	8, 9	X	Surge Line
7	40.35	10	X	Surge Line
8	73.64	5	Z	Surge Line
9	108.52	3	X	Surge Line
10	129.12	10	Y	Surge Line
11	151.51	7, H3	X	Surge Line
12	155.03	7	Y	Surge Line
13	174.76	7, H1	Y	Surge Line
14	186.39	H1	X	Surge Line
15	229.90	10, 11	Z	Surge Line
16	260.38	11	X	Surge Line
17	304.86	3	Y	Surge Line
18	320.52	7	X	Surge Line
19	503.60	H1	X	Surge Line
20	525.21	8	Y	Surge Line
21	535.87	8	X	Surge Line
22	542.64	11	Z	Surge Line
23	668.06	4	X	Surge Line
24	752.11	4	Z	Surge Line

TABLE 4.A-1 NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

Mode Number	Frequency (cps)	Dominant Degrees of Freedom		
		Names	Directions	Locations
25	938.66	11	Y	Surge Line
26	1327.41	8	Z	Surge Line
27	1807.56	4	Y	Surge Line

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Reactor Vessel Outlet Nozzle	Fx	107.0	69.0
		Fy	3.0	172.0
		Fz	65.0	63.0
		Mx	1237.0	2168.0
		My	7329.0	7521.0
		Mz	382.0	38474.0
		M_R	7443.0	39263.0
	Reactor Vessel Inlet Nozzle	Fx	97.0	75.0
		Fy	98.0	105.0
		Fz	107.0	108.0
		Mx	7719.0	22392.0
		My	5402.0	10085.0
		Mz	16910.0	14087.0
		M_R	19358.0	28312.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined East-West and Vertical	Reactor Vessel Outlet Nozzle	Fx	196.0	241.0
		Fy	39.0	253.0
		Fz	4.0	10.0
		Mx	67.0	1042.0
		My	360.0	972.0
		Mz	3277.0	43773.0
		M_R	3298.0	43796.0
	Reactor Vessel Inlet Nozzle	Fx	81.0	220.0
		Fy	97.0	146.0
		Fz	110.0	63.0
		Mx	5963.0	9638.0
		My	3803.0	5981.0
		Mz	11763.0	12770.0
		M_R	13726.0	17080.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Steam Generator Inlet Nozzle	Fs	88.5	144.0
		Fa	89.0	230.0
		Mb	3901.7	4000.0
		Mt	2584.0	2640.0
	Steam Generator Outlet Nozzles	Fs	134.8	202.0
		Fa	54.0	104.0
		Mb	8990.0	13200.0
		Mt	10517.0	14800.0
Combined East-West and Vertical	Steam Generator Inlet Nozzle	Fs	97.1	144.0
		Fa	175.0	230.0
		Mb	4239.0	4000.0
		Mt	113.0	2640.0
	Steam Generator Outlet Nozzle	Fs	126.1	202.0
		Fa	49.0	104.0
		Mb	7681.0	13200.0
		Mt	11224.0	14800.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Pressurizer Surge Line Nozzle	Fx	1.74	2.26
		Fy	0.91	4.85
		Fz	2.71	8.75
		M_R	192.7	690.9
Combined East-West and Vertical	Pressurizer Surge Line Nozzle	Fx	1.07	5.70
		Fy	0.59	4.77
		Fz	1.22	2.71
		M_R	121.5	617.1

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Reactor Coolant Pump Nozzle	Fx	108.0	83.6
		Fy	61.0	93.2
		Fz	53.0	114.1
		Mx	6289.0	12030.0
		My	5632.0	10188.0
		Mz	11867.0	10095.0
		M_R	14564.0	18719.0
	Reactor Coolant Pump Outlet Nozzle	Fx	137.0	93.2
		Fy	98.0	71.3
		Fz	43.0	97.8
		Mx	10457.0	22770.3
		My	2244.0	7514.4
		Mz	2244.0	8957.0
		M_R	7797.0	25596.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined East-West and Vertical	Reactor Coolant Pump Inlet Nozzle	Fx	119.0	143.6
		Fy	50.0	104.5
		Fz	35.0	33.9
		Mx	5291.0	8709.0
		My	4479.0	5238.0
		Mz	12263.0	16662.5
		M_R	14087.0	19517.0
	Reactor Coolant Pump Outlet Nozzle	Fx	134.0	263.8
		Fy	97.0	145.8
		Fz	37.0	109.9
		Mx	5099.0	4072.0
		My	2964.0	13317.0
		Mz	6269.0	16648.0
		M_R	8608.0	21796.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Reactor Vessel Outlet Piping	M_R	7443.0	11514.0
	Steam Generator Inlet Piping	M_R	4680.0	4361.0 *
	* Note: Piping design is controlled by combined East-West and vertical seismic excitation.			
	Steam Generator Outlet Piping	M_R	13836.0	13836.0
	Pump Inlet Piping	M_R	13148.0	13836.0
	Pump Outlet Piping	M_R	13236.0	18528.0
	Reactor Vessel Inlet Piping	M_R	18522.0	18528.0
Combined East-West and Vertical	Reactor Vessel Outlet Piping	M_R	3298.0	11581.0
	Steam Generator Inlet Piping	M_R	4241.0	9658.0
	Steam Generator Outlet Piping	M_R	13601.0	13836.0
	Pump Inlet Piping	M_R	13632.0	13836.0
	Pump Outlet Piping	M_R	8608.0	18528.0
	Reactor Vessel Inlet Piping	M_R	13726.0	18528.0

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Surge Line RCS Nozzle	Fx	4.41	6.0
		Fy	1.45	4.0
		Fz	3.25	4.0
		M_R	277.6	468.0
Combined East-West and Vertical	Surge Line RCS Nozzle	Fx	2.55	6.0
		Fy	0.84	4.0
		Fz	1.31	4.0
		M_R	155.5	468.0
Combined North-South and Vertical	Surge Line Hanger H4	Fx	0.150	0.18
		Fy	0.439	2.0
	Surge Line Hanger H2	Fy	0.209	1.0
		Fz	0.078	0.11
Combined East-West and Vertical	Surge Line Hanger H4	Fx	0.083	0.18
		Fy	0.335	2.0
	Surge Line Hanger H2	Fy	0.145	1.0
		Fz	0.016	0.11

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined North-South and Vertical	Reactor Vessel Outlet Support	Fy	165.0	392.0
		Fz	325.0	663.0
	Reactor Vessel Inlet Support	Fy	314.0	692.0
		F_H	191.0	304.0
	Steam Generator Lower Support	Fy	219.0	431.0
		Fz	195.0	397.0
		Mx	20345.0	24422.0
		My	2335.0	10332.0
	Steam Generator Upper Support	Fx	20.0	24.0
		Fz	219.0	240.0
	Pump Support	Fy	2.6	9.2
	Replacement Pressurizer Support	Fy	14.1	80.0 [a]
		Fz	80.1	84.0 *
		Mx	15107.6 22768.9	17036.0 *

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

TABLE 4.A-2 SEISMIC LOADS ON REACTOR COOLANT SYSTEM COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE (CONTINUED)

SEISMIC EXCITATION	COMPONENT AND DESIGN LOCATION	SEISMIC LOAD		
		COMPONENT	CALCULATED MAXIMUM	DESIGN BASIS
Combined East-West and Vertical	Reactor Outlet Support	Fy	309.0	473.0
		Fz	47.0	42.0
	Reactor Vessel Inlet Support	Fy	227.0	469.0
		F_H	375.0	1020.0
	Steam Generator Lower Support	Fy	278.0	624.0
		Fz	32.0	29.0
		My	356.0	455.0
		Mz	8792.0	24383.0
	Steam Generator Upper Support	Fx	281.0	296.0
		Fz	12.0	10.0
	Reactor Coolant Pump Support	Fy	1.6	4.3
	Replacement Pressurizer Support	Fx	55.6	81.0 *
		Fy	14.1	80.0 *
		Mz	15746.9	16949.0 *

Forces = Kips; Moments = Inch - Kips

$M_R = [Mx^2 + My^2 + Mz^2]^{1/2}$

$F_H = [Fx^2 + Fz^2]^{1/2}$

a. Note that the design basis values are based on original pressurizer design and are retained here for historical purposes only.

FIGURE 4.A–1 REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL MS2

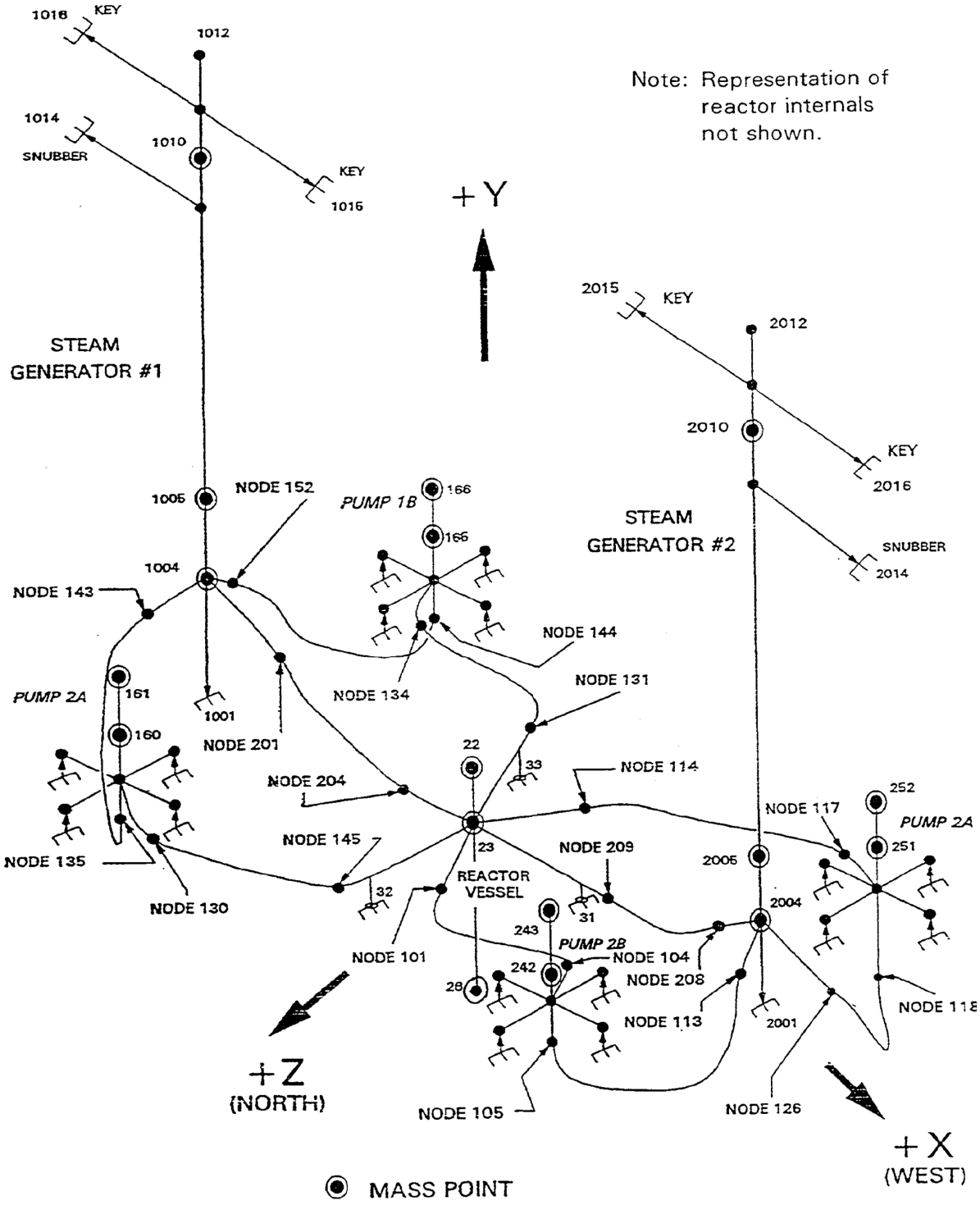


FIGURE 4.A–1A REACTOR COOLANT SYSTEM - SEISMIC ANALYSIS MODEL MS2 AND RV14

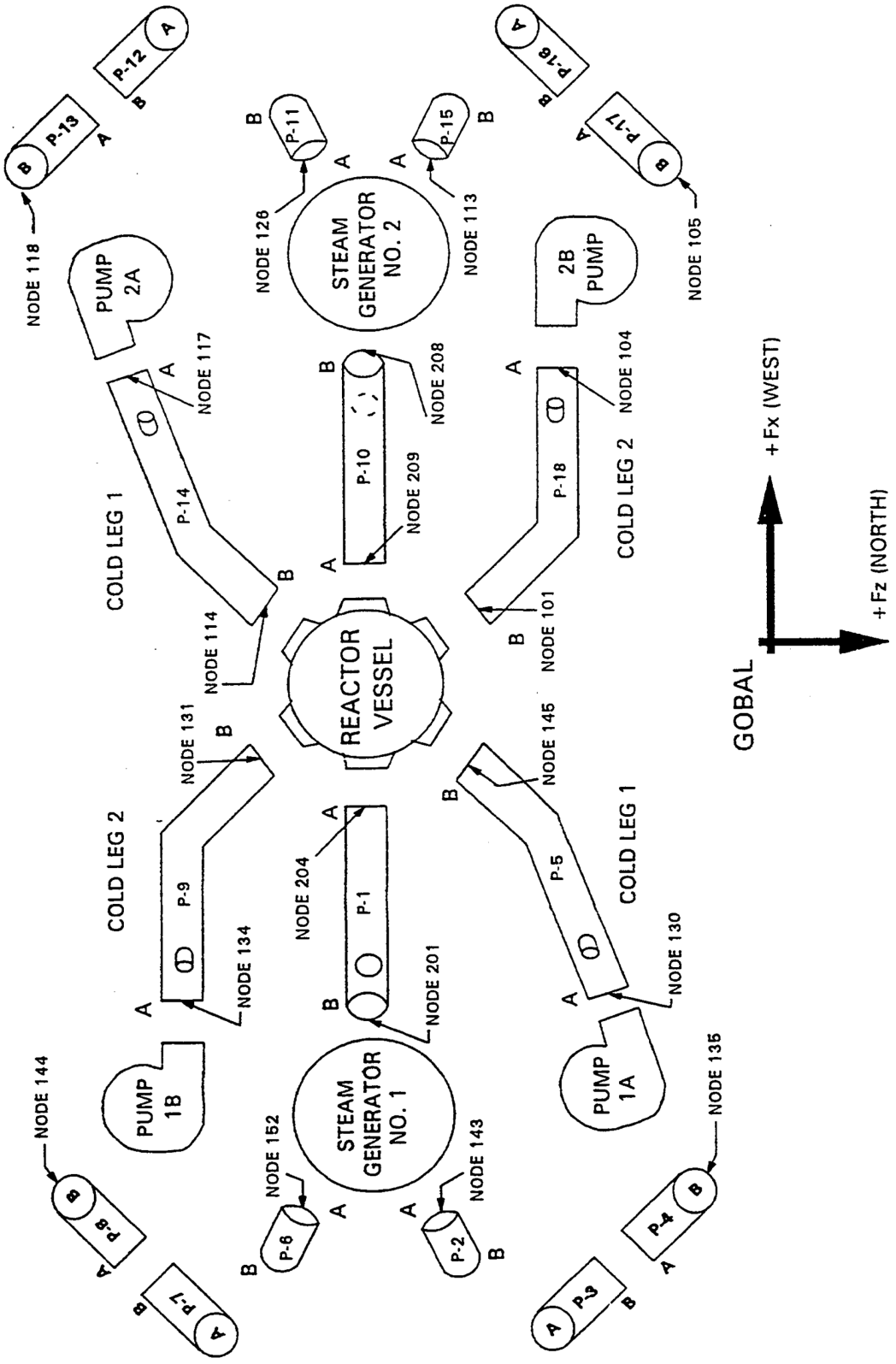


FIGURE 4.A–2 RV14 REACTOR AND INTERNALS SEISMIC ANALYSIS MODEL

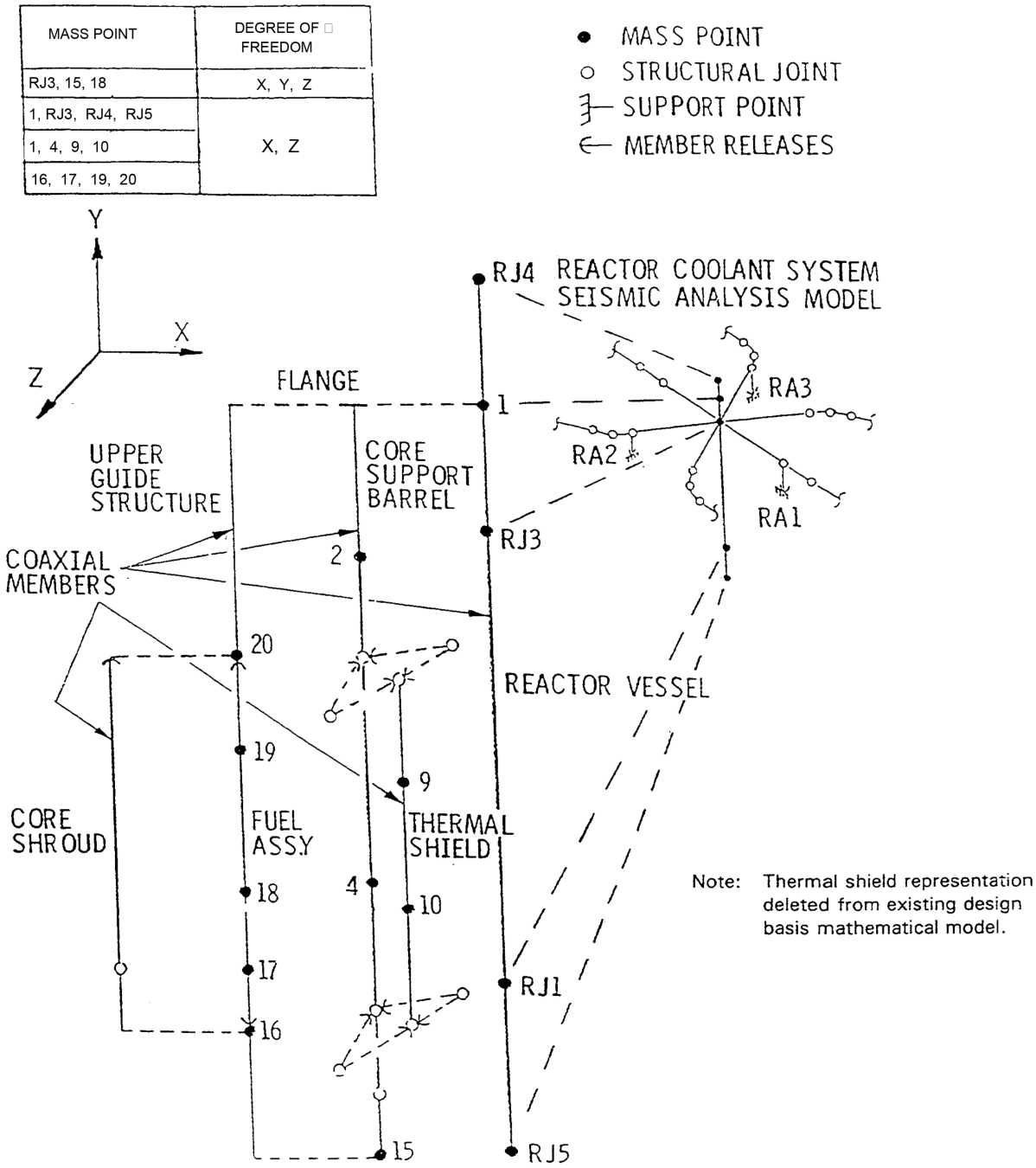
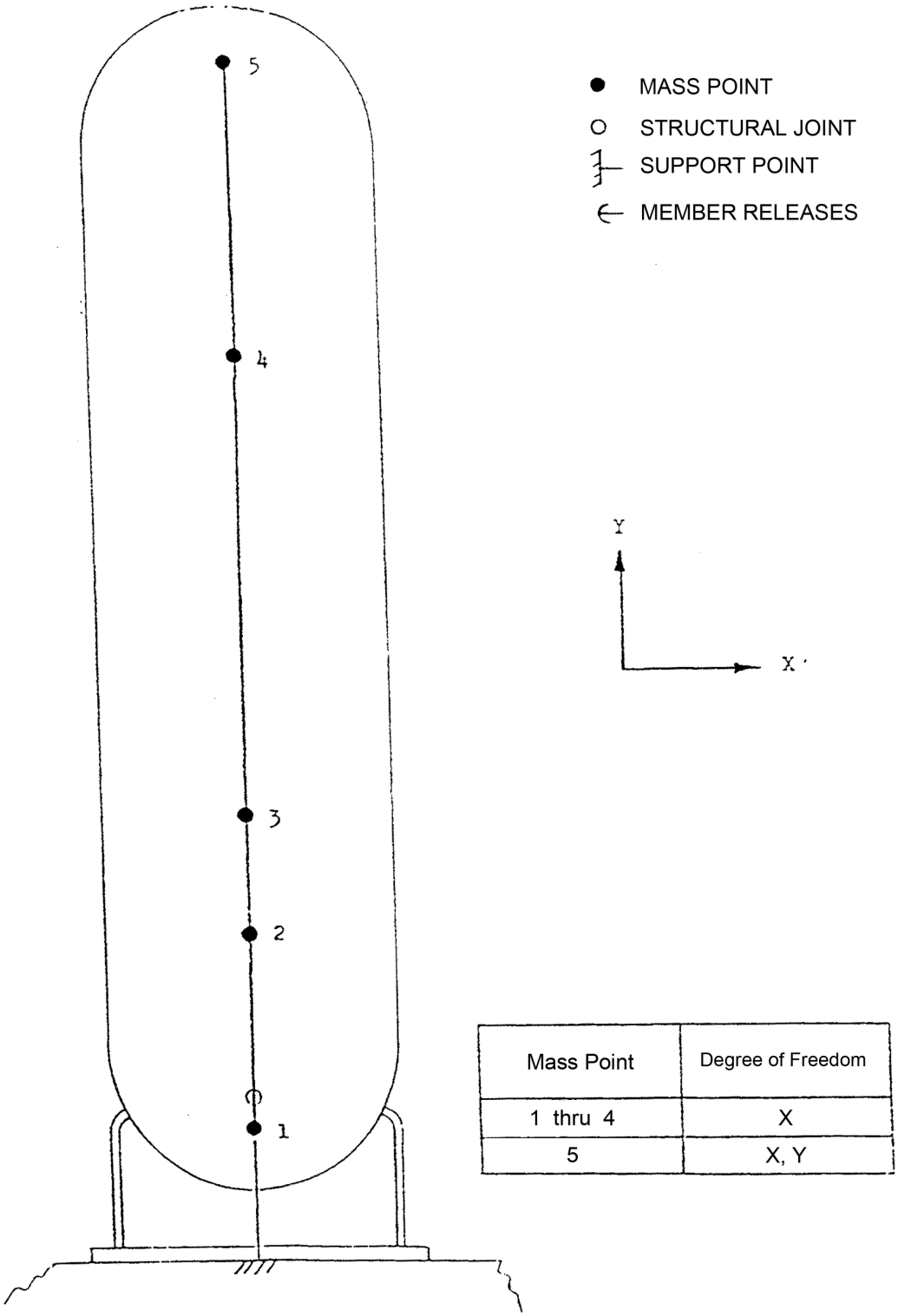
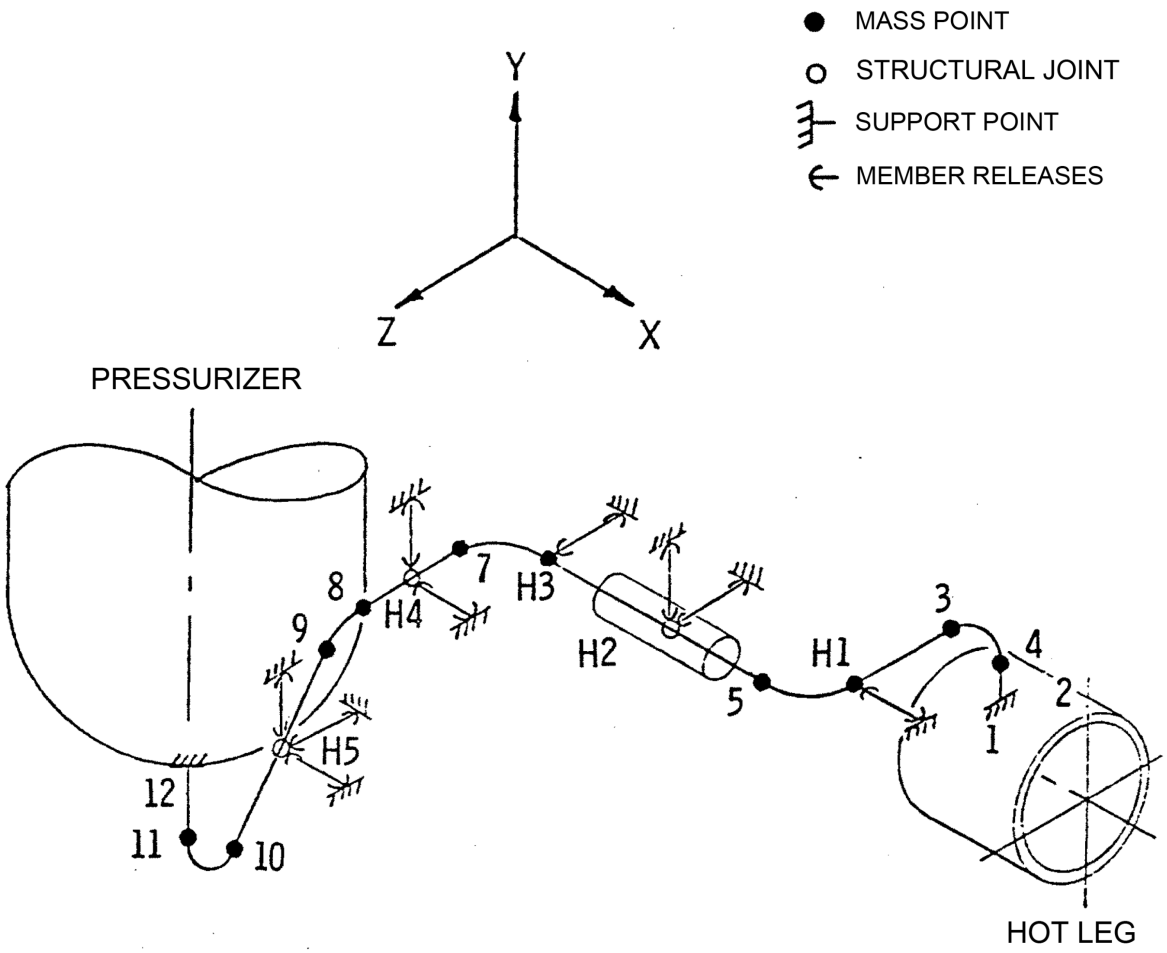


FIGURE 4.A–3 PRESSURIZER SEISMIC ANALYSIS MODEL



Mass Point	Degree of Freedom
1 thru 4	X
5	X, Y

FIGURE 4.A–4 SURGE LINE SEISMIC ANALYSIS MODEL



Mass Point	Degree of Freedom
3, 7	X, Y
H3	Z, Y
4, H1, 5, 8, 9, 10, 11	X, Y, Z

Support Name	Support Restraint Directions
H1	Fx
H2	Fy, Fz
H3	Fz
H4	Fx, Fy
H5	Fx, Fy, Fz

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 5: Structures

CHAPTER 5—STRUCTURES

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CHAPTER 5 – STRUCTURES

5.1 GENERAL

The design bases for structures required for the normal operating conditions are governed by the building design codes and specifications listed in Section 5.1.2. The basic design criterion for the design basis accident and seismic conditions specified that there shall be no loss of any function of the structures which can cause danger to the safety of the public. It should be noted that the terms “Category” and “Class” are used interchangeably throughout the MP2 FSAR in defining seismic design classifications of Structures, Systems and Components.

5.1.1 CLASSES OF STRUCTURES

Structures are grouped into two classes, depending on how their functions relate to public safety and plant operation.

5.1.1.1 Class I Structures

Class I structures are those structures whose loss of function could:

a. Cause or increase the severity of an accident.

b. Preclude establishing and maintaining safe shutdown.

c. Result in a release of radioactivity to the site boundary in excess of the l0 CFR l00 guidelines.

Class I structures are designed to withstand the appropriate seismic and other applicable loads without loss of function. These Class I structures are sufficiently isolated or protected from Class II structures to ensure that their integrities are maintained at all times.

The following are Class I structures:

a. The containment shell and internals

b. The enclosure building

c. The auxiliary building

d. The warehouse (eastern portion of the auxiliary building)

e. The turbine building, except turbine pedestals

f. The intake structure

g. The supports for all Class I system components

Note: See Table 1.4-1 for complete listing of Class I structures, systems, and components that follow the guidance of Regulatory Guide 1.29, Seismic Design Classification.

5.1.1.2 Class II Structures

Class II structures are those whose failure would not result in the release of radioactivity beyond the site boundary in excess of the l0 CFR 20 annual limits and would not prevent safe shutdown of the reactor. The failure of Class II structures, however, may interrupt power generation.

All structures that are not listed under Class I are Class II structures.

5.1.2 CODES AND SPECIFICATIONS

The following codes and specifications, where applicable, were used as the bases for the design and construction of all structures. Modifications to these codes and specifications are noted in the appropriate sections that describe the details of the structures, materials, and construction practices. Later editions of the AISC Manual of Steel Construction (7th, 8th, and 9th edition - Allowable Stress Design) and ACI 318 Code (1971, 1977, 1983, 1989, and 1995) were used for the design and construction of new and modified portions of structures.

Subsequent editions of the AISC Manual of Steel Construction and the ACI 318 Code, as governed by the plant design change process, may be used for the design and construction of new and modified portions of structures.

a. Uniform Building Code (l967 Edition)

b. Building Code Requirements for Reinforced Concrete (ACI 3l8-63)

c. Specifications for Structural Concrete for Buildings (ACI 30l-66)

d. Manual of Steel Construction (AISC, 6th Edition, 1963)

e. State of Connecticut Building Code

f. ASME Boiler and Pressure Vessel Code (1968 Edition)

g. ASTM Standards – Materials and testing procedures used are referenced in the appropriate sections. Whenever possible, ASTM or ASME material and testing procedures are used. Materials and testing procedures for new and modified portions of structures conform to ASTM or ASME Standards as implemented by plant design change documents.

5.1.3 REGULATORY GUIDES

The following Regulatory Guides that were in effect at the time of application for an Operating License were used, where applicable, to establish the bases for the design and construction of all structures. Every effort was made to follow the guidance of the documents. Any areas where the guidance was not followed are detailed in the specified subsections.

Regulatory Guide Number	Title	Subsection
1.10	Mechanical (Cadwell) Splices in Reinforcing Bars of Concrete Structures	5.9.3.2.3
1.11	Instrumentation Lines Penetrating Primary Reactor Containment	5.2.7.2.1
1.13	Fuel Storage Facility Design Basis	5.4.1.1.3
1.15	Testing of Reinforcing Bars for Concrete Structures	5.9.3.2.2
1.19	Nondestructive Examination of Primary Containment Liners	5.9.3.5.3
1.35	In-service Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures	5.2.8.4
1.12	Instrumentation for Earthquakes	5.8.6
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	5.2.8.2

5.2 CONTAINMENT GENERAL DESCRIPTION

The containment system used for Millstone Unit 2 consists of a concrete cylindrical structure, hereinafter referred to as the containment, and a steel framed structure called the enclosure building, which completely surrounds the containment. The spaces between the enclosure building and the containment, together with selected areas of the auxiliary building such as the penetration rooms and rooms containing the engineered safety features, are referred to as the enclosure building filtration region (EBFR). In the event of a loss-of-coolant accident (LOCA), the EBFR is maintained at a slight vacuum by the enclosure building filtration system. Air from the EBFR is processed through charcoal filters and released through the Millstone stack.

The containment consists of a prestressed, reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab. The cylindrical portion is prestressed by a post-tensioning system composed of horizontal and vertical tendons, with the horizontal tendons placed in three 240 degree systems using three buttresses as supports for the anchorages. The dome has a three-way post-tensioning system. The concrete foundation slab is conventionally reinforced with high strength reinforcing steel. A continuous access gallery is provided beneath the base slab for installation of vertical tendons. A one-quarter inch thick welded steel liner is attached to the inside surface of the concrete shell to ensure a high degree of leak-tightness. The floor liner is installed on top of the structural slab and is then covered with concrete.

The containment completely encloses the reactor, reactor coolant system, and portions of the auxiliary and engineered safety features systems. It ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the reactor coolant system occurs.

Principal nominal dimensions of the containment are as follows:

Inside diameter (feet)	130
Inside height (feet)	175
Cylindrical wall thickness (feet)	3.75
Dome thickness (feet)	3.25
Foundation slab thickness (feet)	8.5
Liner plate thickness (inches)	0.25
Internal free volume (cubic feet)	1,899,000

The containment is shown in Figures 5.2–1 and 5.2–2.

5.2.1 CONSTRUCTION MATERIALS

The following materials are used in the construction of the containment.

a. Structural and Miscellaneous Steel

Rolled shapes, plates, and bars	ASTM A-36
Crane rails	ASTM A-1
High strength bolts	ASTM A-325 or ASTM A-490
Stainless steel	ASTM A-240, Type 304

b. Concrete

Base slab (psi)	5000
Cylindrical wall and dome (psi)	5000
Tendon access gallery (psi)	3000
Floor slabs above floor liner (psi)	3000
Primary shield walls and steam generator pedestals (psi)	5000
All other internal structures (psi)	4000

c. Reinforcing Steel

Deformed bars	ASTM A-615, Grade 60
Spiral bars	ASTM A-82

d. Prestressing Steel Tendons, Anchorage, and Sheaths

Wires	ASTM A-421, Type BA
Bearing plates	Armco VNT
Stressing washers	ASTM A-4330
Shims	Armco VNT
Sheaths	Galvanized corrugated steel tubing, 22 Gauge

e. Containment Steel Liner Plate

one-quarter inch plates	ASTM A-285, Grade A
Insert plates	ASTM A-516, Grade 60
Penetration sleeves:	
Pipes	ASME SA-333, Grade 6
Plates	ASME SA-516, Grade 60

Exception	ASME SA-36 for Equipment Hatch External Bolting Attachments

f. Interior Coating (Original Construction)

Steel liner plate

Primer	Carbo-Zinc 11
Finish coat	Phenoline Number 305

Concrete and masonry surfaces

Surfacer	Keeler & Long, Number 6548 epoxy block filler
Primer	Keeler & Long Number 7107 epoxy white primer
Finish coat	Keeler & Long, epoxy enamel

g. Interior Maintenance Coatings (First implemented in Mid Cycle 13, 1997)

All coating materials applied to surfaces inside or to be installed in the reactor containment are epoxy materials tested to withstand Millstone Unit 2 design basis loss of coolant accident (DBA-LOCA) conditions. The coating materials and their application comply with the requirements of Regulatory Guide 1.54. Each coating was tested in accordance with ASTM D3911, "Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," to the DBA conditions represented by the pressure (70 psig) and temperature (340°F) curve of Figure 1, therein. Prior to exposure to the simulated DBA conditions, each coating was irradiated to an accumulated dose of at least $1x10^9$ Rads in accordance with ASTM D4082, "Effect of Gamma Radiation on Coatings for Use in Light-Water Nuclear Power Plants."

h. Waterproofing Membrane

The waterproofing membrane that was installed during construction of the containment is a continuous plain sheet of polyvinyl chloride applied to the concrete surface with an adhesive. The membrane was applied after the forms were stripped. The membrane is composed of an elastomeric material having thickness of 40 mils (minimum), a minimum tensile strength of 2,000 psi, a minimum elongation of 200 percent at 75-80°F, and the water absorption is less than 0.1%. The extent of the waterproofing membrane is shown in Figures 5.3–2 through 5.3–4 of the FSAR. All joints are lapped and the adhesive is applied continuously to the contact surface.

5.2.1.1 Corrosive Protection

The reinforcing steel and tendon sheaths are cast in the concrete walls and base slab, which is a corrosion inhibitive environment. The steel liner is in direct contact with, and anchored to, the inner surface of the concrete wall. Concrete will passivate the steel surfaces, thereby lowering the

galvanic potential and making it compatible with any buried copper ground conductors. Measurements taken in the area show that the earth has a very high resistivity. This further diminishes the possibility of galvanic corrosion. It is therefore believed that there is no need for a cathodic protection system for these steel members.

A refined petroleum oil based product is used as a protective compound for the tendons. The electrical resistivity of this compound is relatively high, which makes it a poor electrolyte. This prevents the possibility of galvanic corrosion that could be detrimental to the tendons.

Note: For description of water intrusion into the tendon gallery during construction and methods of repair, see Appendix 5.F.

5.2.2 DESIGN BASES

The design of the containment structure provides the required features as outlined in Criteria 1, 2, 3, 4, 5, 16, 50, 51, 52, 53, 54, 55, 56, 57, 60, 61, Appendix A of 10 CFR Part 50.

5.2.2.1 Bases for Design Loads

The containment is designed for all credible loads and load combinations. These load combinations consist of loads under normal operation, loads during a LOCA, test loads, and loads due to adverse environmental conditions. The following loads are considered:

a. Dead loads

b. Live loads

c. Loads caused by the pressure and temperature transients of a LOCA

d. Thermal loads

e. Earthquake loads

f. Wind and tornado loads

g. Uplift loads due to buoyant forces

h. External pressure loads

i. Prestressing loads

j. Test loads

5.2.2.1.1 Dead Loads

Dead loads consist of the weight of the containment wall, dome, base mat, interior framing and slabs, and all interior structures and equipment. Equipment dead loads are those specified on the

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

5.2.2.1.2 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

5.2.2.1.3 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.

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.

5.2.2.1.4 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.

5.2.2.1.5 Earthquake Loads

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 to ensure that loss of structural functions would not occur. The seismic design spectrum curves are given in Section 5.8.1.1. A vertical component two-thirds of the magnitude of the horizontal component at the ground surface is applied simultaneously as a static coefficient throughout the height of the structure. A dynamic analysis, utilizing the response spectrum technique, is used to obtain the earthquake loads for design.

5.2.2.1.6 Wind and Tornado Loads

Wind loads for the containment are determined on the basis of the ASCE Paper 3269, "Wind Forces on Structures," using the highest wind velocity at the site for a 100 year recurrence period. The ASCE Paper 3269 is used mainly to determine the shape factors. Based upon the site location and the structure classification, the design wind velocity is taken to be 115 mph with gusts up to 140 mph.

The containment has been analyzed for tornado loads (not coincident with a LOCA or earthquake) on the following basis:

a. Differential bursting pressure between the interior and exterior of the containment is assumed to be 3 psi pressure occurring in the three seconds (1 psi/sec), followed by a calm for two seconds and a re-pressurization.

b. Lateral loads on the containment are based on a tornado funnel which is conservatively assumed to have a peripheral tangential velocity of 300 mph and a forward translation of 60 mph. These velocities are added together, resulting in a design basis tornado wind velocity of 360 mph. The applicable portions of the wind design methods described in the ASCE Paper 3269 are used, particularly for the shape factors. The provisions in the paper for gust factors and variations in wind velocity with respect to height are not applied. The wind velocity is assumed to be uniformly distributed over the height of the structure.

c. A tornado-borne missile as defined in Section 5.2.5.1.2.

With the exception of the missile impact area, the allowable stresses necessary to resist the effects of tornadoes are 90 percent of the yield strength of the reinforcing steel and 85 percent of the ultimate strength of the concrete.

A discussion of the probability of tornado occurrence is presented in Section 2.3 of the Millstone Unit 3 FSAR (Reference 5.2-13).

5.2.2.1.7 Hydrostatic Loads

Buoyant forces resulting from the displacement of ground or flood water by the structure are accounted for in the design of the structure.

The following water levels are considered:

Ground water Elevation	(+) 5-0
Flood water Elevation	(+) 18-1

5.2.2.1.8 External Pressure Loads

An external design pressure, equivalent to a barometric pressure rise to 31 inches of mercury after the containment is sealed at 29 inches of mercury, is considered. For this condition, a differential pressure of 2 psi from the exterior to the interior of the containment is assumed and applied as an external pressure on the containment.

This external design pressure is also adequate to permit the containment to be cooled to 50°F from an initial maximum operating temperature of 120°F.

5.2.2.1.9 Prestressing Loads

Where applicable, prestressing forces are considered in the loading combinations. These include the axisymmetric loads of normal compressive forces in the containment wall and dome, and the local effects at the anchorage zones from stressing and shimming the tendons.

5.2.2.1.10 Test Loads

At the completion of construction, the containment and its penetrations are tested at 115 percent of the design pressure. This test pressure is considered in the load combinations to ensure the structural integrity of the containment.

5.2.2.2 Load Combinations

To ensure the structural integrity, both the working stress method and the ultimate strength method are used in the design of the containment for various loading combinations. The containment is examined with respect to strength, the nature, and the amount of cracking, the magnitude of deformation, and the extent of corrosion so as to ensure proper performance. The structure is designed to meet the performance and strength requirements under the following conditions:

a. Prior to prestressing

b. At transfer of prestress

c. Under sustained prestress

d. At design loads

e. At factored loads

All design criteria are in accordance with ACI-318-63 unless stated otherwise herein. Members subject to stresses produced by temperature forces combined with other loads in design load combinations may be proportioned for reinforcing steel stresses 33-1/3% greater than those specified.

5.2.2.2.1 Load Prior to Prestressing

Under this condition, the structure is designed as conventionally reinforced concrete. It is designed for dead loads, live loads (including construction loads), and wind loads. Allowable stresses are in accordance with ACI-318-63.

5.2.2.2.2 Loads at Transfer of Prestress

The containment is checked for prestress loads and the resulting stresses are compared with those allowed by ACI-318-63, with the following exception:

> The membrane compressive stresses are limited to 0.30 f'_{ci} whereas in combination with flexural compressive stresses, the maximum allowable compressive stresses are limited to 0.60 f'_{ci} in accordance with the ACI Code.

For local stress concentrations with nonlinear stress distribution, as predicted by the finite element analysis, a compressive stress of 0.75 f'_c is permitted when reinforcing steel is utilized to distribute and control these localized strains. These high stresses are allowed since they occur only in small localized areas which are confined by material at lower stresses, and would have to be considerably greater than the allowable stresses of the material before significant local plastic yielding takes place.

Membrane and flexural tensile stresses in concrete are permitted provided that they do not jeopardize the integrity of the steel liner plate. Membrane tensile stresses in concrete are permitted to occur during the post-tensioning sequence, but are limited to 1.0 f'_{ci}. When there are flexural tensile stresses but no membrane tensile stresses, the section is designed in accordance with Section 2605(a) and the ACI Code. The stresses in the liner plate due to the combined membrane tensile stress and flexural tensile stresses are limited to 0.5 f_y. The effects of the prestressing sequence are considered.

Design criteria for shear are in accordance with ACI-318-63, Chapter 26, as modified by the equations shown in Section 5.2.2.2.5. For ultimate strength design, a load factor of 1.5 is used.

5.2.2.2.3 Loads Under Sustained Prestress

The design conditions and the allowable stresses for this case are the same as those stated in Section 5.2.2.2.2 except that the allowable tensile stresses in nonprestressed reinforcing are limited to 0.5 f_y and no membrane tensile stresses in concrete are permitted. When membrane stresses are combined with flexural stresses, tension is permitted provided it does not jeopardize the integrity of the liner plate. Where the flexural tensile stresses exist, the section is designed in accordance with Section 2605(b) of the ACI Code.

5.2.2.2.4 At Design Loads

The containment is designed by the working stress method for the following loading combinations:

a. $D + F + L$ Construction case

b. $D + F + L + T_o + E$ Operating case

c. $D + F + L + P + T_i$ Design incident case

d. $D + F + L + T_s + E$ Prolonged shutdown case

e. $D + F + L + 1.15\ P$ Test case

where:

D = dead loads

L = live loads

F = prestressing loads

P = design pressure

T_i = thermal loads due to the loss-of-coolant incident

T_o = thermal loads due to the operating temperature

T_s = thermal loads due to transient wall temperature over a prolonged shutdown (20°F at exterior face, 70°F at center, 50°F at interior face)

E = operating basis earthquake loads (0.09 g)

Long term effects of creep and shrinkage in concrete are considered in all loading combinations.

When determining the value of the modulus of elasticity of concrete to be used in the containment analysis, instantaneous loads such as internal pressure are differentiated from sustained loads such as dead, prestress, and thermal loads. This distinction is necessary to evaluate the effects of creep and shrinkage deformations on the concrete.

The following equation is used to include the effect of creep and shrinkage in the modulus of elasticity:

$$E_{cs} = E_{ci} \times \frac{E_i}{E_s + E_i}$$

where:

E_{cs} - sustained concrete modulus of elasticity

E_{ci} - instantaneous concrete modulus of elasticity

E_s - concrete strain from sustained loads

E_i - concrete strain from instantaneous loads

The modifications described above are used in the analysis of the containment shell for both the design and factored load conditions.

No modification is made to Poisson's ratio for concrete for either sustained or instantaneous loads.

Sufficient prestressing is provided in the cylindrical and dome portions of the containment to eliminate membrane tensile stresses under design load combinations. Flexural tensile cracking is permitted, but is controlled by unprestressed reinforcing steel.

Under the design load combinations, the same performance criteria as specified in Section 5.2.2.2.2 are applied with the following exceptions:

a. When the net membrane compressive stresses are below 100 psi, they are neglected and a cracked section is assumed in the concrete for the computation of unprestressed reinforcing steel for flexural tension. Flexural tensile stress of $0.5f_y$ in unprestressed reinforcing steel are allowed.

b. When the maximum flexural tensile stress does not exceed $6\sqrt{f'_c}$ and the extent of the tension zone is no more than one-third the depth of the section under consideration, unprestressed reinforcing steel is provided to carry the entire tensile forces in the tension block. Otherwise, a cracked section is assumed in the design of the unprestressed reinforcing steel. When the tensile stresses due to the bending moment are additive to the thermal tensile stresses, the allowable tensile stresses in the unprestressed reinforcing steel is $0.5\ f_y$.

c. The problems of shear and diagonal tension in a prestressed concrete structure are considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stresses at design loads, membrane principal tension is not a critical design case. Membrane

principal tension due to combined membrane tension and membrane shear is discussed in Section 5.2.2.2.5.

Flexural principal tension is associated with bending in planes perpendicular to the surface of the shell and shear acting along these planes (radial shear). The present ACI-318-63 provisions of Chapter 26 for shear are adequate for design purposes with the modifications as discussed in Section 5.2.2.2.5.

Crack control in concrete is accomplished through the use of reinforcing steel in accordance with the ACI-ASCE Code Committee Standards. These criteria are based on recommendations of the Prestressed Concrete Institute. The minimum reinforcing provided in terms of the gross concrete cross-sectional area is as follows:

a. 0.25 percent at the tension face for small members.

b. 0.20 percent for medium size members.

c. 0.15 percent for large members.

d. 0.25 percent at the exterior faces of the containment.

5.2.2.2.5 At Factored Loads

Load factors are the coefficients by which loads are multiplied for analysis purposes to ensure that the load-deformation behavior of the structure is elastic with low strain. The load factor approach is used in the design as a means of making a rational evaluation of the isolated loads which must be considered to assure an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads which have the most variation and which most affect the overall safety of the structure. Consequently, a smaller load factor is used for the fixed gravity loads and larger load factors are employed for the LOCA, earthquake, and tornado loads. Justification of the load factors is discussed in Appendix 5.B.

The design of the containment satisfies the following load factors and load combinations:

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.5P + 1.0T_I + 1.0F] \quad \text{Eq (1)}$$

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.25P + 1.0T_I + 1.25H + 1.25E + 1.0F] \quad \text{Eq (2)}$$

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.25H + 1.0R + 1.0F + 1.25E + 1.0T_o] \quad \text{Eq (3)}$$

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.25H + 1.0F + 1.25W + 1.0T_o] \quad \text{Eq (4)}$$

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.0P + 1.0T_I + 1.0H + 1.0E' + 1.0F] \quad \text{Eq (5)}$$

$$C = 1/\phi\ [(1.00 \pm 0.05)\ (D) + 1.0H + 1.0R + 1.0E' + 1.0F + 1.0T_o] \quad \text{Eq (6)}$$

where:

C = required capacity of the structure to resist factored loads

ϕ = capacity reduction factor

D = dead loads of structures and equipment plus any other permanent loads which contribute stresses, such as hydrostatic pressure or soil pressure. In addition, portions of the live loads are added when these include items such as piping, cables, and trays suspended from floors. An allowance is also made for future additional permanent loads.

P = design accident pressure loads

F = final prestress loads

R = forces or pressures on structure due to the rupture of any one pipe

H = forces on structure due to the thermal expansion of pipes

T_I = thermal loads due to the temperature gradient through the walls, based on temperature corresponding to unfactored design accident pressure

T_o = thermal loads due to the normal operating temperature gradient through the walls

E = operating basis earthquake loads

E′ = design basis earthquake loads

W = normal wind or tornado loads (use 1.0 factor for tornado loads)

Equation (1) assures that the containment has the capacity to withstand pressure loadings at least 50 percent greater than those calculated for the postulated LOCA alone.

Equation (2) assures that the containment has the capacity to withstand loadings at least 25 percent greater than those calculated for the postulated LOCA with a coincident operating basis earthquake.

Equation (3) assures that the containment has the capacity to withstand earthquake loadings 25 percent greater than those calculated for the operating basis earthquake coincident with the associated rupture of any one of the attached piping.

Equation (4) assures that the containment has the capacity to withstand wind loadings at least 25 percent greater than the design wind loadings or full tornado loadings.

Equations (5) and (6) assure that the containment has the capacity to withstand either a postulated LOCA or the rupture of any one of the attached piping coincident with the design basis earthquake.

The yield strength of the structure is defined as the upper limit of the proportional stress-strain behavior of the effective load carrying capacities of the structural materials. The stresses from these load combinations, with the load factors given therein, are less than the yield strength of the structure. For steel (both prestressed and unprestressed), the upper limit is taken to be the guaranteed minimum yield strength, as specified in the appropriate ASTM specifications. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond as specified

by ACI-318-63 Code, as well as the 28 day ultimate compressive strength for concrete in flexure (f'_c).

The peak strain in the concrete due to secondary moments, membrane loads, local loads, and thermal loads is limited to 0.003 inch/inch.

The predicted strain in the steel liner plate for all loading combinations does not exceed 0.005 inch/inch.

Principal concrete tension due to the combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3((f'_c)^{1/2})$.

Principal concrete tension due to the combined membrane tension, membrane shear, and flexural tension caused by the bending moments or thermal gradients is limited to $6((f'_c)^{1/2})$. When the principal concrete tension exceeds the limit of $6(f'_c)^{1/2}$, reinforcing steel is provided in the following manner:

a. Thermal flexural tension – Reinforcing steel is provided in accordance with ACI-505. The minimum area of steel provided is 0.25 percent of the gross concrete cross-sectional area in each direction.

b. Bending moment tension – Sufficient reinforcing steel is provided to resist the moment on the basis of the cracked section theory. The exception is when the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel is 0.5 f_y.

Shear stress limits and reinforcing for radial shear are in accordance with Chapter 26 of ACI-318 with the following exceptions:

a. Formula 26-12 of the Code is replaced by:

$$V_{ci} = Kb'd(f'_c)^{1/2} + M_{cr}(V/M') + V_i$$

where:

$K = 1.75 - (0.036/np') + 4.0\ np'$ but not less than 0.6 for $p' > 0.003$.

For $p' < 0.003$, the value of K is zero.

$$n = (505/(f'_c)^{1/2})$$

$p' = (A'\ s/bd)$

$M_{cr} = (1/y)(6((f'_c)^{1/2}) + f_{pe} + f_n + f_i)$

and

f_{pe} = compressive stresses in concrete due to prestress only applied normal to the cross-section after all losses (including the stresses induced by any secondary moments), at the extreme fiber of the section at which tensile stresses are caused by live loads. (f_{pe} is negative for tensile stresses; positive for compressive stresses)

f_n = stresses due to applied axial loads (f_n is negative for tensile stresses; positive for compressive stresses)

f_i = stresses due to the initial loads at the extreme fiber of a section at which tensile stresses are caused by the applied loads (including the stresses induced by any secondary moments, f_i is negative for tensile stresses; positive for compressive stresses)

V = shear at the section under consideration due to the applied loads

M′ = moment at the critical section under consideration, in the direction of decreasing moment, due to the applied loads. The critical section is considered to be at a distance of 1/2 (d) from the support, where d is the effective depth of the concrete section.

V_i = shear due to the initial loads (positive when the initial shear is in the same direction as the shear due to the applied loads).

The lower limit given by the ACI-318-63 on V_{ci} as $1.7b'd(f'_c)^{1/2}$ is not applied.

b. Formula 26-13 of the Code is replaced by:

$$V_{cw} = 3.5b'd(f'_c)^{1/2} [1 + (fpe + fn)/(3.5(f'_c)^{1/2})]^{1/2} + V_p \qquad \text{Eq (8)}$$

where V_p = radial shear component of effective prestress due to the curvature of tendon at the section considered, and the term, f_n, is as defined in Equation (7). All other notations are in accordance with Chapter 26 of ACI-318-63.

Equation (7) is based on test and work done by Dr. A. H. Mattock of the University of Washington. Equation (8) is based on the commentary for the Proposed Redraft of Section 2610 of ACI-318 by Dr. A. H. Mattock, dated December 1962.

When these equations indicate that the allowed shear stress in concrete is zero, radial shear ties are provided to resist all the calculated shear.

5.2.2.2.6 Prestress Losses

In accordance with ACI-318-63, the design provides for prestress losses caused by the following effects:

a. Seating of anchorage

b. Elastic shortening of concrete

c. Creep of concrete

d. Shrinkage of concrete

e. Stress relaxation of steel

f. Frictional losses due to intended or unintended curvature in the tendon

All of these losses have been predicted with a reasonable degree of accuracy.

In the present application, the environment of the prestressing system and concrete is not appreciably different from that found in conventional prestressed concrete structures such as bridges and buildings. Data from research and practical experience with this type construction have made it possible to evaluate conservatively the allowances for all the prestress loss due to various causes.

5.2.2.2.7 Capacity Reduction Factors

The capacity of all load-carrying structural elements is reduced by a capacity reduction factor (ϕ) as given below. The justification for these numerical values is given in Appendix 5.C. These factors provide for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision, while individually within required tolerances and the limits of good practice, may occasionally combine to result in undercapacity.

The capacity reduction factors used are as follows:

$\phi = 0.90$ for concrete in flexure

$\phi = 0.85$ for tension, shear, bond, and anchorage in concrete

$\phi = 0.75$ for spirally reinforced concrete compression members

$\phi = 0.70$ for tied compression members

$\phi = 0.90$ for fabricated structural steel

$\phi = 0.90$ for mild reinforcing steel

$\phi = 0.95$ for prestressed tendons in direct tension

5.2.2.3 Structural Analysis

The containment is analyzed by a finite element computer program for individual loading cases of dead loads, live loads, winds and tornadoes, temperatures, pressures, and prestresses. A dynamic analysis for seismic loads is performed. The results of the various loadings are superimposed according to the design and factored equations as stipulated in Sections 5.2.2.2.4 and 5.2.2.2.5.

The ACI-318-63 design methods and allowable stresses are used for the concrete as well as for the prestressed and unprestressed reinforcing steel, except as noted herein.

5.2.2.3.1 Critical Areas of Analysis

The main areas for design analysis are:

Restraints at the top and bottom of the cylinder

Restraints at the edge of the dome

Ring girder

Behavior of the base slab relative to an elastic foundation

Transient temperature gradients in the steel liner plate and concrete

Large penetrations

Tendon anchorage zones

Concentrated loads

Seismic loads

5.2.2.3.2 Analytical Techniques

The analysis of the containment consists of two parts: the axisymmetric analysis and the nonaxisymmetric analysis. The axisymmetric analysis is performed by utilizing a finite element computer program for the individual loading cases of dead loads, live loads, temperatures, pressures, and prestresses. The axisymmetric finite element representation of the containment assumes that the structure is axisymmetric and does not take into account the buttresses, penetrations, brackets, and anchors. These items, together with the lateral loads due to earthquakes, winds, tornadoes, and various concentrated loads, are considered in the nonaxisymmetric analysis.

5.2.2.3.2.1 Axisymmetric Analysis

The axisymmetric analysis is performed by using a finite element computer program developed by Dr. E. L. Wilson under the sponsorship of the National Science Foundation. Such a method of analysis is normally used for thick-walled structures where a conventional shell analysis may yield less accurate results. Good correlation has been demonstrated between the finite element analysis method and the test results for thick wall model vessels.

The finite element technique is a general method of structural analysis in which a continuous structure is replaced by a system of elements connected at a finite number of nodal points. In applying this method to an axisymmetric solid structure, such as a containment shell, the continuous structure is replaced by a system of rings of quadrilateral cross-sections which are interconnected at the circumferential joints. Based on energy principles, a set of force equilibrium equations is formulated in which the radial and axial displacements at the circumferential joints

are the unknowns. The results obtained by solving these equations are the deformations of the structure under the given loading conditions.

The finite element mesh used to describe the structure is shown in Figures 5.2–16 through 5.2–18. The upper and lower portions of the containment are analyzed separately to permit use of a greater number of elements for those areas of the structure which are of major concern, e.g., the ring girder area and the haunch connecting the cylindrical shell to the base slab. The finite element mesh of the base slab is extended into the foundation to give consideration to the elastic nature of the foundation material and its effect upon the behavior of the base slab. The tendon access gallery is designed as a separate structure.

The finite element analysis produces stresses due to axisymmetric loads. The stresses from the earthquake loads, as well as wind and tornado loads, are obtained by the nonaxisymmetric analysis and then superimposed on the stresses obtained from the finite element analysis. The final summation of all the stresses is used in the design of the base slab, shell, and dome. The liner plate is considered as an integral part of the structure and is included in the finite element mesh of the containment.

Thermal loads result from temperature differentials across the cylindrical wall. The design temperature gradients for the containment are shown on Figure 5.2–3. In the finite element analysis, when temperatures are specified at every nodal point, thermal stresses are obtained at the center of each element.

The computer program used in the finite element analysis is capable of handling the following inputs:

Eight different materials

Nonlinear stress-strain curves for each material

Axisymmetric loadings of any shape

The program outputs are:

C

ϕ

D

P

An auxiliary computer program plots the isostress curves based on the outputs of the aforementioned program.

5.2.2.3.2.2 Nonaxisymmetric Analysis

The nonaxisymmetric configurations and loadings require various methods of analysis. The descriptions of the methods used, as applied to the different parts of the containment, are given in the following sections.

5.2.2.3.3 Buttress and Tendon Anchorage Zone Analyses

The containment has three buttresses. At each buttress, two out of any group of three hoop tendons are spliced by anchoring on the opposite faces of the buttress, and the third tendon is continuous through the buttress.

Between the opposite anchorages in the buttress, the compressive forces exerted by the spliced tendons are twice as great as elsewhere on the shell. This value, combined with the effect of the tendon which is not spliced, is 1.5 times the prestressing force acting outside of the buttresses. The thickness at the buttress is about 1.5 times that of the wall. Thus, the hoop stresses as well as the hoop strains and radial displacements can be considered as being nearly constant all around the structure.

The vertical stresses and strains, caused by the vertical post-tensioning, become constant a short distance from the anchorages because of the stiffness of the cylindrical shell. The stresses and strains remain nearly axisymmetric despite the presence of the buttresses. The effect of the buttresses on the overall vessel behavior is negligible, whether the structure is under dead loads or prestress loads.

The analysis of the anchorage zone stresses at the buttresses is the most critical of the various types of anchorage areas on the shell. The local stress distribution in the immediate vicinity of the bearing plates is investigated using the following procedures:

a. The Guyon’s Equivalent Prism Method: This method is based on the experimental photoelastic results as well as the equilibrium considerations of homogeneous and continuous media. It also considers the relative bearing plate dimensions of the anchorages. (Reference 5.2-1)

b. The experimental test data presented by S. J. Taylor of the March 1967 London Conference of the Institute of Civil Engineers (Group H, Paper 49): these data are used to evaluate the effect of the biaxial stresses at the anchorages, including the effects of the trumpet welded to the bearing plate.

c. F. Leonhardt’s formula for determining the bursting force in the anchorage zone of a prestressed concrete member (Reference 5.2-2).

d. The three-dimensional stress distribution in the anchorage zones is analyzed in sufficient detail to permit a rational evaluation of the stress concentrations. A conical wedge segment is used as the basic design element and the radial splitting tension is determined as a tangential distribution function. The summation of the splitting stresses through the entire volume of the leading zone establishes the

value of the splitting force. This force is a function of the ratio of the base dimensions of the cone; i.e., a/b, and the height, h, of the cone. Several different combinations of the values are analyzed and the most critical values selected.

Note: For description of water intrusion into the tendon gallery from around the bearing plates during construction and methods of repair, see Appendix 5.F.

Transient thermal gradients are used in the analysis and the resulting stresses are superimposed on the bursting stresses obtained from the triaxial stress analysis.

The possibility of a torsional load being applied to the anchors because the tendon wires are twisted was considered. It was determined that no such load occurs since the ram pull rod is free to rotate during the stressing operation.

The design of the anchorage zone reinforcing is based on the results of these analyses, and the following considerations:

a. Bechtel Topical Report BC-Top-7, “Full Scale Buttress Test for Prestressed Nuclear Containment Structures.”

b. Design of similar anchorages.

c. Rebar spacing determined to allow ease in placing of sound concrete behind the anchorage bearing plates.

d. Review of the reinforcing details from earlier projects undertaken by the consulting firm of T. Y. Lin, Kulka, Yang and Associates.

5.2.2.3.4 Stresses Near Large Openings

The analytical solutions for determining the state of stresses in the vicinity of large openings are based on the procedure described in the Welding Research Council (WRC) Bulletin Number 102, entitled “State of Stress in a Circular Cylindrical Shell with a Circular Hole.” (Reference 5.2-3)

The analysis of the containment, as a whole, was first carried out without considering any openings. This analysis has been done by using the finite element program.

The containment, considering openings, is then analyzed as follows:

a. Formulate differential equations for the shell in a complex variable form with the center of the hole as the origin.

b. Solve the differential equations.

c. Evaluate parameters in the solution.

d. Formulate the boundary conditions based on the stresses obtained from the shell analysis without the hole.

e. Calculate membrane forces, moments, and shears around and at the edge of the opening.

f. Increase and reinforce the wall thickness around the opening to carry the higher forces, moments, and shears. The effects of stress concentration due to the thickening of the wall are considered.

g. Evaluate the effects of prestressing that are not handled in the WRC Bulletin Number 102.

h. Check the design to ensure that the strength of the reinforcement provided is adequate to replace the strength removed by the opening. This check is done to assure a good degree of compatibility between the general containment shell and the areas around the opening.

To analyze the thermal stresses around the openings the following procedure is used. At the edge of the opening a uniformly distributed moment equal but opposite to the moment existing on the rest of the shell is applied. The opening is then analyzed using the methods of the WRC Bulletin Number 102. The effects are superimposed on the stresses calculated by the finite element method.

The membrane stresses resulting from the seismic loads around the openings are modified by appropriate stress concentration factors.

Typical details of reinforcing around the equipment hatch and personnel lock are shown on Figures 5.2–4 and 5.2–5.

5.2.2.3.5 Seismic Analysis

The seismic loads on the containment are determined from a dynamic analysis of the structure. The method of analysis is presented in Section 5.8.3.

5.2.2.3.6 Wind and Tornado Analyses

The design wind loads on the containment are a function of the kinetic energy per volume of the moving air mass. The product of one-half of the air density and the square of the resultant design velocity results in a pressure corresponding to the design wind.

Determination of the design wind pressure on the containment is in accordance with the ASCE Paper 3269, “Wind Forces on Structures.”

The pressure corresponding to the standard air at 0.07651 pcf at 15°C and 760 mm of mercury in terms of the velocity at the appropriate height zone is given by:

$$q = 0.002558V^2$$

Similarly, the design pressure, including the effect of the shape coefficient, C_d, is given by:

$$p = q \times C_d = 0.002558V^2 C_d$$

The design wind velocity for the containment, with the enclosure building attached to it, is taken to be 115 mph with gusts up to 140 mph. The shape coefficient for the enclosure building is found to be:

$$C_d = 1.30$$

The design wind pressure based on the above shape coefficient and the wind velocity of 140 mph is:

$$P = 65.2 \text{ psf}$$

The design pressure is assumed to be constant throughout the height of the enclosure building and is being resisted entirely by the containment.

The design tornado loads on the containment are analyzed on the following basis.

a. Tornado loads are not coincident with an accident or earthquake.

b. Differential bursting pressure between the inside and outside of the containment is assumed to be 3 psi pressure occurring in three seconds (1 psi/second), followed by a calm for two seconds and a repressurization outside.

c. The wind force on the containment is considered as a uniform static load caused by 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 wind velocity of 360 mph. The applicable portions on wind design methods described in the ASCE Paper 3269 are used, particularly for shape factors. The provisions in the ASCE Paper 3269 for gust factors and variation of wind velocity with height do not apply. The wind velocity is assumed to be uniformly distributed over the height of the structure.

d. The metal sidings (but not the supporting frame) of the enclosure are designed to be blown away by the differential bursting pressure, thereby, subjecting the containment to the lateral forces resulting from the 360 mph wind.

e. Tornado-borne missiles is defined in Section 5.2.5.1.2.

f. Except for local crushing at the missile impact area, the allowable stresses available to resists the effects of tornadoes are 90 percent of the yield strength of the reinforcing steel and 85 percent of the ultimate strength of the concrete.

Based on an aspect ratio (h/d) of 1.12 and a surface smoothness (t/d) of 1.8 percent, the shape factor for the containment in the tornado analysis is found to be:

$C_d = 0.70$

where:

h = containment height above ground

t = projection of buttress

d = maximum outside diameter of the containment

The maximum pressure is found to be 274 psf (negative) and occurs at 90 degrees to the direction of wind. Since the analysis of the containment is limited to uniform pressure loading, 80 percent of the maximum pressure is assumed to act uniformly across the horizontal projection of the containment. This is believed to be a conservative approach.

5.2.2.3.7 Results of Structural Analyses

Scaled plots for the results of structural analyses on moments, shears, normal forces, and deflections of all loading conditions are shown in Figures 5.2–19 through 5.2–26. Corresponding stresses at various locations on the containment are shown in Tables 5.2-1 through 5.2-9.

Deformations are consistent with the elastic strains; i.e., design is not governed by deformation. The deformations will not affect the continued functional capability of the containment structure or any other Class I structure which might interact with it. Measurable relative displacements are expected between the containment and other components and structures due to the loads imposed. Those displacements have been accounted for by providing expansion joints or, in the case of most pipes, imposing those deflections on the interfacing components.

5.2.3 STEEL LINER PLATE AND PENETRATION SLEEVES

The containment steel liner plate and penetration sleeves are designed to serve as the primary leakage barrier for the containment. Typical details of the liner are shown on Figures 5.2–6 and 5.2–7. The design considered the composite action of the liner and the concrete structure and includes the transient effects of the liner due to temperature changes during construction, normal operation, and the LOCA. The changes in strain to be experienced by the liner due to these effects, as well as those at the pressure testing of the containment, are considered.

The stability of the liner is achieved by anchoring it to the concrete structure. At all penetration sleeves, the liner is thickened to reduce stress concentrations, based on the 1968 ASME Code, Section III, for Class B vessels. The thickened portions of the liner are then anchored to the concrete. All weldments associated with the penetration sleeves are designed to resist the full applied loads. Typical details of the penetration sleeves are shown on Figure 5.2–8.

All components of the liner which must resist the full design pressure, such as penetration sleeves, personnel lock, and equipment hatch, are designed to meet the requirements of paragraph N-1211, of Section III, Nuclear Vessels, 1968 Edition through the summer 1969 addenda of the ASME Code, except the external bolting attachments to the equipment hatch which were designed to meet the requirements of Section III, Subsection NE, 1986 Edition.

In isolated areas, the liner has an initial inward curvature due to fabrication and erection tolerances and inaccuracies. The anchors are designed to resist the forces and moments induced when a section of liner between anchors has initial inward curvature while the adjacent panels have no such imperfections. As a result, inward deformation of the liner between anchors may occur under both operating and accident conditions. The liner and the anchors are designed with sufficient ductility to undergo displacement to relieve the loads without rupturing under these conditions.

With the exception of the containment spray piping supports, an insert plate is provided to transmit the load through the liner at each location where a load is transferred to the walls, slabs, or dome of the containment. The insert plate is anchored to the concrete by appropriate anchors and shear connections. Examples of such insert plates are the polar crane brackets and the floor beam brackets at the operating deck. Typical details of these brackets are shown in Figure 5.2–9.

5.2.3.1 Construction Materials

Materials used for the construction of the steel liner plate and penetration sleeves are listed in Section 5.2.1.

5.2.3.2 Design Criteria

The design criteria applied to the containment steel liner plate to meet the specified leak rate under the operating and accident conditions are as follows:

a. The liner is protected from damage by potential missiles generated from a LOCA and main steam pipe break.

b. The liner strains are limited to those values that have been shown by past experience to result in leaktight pressure vessels and piping.

c. The liner is prevented from developing distortions sufficient to impair leak tightness.

5.2.3.3 Design Loads

The liner is designed with the capability to resist, without rupture, the compressive stresses due to the following loads:

a. Construction loads, particularly those which are applied to the liner before the concrete is placed.

b. Local thermal loads at hot process penetrations.

c. Thermal gradients.

d. Thermal shock loads due to cold sprays.

e. Local loads, such as structural supports, pipe supports, and restraints, etc.

f. Prestress loads.

g. Creep and shrinkage loads.

The fatigue analysis of the liner plate is made on the basis of the following considerations:

a. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 for the plant life of 40 years. (Daily temperature variations do not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner, and therefore, are not considered.)

b. Thermal cycling due to the containment interior temperature variations during heatup and cooldown of the reactor system. The number of cycles for this loading is assumed to be 500.

c. Thermal cycling due to the LOCA is assumed to be one cycle.

5.2.3.4 Permissible Stresses and Strains

The basis for establishing the allowable liner strains is the 1968 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4.

The thermal stresses in the liner fall into the categories as considered in Article 4 of Section III. The allowable stresses in Figure N-415 (A) of the Code are for alternating stress intensity for carbon steels, with the temperature not exceeding 700°F.

To fulfill the criteria set forth in the 1968 ASME Code, Paragraph N-412 (m) 2, the liner is restrained against significant distortion of an angle grid anchor system. Materials are expected to be exposed to a maximum temperature of approximately 289°F under a LOCA condition which is well below the 700°F limit. The liner design also satisfies the criteria for limiting the strains on the basis of fatigue consideration. Figure N-415 (A) of Paragraph N-412 (n) of the ASME Code and its appropriate limitations are used as the bases for establishing the allowable strains for the liner.

Since the graph in Figure N-415 (A) does not extend below 10 cycles, 10 cycles were used for the LOCA conditions instead of one cycle.

The allowable strain from Figure N-415 (A), based on 10 significant thermal cycles on the LOCA conditions, would be approximately two percent. However, the maximum allowable membrane

tensile or compressive strain is conservatively set at 0.5 percent. The maximum predicted strain in the liner during the LOCA conditions is found to be 0.25 percent in compression.

No maximum allowable compressive strain is set for the test condition because it is expected the value will be less than that experienced under the LOCA conditions. The maximum allowable tensile strain is 0.2 percent for the test conditions; the predicted value is nearly zero.

5.2.3.5 Design of Liner Plate Anchorage

The anchors are designed to preclude failure when subjected to the most severe loads or deformations. They are designed so that a missing or defective anchor will not jeopardize the overall integrity of the liner and anchorage system.

The following factors are considered in the design of the anchorage system:

a. The initial inward curvature of the liner between anchors due to fabrication and erection tolerances and inaccuracies.

b. Variations of anchor spacing.

c. Misalignment of liner seams.

d. Variations of plate thickness.

e. Variations of the yield strength of the liner plate materials.

f. Variations of the Poisson’s ratio for the liner plate materials.

g. Variations of the anchor stiffness.

The anchorage system satisfies the following conditions:

a. The anchors have sufficient strength and ductility so their energy absorbing capability is sufficient to restrain the maximum force and displacement resulting from the condition where a panel with an initial outward curvature is adjacent to a panel with an inward curvature.

b. The anchors have sufficient strength to resist the bending moment that results when a panel with an initial outward curvature is adjacent to a panel with an inward curvature.

c. The anchors have sufficient strength to resist the radial pullout forces.

The proprietary topical report, “Consumer Power Company Palisades Nuclear Power Plant Containment Building Liner Plant Design Report B-TOP-1,” constitutes the basic design approach used in Millstone Unit 2.

The following are the minor differences between the Millstone Unit 2 design and the one presented in the topical report.

a. The welding of the stiffeners is 3/16-6 x 12 rather than 3/16-4 x 12. This does not invalidate the analysis, because the spring constants used in the analysis are similar.

b. The stiffeners on the thickened plates are not welded with a double fillet weld as stated in the topical report. The 3/16-6 x 12 welding is used on all stiffeners. The topical report indicates that additional welding is not required to resist the loads.

c. The 1/4 inch liner material is ASTM A-285 Grade A. This plate has a specified yield strength of 24,000 psi which is lower than the values used in Topical Report, B-TOP-1. This would only tend to decrease the loads on the anchors, as stated in Section 3.4 of the report.

d. A self supporting dome is used on Millstone Unit 2. It is stiffened in two directions instead of one as stated in Section 2.2.2 of the report. Details of the dome are shown on Figure 5.2–6.

5.2.3.6 Design of Weldments

Paragraphs of UW-8 to UW-19, Subsection B, Section VIII of the 1968 ASME Code are used as a guide in the design of the weldment.

Inspection and testing of liner plate weldments during and after erection are discussed in Section 5.9.3.5.3.

Quality control of field welding electrodes are presented in Section 5.9.3.5.4.

Quality control procedures for field welding and nondestructive examination are defined in Section 5.9.4.

5.2.4 INTERIOR STRUCTURES

5.2.4.1 General

Design of the containment interior structures evolves from four basic systems: reactor coolant, main steam, engineered safety features and fuel handling.

The structures which house or support the basic systems are designed to sustain the loading cases as outlined in Sections 5.2.4.4.1 and 5.2.4.4.2.

The design bases are as follows:

a. The structures are capable of sustaining all operating loads, seismic loads, and thermal deformations.

b. Loads and deformations resulting from an LOCA and the associated effects on any of the basic systems are sustained and restricted so that propagation of the failure to any other system is prevented. In addition, a single failure in one of the cooling pipes of the nuclear steam supply system is restricted such that propagation of the failure to the other cooling piping is prevented.

Structural details for the supports for major Class I equipment such as the reactor vessel and steam generators, are shown in Figure 5.2–11 through 5.2–13. Typical details for the primary and secondary shield walls are shown on Figure 5.2–14.

5.2.4.2 Construction Materials

The following materials are used in the construction of the containment interior structures:

Concrete:

Primary shield walls	5,000 psi
Steam generator supports	5,000 psi
Secondary shield walls	4,000 psi
Refueling pool walls	4,000 psi
Reinforcing steel	ASTM A-615, Grade 60
Carbon steel plates	ASTM A-302, Grade B, A-441, and A-569
Stainless steel plates	ASTM A-240, Type 304
Stainless steel tubes	ASTM A-358, Type 304
Structural and miscellaneous steel	ASTM A-36 and A-441
Anchor bolts	ASTM A-307, A-325, and A-490

5.2.4.3 Design Loads

The following loads are considered in the design of the interior structures:

a. Dead loads

b. Live loads

c. Earthquake loads

d. Loss-of-coolant accident (LOCA) loads

e. Pipe rupture loads

5.2.4.3.1 Dead Loads

Dead loads consist of the weight of the concrete, structural steel, equipment, major piping, and electrical conductors. Equipment dead loads are those specified on the drawings supplied by the manufacturers of the various pieces of equipment. Major equipment supported by the interior structures are reactor vessel, steam generators, pressurizers, and the reactor coolant pumps.

5.2.4.3.2 Live Loads

Live loads for the design of the interior structures are:

Floor and equipment area	250 psf
Containment laydown area	1,000 psf

Equipment live loads are those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

5.2.4.3.3 Earthquake Loads

Earthquake loads are predicated upon an operating base earthquake at the site having a horizontal ground acceleration of 0.09 g and a vertical acceleration of 0.06 g. In addition, a design basis earthquake having a ground acceleration of 0.17 g and a vertical acceleration of 0.11 g is used to check the design to ensure that there will be no loss of function.

Seismic response spectrum curves are given in Section 5.8.1.1, for both horizontal and vertical ground motions.

5.2.4.3.4 Loss-of-Coolant-Accident (LOCA) Loads

The maximum forces which result from a pipe rupture are based on the following:

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. 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 secondary shielding room, as described in Section 5.2.4.4.4.

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 Section 5.2.5.1.1.

Leak-before-break (LBB) analyses of the reactor coolant system (RCS) main coolant loops, the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping allows the exclusion of the dynamic effects associated with pipe ruptures in the above piping segments from the design basis. Dynamic effects of pipe rupture include the effects of pipe whipping, subcompartment pressurization and discharging fluids.

5.2.4.4 Design Criteria

In establishing the structural design criteria for the interior structures, consideration was given to a structure which would withstand the differential pressure within the cavities in the event of an accident, and to minimizing the effects of the pipe rupture force by the use of supports and restraints.

The ACI-318-63, “Building Code Requirements for Reinforced Concrete,” and AISC “Manual of Steel Construction,” 6th Edition, are used as design criteria for reinforced concrete and structural steel, respectively, except as noted in the following:

Localized concrete yielding is permitted when it can be demonstrated that the yield capacity of the component is not affected, and that this small localized yielding does not generate missiles which could damage the structure. Full recognition is given to the time increments associated with these postulated failure conditions, and yield capacities are appropriately increased when a transient analysis demonstrates that the rapid strain rate justifies this approach. The walls are also designed to provide adequate protection for potential missile generation which could damage the containment liner.

The yielding of concrete discussed above concerns itself with localized stress concentration under the load associated with a LOCA. The load due to pipe whipping followed a pipe break in the reactor coolant system is considered a concentrated load and is analyzed to ascertain that its local effect on the concrete surface will not result in changes in the member capacity. These high local stresses are not identified because of simplifications made in the design analysis. These high stresses are allowed because they occur in a very small percentage of the cross-section, are confined by material at lower stress, and would have to be considerably greater than the values allowed before significant local plastic yield would result.

The outline of the reactor coolant system and the primary supporting structure are presented in Figures 5.2–35 and 5.2–36. Localized concrete yielding is permitted at the point of impact resulting from any high energy pipe rupture. The high concrete stress is limited to the area of impact and is confined by concrete in compression at lower stresses, and is estimated conservatively to be not more than one inch in thickness within the secondary wall boundary. Since the area of high concrete stresses is in compression, no significant concrete cracking is expected. Pipe rupture is postulated at locations which result in the most critical conditions for designing the structures. A typical case is shown in Figures 5.2–35 and 5.2–36. Under this load, combined with other loads in the design load combination, a strip of concrete wall of width L as shown in Figure 5.2–36 was analyzed. A continuous span of secondary shield wall, supported at the buttresses and fixed at the refueling pool walls was assumed. Reinforcing steel was proportioned according to the results obtained.

The strength of the structures at working stress and overall yielding is compared to various loading combinations to ensure safety. The structures are designed to meet the performance and strength requirements under the following conditions:

a. At design loads.

b. At factored loads.

5.2.4.4.1 At Design Loads

This loading is the basic “working stress” design. The structure is designed for the following loading cases:

a. D + L

b. D + L + H

c. D + L + H + E

where:

D = dead loads

L = live loads

H = thermal loads under operating conditions

E = operating basis earthquake

5.2.4.4.2 At Factored Loads

The structure is checked for the factored loads and load combinations as follows:

a. $C = 1/\phi\ (1.25D + 1.25L + 1.0\ R + 1.25E)$

b. $C = 1/\phi\ (1.25D + 1.25L + 1.25\ H + 1.25E)$

c. $C = 1/\phi\ (1.0D + 1.0L + 1.0\ R + 1.0T_I + 1.0E')$

d. $C = 1/\phi\ (1.0D + 1.0L + 1.0\ H + 1.0E')$

e. $C = 1/\phi\ (1.0D + 1.0L + 1.0\ P + 1.OT_I + 1.0E')$

where:

C = required capacity of the structure to resist factored loads

ϕ = capacity reduction factor

D = dead loads

L = live loads

H = thermal loads under operating conditions

P = differential pressure due to a double-ended pipe rupture

T_I = loads due to thermal gradient produced by a double-ended pipe break

E = operating basis earthquake loads

E′ = design basis earthquake loads

R = loads due to pipe rupture (includes both jet and pressure differential forces)

5.2.4.4.3 Thermal Gradients

The thermal gradients in the interior structures are maintained at a low level so they have very small structural effects on the concrete walls. Nevertheless, these effects are considered in the design.

a. Primary shield walls.

Based on the reactor vessel and cooling piping heat loads and 100 percent of cooling air, the temperature within the primary shield walls is approximately 7°F higher than the air temperature, due to nuclear heating. Cooling air from the CAR coolers is supplied to the bottom of the reactor cavity, primary shield walls, reactor vessel supports and the ex-core detectors. Calculations, which support the permanent reactor cavity seal project and neutron shield modification, have estimated that the cavity cooling air limits the maximum temperature of the primary shield walls to less than 150°F.

b. Secondary shield walls, steam generator pedestals, and refueling pool walls.

The ventilation system will eliminate the temperature gradients across the secondary shield walls, the refueling pool walls, and the operating floor. The ambient temperature inside the containment varies between 105°F and 120°F under operating conditions.

5.2.4.4.4 Differential Pressures

The differential pressure-time curves across the primary shield wall and the annular space between the reactor coolant pipe and the pipe sleeve extending through the reactor cavity wall (biological shield) have been excluded from the design basis, as all high energy lines within the reactor vessel cavity have supporting NRC approved leak before break analyses. The Bechtel COPRA computer program (NS731-NE576) is used to calculate the differential pressure-time curves across the secondary shield walls. The calculations are based on conservation of mass, momentum, and energy.

The ensuing flow from the compartment follows the orifice flow relations with the entrance and friction losses included in the flow coefficient for each case.

The following is a summary for the input parameters used for the computer analysis of steam generator compartment pressurization:

Containment free volume (cubic feet)	1.899×10^6
Initial containment temperature (°F)	120
Initial containment pressure (psia)	14.7
Initial containment humidity (%	30
East Steam Generator Compartment	
Volume (cubic feet)	62,000
Vent Area (square feet)	2,800
West Steam Generator Compartment	
Volume (cubic feet)	54,100
Vent Area (square feet)	2,620

Neither steam line nor feedwater line breaks were analyzed in the steam generator compartments. Therefore a double-ended guillotine break in the hot leg, with a restricted break area of 10.78 square feet, is considered, which provides the bounding rate of energy and mass releases.

The Pressurizer Compartment was originally designed for a 22 psi differential pressure based on its contiguous boundary with the east steam generator cavity. However, due to a modified, semi-open blockhouse design that was implemented to support the replacement of the original pressurizer and the adoption of Leak-Before-Break methodology to eliminate pressurization effects due to the primary coolant piping, the design differential pressure for this structure is currently based on a double-ended feedwater break. Utilizing a break size comparison between the less energetic feedwater break case (2.97square feet) and the previously analyzed hot leg break case (10.78 square feet) provides a bounding estimate for the differential pressure effects from the feedwater break.

Results of compartment pressurization analyses are listed below. Conservative estimates of compartment geometries, combined with the conservative calculational model assumed in COPRA, result in differential pressures across compartment walls that are inherently larger than can actually occur. Consequently, no additional safety margins are necessary.

COMPARTMENT	PRESSURES	
	DIFFERENTIAL (PSI)	DESIGN DIFFERENTIAL (PSI)
Steam Generator Cavity (East)	8.7	22
Steam Generator Cavity (West)	9.95	22
Pressurizer	2.4	3

5.2.5 SPECIFIC DESIGN TOPICS

5.2.5.1 Missile Protection

5.2.5.1.1 Design Criteria Inside the Containment

High pressure reactor coolant system components, which could be the source of missiles, are screened either by the concrete shield walls enclosing the reactor coolant loops, by the concrete operating floors, or by special missile shields to block the passage of any missiles and prevent them from striking the wall and dome of the containment. All potential missile sources are oriented so that the missiles are intercepted by the shields or structures provided. A shield is provided over the control rod driving mechanism to block the passage of any missiles generated as a result of a postulated fracture of the nozzle.

Missile protection inside the containment is provided to comply with the following criteria:

a. The containment liner is protected from a loss of function due to damage by missiles as might be generated in a LOCA for piping break sizes up to and including the double-ended severance of the largest reactor coolant pipe.

b. The engineered safety features, systems, and components required to maintain the integrity of the containment are protected from a loss of function due to damage by the following missiles:

    1. Valve stems

    2. Valve bonnets

    3. Instrument thimbles

    4. Various types and sizes of nuts and bolts

Table 5.2-10 lists the spectrum of potential missiles from inside the containment, their kinetic energies, weights and leading cross-section configurations:

The following methods are used to implement the missile protection criteria:

a. Components of the reactor coolant system are examined to identify and classify missiles according to the sizes, shapes, and kinetic energy so as to analyze their effects.

b. Missile velocities are calculated considering both the fluid and mechanical driving forces that exist during missile generation.

c. The reactor coolant system is surrounded by reinforced concrete and steel structures which are designed to withstand the forces associated with the double-ended rupture of a reactor coolant pipe and to stop the missiles.

The structural design of the missile shields takes into account both the static and the impact loads and is based upon the state-of-the-art of missile penetration data.

Leak Before Break (LBB) analyses of the reactor coolant system (RCS) main coolant loops, the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping allows the exclusion of the dynamic effects associated with pipe ruptures in the above piping segments from the design basis. Dynamic effects of pipe rupture include the effects of pipe whipping, subcompartment pressurization and discharging fluids.

Certain postulated incidents such as the massive, rapid failure of the reactor vessel, steam generators, pressurizers, and the main coolant pump flywheels and casings are considered incredible because of the material characteristics, inspections, quality control during fabrication, and the conservatism in design as applied to the particular components. These same factors also apply to the stems and bonnets of both motor-operated shutdown cooling suction valves located inside containment. Both valves have been subjected to detailed structural and functional analysis, and the stem and bonnets have been found to not be credible missiles.

In establishing the credibility of any missile source, the use of redundancy of load paths, such that no single failure could lead to a missile ejection, has been credited as the basis for adequate protection from missile generation. In the case of missiles originating from valves, the valve stem is considered a potential missile only when it is not back seated and where no air or motor operator exists that would interfere with the ejection of the valve stem. Valve bonnets are not considered as a source of missiles when the flanges and bolts are designed in accordance with ASME Section III and the torque is controlled during the tightening process by approved plant procedures. Valve bonnet missiles are also not considered credible when the bonnet is welded to the valve body or in cases where the bonnet is integral with the body of the valve. While the failure of single bolts and nuts is considered credible, they are considered a negligible concern due to the small amount of stored elastic energy that they process.

5.2.5.1.2 Design Criteria Outside the Containment

Missile protection outside the containment is provided to comply with the following requirements:

a. The containment steel liner plate and penetrations are protected from the loss of function due to damage by tornado borne missiles.

b. All engineered safety features piping which penetrates the containment, and which is required to maintain the containment integrity, is protected from a loss of function due to tornado borne missiles.

c. All components required to maintain the containment integrity, or whose failure would result in the uncontrolled release of radioactivity, are protected from a loss of function due to damage by tornado borne missiles.

Protection is provided for the following three types of tornado borne missiles:

a. A fir plank, 4 inches by 12 inches by 12 feet., weighing 105 pounds and traveling end-on at a speed of 250 mph.

b. A passenger auto (4,000 lb.) impact velocity of 50 mph not more than 25 feet above grade with a contact area of 20 square feet.

c. A 3 inch by 10 foot long (ASA Schedule 40) pipe (72 pounds) traveling end-on at a speed of 100 mph at any elevation on the structure.

Appendix 5.D addresses an expanded spectrum of tornado missiles which are additionally considered in the design of Tornado Missile Protection that was requested by the NRC. The design criteria presented in Appendix 5.D is an expansion of the tornado missile protection criteria of this Section and does not delete any previous requirements. The fir plank of this Section and the wooden beam of Appendix 5.D are the same missiles.

Analysis of the effect of the impact of the missiles on structures is based on the methods presented in the NavDocks P-51, “Design of Protective Structures-a New Concept of Structural Behavior,” published by U.S. Bureau of Yards and Docks, August 1950, Washington, D.C.

Provisions to tie down all slabs, blocks, or partitions outside of containment which are potential seismic or tornado missiles are described as follows:

1. Slabs and Blocks - Slabs and blocks which are potential seismic or tornado missiles are those items which fall into the category of hatch covers or removable partitions and lie within the Class I structures in areas containing Class I equipment or components.

   All removable wall panels are tied structurally to the building by retaining members and reinforcing within the wall panel. In all cases, removable wall panels

are designed to remain in place and intact, sustaining seismic or pressure loadings appropriate to the elevation within the buildings. Hatch covers which do not serve as vents during build-up and decay of pressures which would possibly occur during a tornado, are secured with fastening devices which will resist all design forces due to such loading. Hatch covers which serve as vents are designed to open to relieve internal pressures, but are provided with mechanical retaining devices which prevent the element from becoming a missile during seismic or tornado occurrences.

2. Partitions - The partitions and walls that are located within areas housing Class I equipment or components are reinforced vertically and horizontally and are anchored around the perimeter of the elements to the building structure. All partitions within these areas are constructed of either reinforced concrete or reinforced concrete masonry units. The design provides structural adequacy to sustain appropriate seismic or differential pressures resulting from a tornado occurrence.

   Earthquake loads are defined in Sections 5.2.2 and 5.8 of the FSAR.

   Tornado pressure design criteria is defined in Section 5.2.2 of the FSAR.

5.2.5.1.3 Turbine Missile Consideration

The provisions to control turbine overspeed for the Millstone Unit Number 2 turbine generator are documented in Amendment 11 to the Millstone Unit Number 2 Preliminary Safety Analysis Report. Amendment 11 was filed with the Commission on June 12, 1970. Based on the overspeed protection and controls provided, the applicants conclude that the probability of destructive overspeed as a possible cause of turbine failure is extremely low.

Turbine missiles were discussed in Amendment 7 to the Millstone Unit Number 2 Preliminary Safety Analysis Report filed with the Commission on April 24, 1970.

The MP2 LP rotors, which were of the original shrunk-on wheel design, were replaced during 2R15 with monoblock rotors. The monoblock design eliminates the keyway between the rotor shaft and the shrunk-on wheels. In accordance with NRC approved GE methodology (ref. NUREG-1048), the dominant mode of wheel failure and resulting missile generation was identified to be brittle fracture, due to keyway stress corrosion cracking occurring at or near normal operating speed. The requirement for the unfavorably oriented turbine (the case of MP2), in NUREG-1048 is that the probability be less than 10^{-5} per year. Since the monoblock rotor design eliminates the brittle fracture mode, the probability for a burst and a missile at normal operating speed is negligible. The remaining probability is a function of a ductile failure that is dependent upon the probability of a turbine overspeed event. The probability of a complete control system failure was determined to be 1.7×10^{-6} with a GE Mark I control system (reference DCR M2-03001, Rev. 0, Attachment 2). As part of the GE Mark VIe turbine controls digital upgrade (reference Design Change MP2-10-01016, Attachment 15), GE determined that the

probability of an overspeed event is less with a Mark VIe control system than with a Mark I control system.

Refer to Figure 1.2–15 of the FSAR, Section E-E through the turbine building. When viewed from the south, the turbine generator unit rotates in a counter-clockwise direction. As can be seen in this figure, the safety-related structures such as the containment, diesel generator rooms, auxiliary and control buildings, are protected from low angle missiles by the massive turbine generator pedestal. The probability of a high angle missile returning under gravity forces to strike any portion of the plant is extremely low. This coupled with a review of the plant layout containing the components and systems required to bring the plant to a safe shutdown condition without off site power available indicates that turbine generated missile damage will not preclude the safe shutdown of the plant.

5.2.5.2 Post-Tensioning Sequence

The detailed stressing sequence is based on the design requirements to limit the membrane tension in the concrete to $1.0(f'_{ci})^{1/2}$ and to minimize unbalanced loads which would produce differential stresses in the structure. Finite element mesh of the containment shell used to establish the post-tensioning sequence is shown on Figure 5.2–18.

Bechtel provides the post-tensioning system vendor with the prestressing force requirements, the anticipated concrete elastic shrinkage and creep, and the numerical order in which the tendons are to be stressed. The vendor then incorporates all this information, along with losses due to tendon relaxation, friction, and anchorage losses, if any, to establish the initial jacking force for each sequential operation.

Force measurements are obtained by measuring the elongation of the prestressing tendons and comparing it with the calculated forces indicated by the jack-dynanometer or pressure gauge. The latter represents the pressure in the jack with a tolerance of ± 2 percent. The calibration of the force-jack pressure gauge or dynanometer combination is traceable to the National Bureau of Standards and is so certified prior to the application of prestressing forces. Pressure gauges and jacks so calibrated are always used together.

During stressing, records are kept on the pressures applied and the corresponding elongations obtained. Jack-dynanometer or gauge combinations are checked against elongation of the tendons and any discrepancy exceeding ± 5 percent of the computed values utilizing the average load elongation curve is corrected and documented.

5.2.5.3 Differential Displacement Between Structures

A differential settlement of one-eighth inch between the containment and the auxiliary building foundations is assumed for design. Effects of the dynamic displacement of adjacent structures due to seismic disturbances are included in the analysis described in Section 5.8. It also includes the rocking of structures on dynamic elasticity of the foundations. The maximum and minimum values of displacement are taken as the separation of structures due to movement in opposite directions, both vertically and horizontally. The maximum differential movement at the level of

the auxiliary building roof is expected to be in the order of ± 1.5 inches. The maximum displacements at the level of the auxiliary building roof and the containment structure at the same elevation were calculated to be three-eighth and one-eighth inches, respectively, when subject to a design-basis earthquake. Provisions for 1.5 inches at all junctions between the containment structure and the auxiliary building were considered to be a conservative value. Provisions are made at all junctions between the containment and the auxiliary building to permit the differential movement to take place with no significant transfer of loads to the containment.

Piping flexibility analyses include the effect of the differential movement between structures, as well as the effect of seismic acceleration on the piping and its contained masses.

5.2.5.4 Polar Crane for the Containment

The polar crane is designed to meet the loading requirements of the applicable portions of the Electric Overhead Crane Institute (EOCI) "Specification Number 61 for Electric Overhead Traveling Cranes," except that the earthquake loading is the seismic response of the containment at the crane supports. In addition, the crane bridge and trolley are provided with mechanical guides to prevent possible derailment at the design basis earthquake. Furthermore, the polar crane is designed so that even in the unlikely event of the failure of a rail, the crane will remain on the supports.

Typical details for the polar crane supports are shown in Figures 5.2–9 and 5.2–15.

5.2.5.5 Containment Maintenance Truss

A maintenance truss is provided in the containment for use in the maintenance of containment spray piping and ease of inspection of the interior of the dome liner plate.

It is supported at the top by a large pin embedded in the containment dome. The bottom of the truss rests on the polar crane runway rail. The truss is moved around the containment by the polar crane.

The truss is a Class I item designed for all postulated load conditions that are used for the containment design.

5.2.5.6 Unit 2 Stack

The Unit 2 stack extends approximately 12 feet above the enclosure building and serves as a ventilation discharge duct. It is designed to the following factored load equations. The seismic response of the stack is taken as that of the enclosure building in which the stack is attached:

$$C = 1/\phi\ [(1.00 + 0.05)\ (D) + 1.25\ E]$$

$$C = 1/\phi\ [(1.00 + 0.05)\ (D) + 1.25\ W]$$

$$C = 1/\phi\ [(1.00 + 0.05)\ (D) + 1.0\ E']$$

where:

C = required capacity of the structure to resist factored loads

ϕ = capacity reduction factor

D = dead loads of structure

E = operating basis earthquake

E′ = design basis earthquake

W = wind or tornado loads

5.2.5.7 Pipe Whip Protection Criteria

Details may be found in Section 6.1.4.1.1.1.

5.2.5.7.1 Methods of Protection Against Pipe Whip

Protection against critical pipe failures as defined in Section 6.1.4.1 of critical safety-related targets is assured by one of the following means:

a. Plant layout is arranged such that the targets are physically separated from the effects of potential pipe failures. (See Section 6.1.4.1.1.2.)

b. Either functional structure (such as walls, floors, etc.) or specially designed barriers are provided. Such barriers are also used to protect against the effects of jet impingement forces.

c. In areas where unrestrained motion of a pipe failure as defined in Section 6.1.4.1.1.2 could hit a safety-related target, pipe restraints are employed. Spacing of the restraints is as follows:

(i) Along a continuous run of restraints the spacing of restraints shall be:

$$L = (4M_p/F_j)$$

where:

L = center-to-center span of restraint

M_p = plastic moment of the pipe, considering the effect of longitudinal stresses resulting from the internal pipe pressure. Allowable stress is taken 1.2 times the lower limit yield stress as defined in ASME Section III, “Nuclear Power Plant Components,” Paragraph NB-3225.

F_j = jet impingement force as defined in Subsection d.

(ii) The maximum distance of a restraint from the working point of an elbow for a circumferential break is:

$$L = (M_p/F_j)$$

d. If an analysis demonstrates a component can withstand the loading generated by either the jet impingement force, or the effect of a pipe whip impact, protection is assured.

The jet impingement load acting on a target shall be taken as:

$$F_j = (DLF)\ PK_\phi A_{TE}$$

where:

F_j = jet impingement force

DLF ≤ 2.0, Dynamic Load Factor to account for the amplification resulting from the sudden impulse nature of the jet force. The use of DLF values lower than 2.0 require justification.

P = the steady state blowdown pressure established for the specific break location, with consideration of internal fluid parameters based on the maximum Normal Operating Condition and piping internal friction, as appropriate.

$K\phi$ = 1.0, for cold water and non-flashing/subcooled fluid blowdown.

$K\phi$ = jet dispersion correction factor for steam and steam/water mixtures.

$= [1 + (2x/D_e) \tan \phi]^{-2}$, for $x < 5\ D_e$

$= 0.131$, for $x \geq 5\ D_e$

x = distance from pipe break to target

D_e = effective diameter of break taken as internal area of pipe

ϕ = dispersion half angle-taken as 10°

A_{TE} = effective impingement area of target; i.e., area perpendicular to axis of jet spray. The value depends upon the relative size and orientation of the jet and the target. For pipes and targets with curved surfaces, an appropriate shape factor may be used.

5.2.5.7.2 Design Procedures for Restraints and Barriers

a. Pipe Restraints:

Reaction forces on restraints resulting from pipe breaks are taken to be:

$$F_r = 2.0\ P \times A$$

where

P = maximum normal operating pressure

A = internal cross-sectional area of pipe

The factor of two accounts for the dynamic amplification resulting from the sudden impulse nature of the jet force and its effect on supporting restraints. A lower value may be used if justified.

All restraints are designed using the working stress method. Allowable bending and tension stresses are taken as 1.2 times the lower limit yield stress, after the ASME Section III, "Nuclear Power Plant Components," Paragraph NB-3225. Other stresses are based upon AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," 1963, Section 2.0. However, allowable stresses for welds and bolt loads are reduced to 90 percent the specified allowable values.

The structures supporting the pipe restraints are designed for the reaction load (F_r) as described in Section 5.2.2 for the Containment, Section 5.2.4 for the Containment Internals, and Section 5.4.3 for the Auxiliary Building.

b. Pipe Impact

Impact from unrestrained pipe motion is analyzed as follows:

The pipe is assumed to swing about its nearest restraint in a configuration that produces the worst analytical condition. Impact energy is taken to be:

$$E_I = KE - E_H$$

where:

KE = (jet force) x (arc of swing)

E_H = (plastic moment of pipe) x (angle of swing)

This information is then utilized to compute the structural effects on barriers. To account for local deformation the methods presented in the NAVDOCKS P-51, "Design of Protective Structures - a New Concept of Structural Behavior," published by the U.S. Bureau of Yards and Docks, August 1950, Washington, D.C., and the Stanford Research Institute Formula as presented in Reference 5.2-5.

Design criteria for barriers are based upon energy methods, utilizing ductility ratios of 10 and 5, respectively, for steel and concrete in tension or flexure. The method of analysis is similar to that presented in Reference 5.2-6. Allowable stresses are taken as defined in AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," Section 2.0 and

ACI 318-63, Part IV-B. In those instances where functional structure is utilized as a barrier, the loading condition is as determined in either Section 5.2.2 for the Containment, Section 5.2.4 for the Containment Internals, and Section 5.4.3 for the Auxiliary Building.

5.2.5.8 Jib Crane for Containment

The Jib Crane for Containment is designed and fabricated in accordance with the requirements of ASME NOG-1 “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)”, 2004 Section 4100 and 4300 for structural and seismic analysis, Section 4200 for materials, welded and bolted connections, and Section 7000 for inspection and nondestructive examination testing. Areas not covered by ASME NOG-1-2004 conform to the requirements of CMM 70-2010 to the extent applicable to a jib crane.

The mounting structure attaching the crane to the pressurizer cubicle Is designed to meet specific seismic requirements (Seismic II/I) such that in the event of a design basis earthquake, the crane will retain its structural integrity. A boom support structure, also designed to meet seismic criteria (Seismic II/I), provides the support to secure the crane boom in the stored position to the top of the pressurizer cubicle In the event of a design basis earthquake. The design of the Jib Crane for Containment on the pressurizer cubicle is based on the following criteria.

- The crane is only operated during outage conditions. This dictates that the 3 psi accident condition for which the pressurizer cubicle is designed will not occur concurrently with the crane operation of the Jib Crane.

- During plant operation, the crane is in a stored position, tied down, and that seismic considerations are accounted for.

- The crane is operated only during outage conditions and utilizes predefined safe load paths approved in station heavy load procedures which were developed to meet NUREG 0612 requirements. Components/parts need not be operational following a seismic event, but are designed to retain structural integrity during an OBE or SSE.

5.2.6 CONTAINMENT PENETRATIONS

5.2.6.1 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.

5.2.6.1.1 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 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} standard cc/second of dry helium under post-accident environmental conditions, seismic integrity, thermocycling, and simulated field installation. Replacement medium voltage electrical penetration modules are designed to meet or exceed all requirements pursuant to IEEE 317, 1976.

Electrical aspects of these penetrations are described in Section 8.7.2.2.

5.2.6.1.2 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 Section 5.2.3.

5.2.6.1.3 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 non-power 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.

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.

5.2.6.1.4 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 sleeve is welded to the containment liner. A bellows expansion joint is provided on the 42 inch sleeve to compensate for differential movement. Typical details of the fuel transfer tube are shown in Figure 5.2–10.

5.2.6.2 Design of Penetrations

5.2.6.2.1 Design Criteria

All piping passing through the containment walls are permanently welded to the wall sleeves, forming an extension of the containment:

To preserve the integrity of the containment, provisions are incorporated to prevent internal and external forces exerted by connecting piping on the wall sleeves from fracturing or breaching the containment pressure boundary. Additionally, protection against missiles is provided for the penetration piping and valving inside and outside the containment.

Further protection of each line, necessary to preclude the loss of pipe structural integrity between penetration and the first valve, is accomplished by shortening the exposed length of pipe and installing the first valve as close as possible to the internal or external wall of the structure, depending upon valve operating and maintenance clearances. Design bases which apply to the provision of automatic and manual isolation valves in the penetration lines are contained in Section 5.2.6.2.4.

5.2.6.2.2 Design of High-Temperature Penetrations

High-temperature piping penetrations consist of two for feedwater, two for main steam, and two for steam generator blowdown. These have a maximum operating temperature range between 435°F and 550°F. Thermal insulation is provided in the air gap between the pipe and penetration liner sleeve. The combination of insulation and penetration cooling is designed to restrict maximum temperature in the concrete to 150°F.

For the condition created by loss of penetration cooling, the maximum steady state temperature in the concrete is 300°F at the penetration surface and decreases to 120°F at a maximum radial depth of 48 inches in the containment wall (Section 9.9.4.4.1). This thermal gradient produces localized compressive stresses in the concrete immediately surrounding the penetration and low tensile stresses distributed some distances from the sleeve. These compressive stresses plus the restraint provided by the prestress loads minimize the effect of elevated temperatures on the concrete (Reference 5.2-4).

In addition, the normal operating temperature of 150°F continues to cure and dry concrete near the high-temperature penetrations. Since dry concrete is only slightly affected by high temperatures, normal operation is beneficial and reduces strength losses from a temperature rise.

On this basis, the concrete in the localized area around the penetrations can withstand 300°F without significant strength loss.

5.2.6.2.3 Penetration Materials

The materials for containment penetrations, which include mechanical, electrical, and access penetrations, conform to the requirements of the ASME Nuclear Vessel Code and ANSI B31.7-68 including Code Case 1427.

As required by these codes, carbon steel penetration materials, which form the containment pressure boundary meet the necessary Charpy V-notch impact test values at a temperature 30°F below the lowest service temperature. Impact testing for mechanical piping systems is performed at + 20°F, for uniformity.

a. Mechanical Penetration Material Specification

Penetration Sleeve (Pipe)	SA-333, Grade 6
(Rolled Plate)	SA-516, Grade 60
Penetration Reinforcing Rings	ASTM A-516, Grade 60
Penetration Sleeve Reinforcing	ASTM A-516, Grade 60
Bar Anchoring Rings and Plates	ASTM A-516, Grade 60
Rolled Shapes (Nonpressure Boundary)	ASTM A-36

System Piping	ASTM A-333, Grade 6 ASME SA 335 GR P22
	ASTM A-312 TP 304 ASME SA 234 GR WP22
	ASTM A-376 TP 316
	ASTM A-376 TP 304
	ASTM A-155 Gr KCF 70
	ASTM A-106, Grade B
Isolation Valves	ASTM A-182, Grade F316
	ASTM A-105, Grade II
	ASTM A-351, Grade CF8, CF8M
	ASTM A-216, Grade WCB
	ASTM A-350, Grade LFI
	ASTM A-516, Grade 70
	ASTM A-515, Grade 70

b. Electrical Penetrations

The penetration sleeves to accommodate the electrical penetration assembly canisters are SA-333 Grade 6 carbon steel, pipe schedule 80.

c. Access Penetration Materials

The containment equipment hatch insert, flanges, and head are fabricated of ASTM A-516 Grade 60 carbon steel plate.

The personnel lock is fabricated on ASTM A-516 Grade 60 carbon steel plate.

5.2.6.2.4 Provisions for Isolation Valves

Those piping systems which penetrate the containment are provided with isolation valves which conform with the requirements of 10 CFR Part 50, General Design, Criteria 54, 55, 56, and 57.

These provisions are described in detail in Section 5.2.7.

5.2.6.3 Installation of Penetrations

Penetration sleeves are welded to the liner plate and embedded in the concrete containment wall as shown in Figure 5.2–8. All welding and approved welding procedures used are in conformance with the requirements of ASME Section IX and ASME Section III, 1971, Nuclear Vessels.

Piping systems passing through the sleeves are installed as shown in Figure 5.2–8. The single barrier closure provided is welded to the sleeve, using the same requirements noted previously.

5.2.6.4 Testability of Penetrations

All containment penetrations are subject to an initial leak rate test and periodic leak rate testing in accordance with the requirements of 10 CFR Part 50, Appendix J.

Provisions have been made to pressurize the containment pressure boundary for leak rate testing as described in Section 5.2.8.1.

Valve arrangements incorporate provisions for leak testing of the piping systems penetrations, the containment and the isolation valves for these systems, in accordance with Criterion 54 of 10 CFR Part 50, Appendix A, as necessary to perform the leak rate testing required by 10 CFR Part 50, Appendix J.

These provisions are defined in Sections 5.2.7.1 and 5.2.7.2.

5.2.7 CONTAINMENT ISOLATION SYSTEM

5.2.7.1 Design Bases

5.2.7.1.1 Functional Requirements

The containment isolation system functions to provide a double barrier to prevent leakage through all containment fluid penetrations. As a result, no single, credible failure, or malfunction of an active component can result in loss-of-isolation capabilities or intolerable leakage.

5.2.7.1.2 Design Criteria

The following criteria have been used in the design of the containment isolation system according to 10 CFR Part 50:

a. Piping systems penetrating primary reactor containment shall be provided with leak detection and isolation.

b. Piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits as necessary per the testing requirements of 10 CFR Part 50, Appendix J.

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

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

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

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

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

d. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. The valves which are used are:

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

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

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

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

e. Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

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

g. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as

necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against most severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

5.2.7.2 System Description

5.2.7.2.1 System

Containment isolation data is listed in Table 5.2-11. The containment isolation system is shown schematically in Figures 5.2–27 through 5.2–34.

The containment isolation system closes all fluid penetrations not required for operation of the engineered safety feature systems, to prevent the leakage of radioactive materials to the environment. Fluid penetrations serving engineered safety feature systems also meet the design basis double barrier criteria, but are not closed by the containment isolation signal.

The location and types of containment isolation valves meet the requirements of Appendix A of 10 CFR Part 50, General Design Criteria 54, 55, 56, and 57 as noted below in parentheses.

The fluid penetrations which may require isolation after an accident are categorized (and listed in Table 5.2-11 under the column heading “Pene. Category type”) as follows:

Type P - Lines that connect to the reactor coolant pressure boundary (Criterion 55 of 10 CFR Part 50, Appendix A).

These lines are provided with two isolation valves. One valve is inside the containment and the other is outside, located as close as practical to the containment structure. For safety injection/ shutdown cooling where flow direction is inward, the inside valve is a check valve and the outside isolation valve is capable of remote operation. Otherwise, the inside valves are either remotely operated or locked closed valves. The outside valves are always either remotely operated or locked closed.

Type O - Lines that are open to the containment internal atmosphere (Criterion 56 of 10 CFR Part 50, Appendix A).

These lines are provided with similar containment isolation valve arrangements as described for Type P penetrations.

Type N - Lines that are neither connected to the reactor coolant pressure boundary nor open to the containment atmosphere but do form a closed system within the containment structure (Criterion 57 of 10 CFR Part 50, Appendix A).

These lines are provided with single isolation valves located outside the containment as close to the containment structure as practical. These valves are either remotely operated or are locked closed manual valves.

Prior to and during the construction and licensing of Millstone Unit 2, a system inside containment was defined as a closed system if the system neither communicated with the primary coolant nor the containment atmosphere. All Type N penetrations listed in Table 5.2-11 are categorized based on this definition and are GDC compliant. However, some of the systems associated with Type N penetrations (primary make-up water system, nitrogen supply system, waste gas header, and RBCCW system) are not in strict compliance with the criteria for a closed system as defined by the current standards (SRP 6.2.4 and ANS 56.2/ANSI N271-76). The current standards for a closed system inside containment call for them to be Class 2. The RBCCW system is fabricated to Safety Class 3 requirements in accordance with the acceptance criteria for the systems in effect at the time it was designed. NRC has accepted that for a low energy system such as RBCCW, the differences in safety classes 2 and 3 in terms of fabrication and surveillance requirements is sufficiently small that there is good likelihood that the system will remain intact during an accident (NRC Letter from John F. Stolz to Edward J. Mroczka, dated January 15, 1991, “Issuance of Exemption to 10 CFR Part 50, Appendix J, Sections III.A and III.C for the Millstone Nuclear Power Station, Unit Number 2 (TAC Number 75970)”).

Due to the various acceptable options in arranging containment isolation valves, the specific valve arrangements have been given for each penetration in Figures 5.2–27 through 5.2–34. Complete descriptions of these isolation valve arrangements are summarized in Table 5.2-11 indicating each penetration by number, the valve arrangement identification, the penetration category, the testing requirements, the size and type of the valves, the mode actuation, and the valve positions in respect to normal and/or isolation conditions including the positions during air or power failure.

The piping penetrations forming an extension of the containment are designed in accordance with ANSI B-31.7, Nuclear Piping Code, Class I or II as a minimum and are installed in accordance with ASME Section III, Nuclear Power Plant Components Code. These penetrations are described in Section 5.2.6.1.2 and 5.2.6.3. The piping is subjected to a stress report as prescribed in ASME, Section III, which includes its being subjected to a seismic analysis in accordance with the State of Connecticut Building Code, Class I, as described in Section 5.2.4.3.3. The penetration is subjected to the missile and whip protection criteria described in Section 5.2.5.

Containment isolation valves are designed in accordance with the Draft ASME Code for Pumps and Valves for Nuclear Power or ASME Section III (1971 Edition). Isolation valves are subjected to a stress report or analysis which includes seismic analysis as prescribed in Section 5.2.4.3.3. Installation of these valves is subjected to the same requirements imposed on the containment penetration piping as described above.

There is sufficient redundancy in the instrumentation circuits of the engineered safety features protection system to minimize the possibility of inadvertent tripping of the isolation system. Further discussion of this redundancy and the instrumentation signals which trip the isolation system is presented in Section 7.3.

The containment pressure instrumentation is physically located close to the containment and installed using short couplings between the containment penetrations and instrumentation provided. All instrumentation provided is designed as a pressure containing device, whereby rupture of the sensing bellows will not release radioactivity to the environment but will be contained within the housing of the instrument itself. The instrument lines are sized or orificed on the inside of the containment such that the response time of the transmitters remains within an acceptable level while in the unlikely event of instrument line or transmitter housing failure, the leakage is reduced to the minimum extent practical. One shutoff valve is provided in each line for test purposes.

The instrument lines, up to and including the pressure retaining parts of instruments, are Seismic Class I, subject to quality assurance surveillance, and conservatively designed to a quality equivalent to the containment penetrations or better. All containment pressure instrumentation equipment is located in an enclosed area and protected against physical damage due to pipe whip or missiles. Provisions are included in plant design for periodic visual in-service inspection of lines from the outside of containment up to an including the pressure instruments. The single manual shutoff valve in each line will enable periodic pressurization of the impulse line from the shutoff valve up to and including the pressure transmitter for leak testing and instrument calibration.

All containment atmosphere sampling lines are designed in accordance with Regulatory Guide 1.11.

All containment penetrations except the containment sump recirculation lines, Numbers 12 and 13, and the containment pressure transmitter penetration lines, Numbers 47, 69, 70, and 71, comply with NRC General Design Criteria 55, 56, and 57 or to Regulatory Guide 1.11. Valving inside the containment is eliminated on these penetrations because of the critical nature of these penetrations.

The containment sump recirculation lines are embedded in concrete inside the containment and protected by a guard pipe outside the containment to maintain the containment boundary and system integrity. The system is required to function during the post-accident condition; therefore, if valving inside the containment existed, it would be required to be open.

The containment pressure transmitter installation is described in Table 5.2-11 and is illustrated in Figure 5.2–34 (Arrangement 30).

5.2.7.2.2 Components

The major system components and materials of fabrication for the containment isolation system are indicated in Tables 5.2-12 and 5.2-13 for piping and valves, respectively.

5.2.7.3 System Operation

5.2.7.3.1 Emergency Operation

In the unlikely event of a LOCA, containment isolation system automatically isolates the nonessential process lines coincident with the containment isolation actuation signal (CIAS) as described in Section 7.3. Containment penetrations that are opened directly to the containment, such as the normal sump drain, are also automatically isolated by the CIAS as described in Section 7.3. Other process lines which are not opened 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.

5.2.7.4 Availability and Reliability

5.2.7.4.1 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/hour per inch of seat diameter. Butterfly valves used for isolation are purchased and tested to ensure essentially zero seat leakage preinstallation under the test pressure conditions. Subsequent to installation, leak tightness is confirmed by testing as described in Section 5.2.7.1.2.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.

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. These valves are locked closed, pneumatically isolated and electrically disconnected when the plant is in modes 1 through 4. This is accomplished by closing the instrument air isolation valves and pulling the control power fuses for each of the valves. The associated instrument air isolation valves and fuse blocks are then locked. By locking them closed in this manner, these valves are considered sealed closed isolation valves.

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.

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. Valves 2-MS-65A, 2-MS-65B, 2-MS-202, and 2-SI-651 have disconnect switches in their power circuits to ensure the valves are in their proper position in the event of an Appendix R fire, and 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.

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. However, the twelve RBCCW CAR cooler inlet and outlet containment isolation valves are not equipped with reserve air supply tanks. The RBCCW System was designed (back fitted) with the 12 containment isolation valves as a defense in depth design feature for containment integrity to meet the requirements of GDC 57. These valves fail open (which is their accident position) on loss of air. They are located outside containment and are equipped with a hand wheel as a secondary mode of operation.

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.

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 Number 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.

Penetration Number 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.

The safety injection test line, penetration Number 11, conforms to General Design Criteria Number 57 of 10 CFR Part 50, Appendix A. This line is isolated from the reactor coolant pressure boundary by two valves, both closed, as allowed by 10 CFR 50.55a (January 1972 edition), footnote 1(b) for exclusion from the reactor coolant system. Therefore, the safety injection test line is neither open directly to the containment atmosphere nor part of the reactor coolant pressure boundary, as provided by Criterion Number 57.

Penetration Numbers 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.

Penetration Numbers 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.

5.2.7.4.2 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.

RBCCW CAR cooler inlet outlet containment isolation valves (2-RB-28.1A to D, 2-RB-28.2A to D and 2-RB-28.3A to D) are exempt from Appendix J Type C testing (NRC Letter from John F. Stolz to Edward J. Mroczka, dated January 15, 1991, "Issuance of Exemption to 10 CFR Part 50, Appendix J, Sections III.A and III.C for the Millstone Nuclear Power Station, Unit Number 2 (TAC Number 75970)"). They do not have leakage criteria. The leakage criteria for CIVs on closed systems that are exempt from Appendix J testing and whose primary safety function is to remain open or open is based on the functional leakage requirements, which is limited to "as low as reasonably attainable."

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.

5.2.8 CONTAINMENT TESTING AND SURVEILLANCE

5.2.8.1 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 10 CFR 50, 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).

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.

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.

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}.

The Type A test and the supplemental verification test are performed according to the requirements of the MP2 Technical Specification and 10 CFR 50, 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 (Reference 5.2-8) and the total time method from BN-TOP-1 (Reference 5.2-9)

Quality-related software employing these techniques calculates the least-squares fit and upper confidence limit containment leakage rate, L_{am}. It automatically checks it against the acceptance criteria (0.75 L_a).

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 Reference 5.2-12 for details), and the superimposed leakage rate, L_o.

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

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.

5.2.8.1.1 Total Time Method for Calculating Containment Leakage Rate

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.

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.

5.2.8.1.2 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).

5.2.8.2 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.

5.2.8.3 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.

5.2.8.4 Tendon Surveillance

5.2.8.4.1 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.

The requirements of 10 CFR 50.55a were amended by the Nuclear Regulatory Commission (NRC) to incorporate Subsections IWE and IWL of Section XI of the ASME B&PV Code, on September 9, 1996. Following this change, DENC revised the containment inservice inspection program for MPS2 to implement the requirements of ASME Section XI, Subsections IWE and IWL. RG 1.35 was determined to be redundant with ASME Section XI, Subsection IWL and was no longer needed. RG 1.35 was subsequently withdrawn by the NRC in August 2015.

5.2.8.4.2 Compliance with Regulatory Guide

The tendon surveillance program was developed and complied with the Regulatory Guide 1.35. However, in 2000 ASME Section XI, Subsections IWE and IWL were also implemented. The NRC determined RG 1.35 was redundant with ASME Section XI, Subsection IWL and subsequently withdrawn in August 2015. All inspections and testing are now performed in accordance with the requirements of ASME Section XI, IWL as amended by 10 CFR 50.55a.

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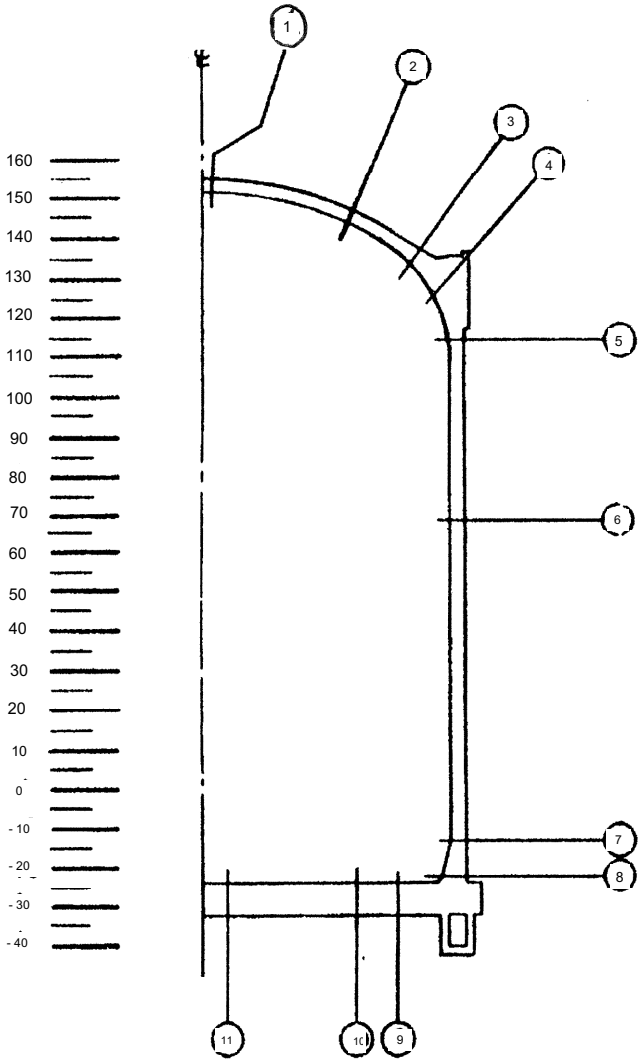
5.2-13 Millstone Unit 3, Final Safety Analysis Report, Section 2.3-Meteorology.

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.

5.2-15 ANSI N45.4-1972, “Leakage Rate Testing of Containment Structures for Nuclear Reactors.”

TABLE 5.2-1 CONTAINMENT STRUCTURE ANALYSIS SUMMARY



KEY ELEVATION SHOWING LOCATION OF REFERENCE SECTIONS

Notes on Table Values

The values shown in the following tables are taken from the cracked section analysis for the containment building. Earthquake forces are added to the forces from other loads and the resulting total is solved to obtain the stresses given.

The allowable stresses are based on the criteria presented herein. The entire containment shell is constructed using 5000 psi concrete and Grade 60 bonded reinforcing steel. The liner plate has a guaranteed minimum yield strength of 24,000 psi.

Values in these tables correspond to Figures 5.2–19 to 5.2–26.

TABLE 5.2-2 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, INITIAL PRESTRESS AND LIVE LOAD (D+KF+L)

Portion	Section	Concrete Stress – Flexural & Membrane: Merdian (PSI)	Concrete Stress– Flexural & Membrane: Hoop (PSI)	Concrete Stress – Membrane: Merdian (PSI)	Concrete Stress – Membrane: Hoop (PSI)	Concrete Stress – Actual Shear Allowable Shear Capacity	Concrete Stress – Reinforcing Stress: Merdian (PSI)	Concrete Stress – Reinforcing Stress: Hoop (PSI)	Concrete Stress – Liner Plate Strain: Meridian %	Concrete Stress – Liner Plate Strain: Hoop %
Allowable		-3750	-3750	-1500	-1500		+30000	+30000	± 0.2%	± 0.2%
DOME	1	-977	-1497	-953	-1438	0.255	-7482	-11131	-0.036	-0.056
	2	-1142	-665	-1098	-594	0.442	-8514	-5246	-0.043	-0.020
	3	-1820	-436	-784	-204	0.557	-11052	-2705	-0.068	-0.016
RING GIRDER	4	-610	-445	-558	-351	0.198	-2302	-1821	-0.023	-0.017
WALL	5	-829	-942	-723	-788	0.709	-6322	-7079	-0.031	-0.035
	6	-1011	-1629	-898	-1374	0.035	-6581	-9612	-0.038	-0.061
	7	-973	-773	-961	-663	0.394	-7765	-5981	-0.036	-0.021
HAUNCH	8	-749	-292	-569	-278	0.364	-5680	-2313	-0.028	-0.011
BASE SLAB	9	-342	-106	-109	-63	0.379	+6781	+2106	-0.013	-0.004
	10	-108	-61	-64	-42	0.138	-807	-466	-0.004	-0.002
	11	-36	-35	-29	-28	0.005	-276	-273	-0.001	-0.001

TABLE 5.2-3 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, INITIAL PRESTRESS AND LIVE LOAD (D+F+L+1.15P)

Portion	Section	Concrete Stress – Flexural & Membrane: Merdian (PSI)	Concrete Stress– Flexural & Membrane: Hoop (PSI)	Concrete Stress – Membrane: Merdian (PSI)	Concrete Stress – Membrane: Hoop (PSI)	Concrete Stress – Actual Shear Allowable Shear Capacity	Concrete Stress – Reinforcing Stress: Merdian (PSI)	Concrete Stress – Reinforcing Stress: Hoop (PSI)	Concrete Stress – Liner Plate Strain: Meridian %	Concrete Stress – Liner Plate Strain: Hoop %
Allowable		-3750	-3750	-1500	-1500		+30000	+30000	± 0.2%	± 0.2%
DOME	1	-70	-145	-24	-38	0.298	+1134	+3855	-0.003	-0.005
	2	-139	0	-88	+56	0.074	-355	+9394	-0.005	-0.003
	3	-173	-202	-89	-101	0.914	-1101	-1277	-0.006	-0.008
RING GIRDER	4	-320	-291	-278	-294	0.088	+13653	-1957	+0.006	-0.011
	5	-559	-412	-191	-410	0.114	+8692	-3304	+0.004	-0.015
WALL	6	-318	-87	-265	-63	0.017	-1829	-374	-0.012	-0.003
	7	-1762	-981	-319	-646	0.400	+11847	-7212	+0.014	-.0012
HAUNCH	8	-1779	-580	-231	-386	0.357	+19385	-4432	+0.020	-0.007
	9	-636	-626	+53	-3	0.585	+17764	+12418	-0.024	-0.023
BASE SLAB	10	-1488	-613	+72	-18	0.154	+29512	+12153	-.0056	-0.023
	11	-64	-63	-44	-43	0.015	-486	-480	-0.002	-0.002

TABLE 5.2-4 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, INITIAL PRESTRESS AND LIVE LOAD, OPERATING TEMPERATURE AND OBE ($D+F+L+T_0+E$) [A]

Portion	Section	Concrete Stress – Flexural & Membrane: Merdian (PSI)	Concrete Stress– Flexural & Membrane: Hoop (PSI)	Concrete Stress – Membrane: Merdian (PSI)	Concrete Stress – Membrane: Hoop (PSI)	Concrete Stress – Actual Shear Allowable Shear Capacity	Concrete Stress – Reinforcing Stress: Merdian (PSI)	Concrete Stress – Reinforcing Stress: Hoop (PSI)	Concrete Stress – Liner Plate Strain: Meridian %	Concrete Stress – Liner Plate Strain: Hoop %
Allowable		-3750	-3750	-1500	-1500		+40000	+40000	±0.2%	±0.2%
DOME	1	-1400	-2124	-840	-1276	0.161	-1925	-3360	-0.063	-0.086
	2	-1745	-1270	-906	-519	0.414	-281	+3032	-0.074	-0.057
	3	-2737	-1456	-618	-302	0.485	+29937	+20938	-0.111	-0.063
RING GIRDER	4	-690	-1662	-491	-542	0.018	-1371	+16143	-0.026	-0.068
	5	-1610	-1930	-615	-708	0.523	+6967	+11510	-0.069	-0.081
WALL	6	-1816	-2557	-770	-1194	0.028	+7085	+13832	-0.077	-0.104
	7	(-600)	-1134	-1015	-546	(0.886)	(+4492)	+554	-0.136	-0.052
		-3428				0.990	+20208			
HAUNCH	8	(-1051)	-845	-601	-356	0.595	(+14090)	+4192	-0.104	-0.037
		-2601					+39554			
	9	(-1142)	(-274)	-294	+7	0.615	(+10720)	(+7443)	(-0.051)	(-0.018)
		-1098	-261				+10308	+7089	-0.049	-0.017
BASE SLAB	10	(-1064)	(-628)	-134	-5	0.350	(+12777)	(+13656)	(-0.050)	(-0.032)
		-858	-616				+10304	+13388	-0.040	-0.031
	11	(-968)	(-855)	-308	-291	0.222	(+5539)	(+3911)	(-0.045)	(-0.040)
		-978	-864				+5595	+3950	-0.045	-0.040

a. Note: Values in parentheses reflect current controlling stresses/strains resulting from the revised seismic analysis performed in 1998-99.

TABLE 5.2-5 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, LIVE LOAD, 100% ACCIDENT PRESSURE AND ACCIDENT TEMPERATURE ($D+F+L+1.0P+T_1$)

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity				
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)		MER. (PSI)	HOOP (PSI)	MER. %	HOOP %
Allowable		-3750	-3750	-1500	-1500		+40000	+40000	± 0.5%	±0.5%
DOME	1	-238	-1312	-125	-204	0.400	+28479	+33263	-0.0104	-0.141
	2	-419	0	-139	-32	0.321	+28276	+28784	-0.110	-0.086
	3	-1089	-2238	-103	-284	0.543	+23322	+20642	-0.136	-0.177
RING GIRDER	4	-1283	-2956	-305	-515	0.038	-3608	+8183	-0.155	-0.201
	5	-2122	-2716	-223	-526	0.193	+27302	+5991	-0.079	-0.196
WALL	6	-2115	-2053	-304	-206	0.036	+14203	+14712	-0.173	-0.171
	7	-1854	-2420	-385	-630	0.199	+5215	+1545	-0.163	-0.185
HAUNCH	8	-305	-2271	-264	-574	0.313	-2402	+840	-0.008	-0.135
	9	-764	-853	+51	-75	0.432	+22444	+13189	-0.027	-.039
BASE SLAB	10	-1164	-847	+89	-110	0.119	+34050	+11248	-0.036	-0.039
	11	-708	-714	-265	-266	0.051	+2269	+2272	-0.034	-0.035

TABLE 5.2-6 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, OPERATING TEMPERATURE, THERMAL EXPANSION FORCES OF PIPES, PIPE RUPTURE FORCES AND DBE ($D+F+T_0+H+R+E^1$)[A]

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity	MER. (PSI)	HOOP (PSI)	MER. %	HOOP %
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)					
ALLOWABLE		-4250	-4250	-4250	-4250		+54000	+54000	± 0.5%	± 0.5%
DOME	1	-1400	-2124	-840	-1276	0.127	-1925	-3360	-0.063	-0.086
	2	-1745	-1270	-906	-519	0.371	-281	+3032	-0.074	-0.057
	3	-2737	-1456	-618	-302	0.430	+29937	+20938	-0.111	-0.063
RING GIRDER	4	-690	-1662	-491	-542	0.014	-1371	+16143	-0.026	-0.068
	5	-1610	-1930	-615	-708	0.442	+6967	+11510	-0.069	-0.081
WALL	6	-1816	-2557	-770	-1194	0.023	+7085	+13832	-0.077	-0.104
	7	(-949)	-1134	-1091	-546	(0.720)	(+7340)	+554	-0.155	-0.052
		-3935				0.941	+26160			
HAUNCH	8	(-1601)	-845	-641	-356	0.726	(+22520)	+4192	-0.124	-0.037
		-3142					+50111			
	9	(-1330)	(-77)	-314	+65	0.562	(+13639)	(+11574)	(-0.058)	(-0.000)
		-1265	-72				+12887	+10817	-0.055	-0.000

TABLE 5.2-6 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, OPERATING TEMPERATURE, THERMAL EXPANSION FORCES OF PIPES, PIPE RUPTURE FORCES AND DBE ($D+F+T_0+H+R+E^1$)[A] (CONTINUED)

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity	MER. (PSI)	HOOP (PSI)	MER. %	HOOP %
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)					
BASE SLAB	10	(-1230)	(-710)	-118	-3	0.334	(+16530)	(+15531)	(-0.056	(-0.035)
		-939	-657				+12618	+14381	-0.043	-0.032
	11	-1043	(-873)	-306	-283	0.224	+6958	(+4664)	-0.047	(0.040)
			-891					+4759		-0.041

a. Note: Values in parentheses reflect current controlling stresses/strains resulting from the revised seismic analysis performed in 1998-99.

TABLE 5.2-6A DELETED BY FSARCR 04-MP2-016

TABLE 5.2-7 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, 100% ACCIDENT PRESSURE, THERMAL EXPANSION FORCES OF PIPES, PIPE RUPTURE FORCES AND DBE ($D+F+1.0P+H+T_1+E^1$) [A]

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity				
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)		MER. (PSI)	HOOP (PSI)	MER. %	HOOP %
ALLOWABLE		-4250	-4250	-4250	-4250		+54000	+54000	± 0.5%	± 0.5%
DOME	1	-238	-1312	-125	-204		+28479	+33263	-0.104	-0.141
	2	-419	0	-139	-32	0.271	+28276	+28784	-0.110	-0.086
	3	-1089	-2238	-103	-284	0.481	+23322	+20642	-0.136	-0.177
RING GIRDER	4	-1283	-2956	-305	-515	0.035	-3608	+8183	-0.155	-0.201
	5	-2122	-2716	-223	-526	0.171	+27302	+5991	-0.079	-0.196
WALL	6	-2115	-2053	-304	-206	0.032	+14203	+14712	-0.173	-0.171
	7	-691	-2420	-622	-630	0.921	+16649	+1545	-0.231	-0.185
HAUNCH	8	-1972	-2271	-405	-574	0.837	+17453	+840	-0.120	-0.135
	9	(-1638)	(-511)	-52	+234	0.816	(+30071)	(+33602)	(-0.070)	(+0.047)
		-1575	-487				+28914	+32002	-0.067	+0.045
BASE SLAB	10	(-3825)	(-803)	+172	+137	0.328	(+52609)	(+32029)	(-0.010)	(-0.004)
		-1591	-750				+51617	+29934	-0.009	-0.004
	11	-948	(-784)	-253	-232	0.237	+6921	(+4659)	-0.044	(-0.037)
			-800					+4754		-0.038

a. Note: Values in parentheses reflect current controlling stresses/strains resulting from the revised seismic analysis performed in 1998-99.

TABLE 5.2-8 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, 125% ACCIDENT PRESSURE, 125% THERMAL EXPANSION FORCES OF PIPES, ACCIDENT TEMPERATURE AND 125% DBE ($D+F+1.25P+1.25H+T_1+1.25E$)[A]

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity				
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)		MER. (PSI)	HOOP (PSI)	MER. %	HOOP %
ALLOWABLE		-4250	-4250	-4250	-4250		+54000	+54000	± 0.5%	± 0.5%
DOME	1	-233	-374	+51	+60	0.359	+34254	+44242	-0.079	0.097
	2	-91	0	+50	+92	0.252	+44348	+38368	-0.033	-0.041
	3	-325	-2125	+26	-269	0.456	+22511	+19599	-0.107	-0.173
RING GIRDER	4	-211	-2903	-260	-514	0.035	-2478	+5846	-0.008	-0.199
	5	-699	-2559	-125	-468	0.070	+14752	+7094	-0.117	-0.190
WALL	6	-1247	-14	-191	+41	0.032	+20789	+39086	-0.141	-0.040
	7	-2908	-2361	-454	-606	0.947	+11400	+1640	-0.202	-0.183
HAUNCH	8	-2292	-2264	-93	-525	0.859	+40726	+312	0.000	-0.135
	9	(-1580)	(-592)	+9	+155	0.870	(+33430)	(+28651)	(-0.068)	(+0.014)
		-1463	-564				+30954	+27287	-0.063	+0.013
BASE SLAB	10	(-1767)	(-806)	+67	+42	0.277	(+46578)	(+23033)	(-0.064)	(-0.039)
		-1485	-753				+39141	+21526	-0.054	-0.036
	11	(-961)	(-829)	-279	-263	0.177	(+5903)	(+4143)	(-0.044)	(-.039)
		-933	-821				+5731	+4102	-0.043	0.039

a. Note: Values in parentheses reflect current controlling stresses/strains resulting from the revised seismic analysis performed in 1998-99.

TABLE 5.2-9 CONTAINMENT STRUCTURE ANALYSIS SUMMARY - DEAD LOAD, PRESTRESS, 150% ACCIDENT PRESSURE, AND ACCIDENT TEMPERATURE ($D+F+1.5P+T_1$)

Portion	Section	Concrete Stress					Reinforcing Stress		Liner Plate Strain	
		Flexural & Membrane		Membrane		Actual Shear Allowable Shear Capacity				
		MER. (PSI)	HOOP (PSI)	MER. (PSI)	HOOP (PSI)		MER. PSI)	HOOP (PSI)	MER. %	HOOP %
ALLOWABLE		-4250	-4250	-4250	-4250		+54000	+54000	± 0.5%	± 0.5%
DOME	1	-249	-366	+226	+324	0.359	+45769	+54195	-0.029	-0.020
	2	-114	0	+239	+216	0.239	+43842	+34850	-0.039	-0.058
	3	0	-1995	+155	-252	0.456	+30200	+18706	-0.018	-0.168
RING GIRDER	4	-356	-2856	-215	-512	0.028	-2855	+3587	-0.003	-0.197
	5	-47	-2452	-30	-409	0.019	+26249	+8102	-0.092	-0.186
WALL	6	-166	0	-78	+287	0.032	+28588	+33155	-0.100	-0.065
	7	-895	-2298	-157	-581	0.424	+9106	+1776	-0.127	-0.180
HAUNCH	8	-607	-2255	-145	-549	0.503	+9998	-224	0.000	-0.124
	9	-1121	-1094	+135	-115	0.629	+37848	+15961	-0.001	-0.048
BASE SLAB	10	-1597	-1004	-88	-208	0.284	+27145	+9735	-0.068	-0.045
	11	-795	-797	-311	-313	0.023	+1981	+2002	-0.038	-0.038

TABLE 5.2-10 SPECTRUM OF POTENTIAL MISSILES FROM INSIDE THE CONTAINMENT

Table 1: I. Reactor Vessel

ITEM	KINETIC ENERGY (ft-lb.)	Weight (lb.)	LEADING SECTION	POINT OF IMPACT
A. Closure Head Nut	2,022	116	Annular OD = 10-9/16 inch ID = 6.8 inch	Overhead missile shield
B. Closure Head Nut & Stud	4,932	710	Solid Circle 7 inch Diameter	Overhead missile shield
C. Instrumentation Assembly	12,700	335	Solid Disk 6.5 inch Diameter and 3 inch Thick	Overhead missile shield
D. Instrumentation from Flange Up	14,000	165	Solid Disk 6.5 inch Diameter and 3 inch Thick	Overhead missile shield
E. Instrument Flange Stud	14.3	6.5	Solid Circle 1.5 inch Diameter	Overhead missile shield

Table 2: II. Steam Generator

ITEM	KINETIC ENERGY (ft-lb.)	Weight (lb.)	LEADING SECTION	POINT OF IMPACT
A. Steam Generator Primary Manway Stud and Nut	48.53	9	Solid Circle 1.338 inch Diameter	Containment Floor or Secondary Shield Wall
B. Steam Generator Secondary Manway Stud and Nut	10.0	5	Solid Circle 1.25 inch Diameter	Steam Generator Blockhouse Wall
C. Steam Generator Secondary Handhole Stud and Nut	3.98	1.7	Solid Circle 0.82 inch Diameter	Primary or Secondary Shield Wall

TABLE 5.2-10 SPECTRUM OF POTENTIAL MISSILES FROM INSIDE THE CONTAINMENT (CONTINUED)

Table 3: III. Pressurizer

Item	Kinetic Energy (ft-lb.)	Weight (lb.)	Leading Section	Point of Impact
A. Pressurizer Manway Stud & Nut [NOTE: This missile envelops the Pressurizer vent port stud missile, which would strike the south wall.]	57.16	6.8	Solid Circle 1.0 inch Overhead missile shield Diameter	Upper west wall of Pressurizer Blockhouse
B. Pressurizer Temperature Element	290	3	Solid Disk 2.75 inch Overhead missile shield Diameter and 0.5 inch Overhead missile shield Thick	Pressurizer Blockhouse wall, roof slab or platform
C. Pressurizer Instrument Nozzle and Instrument (Based on V & VII Nozzle & RTD cases.)	1,125	11.1	Solid Disk 2.75 inch Overhead missile shield Diameter and 0.5 inch Overhead missile shield Thick	Pressurizer Blockhouse wall, roof slab or platform
D. Safety Valve Flange Bolt [NOTE: This missile envelops stud missiles from either flange or from the bonnet bolted connection.]	15	3.7	Solid Circle 1.25 inch Overhead missile shield Diameter	Pressurizer Blockhouse Wall or Floor

Table 4: IV.

Item	Kinetic Energy (ft-lb.)	Weight (lb.)	Leading Section	Point of Impact
Control Element Drive Mechanism (Magnetic Jack	47,800	2,100	Solid Circle 11 inch Overhead missile shield Diameter	Overhead missile shield

TABLE 5.2-10 SPECTRUM OF POTENTIAL MISSILES FROM INSIDE THE CONTAINMENT (CONTINUED)

Table 5: V.

Item	Kinetic Energy (ft-lb.)	Weight (lb.)	Leading Section	Point of Impact
Main Coolant Piping Temperature Nozzle with RTD	1,125	11.1	Solid Disk 2.75 inch Overhead missile shield Diameter and 0.5 inch Overhead missile shield Thick	Secondary shield wall

Table 6: VI.

Item	Kinetic Energy (ft-lb.)	Weight (lb.)	Leading Section	Point of Impact
Surge and Spray Piping Thermal Wells with RTD Assembly	277	1/34	Solid Disk 2.75 inch Overhead missile shield Diameter and 0.5 inch Thick	Secondary shield wall

Table 7: VII.

Item	Kinetic Energy (ft-lb.)	Weight (lb.)	Leading Section	Point of Impact
Main Coolant Pump Thermal Well with RTD	1,125	11.1	Solid Disk 2.75 inch Overhead missile shield Diameter and 0.5 inch Overhead missile shield Thick	Secondary shield wall

The basic formulas used for the calculation of missile penetration are those presented in NavDocks P-51, “Design of Protective Structures - a New Concept of Structural Behavior,” published by the U.S. Bureau of Yards and Docks, August, 1950, Washington, D.C.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Valve Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
1	Demineralized Water	PMW	IA	N [12]	IN	1B	Yes	2-PMW-43	Outside	2 inch	Globe	1	2 inch	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-PMW-165	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-PMW-3	Inside	2 inch	Check	1		–	–	–	–	No	–
2	Letdown Line to Purification Demineralizer	CVCS	IA	P	OUT	7	Yes [4]	2-CH-516	Inside	3 inch	Globe	1	2 inch	Diaphragm	CIAS	Open	Closed	Yes	Closed
							No	2-CH-006 [11]	Inside	2 inch	Gate	1		Manual	----	Open	As Is	No	Open
							Yes	2-CH-089	Outside	2 inch	Globe	1		Diaphragm	CIAS	Open	Closed	Yes	Closed
							No	2-CH-763, 2-CH-658, 2-CH-991	Inside	1 inch	Gate	3		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-260, 2-CH-082, 2-CH-083	Inside	0.75 inch	Globe	3		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-067	Outside	0.75 inch	Gate	1		Manual	----	Locked Closed	As Is	No	Closed
							Yes [4]	2-CH-515	Inside	3 inch	Globe	1		Diaphragm	SIAS	Open	Closed	Yes	Closed
3	Reactor Coolant Charging Line	CVCS	IA	P	IN	9	Yes [4]	2-CH-518, 2-CH-519	Inside	2 inch	Globe	2	2 inch	Diaphragm	Remote	Open	Open	Yes	Open
							Yes [4]	2-CH-517	Inside	2 inch	Globe	1		Diaphragm	Remote	Closed	Closed	Yes	Closed
							Yes [4]	2-CH-434	Inside	2 inch	Gate	1		Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-CH-429	Outside	2 inch	Gate	1		MOV	Remote	Open	As Is	Yes	Open
							No	2-CH-004, 2-CH-003	Inside	0.75 inch	Gate	2		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-001, 2-CH-002, 2-CH- 443, 2-CH-714	Inside	0.75 inch	Globe	4		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-710	Outside	1 inch	Gate	1		Manual	----	Locked Closed	As Is	No	Closed
							Yes [4]	2-RC-71	Inside	0.75 inch	Globe	1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-661	Inside	1 inch	Gate	1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-435 [11]	Inside	2 inch	Spring Check	1 [5]		–	–	–	–	No	–

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category [2]	Flow Direction	Valve Arrangement	Type C Testing Requirements [3]	Valve Identification		Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							Yes	2-CH-986		Outside	3/4 inch x 1inch	Relief	1		–	–	–	–	No	–
4	Containment Spray Water	CSS	IA	O	IN	17A	Yes	2-CS-5A		Inside	8 inch	Check	1	8 inch	–	–	–	----	No	–
							Yes	2-CS-4.1A		Outside	8 inch	Gate	1		MOV	CSAS	Closed	As Is	Yes	Open
							No	2-CS-049C		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CS-049A		Outside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
5	Containment Spray Water	CSS	IA	O	IN	17B	Yes	2-CS-5B		Inside	8 inch	Check	1	8 inch	–	–	–	–	No	–
							Yes	2-CS-4.1B		Outside	8 inch	Gate	1		MOV	CSAS	Closed	As Is	Yes	Open
							No	2-CS-101		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
6,8	Safety Injection Low & High Pressure	SIS	IB	P	IN	10A (Penetration 6)	No	Penetration 6 2-SI-645 2-SI-160 2-SI-161	Penetration 8 2-SI-635	Outside	6 inch	Globe	1	6 inch	MOV	SIAS	Closed	As Is	Yes	Open
								2-SI-163		Outside	3 inch	Gate	2	6 inch	Manual	-	Locked Closed	As Is	No	Locked Closed
											0.75 inch	Globe	1	6 inch	Manual	-	Locked Closed	As Is	No	Locked Closed
						10B (Penetration 8)	No	2-SI-646, 2-SI-647	2-SI-636, 2-SI-637	Outside	2 inch	Globe	2		MOV	SIAS	Throttled	As Is	Yes	Open
							No		2-SI-733, 2-SI-041E	Outside	(Penetration 8 only) 1inch	Globe	2		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-144 [11]	2-SI-134 [11]	Outside	6 inch	Check	1		–	–	–	–	No	–
							No	2-SI-143 [11], 2-SI-1009 [11]	2-SI-133 [11], 2-SI-1010	Outside	2 inch	Check	2		–	–	–	–	No	–
							No	2-SI-095, 2-SI-1734		Outside	(Penetration 6 only) 1 inch	Globe	2		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-041F, 2-SI-1735, 2-SI-1742D	2-SI-041D, 2-SI-110, 2-SI-742C	Outside	0.75 inch	Globe	3		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Valve Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							No	2-SI-145 [11], 2-SI-135 [11], 2-SI-146 [11] 2-SI-136 [11]	Outside	0.75 inch	Globe	2		Manual	–	Open	As Is	No	Open
							No	2-SI-706D 2-SI-706C	Inside	6 inch	Check	1		–	–	–	–	No	–
							No	2-SI-848 S-SI-846	Outside	0.75 inch	Globe			Manual	–	Locked Closed	–	No	Closed
7	Safety Injection Low & High Pressure	SIS	IB	P	IN	10C	No	2-SI-615	Outside	6 inch	Globe	1	6 inch	MOV	SIAS	Closed	As Is	Yes	Open
							No	2-SI-616, 2-SI-617	Outside	2 inch	Globe	2		MOV	SIAS	Throttled	As Is	Yes	Open
							No	2-SI-114 [11]	Outside	6 inch	Check	1		–	–	–	–	No	–
							No	2-SI-012 [11], 2-SI-113 [11]	Outside	2 inch	Check	2		–	–	–	–	No	–
							No	2-SI-041A, 2-SI-107, 2-SI-716, 2-SI-715, 2-SI-742A	Outside	0.75 inch	Globe	5		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-717, 2-SI-718	Outside	1 inch	Gate	2		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-115 [11], 2-SI-116 [11]	Outside	0.75 inch	Globe	2		Manual	–	Open	As Is	No	Open
							No	2-SI-706A	Inside	6 inch	Check	1		–	–	–	–	No	–
9	Safety Injection Low & High Pressure	SIS	IB	P	IN	10D	No	2-SI-625	Outside	6 inch	Globe	1	6 inch	MOV	SIAS	Closed	As Is	Yes	Open
							No	2-SI-626, 2-SI-627	Outside	2 inch	Globe	2		MOV	SIAS	Throttled	As Is	Yes	Open
							No	2-SI-124 [11]	Outside	6 inch	Check	1		–	–	–	–	No	–
							No	2-SI-123 [11], 2-SI-011 [11]	Outside	2 inch	Check	2		–	–	–	–	No	–
							No	2-SI-722, 2-SI-723, 2-SI-720, 2-SI-721, 2-SI-742B	Outside	0.75 inch	Globe	5		Manual	----	Locked Closed	As Is	No	Closed
							No	2-SI-125 [11], 2-SI-126 [11]	Outside	0.75 inch	Globe	2		Manual	–	Open	As Is	No	Open
							No	2-SI-706B	Inside	6 inch	Check	1			–	–	–	No	–
10	Reactor Coolant Shutdown Cooling	SIS	IB	P	OUT	11	No	2-SI-709	Outside	12 inch	Gate	1	12 inch	Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification		Location Reference to Containment Structure	Valve Size	Valve Type	Valve Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							No	2-SI-651		Inside	12 inch	Gate	1		MOV	Remote[7]	Closed	As Is	Yes	Closed
							No	2-SI-101A		Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-102A		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SI-043A		Inside	0.75 inch	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 inch	Gate	1	2 inch	Manual	–	Locked Closed	As Is	No	Closed
12 & 13	Containment Sump Recirculation Line	SIS	Special	Special	OUT	16	No	Penetration 12 2-CS-16.1A	Penetration 13 2-CS-16.1B	Outside	24 inch	Gate	1	24 inch	MOV	SRAS	Closed	As Is	Yes	Open
							No	2-CS-127 [11]	2-CS-125 [11]	Outside	0.75 inch	Gate	1		Manual	–	Locked Open	As Is	No	Open
							No	2-CS-130	2-CS-135	Outside	0.75 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CS-131 [11]	2-CS-136 [11]	Outside	0.75 inch	Gate	1		Manual	–	Locked Open	As Is	No	Open
							No	2-CS-132	2-CS-137	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CS-133	2-CS-138	Outside	0.75 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CS-134	2-CS-139	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CS-140 [11]	2-CS-141 [11]	Outside	0.75 inch	Check	1		–	–	Closed	As Is	No	Closed
14	Containment Sump to Aerated Drain Tank	RWS	IA	O	OUT	13A	Yes	2-SSP-16.2		Outside	3 inch	Globe	1	3 inch	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-SSP-16.1		Inside	3 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-SSP-51		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-SSP-73		Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category [2]	Flow Direction	Valve Arrangement	Type C Testing Requirements [3]	Valve Identification		Location Reference to Containment Structure	Valve		Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
											Size	Type								
15 & 16	Feedwater & Auxiliary Feedwater	FW	II	N	IN	15A	No	Penetration 15 2-FW-5A	Penetration 16 2-FW-5B	Outside	18 inch	Positive Acting Check Valve	1	18 inch	Backflow	–	Open	Closed	Yes	Closed
						for Penetration 15	No	2-FW-12A	2-FW-12B	Outside	6 inch	Positive Acting Check Valve	1		Backflow	–	Closed	Closed	Yes	Closed
						15B for Penetration 16	No	2-FW-16A [11]	2-FW-16B [11]	Outside	1 inch	Check	1		–	–	–	–	No	–
							No	2-FW-15A	2-FW-15B	Outside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-FW-86	2-FW-182	Outside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-FW-261A		Outside	(Pene. 15 only)	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
19 & 20	Main Steam	MSS	III	N	OUT	23	No	Penetration 19 2-MS-64A	Penetration 20 2-MS-64B	Outside	0.75 inch	Stop check	1	34 inch	Air Cylinder	MSI	Open	Closed	Yes	Closed [6]
							No	2-MS-371	2-MS-369	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-MS-201	2-MS-202	Outside	4 inch	Gate	1		MOV	Remote [7]	Open	As Is	Yes	Open
							No	2-MS-3A [11]	2-MS-3B [11]	Outside	12 inch	Gate	1		Manual	–	Open	As Is	No	Open
							No	2-MS-190A	2-MS-190B	Outside	8 inch	Globe	1		Diaphragm	Steam Generator Pressure	Closed	Closed	Yes	Closed
							No	2-MS-265B	2-MS-266B	Outside	1 inch	Globe	1		Diaphragm	MSI	Open	Closed	Yes	Closed
							No	2-MS-247, 2-MS-248, 2-MS-249, 2-MS-250, 2-MS-251, 2-MS-252, 2-MS-253, 2-MS-254	2-MS-239, 2-MS-240, 2-MS-241, 2-MS-242, 2-MS-243, 2-MS-244, 2-MS-245, 2-MS-246	Outside	6 inch	Relief	8		–	–	–	–	No	–
							No	2-MS-65A	2-MS-65B	Outside	3 inch	Globe	1 ***		MOV	MSI [7]	Closed	As Is	Yes	Closed
							No	2-MS-297	2-MS-296	Outside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification		Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							No	2-MS-265A [11]	2-MS-266C [11]	Outside	1 inch	Gate	1		Manual	–	Open	As Is	No	Open
							No	2-MS-255		Outside	(Pene-tration 19 only) 0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No		2-MS-258	Outside	(Pene-tration 20 only) 1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-MS-41A [11]	2-MS-41B [11]	Outside	0.75 inch	Globe	1		Manual	–	Open	As Is	No	Open
							No	2-MS-461 [11]	2-MS-462 [11]	Outside	0.75 inch	Globe	1		Manual	–	Closed	As Is	No	Closed
							No	2-MS-459	2-MS-458	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Locked Closed
21	Reactor Coolant & Pressurizer Sampling	SS	IA	P	OUT	19	No	2-LRR-265 [11]		Inside	0.5 inch	Check	1	0.5 inch	–	–	–	–	No	–
							Yes	2-LRR-61.1		Inside	0.5 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-RC-45		Outside	0.75 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-RC-001, 2-RC-002		Inside	0.75 inch	Globe	2		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-RC-003		Inside	0.75 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-RC-65 [11]		Inside	3/8 inch	Globe	1		Manual	–	Open	As Is	No	Open
							No	2-RC-434, 2-RC-435		Inside	3/8 inch	Globe	2		Manual	–	Locked Closed	As Is	No	Closed
22 & 23	Steam Generator Bottom Blowdown	SGBS	IA	N	OUT	14	No	Penetration 22 2-MS-220A	Penetration 23 2-MS-220B	Outside	2 inch	Globe	1	2 inch	Diaphragm	AFAIS, CIAS & High Contain-ment Radiation High Radiation	Open	Closed	Yes	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category [2]	Flow Direction	Valve Arrangement	Type C Testing Requirements [3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Valve Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
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 inch	Gate	1	8 inch	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-289	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RB-173A [11]	Outside	0.75 inch	Globe	1		Manual	–	Open	As Is	No	Open
25 & 26	Reactor Building Closed Cooling Water to Containment Air Recirculation Units	RBCCW	IA	N	IN	21A	No	Penetration 25 2-RB-28.1D	Penetration 26 2-RB-28.1B	Outside	10 inch	Butterfly	1	10 inch	Air Cylinder.	Remote	Open	Open	Yes	Open
							No	2-RB-282	2-RB-283	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No		2-RB-345	Outside	(Penetration 26 only) 1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
27 & 28	Reactor Bldg. Closed Cooling Water to Containment Air Recirculation Units	RBCCW	IA	N	IN	21B	No	Penetration 27 2-RB-28.1A	Penetration 28 2-RB-28.1C	Outside	10 inch	Butterfly	1	10 inch	Air Cylinder	Remote	Open	Open	Yes	Open
							No	2-RB-236	2-RB-237	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
29	Reactor Building Closed Cooling Water Outlet from Reactor Coolant Pumps and Other Components [8]	RBCCW	IA	N	OUT	2	Yes	2-RB-37.2A	Outside	8 inch	Gate	1	8 inch	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-297A	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RB-298	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
30, 31, 32 & 33	Reactor Bldg. Closed Cooling Water From Containment Air Recirculation Cooling	RBCCW	IA	N	OUT	22	No	Penetration 30 2-RB-28.3D / Penetration 32 2-RB-28.3A	Outside	10 inch	Butterfly	1	10 inch	Air Cylinder	SIAS	Closed	Open	Yes	Open
								Penetration 31 2-RB-28.3B / Penetration 33 2-RB-28.3C											
							No	Penetration30 2RB-28.2D / Penetration 32 2RB-28.2A	Outside	6 inch	Butterfly	1	6 inch	Air Cylinder	Remote	Open	Open	Yes	Open
								Penetration 31 |2RB-28.2B / Penetration 33 2RB-28.2C											
34	Nitrogen Supply	NS	IA	N [12]	IN	18	Yes	2-SI-312	Outside	0.75 inch	Globe	1	1 inch	Diaphragm	CIAS	Open	Closed	Yes	Closed
							No	2-SI-045	Outside	0.75 inch	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 inch	Globe	1	4 inch	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes[4]	2-LRR-43.1	Inside	3 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-LRR-291	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							No	2-LRR-293, 2-LRR-295	Outside	0.75 inch	Globe	2		Manual	–	Locked Closed	As Is	No	Closed
36	Instrument Air	IA	IA	O	IN	33	Yes	2-IA-566	Outside	0.5 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed [9]
							Yes	2-IA-569	Inside	0.5 inch	Check	1		–	–	–	–	No	–
							No	2-IA-572	Inside	0.5 inch	Gate	1		Manual	–	Locked closed	As Is	No	Closed
37	Instrument Air	IA	IA	O	IN	1A	Yes	2-IA-27.1	Outside	2 inch	Globe	1	2 inch	Diaphragm	Remote	Open	Closed	Yes	Open [9]
							No	2-IA-40	Outside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-IA-43	Inside	2 inch	Check	1		–	–	–	–	No	–
38	Station Air	SA	IA	O	IN	3	Yes	2-SA-19	Outside	2 inch	Gate	1	2 inch	Manual	–	Locked Closed	As Is	No	Closed
							No	2-SA-28	Inside	1 inch	Gate	1	2 inch	Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-SA-22	Inside	2 inch	Check	1		–	–	–	–	No	–
39	Purge Air Inlet	PA	IC	O	IN	4	Yes	2-AC-4	Outside	48 inch	Butterfly	1	48 inch	Air Cylinder	High Contain-ment Radiation	Locked Closed	Closed	Yes	Closed
							Yes[4]	2-AC-5	Inside	48 inch	Butterfly	1		Air Cylinder	High Contain-ment Radiation	Locked Closed	Closed	Yes	Closed
							No	2-AC-21	Outside	0.75 inch	Globe	1		Manual	----	Locked Closed	As Is	No	Closed
40	Purge Air Discharge	PA	IC	O	OUT	5	Yes	2-AC-7	Outside	48 inch	Butterfly	1	48 inch	Air Cylinder	High Contain-ment Radiation	Locked Closed	Closed	Yes	Closed
							Yes[4]	2-AC-6	Inside	48 inch	Butterfly	1		Air Cylinder.	High Contain-ment Radiation	Locked Closed	Closed	Yes	Closed
							No	2-AC-31	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
42	Fuel Transfer Tube	FTS	Special	O	IN/OUT	8	No, Type B	N/A	Inside	Special Closure			36 inch	–	–	Closed	–	–	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							No	2-RW-291	Inside	0.5 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RW-31	Inside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RW-292	Inside	0.5 inch	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	0.75 inch	Globe	1	0.75 inch	Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes	2-CH-198	Outside	0.75 inch	Globe	1		Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes	2-CH-505	Outside	0.75 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-CH-758, 2-CH-768, 2-CH-701	Outside	0.75 inch	Globe	3		Manual	–	Locked Closed	As Is	No	Closed
							No	2-CH-767 [11], 2-CH-766 [11]	Outside	0.75 inch	Globe	2		Manual	–	Locked Open	As Is	No	Open
							No	2-CH-744	Outside	0.75 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
47, 69 70, 71	Pressure Monitoring		IA	Special	IN/OUT	30	No	Penetration 47 2-AC-97 [11] Penetration 70 2-AC-98 [11] Penetration 69 2-AC-99 [11] Penetration 71 2-AC-96 [11]	Outside	0.5 inch	Globe	1	0.5 inch	Manual	–	Open	As Is	Yes	Open
51	Waste Gas Header	RWS	IA	N [12]	OUT	12	Yes	2-GR-11.2	Outside	3 inch	Globe	1	3 inch	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes [4]	2-GR-11.1	Inside	3 inch	Globe	1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-GR-63	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
49	Fire Protection	Fire	IA	O	IN	34	Yes	2-Fire-108	Outside	6 inch	Butterfly	1	6 inch	Manual	–	Locked Closed	As Is	No	Closed
							No	2-Fire-125	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-Fire-109	Inside	6 inch	Check	1		–	–	–	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification	Location Reference to Containment Structure	Valve Size	Valve Type	Valve Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
53	Reactor Building Closed Cooling Water Inlet to Reactor Coolant Pumps and Other Components [8]	RBCCW	IA	N	IN	24	Yes	2-RB-30.1B	Outside	6 inch	Gate	1	6 inch	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-291	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RB-173B [11]	Outside	0.75 inch	Globe	1		Manual	–	Open	As Is	No	Open
54	Reactor Building Closed Cooling Water Outlet from Reactor Coolant Pumps and Other Components [8]	RBCCW	IA	N	OUT	2	Yes	2-RB-37.2B	Outside	6 inch	Gate	1	6 inch	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-300	Outside	1 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RB-299A	Outside	0.75 inch	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 inch	Blind Flange	1		–	–	–	–	No	–
							No, Type B	SF-01	Outside	6 inch	Spectacle Flange	1	6 inch	–	–	–	–	No	–
							No	2-AC-107	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
61 & 86	Containment Air Sample	CAS	IC	O	OUT	26	Yes	Penetration 61 2-AC-12	Penetration 86 2-AC-47	Outside	1.5 inch	Butterfly	1	1 inch	Diaphragm	CIAS	Open	Closed	Yes	Closed[10]
							Yes	2-EB-88	2-EB-89	Inside	1.5 inch	Butterfly	1		Diaphragm	CIAS	Open	Closed	Yes	Closed[10]
							No	2-AC-101	2-AC-102	Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification		Location Reference to Containment Structure	Valve		Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
											Size	Type								
62 & 87	Containment Air Sample	CAS	IC	O	IN	28	Yes	Penetration 62 2-AC-54	Penetration. 87 2-AC-55	Inside	0.5 inch	Check	1	1 inch	–	–	–	–	No	–
							Yes	2-AC-15	2-AC-20	Outside	1.5 inch	Butterfly	1		Diaphragm	CIAS	Open	Closed	Yes	Closed[10]
							No	2-AC-103	2-AC-104	Outside	0.75 inch	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 inch	Gate	1	4 inch	Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-RW-21		Outside	4 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RW-158		Outside	0.75 inch	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 inch	Gate	1	4 inch	Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-RW-63		Outside	4 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-RW-159		Outside	0.75 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
82	Hydrogen Purge	HC	IA	O	OUT	25A	Yes [4]	2-EB-91		Inside	6 inch	Butterfly		6 inch	Air Cylinder	CIAS & High Contain-ment Radiation	Closed	As Is	Yes	Closed
							Yes	2-EB-92		Outside	6 inch	Butterfly	1		Diaphragm	CIAS & High Contain-ment Radiation	Closed	Closed	Yes	Closed
							Yes [4]	2-EB-86		Inside	0.75 inch	Gate	1		Manual	–	Locked Closed	As Is	No	Closed
							No	2-EB-120		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
83	Hydrogen Purge	HC	IA	O	OUT	25B	Yes [4]	2-EB-100		Inside	6 inch	Butterfly	1	6 inch	Air Cylinder	CIAS & High Contain-ment Radiation	Closed	As Is	Yes	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-11 CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION (CONTINUED)

Penetration Number	Service	System	Penetration Type [1]	Penetration Category[2]	Flow Direction	Valve Arrangement	Type C Testing Requirements[3]	Valve Identification		Location Reference to Containment Structure	Valve Size	Valve Type	Number	Penetration Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indication	Post Incident Position
							Yes	2-EB-99		Outside	6 inch	Butterfly	1		Diaphragm	CIAS & High Contain-ment Radiation	Closed	Closed	Yes	Closed
							No	2-EB-121		Outside	0.75 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
65 & 72	Steam Generator Blowdown Sample	SGBS	IA	N	OUT	14	No	Penetration 65 2-MS-191A	Penetration 72 2-MS-191B	Outside	0.5 inch	Globe	1	0.5 inch	Diaphragm	CIAS	Open	Closed	Yes	Closed
63 & 64	Containment Pressure Test Connection	ILRT	IC	O	OUT	31	Yes	Penetration 63 2-AC-114	Penetration. 64 2-AC-112	Outside	1 inch	Globe	1	1 inch	Manual	–	Locked Closed	As Is	No	Closed
							Yes	2-AC-117	2-AC-116	Inside	1 inch	Globe	1		Manual	–	Locked Closed	As Is	No	Closed
							No, Type B	1 inch -Blind Flange TC	1 inch - Blind Flange TC	Outside	1 inch	Blind Flange	2	1 inch	–	–	–	–	No	–

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

NOTES:

1. See Figure 5.2–8.
2. See Section 5.2.7.2.1.
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. Valves 2-MS-65A, 2-MS-65B, 2-MS-202 and 2-SI-651 have disconnect switches in their power circuits to ensure the valves are in their proper position in the event of an Appendix R fire (hot short issue). Once these valves have been positioned and the disconnect switch is placed in the “off” position, the valves will not respond to automated signals. 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 Heat Exchanger.
9. The post incident position 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 opened 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.
11. Valves within containment boundary that are instrument isolation valves or valves not required to go closed or be closed for isolation of the penetration.
12. The allowed outage time of 72 hours as specified in Technical Specification action statement 3.6.3.1.d is applicable for all ‘N’ Type penetrations except penetration numbers 1, 34, 51.

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 3.6-1, Containment Isolation Valve List.

TABLE 5.2-12 CONTAINMENT PENETRATION PIPING

Penetration Number	System	PIPE			Material	Class	Code	Fittings
		Size	Schedule	Line Designation				
1	DWS	2 inch	40S	HCB-4	A-312 TP 304	2	B31.7	3000 pound Socket Weld
2	CVCS	2 inch	160	CCB-5	A-376 TP 316	2	B31.7	6000 pound Socket Weld
3	CVCS	2 inch	160	CCB-6	A-376 TP 316	2	B31.7	6000 pound Socket Weld
4	CSS	8 inch	20	GCB-11	A-312 TP 304	2	B31.7	Butt welded
5	CSS	8 inch	20	GCB-11	A-312 TP 304	2	B31.7	Butt welded
6	SIS	6 inch	120	CCA-6	A-376 TP 316	2	B31.7	Butt welded
7	SIS	6 inch	120	CCA-6	A-376 TP 316	2	B31.7	Butt welded
8	SIS	6 inch	120	CCA-6	A-376 TP 316	2	B31.7	Butt welded
9	SIS	6 inch	120	CCA-6	A-376 TP 316	2	B31.7	Butt welded
10	SIS	12 inch	20	GCB-1	A-376 TP 316	2	B31.7	Butt welded
11	SIS	2 inch	40S	GCB-14	A-312 TP 304	2	B31.7	3000 pound Socket Weld
12	SIS	24 inch	10S	HCB-1	A-312 TP 304	2	B31.7	Butt welded
13	SIS	24 inch	10S	HCB-1	A-312 TP 304	2	B31.7	Butt welded
14	RWS	3 inch	10S	HSB-1	A-312 TP 304	2	B31.7	Butt welded
15	FW	18 inch	60	EBB-6	SA 335 GR P22	2	B31.7	Butt welded
16	FW	18 inch	60	EBB-6	SA 335 GR P22	2	B31.7	Butt welded
19	MSS	34 inch	0.977 inch wall	EBB-2	A-155 GR KCF70	2	B31.7	Butt welded
20	MSS	34 inch	0.977 inch wall	EBB-2	A-155 GR KCF70	2	B31.7	Butt welded

TABLE 5.2-12 CONTAINMENT PENETRATION PIPING (CONTINUED)

Penetration Number	System	PIPE			Material	Class	Code	Fittings
		Size	Schedule	Line Designation				
21	SS	0.75 inch	160	CCB-10	A-376 TP 316	2	B31.7	6000 pound Socket Weld
22	SGBS	2 inch	80	EBB-5	A-106 GR B	2	B31.7	3000 pound Socket Weld
23	SGBS	2 inch	80	EBB-5	A-106 GR B	2	B31.7	3000 pound Socket Weld
24	RBCCW	8 inch	40	HBB-5	A-333 GR 6	2	B31.7	Butt welded
25	RBCCW	10 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
26	RBCCW	10 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
27	RBCCW	10 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
28	RBCCW	10 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
29	RBCCW	8 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
30	RBCCW	10 inch 6 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
31	RBCCW	10 inch 6 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
32	RBCCW	10 inch 6 inch	40	HBB-3	A-333 GR 6	2	B31.7	Butt welded
33	RBCCW	10 inch 6 inch	40	HBB-4	A-333 GR 6	2	B31.7	Butt welded
34	NS	1 inch	40S	GCB-12	A-312 TP 304	2	B31.7	3000 pound Socket Weld
35	RWS	4 inch	10S	HSB-2	A-312 TP 304	2	B31.7	Butt welded
36	IA	0.5 inch	40S	HCB-31	A-312 TP 304	2	B31.7	3000 pound Socket Weld

TABLE 5.2-12 CONTAINMENT PENETRATION PIPING (CONTINUED)

Penetration Number	System	PIPE			Material	Class	Code	Fittings
		Size	Schedule	Line Designation				
37	IA	2 inch	80	HBB-13	A-333 GR 6	2	B31.7	3000 pound Socket Weld
38	SA	2 inch	80	HBB-12	A-333 GR 6	2	B31.7	3000 pound Socket Weld
39	PA	48 inch	0.375 inch wall	HBB-7	A-333 GR 6	2	B31.7	Butt welded
40	PA	48 inch	0.375 inch	HBB-8	A-333 GR 6	2	B31.7	Butt welded
43	CVCS	0.75 inch	160	CCB-9	A-376 TP 316	2	B31.7	6000 pound Socket Weld
49	Fire Protection	6 inch	40	HBB-19	SA-106 GR B	2	ASME	Butt welded
51	RWS	3 inch	40	HRB-1	A-106 GR B	2	B31.7	Butt welded
53	RBCCW	6 inch	40	HBB-5	A-333 GR 6	2	B31.7	Butt welded
54	RBCCW	6 inch	40	HBB-6	A-333 GR 6	2	B31.7	Butt welded
61	CAS	1 inch	40S	HCB-9	A-312 TP 304	2	B31.7	3000 pound Socket Weld
62	CAS	1 inch	40S	HCB-9	A-312 TP 304	2	B31.7	3000 pound Socket Weld
65	SBGS	0.5 inch	80	EBB-8	A-106 GR B	2	B31.7	3000 pound Socket Weld
67	SFPCS	4 inch	10S	HCB-10	A-312 TP 304	2	B31.7	Butt welded
68	SFPCS	4 inch	10S	HCB-11	A-312 TP 304	2	B31.7	Butt welded
72	SGBS	0.5 inch	80	EBB-8	A-106 GR B	2	B31.7	3000 pound Socket Weld
82	H_2 Purge	6 inch	40	HBB-10	A-333 GR 6	2	B31.7	Butt welded
83	H_2 Purge	6 inch	40	HBB-10	A-333 GR 6	2	B31.7	Butt welded

TABLE 5.2-12 CONTAINMENT PENETRATION PIPING (CONTINUED)

Penetration Number	System	PIPE			Material	Class	Code	Fittings
		Size	Schedule	Line Designation				
85	Leak Rate Test	6 inch	40	HBB-1	A-333 GR 6	2	B31.7	Butt welded
86	CAS	1 inch	40S	HCB-9	A-312 TP 304	2	B31.7	3000 pound Socket Weld
87	CAS	1 inch	40S	HCB-9	A-312 TP 304	2	B31.7	3000 pound Socket Weld

Penetration 15 - The double flued head penetration and a portion of piping between the penetration and the containment isolation valve are Carbon Steel (SA-105 and SA-234 Grade WPB, respectively).

Penetration 16 - The double flued head penetration is Carbon Steel (SA-105).

Penetration 17 - Personnel Airlock.

Penetration 18 - Equipment Hatch.

Penetration 42 - Fuel Transfer Tube.

Penetrations 41, 46, 48, 50, 52, 55, 56, 57, 58, 59, 60, 66, 73, 74, 78, 79, 80, 81, 84, 88, 89 - SPARE.

Penetrations 47, 69, 70, 71 - Containment Pressure Transmission.

Penetration 75 - Electrical

Penetrations 63, 64 - Test Connection.

Penetrations 6, 7, 8, 9 were constructed to Class 1 requirements but are considered Class 2 piping upstream of valves 2-SI-706A, B, C, D.

TABLE 5.2-13 MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL [2]	CODE	NUCLEAR CLASS	RATING
2-PMW-43	1	2 inch	Globe	A351, GR CF8M	*	II	150 pound
2-PMW-3	1	2 inch	Check	A182, GR F-316	*	II	600 pound
2-CH-006	2	2 inch	Gate	A182, GR F-316	*	II	1500 pound
2-CH-516	2	3 inch	Globe	A182, GR F-316	*	II	1500 pound
2-CH-089	2	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-CH-515	2	3 inch	Globe	A182, GR F-316	*	II	1500 pound
2-CH-434	3	2 inch	Gate	A182, GR F-316	*	II	1500 pound
2-CH-429	3	2 inch	Gate	A182, GR F-316	*	II	1500 pound
2-CH-518	3	2 inch	Globe	A351, GR CF-8	*	II	1500 pound
2-CH-519	3	2 inch	Globe	A351, GR CF-8	*	II	1500 pound
2-CH-517	3	2 inch	Globe	A351, GR CF-8	*	II	1500 pound
2-CH-435	3	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-CS-5A	4	8 inch	Check	A351, GR CF-8	*	II	300 pound
2-CS-4.1A	4	8 inch	Gate	A351, GR CF-8	*	II	300 pound
2-CS-5B	5	8 inch	Check	A351, GR CF-8	*	II	300 pound
2-CS-4.1B	5	8 inch	Gate	A351, GR CF-8	*	II	300 pound
2-SI-645	6	6 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-144	6	6 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-646	6	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-647	6	2 inch	Globe	A182, GR F-316	*	II	1500 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-SI-009	6	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-143	6	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-706D	6	6 inch	Check	A351, GR CF-8	*	I	1500 pound
2-SI-160	6	3 inch	Gate	SA351, GR CF8M	*	II	1500 pound
2-SI-161	6	3 inch	Gate	SA351, GR CF8M	*	II	1500 pound
2-SI-615	7	6 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-114	7	6 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-616	7	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-617	7	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-012	7	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-113	7	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-706A	7	6 inch	Check	A351, GR CF-8	*	I	1500 pound
2-SI-636	8	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-637	8	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-010	8	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-133	8	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-706C	8	6 inch	Check	A351, GR CF-8	*	I	1500 pound
2-SI-635	8	6 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-134	8	6 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-625	9	6 inch	Globe	A182, GR F-316	*	II	1500 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-SI-124	9	6 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-626, 2-SI-627	9	2 inch	Globe	A182, GR F-316	*	II	1500 pound
2-SI-123	9	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-011	9	2 inch	Check	A182, GR F-316	*	II	1500 pound
2-SI-706B	9	6 inch	Check	A351, GR CF-8	*	I	1500 pound
2-SI-709	10	12 inch	Gate	A351, GR CF-8	*	I	1500 pound
2-SI-651	10	12 inch	Gate	A182, GR F-316	*	I	1500 pound
2-SI-463	11	2 inch	Gate	A182, GR F-316	*	II	CL800
2-CS-16.1A	12	24 inch	Gate	A351, GR CF-8	*	II	150 pound
2-CS-16.1B	13	24 inch	Gate	A351, GR CF-8	*	II	150 pound
2-SSP-16.2	14	3 inch	Globe	A351, GR CF8M	*	II	150 pound
2-SSP-16.1	14	3 inch	Globe	A351, GR CF8M	*	II	150 pound
2-FW-5A	15	18 inch	Check	A216, GR WCB	*	II	600 pound
2-FW-12A	15	6 inch	Check	A216, GR WCB	*	II	600 pound
2-FW-5B	16	18 inch	Check	A216, GR WCB	*	II	600 pound
2-FW-12B	16	6 inch	Check	A216, GR WCB	*	II	600 pound
2-MS-64A	19	34 inch	Check	A216, GR WCB	*	II	600 pound
2-MS-201	19	4 inch	Gate	A105, GR 2	*	II	600 pound
2-MS-3A	19	12 inch	Gate	A105, GR 2	*	II	600 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-MS-190A	19	8 inch	Globe	A216, GR WCB	*	II	600 pound
2-MS-265B	19	1 inch	Globe	A182, GR F-316	*	II	600 pound
2-MS-65A	19	3 inch	Globe	A105, GR 2	*	II	600 pound
2-MS-64B	20	34 inch	Check	A216, GR WCB	*	II	600 pound
2-MS-202	20	4 inch	Gate	A105, GR 2	*	II	600 pound
2-MS-3B	20	12 inch	Gate	A105, GR 2	*	II	600 pound
2-MS-190B	20	8 inch	Globe	A216, GR WCB	*	II	600 pound
2-MS-266B	20	1 inch	Globe	A182, GR F-316	*	II	600 pound
2-MS-65B	20	3 inch	Globe	A105, GR 2	*	II	600 pound
2-LRR-61.1	21	0.5 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-RC-001	21	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-RC-002	21	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-RC-003	21	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-RC-45	21	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-MS-220A	22	2 inch	Globe	A216, GR WCB	*	II	600 pound
2-MS-220B	23	2 inch	Globe	A216, GR WCB	*	II	600 pound
2-RB-30.1A	24	8 inch	Gate	A350, GR LF1	*	II	150 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-RB-28.1D	25	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.1B	26	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.1A	27	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.1C	28	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-37.2A	29	8 inch	Gate	A350, GR LF1	*	II	150 pound
2-RB-28.3D	30	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.2D	30	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.3B	31	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.2B	31	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.3A	32	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.2A	32	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.3C	33	10 inch	Butterfly	A516, GR 70	*	II	150 pound
2-RB-28.2C	33	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-SI-312	34	0.75 inch	Globe	A182, GR F-316	*	II	150 pound
2-LRR-43.1	35	3 inch	Globe	A351, GR CF8M	*	II	150 pound
2-LRR-43.2	35	3 inch	Globe	A351, GR CF8M	*	II	150 pound
2-IA-569	36	0.5 inch	Check	A182, GR F-316	ASME 1983	II	600 pound
2-IA-566	36	0.5inch	Gate	A182, GR F-316	ASME 1983	II	600 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-IA-27.1	37	2 inch	Globe	A216, GR WCB	*	II	600 pound
2-IA-43	37	2 inch	Check	A216, GR WCB	*	II	600 pound
2-SA-19	38	2 inch	Gate	A105, GR 2	*	II	600 pound
2-SA-22	38	2 inch	Check	A105, GR 2	*	II	600 pound
2-AC-4	39	48 inch	Butterfly	A516, GR 70	*	II	150 pound
2-AC-5	39	48 inch	Butterfly	A516, GR 70	*	II	150 pound
2-AC-6	40	48 inch	Butterfly	A516, GR 70	*	II	150 pound
2-AC-7	40	48 inch	Butterfly	A516, GR 70	*	II	150 pound
2-CH-506	43	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-CH-198	43	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-CH-505	43	0.75 inch	Globe	A351, GR CF8M	*	II	2500 pound
2-FIRE-108	49	6 inch	Butterfly	A216, GR WCB	ASME 1977	II	150 pound
2-FIRE-109	49	6 inch	Check	A216, GR WCB	ASME 1977	II	150 pound
2-GR-11.2	51	3 inch	Globe	A216, GR WCB	*	II	150 pound
2-GR-11.1	51	3 inch	Globe	A216, GR WCB	*	II	150 pound
2-RB-30.1B	53	6 inch	Gate	A105, GR 2	*	II	150 pound

TABLE 5.2-13 MAJOR(1) CONTAINMENT ISOLATION VALVES (CONTINUED)

VALVE ID NUMBER	PENETRATION NUMBER	SIZE	VALVE TYPE	BODY MATERIAL (2)	CODE	NUCLEAR CLASS	RATING
2-RB-37.2B	54	6 inch	Gate	A105, GR 2	*	II	150 pound
2-AC-12	61	1.5 inch	Butterfly	A515, GR 70	*	II	300 pound
2-EB-88	61	1.5 inch	Butterfly	A515, GR 70	*	II	300 pound
2-AC-15	62	1.5 inch	Butterfly	A515, GR 70	*	II	300 pound
2-AC-54	62	0.5 inch	Check	A182, GR F-316	*	II	600 pound
2-MS-191A	65	0.5 inch	Globe	A216, GR WCB	*	II	600 pound
2-RW-232	67	4 inch	Gate	A182, GR F-316	*	II	150 pound
2-RW-21	67	4 inch	Gate	A182, GR F-316	*	II	150 pound
2-RW-154	68	4 inch	Gate	A182, GR F-316	*	II	150 pound
2-RW-63	68	4 inch	Gate	A182, GR F-316	*	II	150 pound
2-MS-191B	72	0.5 inch	Globe	A216, GR WCB	*	II	600 pound
2-EB-91	82	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-EB-92	82	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-EB-100	83	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-EB-99	83	6 inch	Butterfly	A516, GR 70	*	II	150 pound
2-AC-47	86	1.5 inch	Butterfly	A515, GR 70	*	II	150 pound
2-EB-89	86	1.5 inch	Butterfly	A515, GR 70	*	II	150 pound
2-AC-20	87	1.5 inch	Butterfly	A515, GR 70	*	II	150 pound
2-AC-55	87	0.5 inch	Check	A182, GR F-316	*	II	600 pound

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.

* ASME Section III, 1971, or Draft ASME Pump and Valve Code Penetration numbers 41, 46, 48, 50, 52, 55, 56, 57, 58, 59, 60, 66, 73, 74, 78, 79, 80, 81, 84, 88, 89 -
SPARE Penetration number 85 - Leak Rate Testing Penetration numbers 63, 64 - Test Connection.

TABLE 5.2-14 TYPICAL LEAK RATE MEASUREMENT SYSTEM INSTRUMENTATION

QUANTITY	DESCRIPTION
18	Temperature Monitoring System: Resistance Temperature Detector; Accuracy: ± 0.5°F; Sensitivity: ± 0.1°F.
6	Dewpoint Temperature Monitoring System: Accuracy: ± 2°F; Sensitivity: ± 0.5°F.
2	Flowmeters: Mass Flow Meters; Accuracy: ± 2.0% full scale; Sensitivity: ± 1.0% full scale.
2	Pressure Monitoring: Precision Pressure Gages; Accuracy: ± 0.02% of reading; Sensitivity: ± 0.001 psi.

NOTE: Instrumentation listed above are typically used during an ILRT. The instrumentation parameters are obtained from ANS 56.8 and are used as guidance in selecting acceptable instruments. Appropriate alternatives to the above instrumentation can be used for the ILRT.

TABLE 5.2-15 TYPICAL CONTAINMENT RESISTANCE TEMPERATURE DETECTORS AND DEWCELL SENSOR VOLUME WEIGHT FRACTIONS

RTD	Elevation (feet)	AZ (Degrees)	Distance From Centerline (feet)	Volume Fraction
9769	150	90	12	0.127
8110	95	220	65	0.091
9767	95	40	65	0.091
8111	90	310	60	0.091
8112	90	130	60	0.091
8084	40	5	45	0.071
8108	44	135	60	0.071
8109	44	265	60	0.071
8097	30	95	20	0.026
8098	30	235	20	0.020
8094	20	350	45	0.029
9770	18	220	55	0.028
9771	18	90	50	0.028
8087	3	5	32	0.020
9765	3	240	65	0.020
9766	3	125	65	0.021
8091	-15	330	35	0.052
9768	-10	115	50	0.052
			TOTAL 1.000	

Dewcells	Elevation (feet)	AZ (Degrees)	Distance From Centerline (feet)	Volume Fraction
8101	105	40	45	0.245
8090	105	220	45	0.245
5458	45	300	45	0.150
8102	45	120	45	0.150
8093	-4	220	45	0.105
5457	-4	40	45	0.105
			TOTAL 1.000	

FIGURE 5.2–1 CONTAINMENT STRUCTURE DETAILS

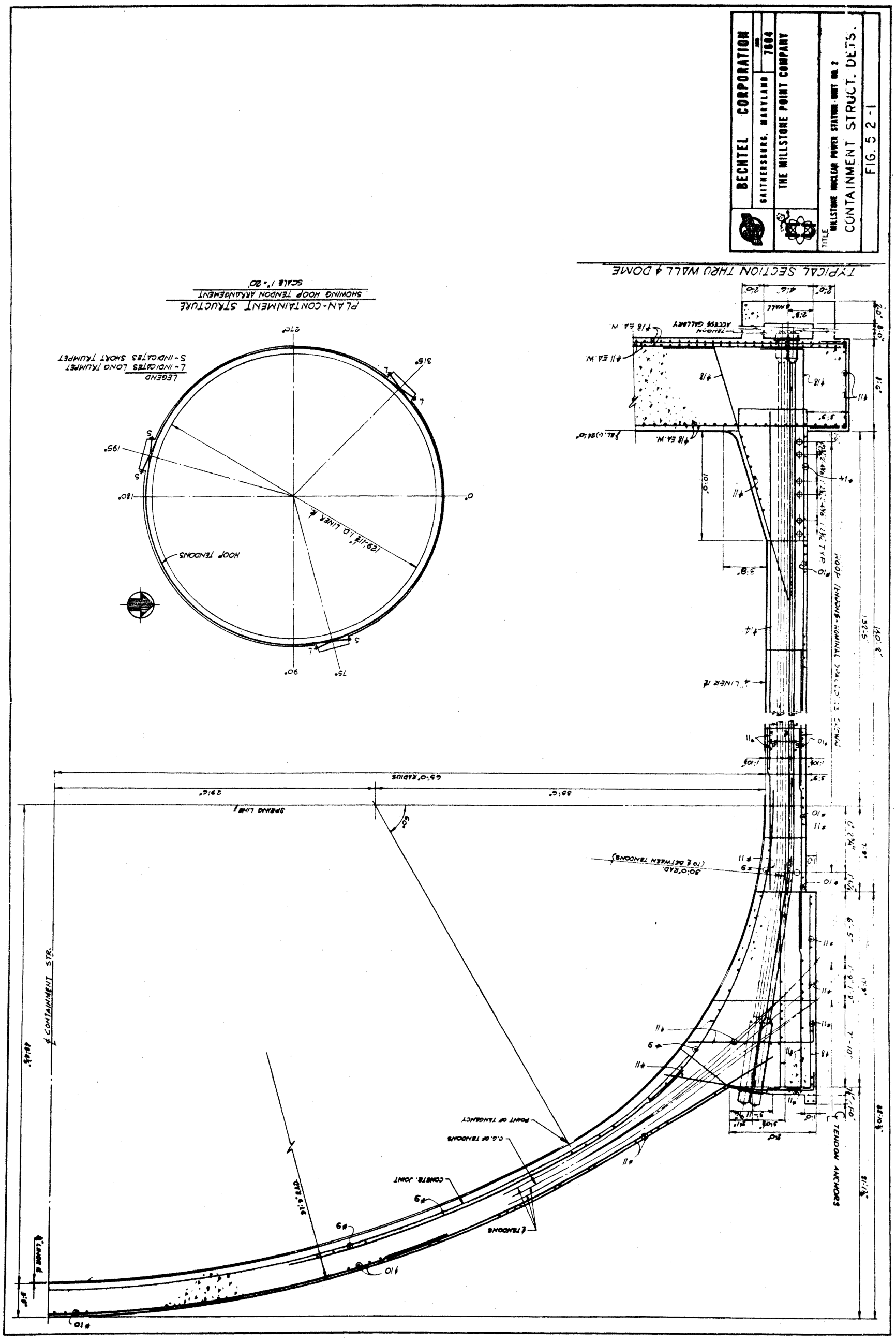
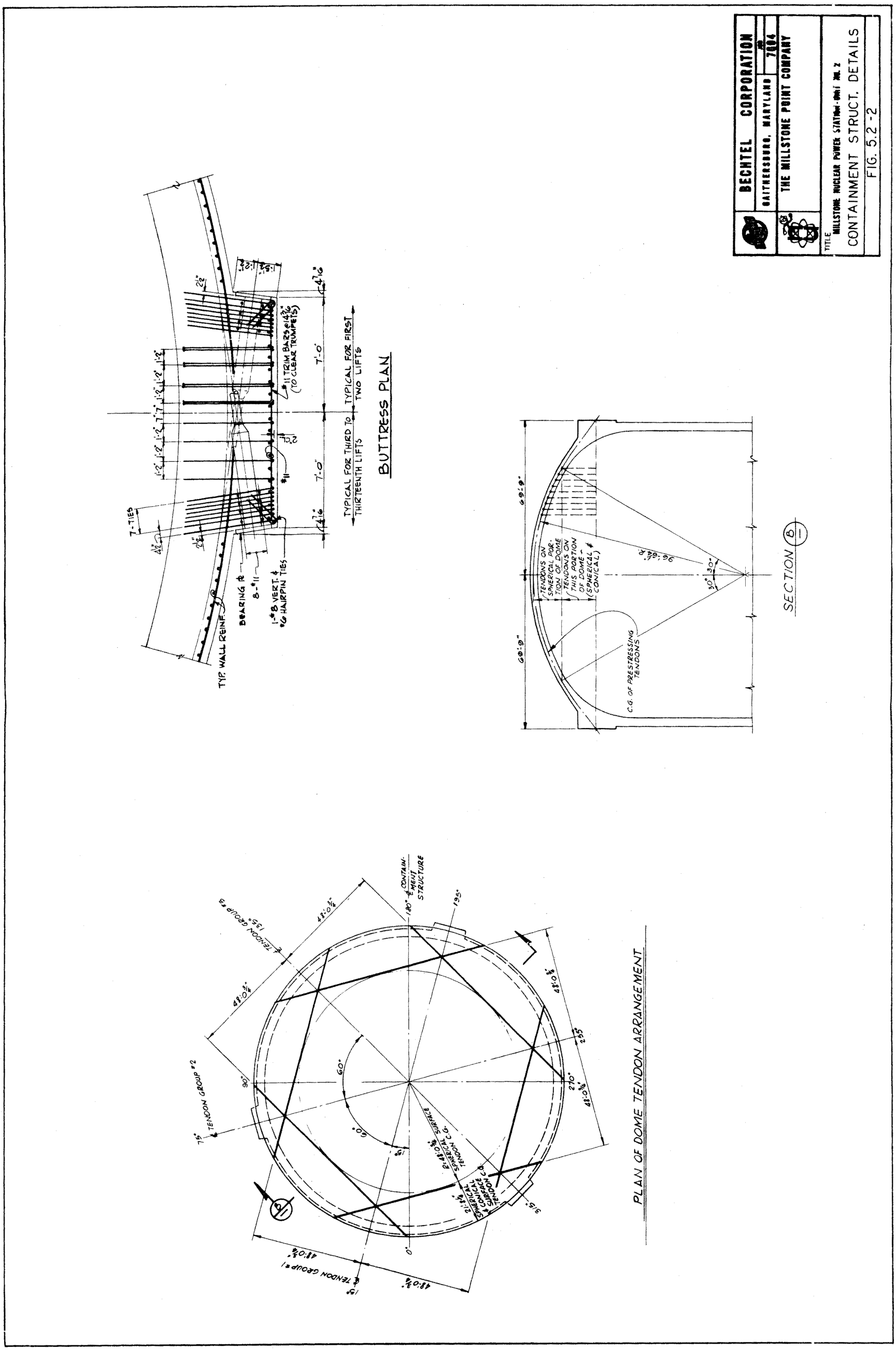


FIGURE 5.2–2 CONTAINMENT STRUCTURE DETAILS



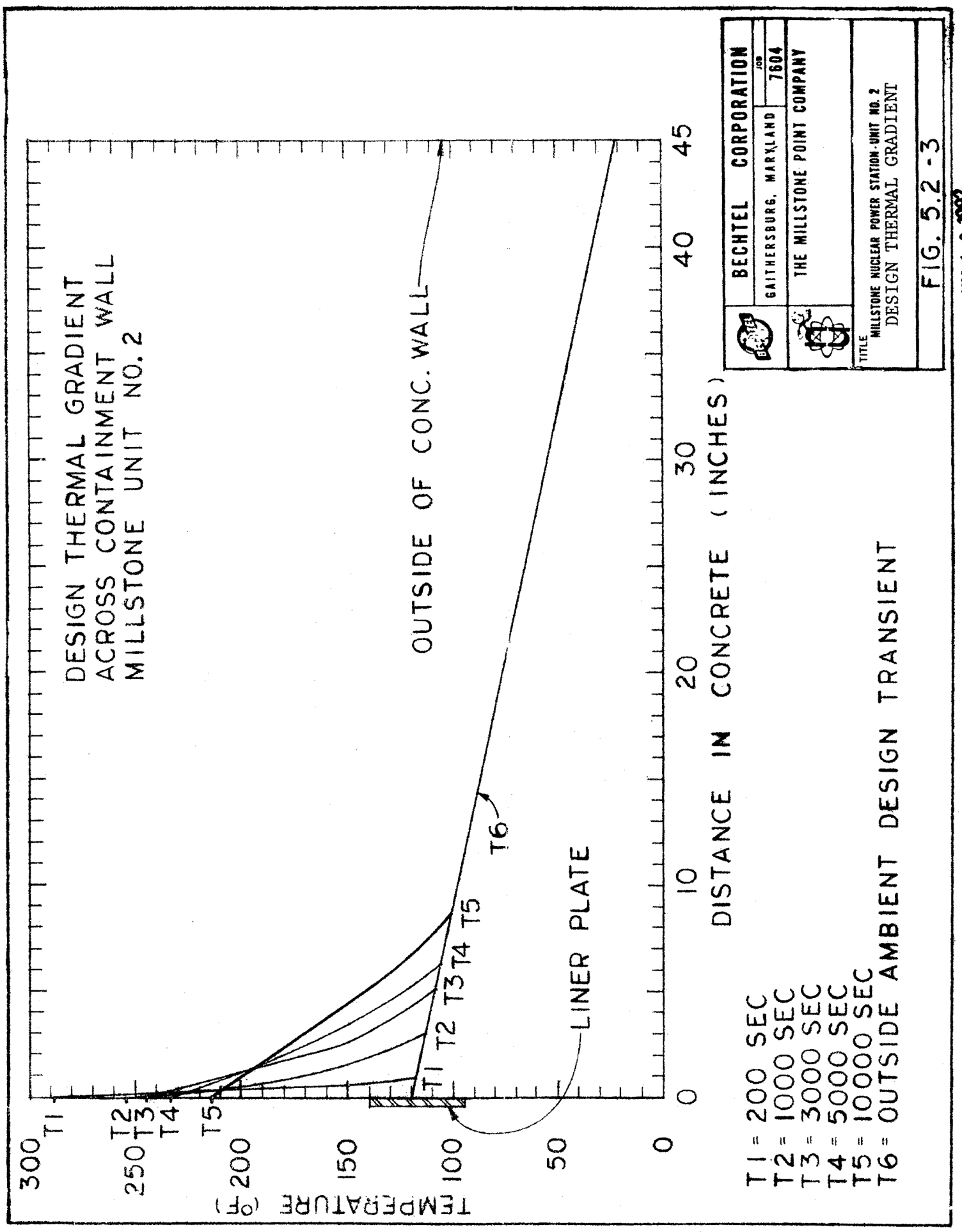


FIGURE 5.2–3 DESIGN THERMAL GRADIENT

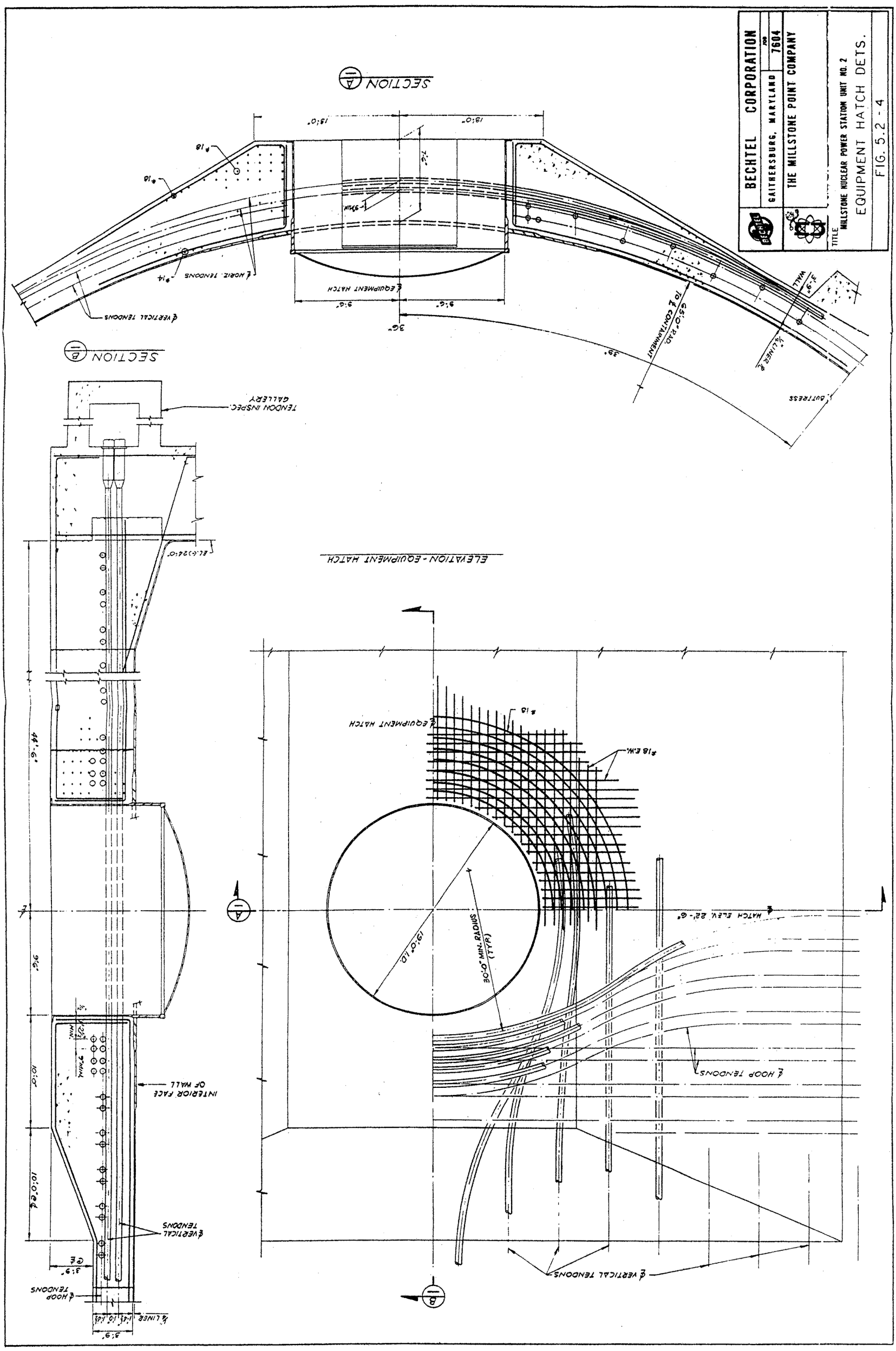


FIGURE 5.2–4 EQUIPMENT HATCH DETAILS

FIGURE 5.2–5 PERSONNEL LOCK DETAILS

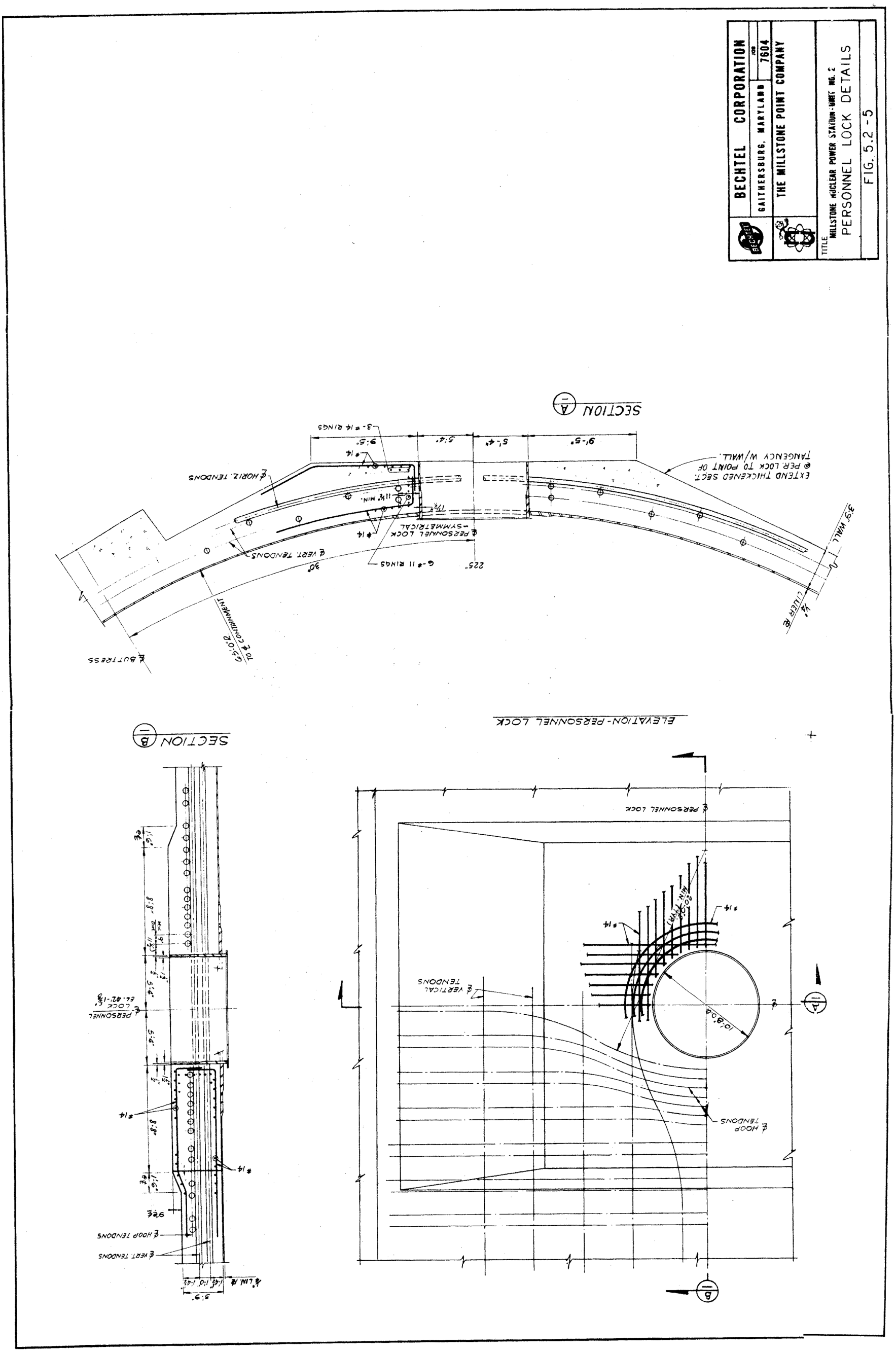


FIGURE 5.2–6 LINER PLATE

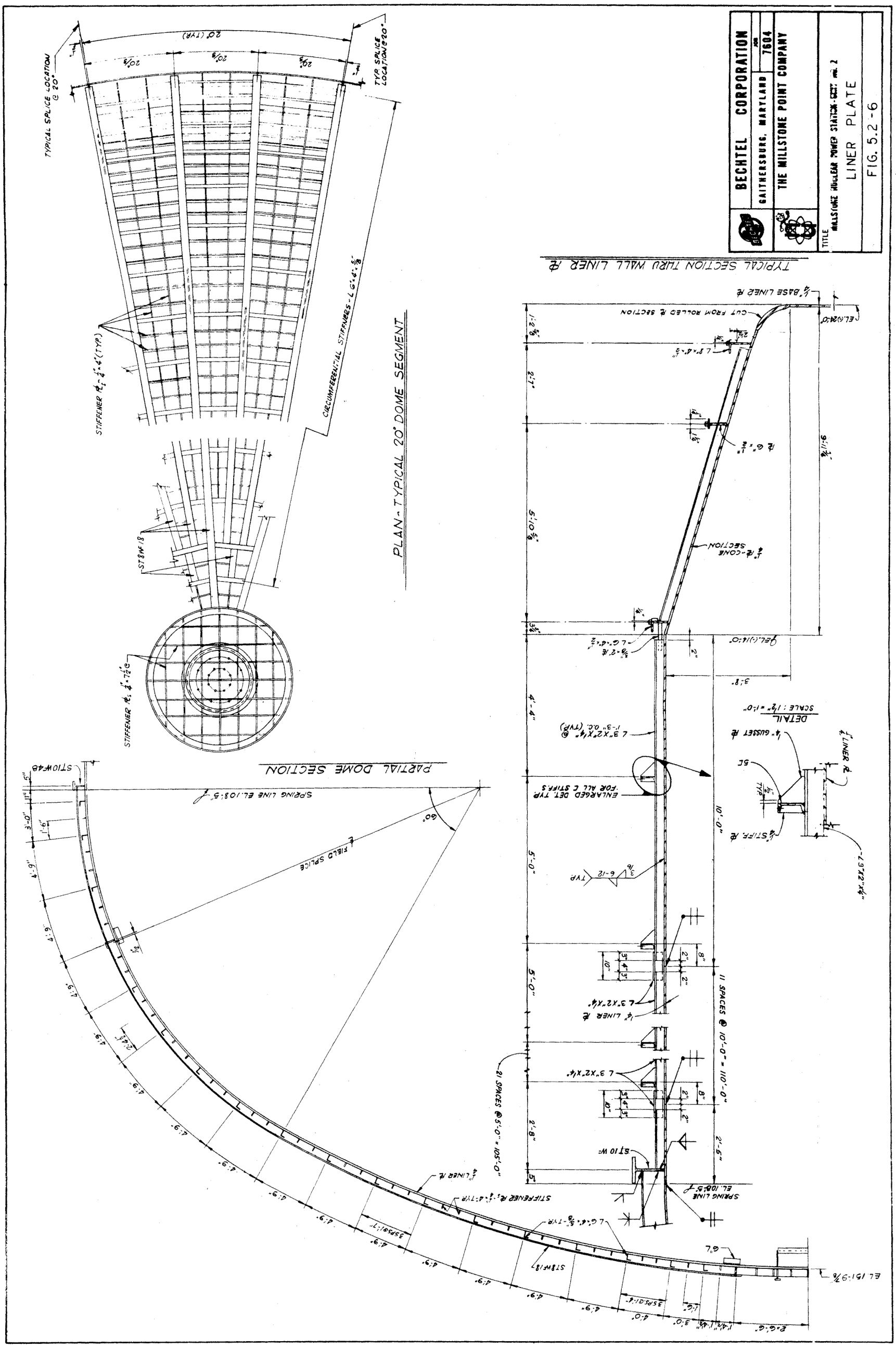


FIGURE 5.2–7 LEAK CHASE CHANNELS

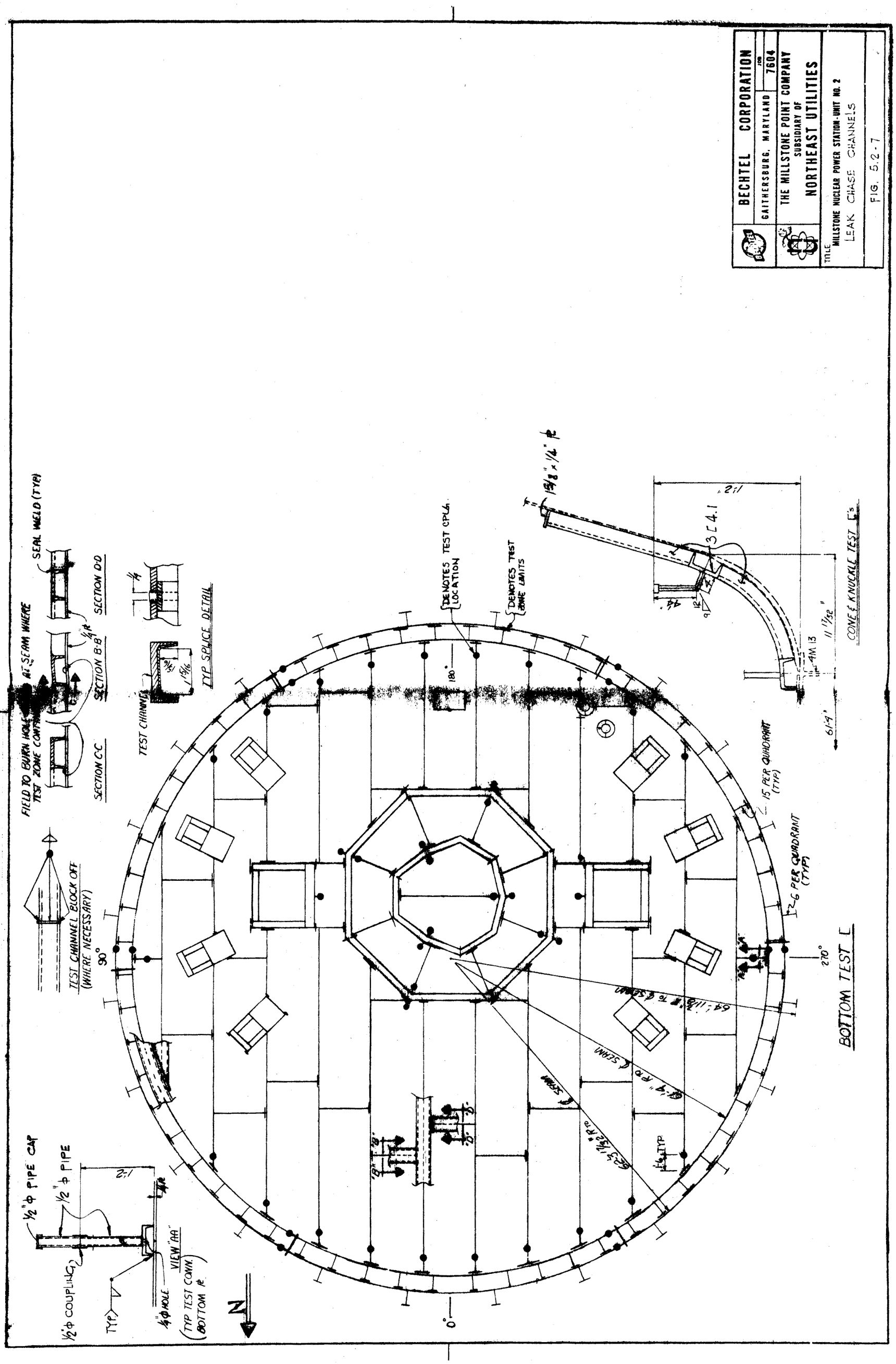


FIGURE 5.2–8 TYPICAL PENETRATIONS

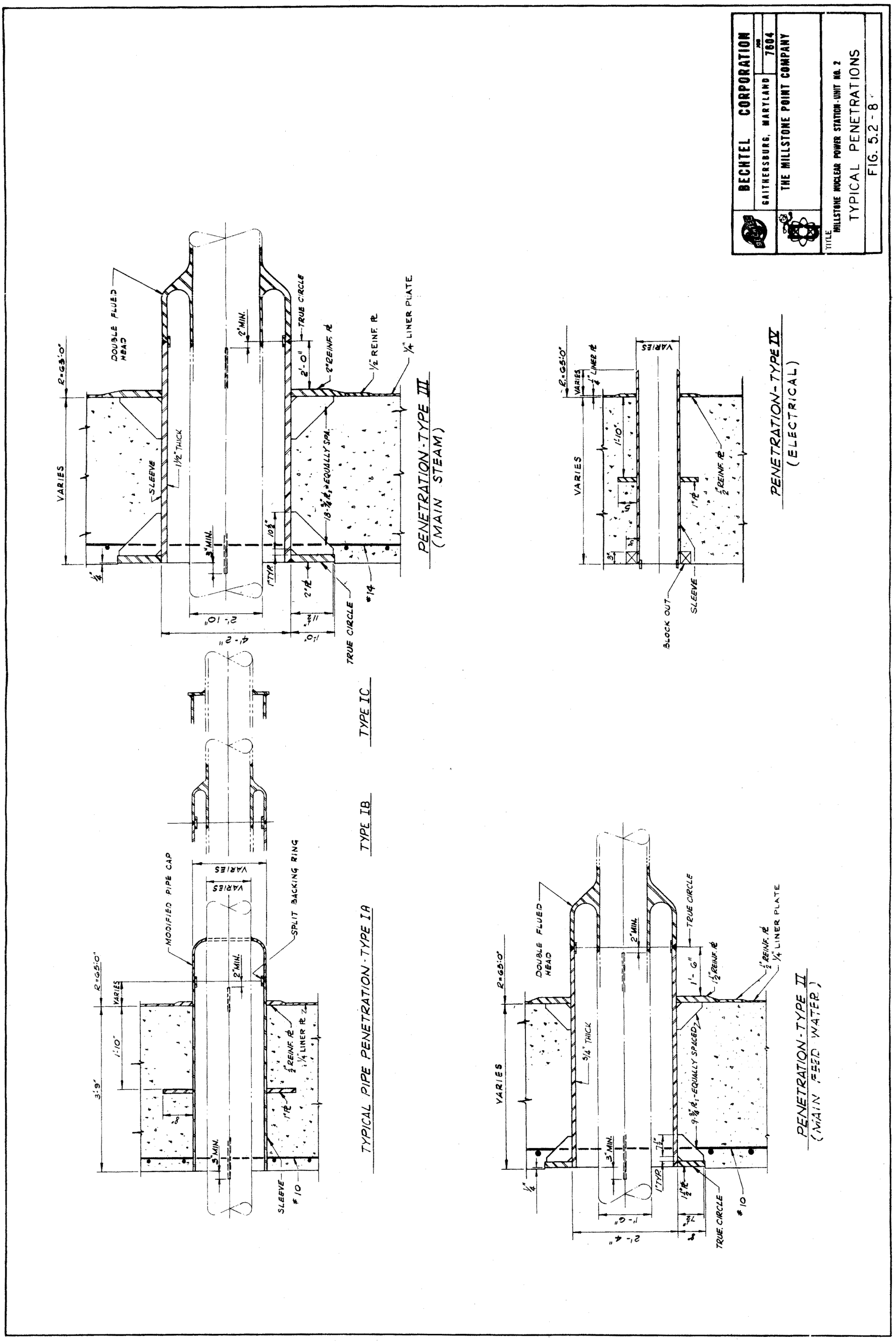
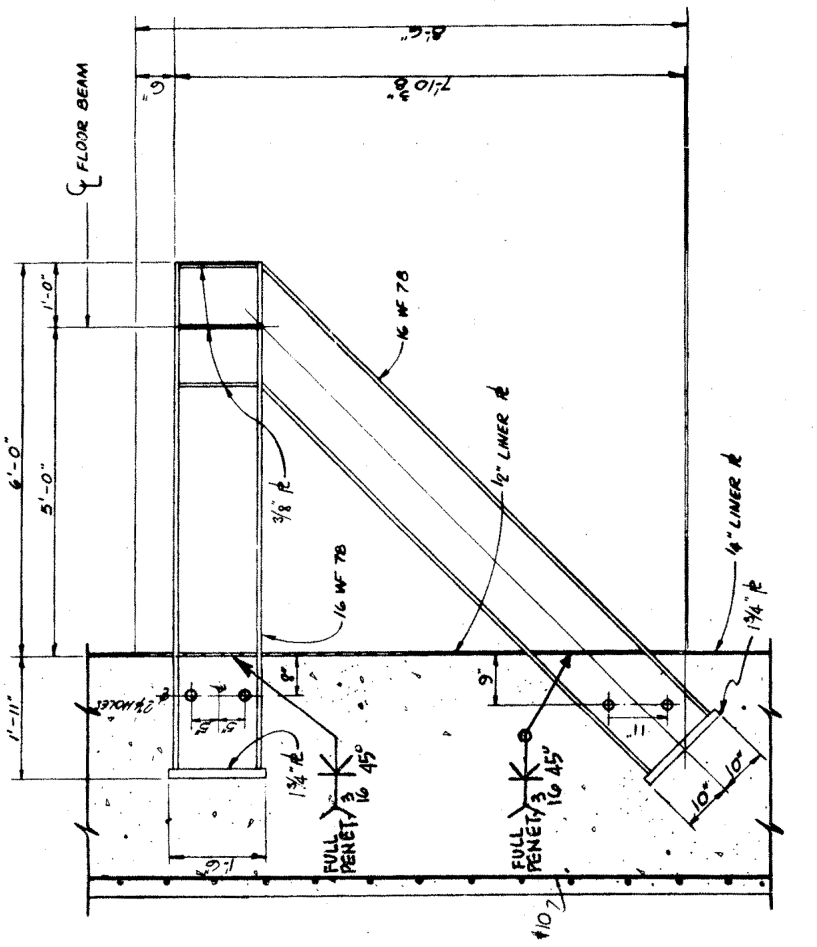


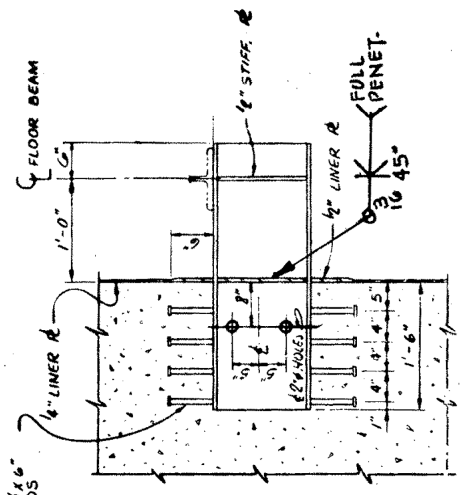
FIGURE 5.2–9 BRACKET DETAILS

BECHTEL CORPORATION
GAITHERSBURG, MARYLAND
JOB 7604
THE MILLSTONE POINT COMPANY
TITLE
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BRACKET DETAILS
FIG. 5.2-9

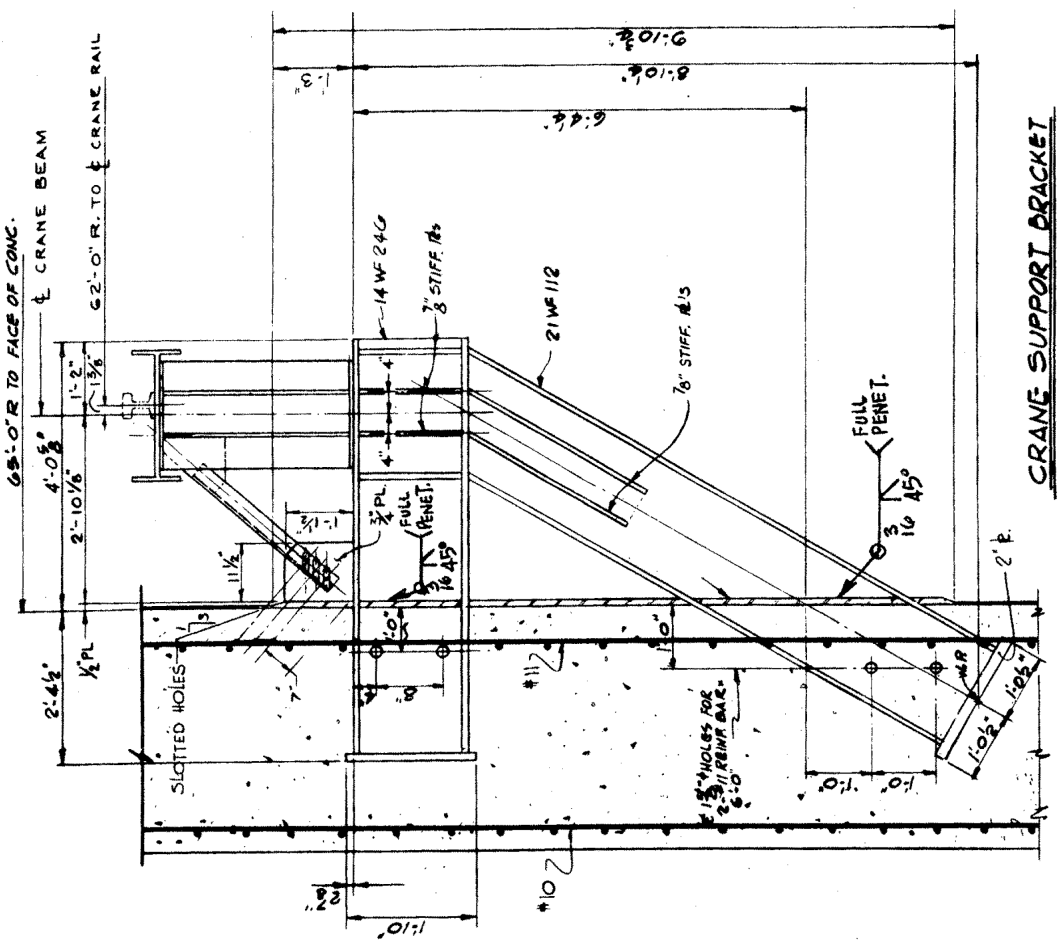
JUN 10 1982



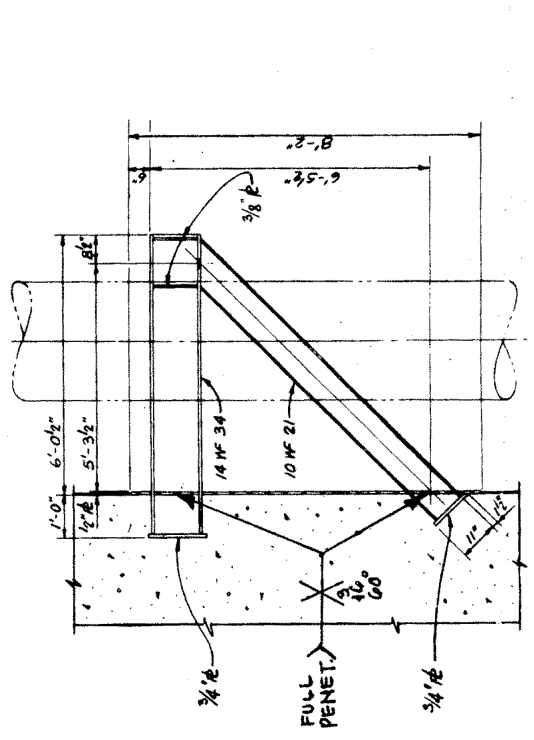
FLOOR BEAM SUPPORT BRACKET



FLOOR BEAM SUPPORT BRACKET



CRANE SUPPORT BRACKET



PIPE SUPPORT — TYPICAL

68-01-50

BECHTEL CORPORATION
GAITHERSBURG, MARYLAND 7604
THE MILLSTONE POINT COMPANY
TITLE: MILLSTONE NUCLEAR POWER STATION - UNIT NO. 2
LINER PLATE DETAILS
FIG. 5.2-10

PERSONNEL LOCK DETAIL

DETAIL SECTION AT EQUIPMENT HATCH

PLAN OF REFUELING TUBE

FIGURE 5.2–10 LINER PLATE DETAILS

FIGURE 5.2–11 REACTOR VESSEL SUPPORT DETAILS

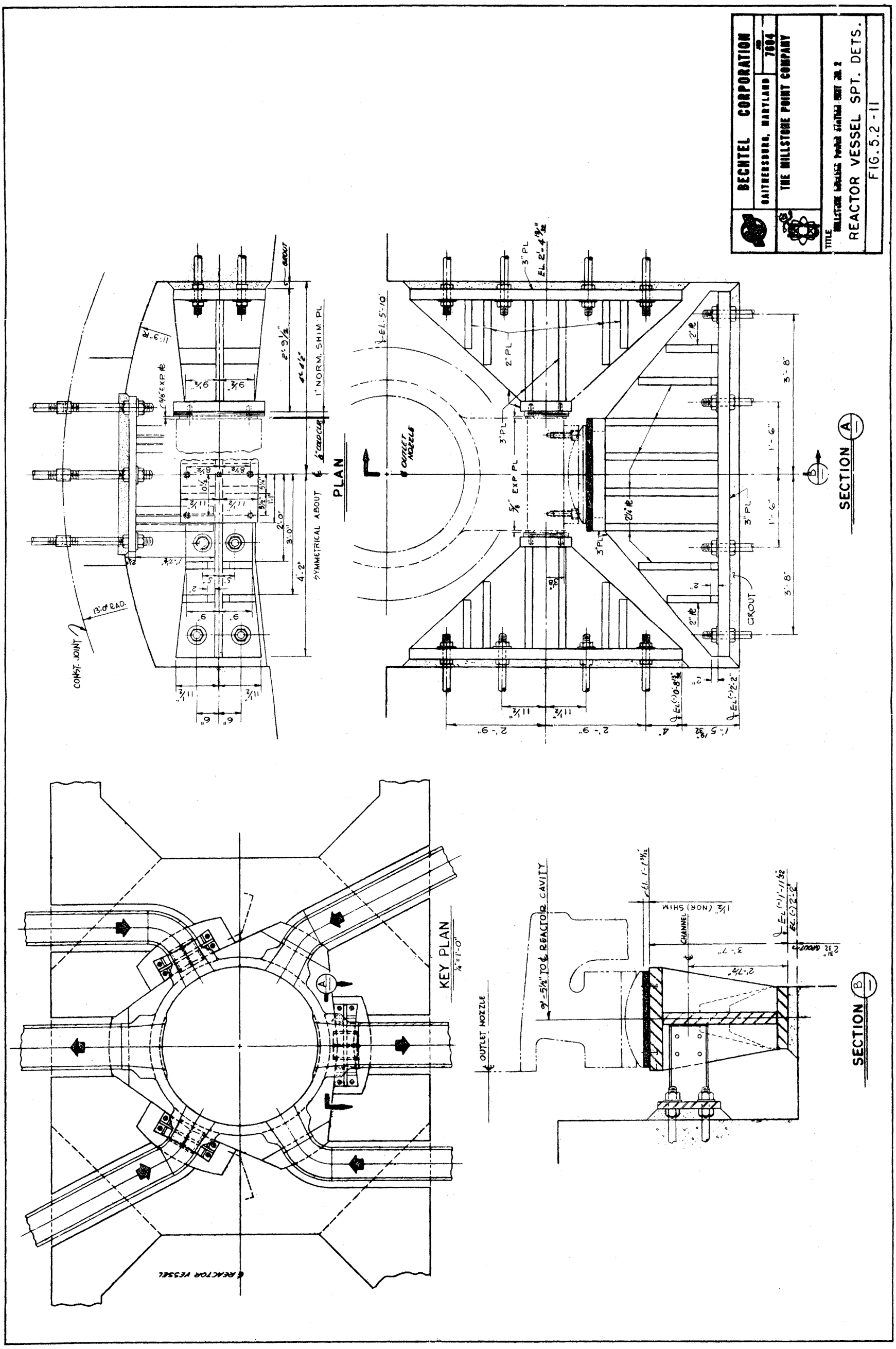


FIGURE 5.2–12 LOWER STEAM GENERATOR SUPPORT DETAILS

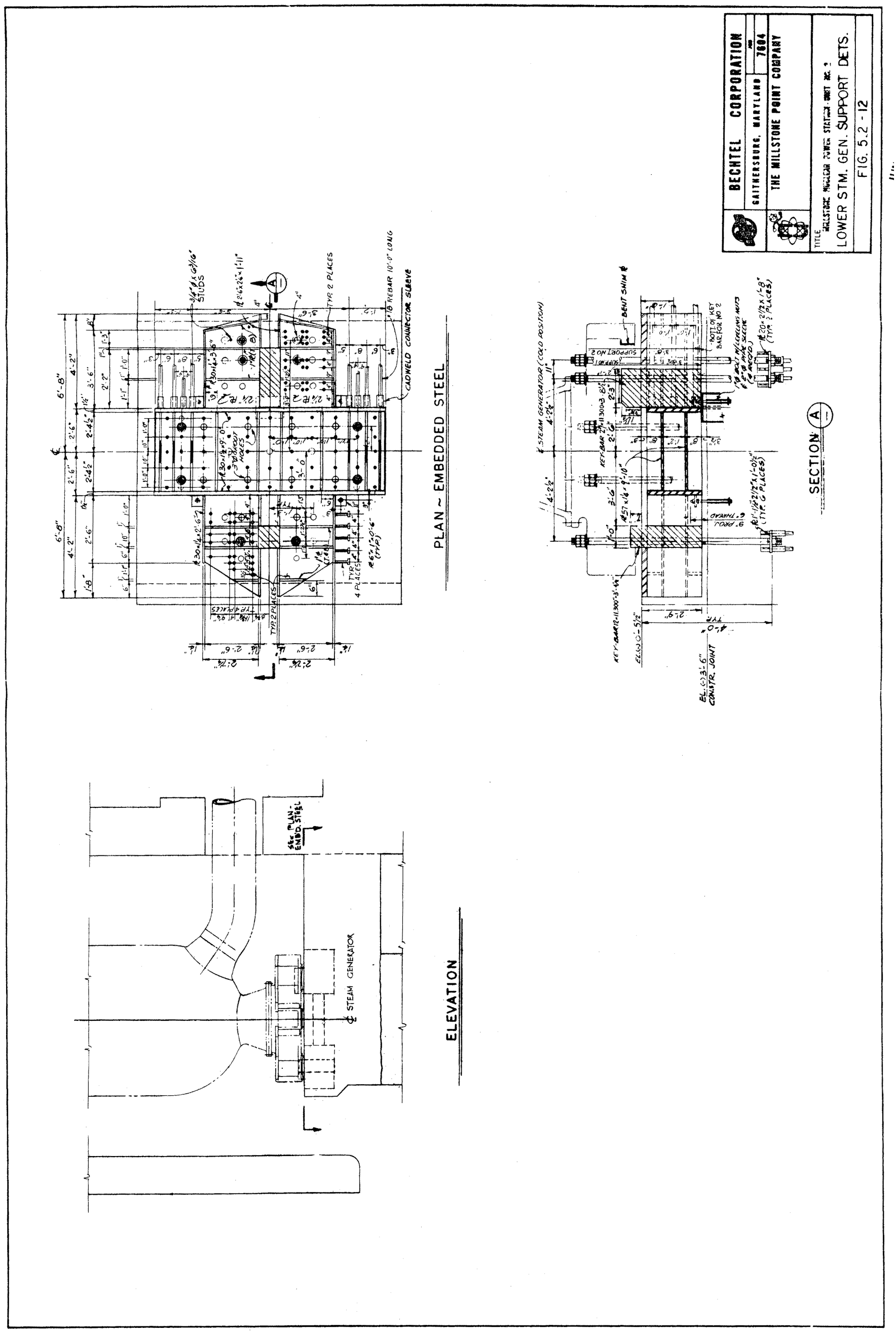
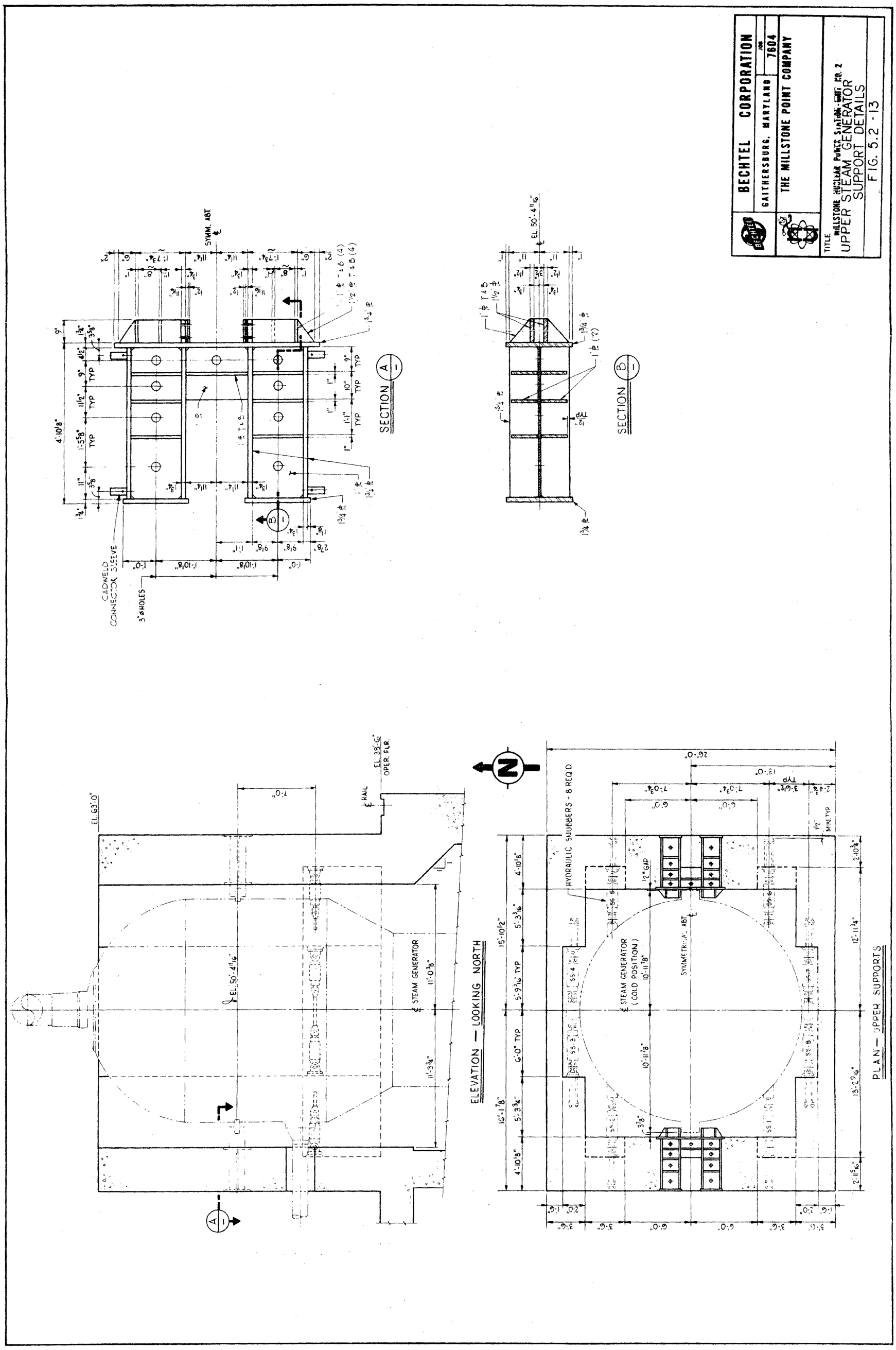


FIGURE 5.2–13 UPPER STEAM GENERATOR SUPPORT DETAILS



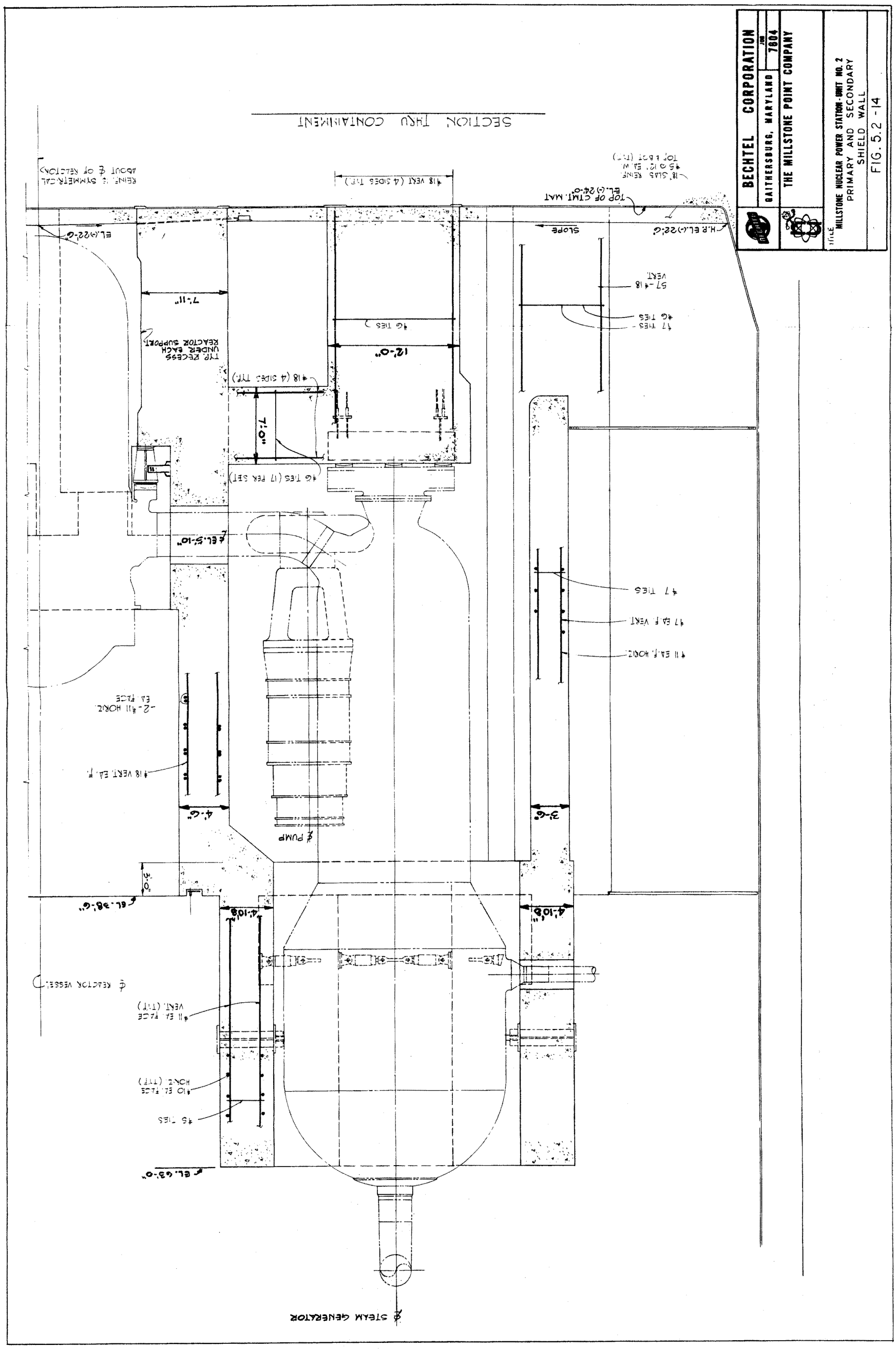


FIGURE 5.2–14 PRIMARY AND SECONDARY SHIELD WALL

FIGURE 5.2–15 DETAIL - SEISMIC RESTRAINT

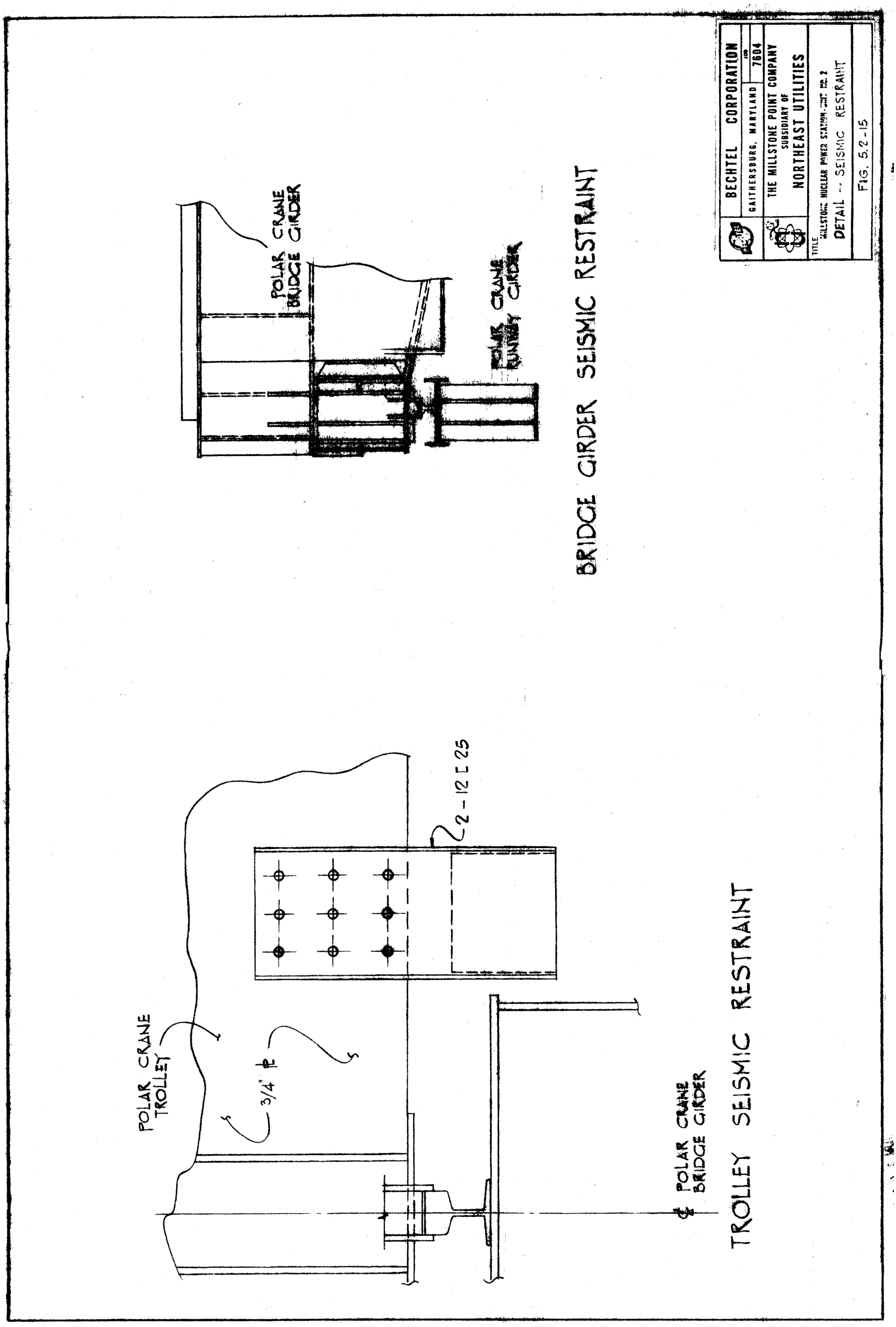


FIGURE 5.2–16 FINITE ELEMENT MESH OF BOTTOM OF CONTAINMENT SHELL

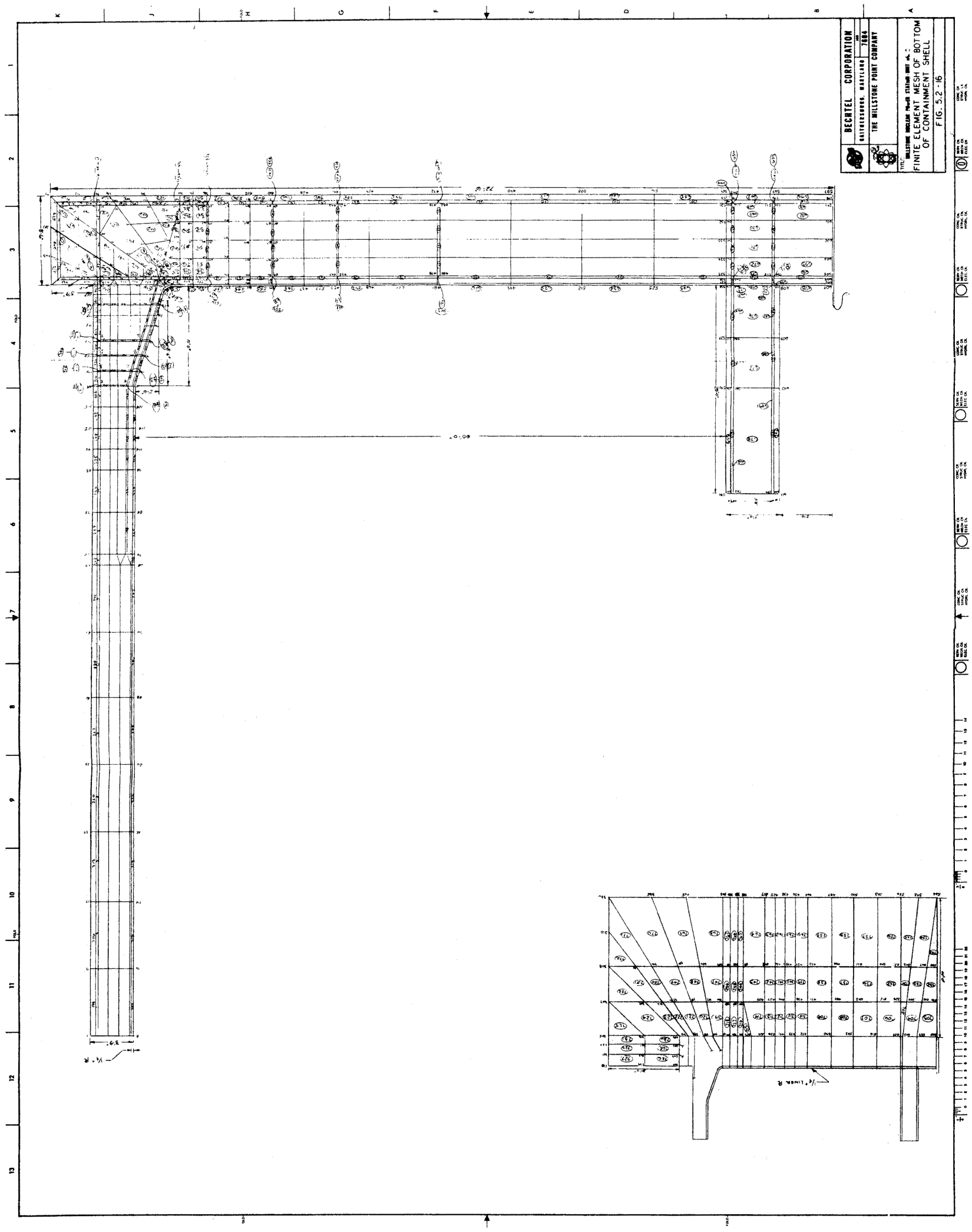
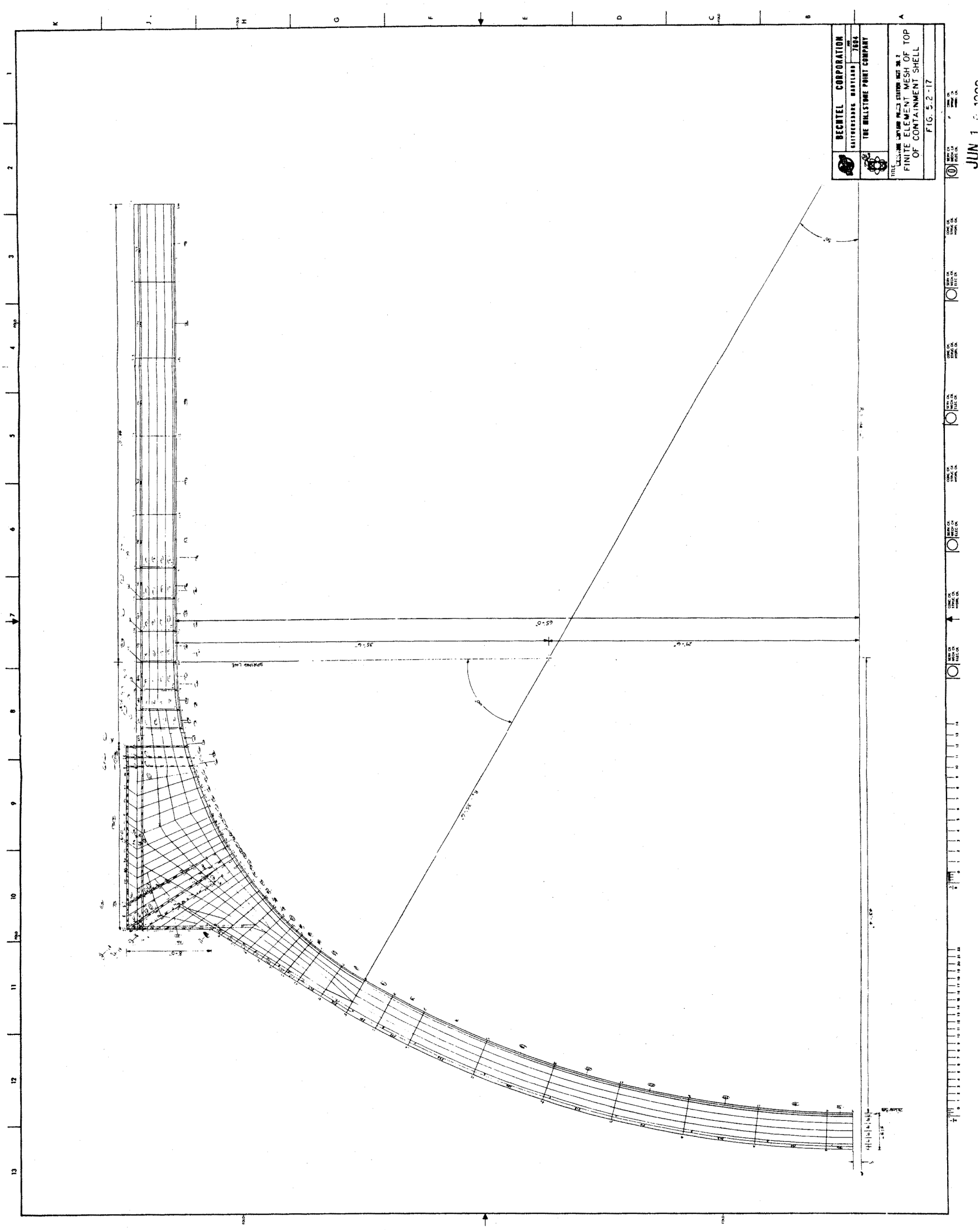


FIGURE 5.2–17 FINITE ELEMENT MESH OF TOP OF CONTAINMENT SHELL



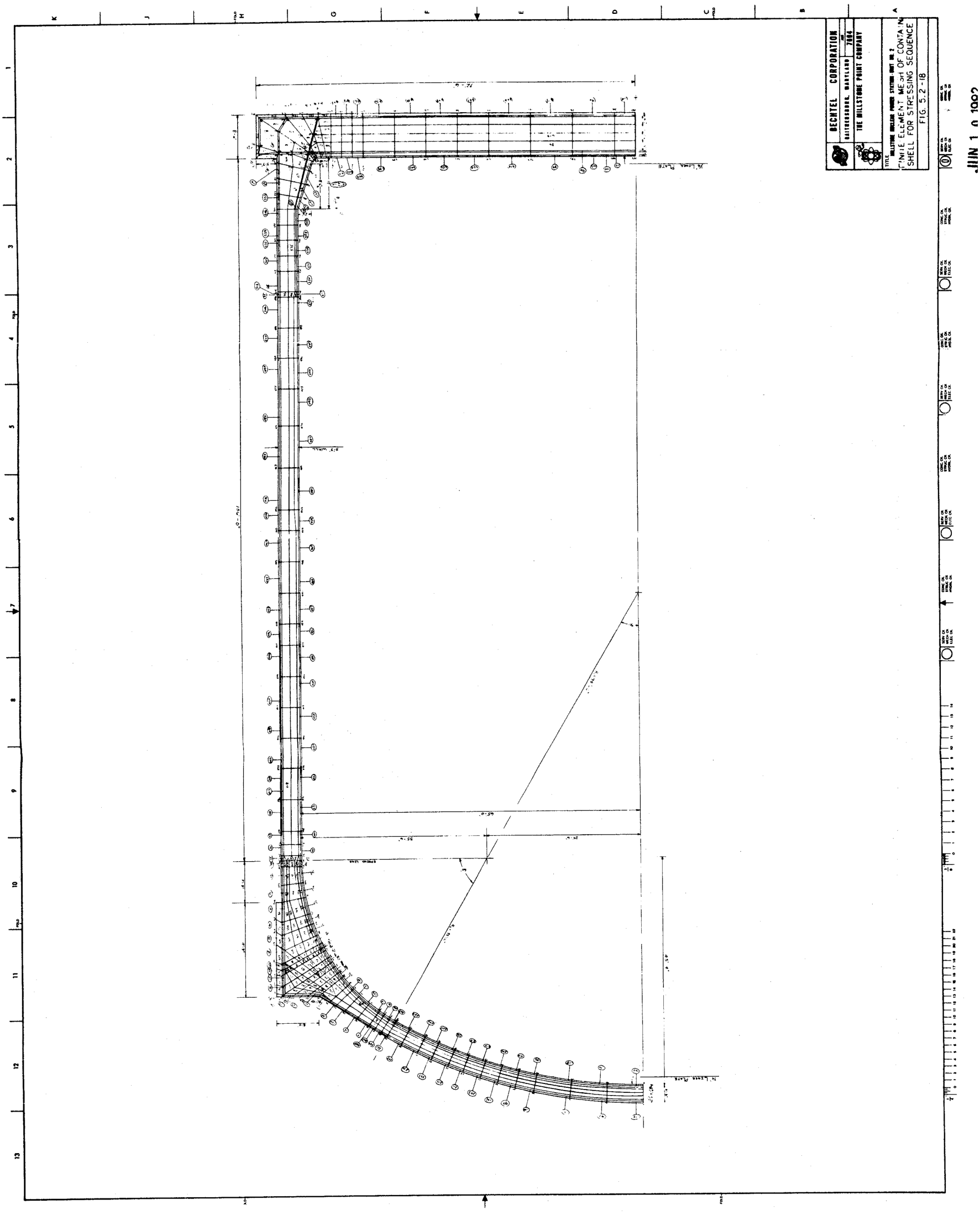


FIGURE 5.2–18 FINITE ELEMENT MESH OF CONTAINMENT SHELL FOR STRESSING SEQUENCE

FIGURE 5.2–19 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD AND INITIAL PRESTRESS, LIVE LOAD

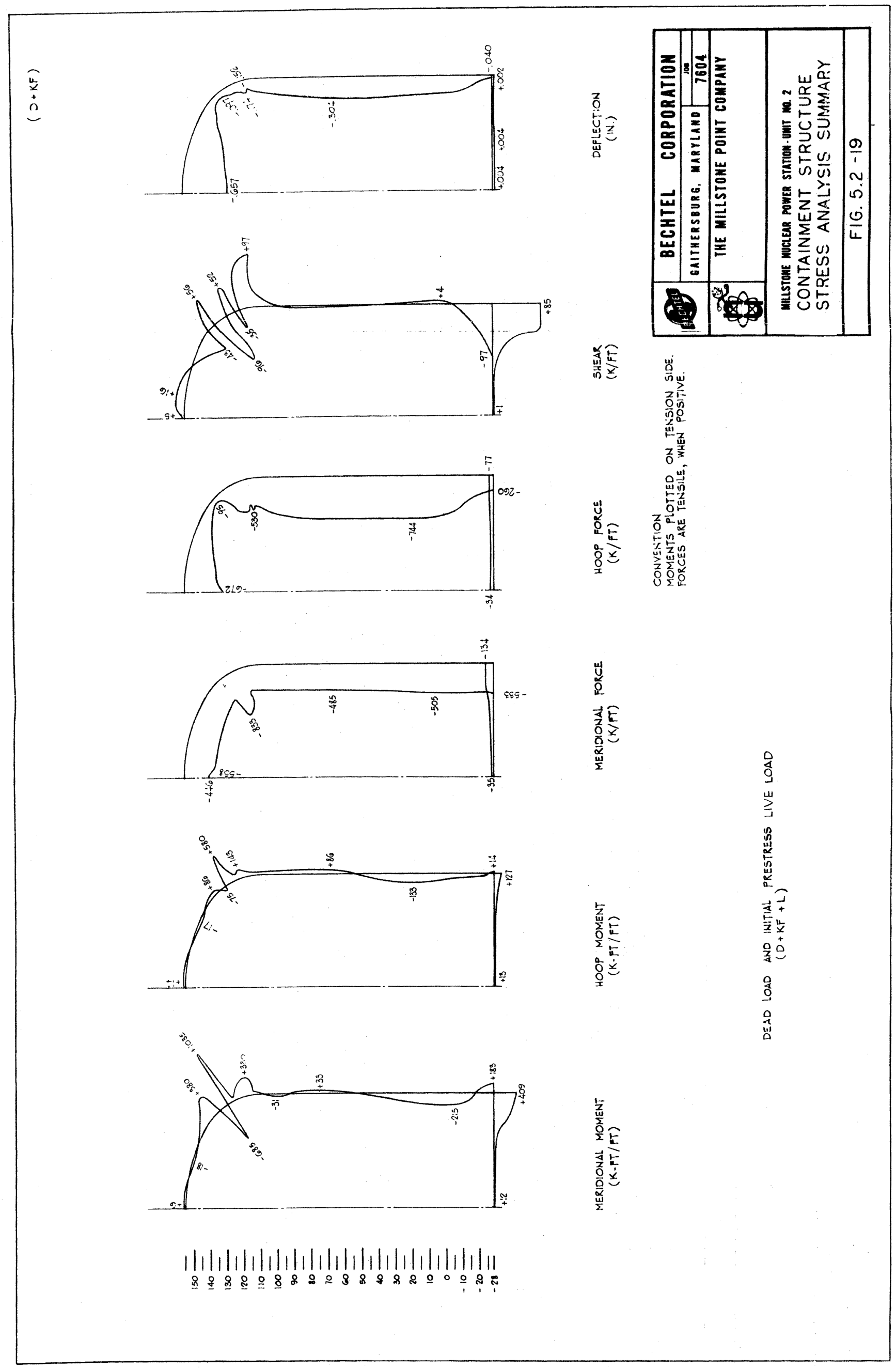
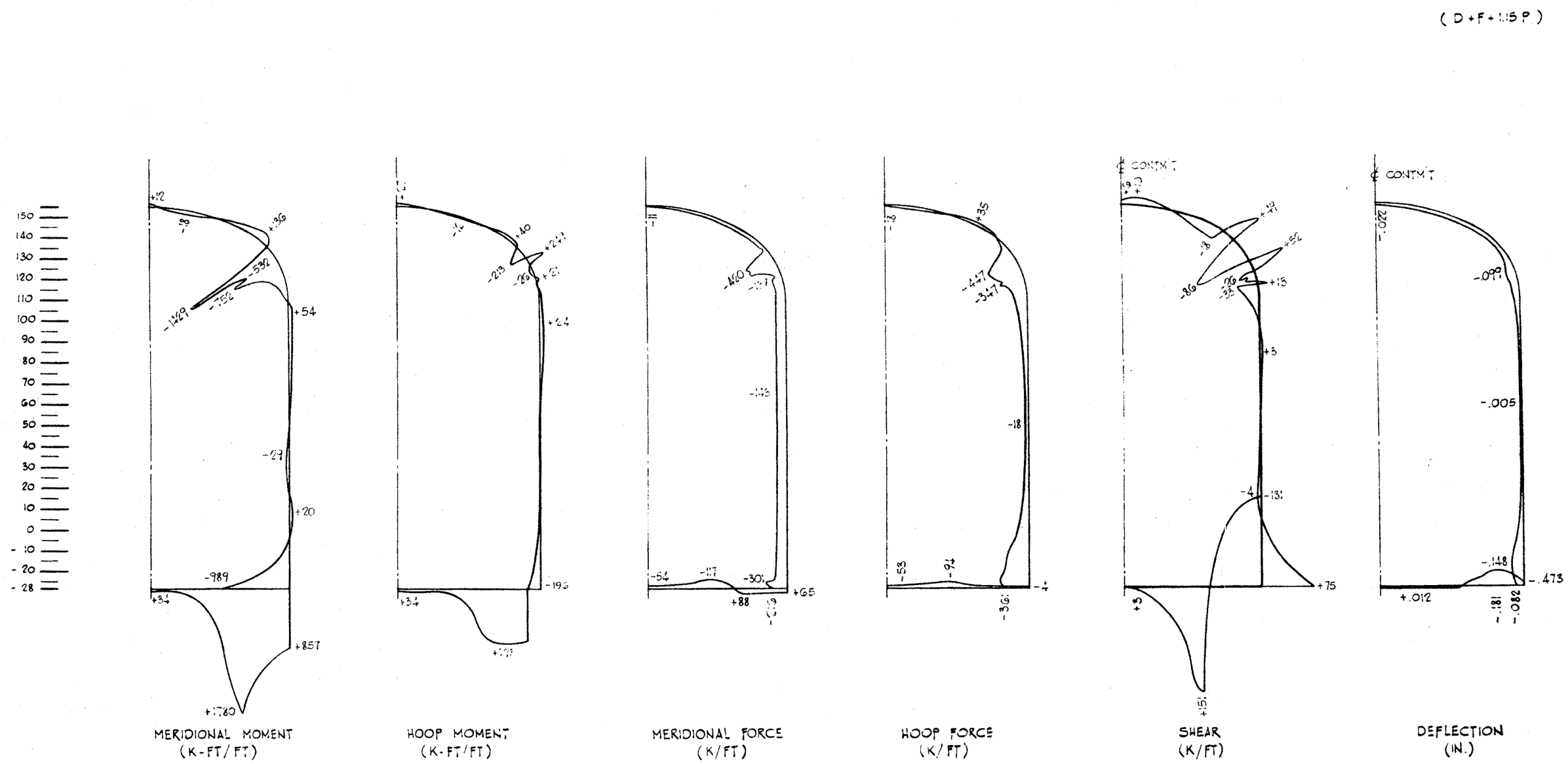


FIGURE 5.2–20 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, LIVE LOAD, PRESTRESS AND TEST PRESSURE



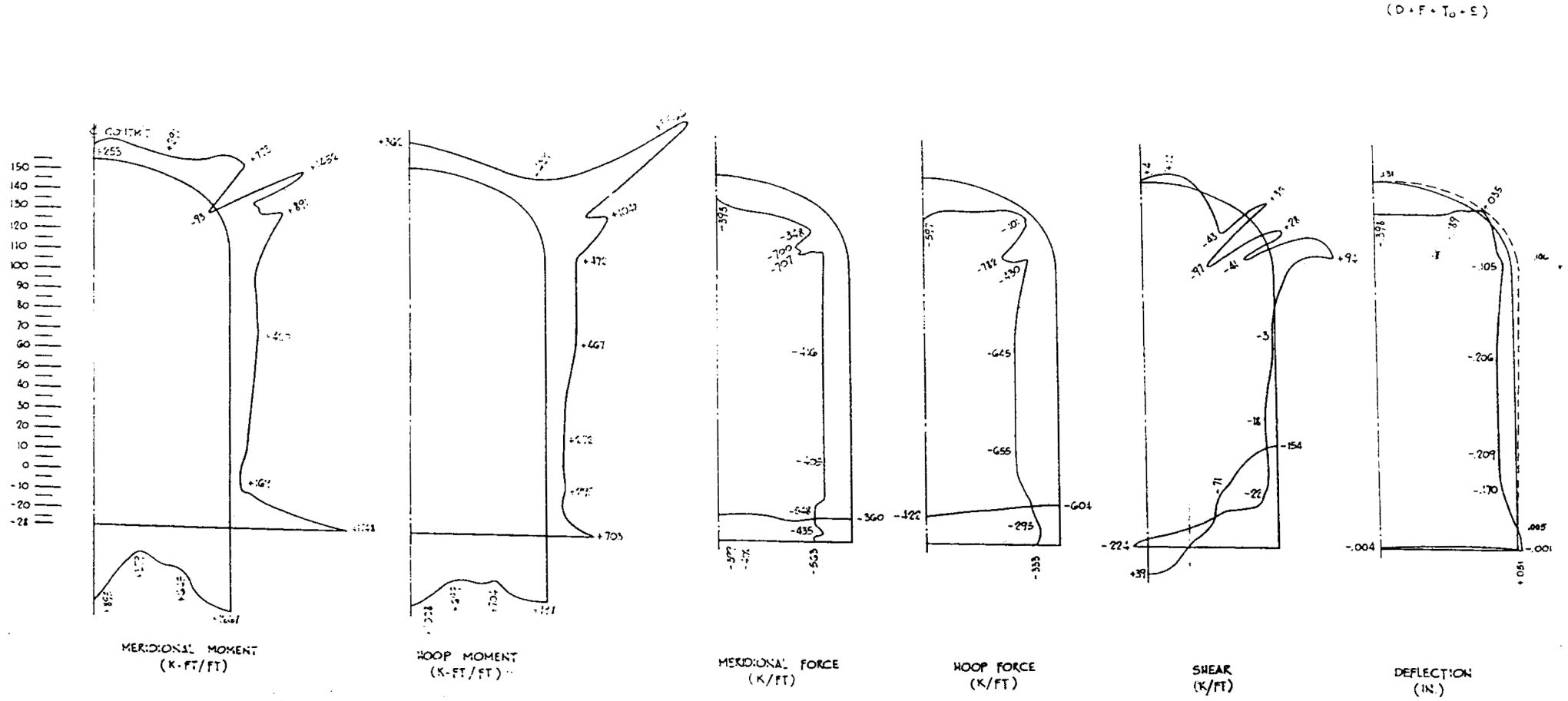
CONVENTION:
MOMENTS PLOTTED ON TENSION SIDE.
FORCES ARE TENSILE, WHEN POSITIVE.

DEAD LOAD, LIVE LOAD, PRESTRESS AND TEST PRESSURE
(D+L+F+1.15P)

BECHTEL CORPORATION	
GAITHERSBURG, MARYLAND	JOB 7604
THE MILLSTONE POINT COMPANY	
MILLSTONE NUCLEAR POWER STATION - UNIT NO. 2 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY	
FIG. 5.2 -20	

JUN 10 1982

FIGURE 5.2–21 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, LIVE LOAD, PRESTRESS, OPERATING TEMPERATURE AND DBE



NOTE:
Changes resulting from the revised seismic analysis performed in 1998-99 have not been incorporated in this Figure, but the controlling stresses/strains resulting from the revision are shown in Table 5.2-4.

FIGURE 5.2-21
CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY
MARCH 1999

FIGURE 5.2–22 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, LIVE LOAD, PRESTRESS, 100% ACCIDENT PRESSURE AND ACCIDENT TEMPERATURE

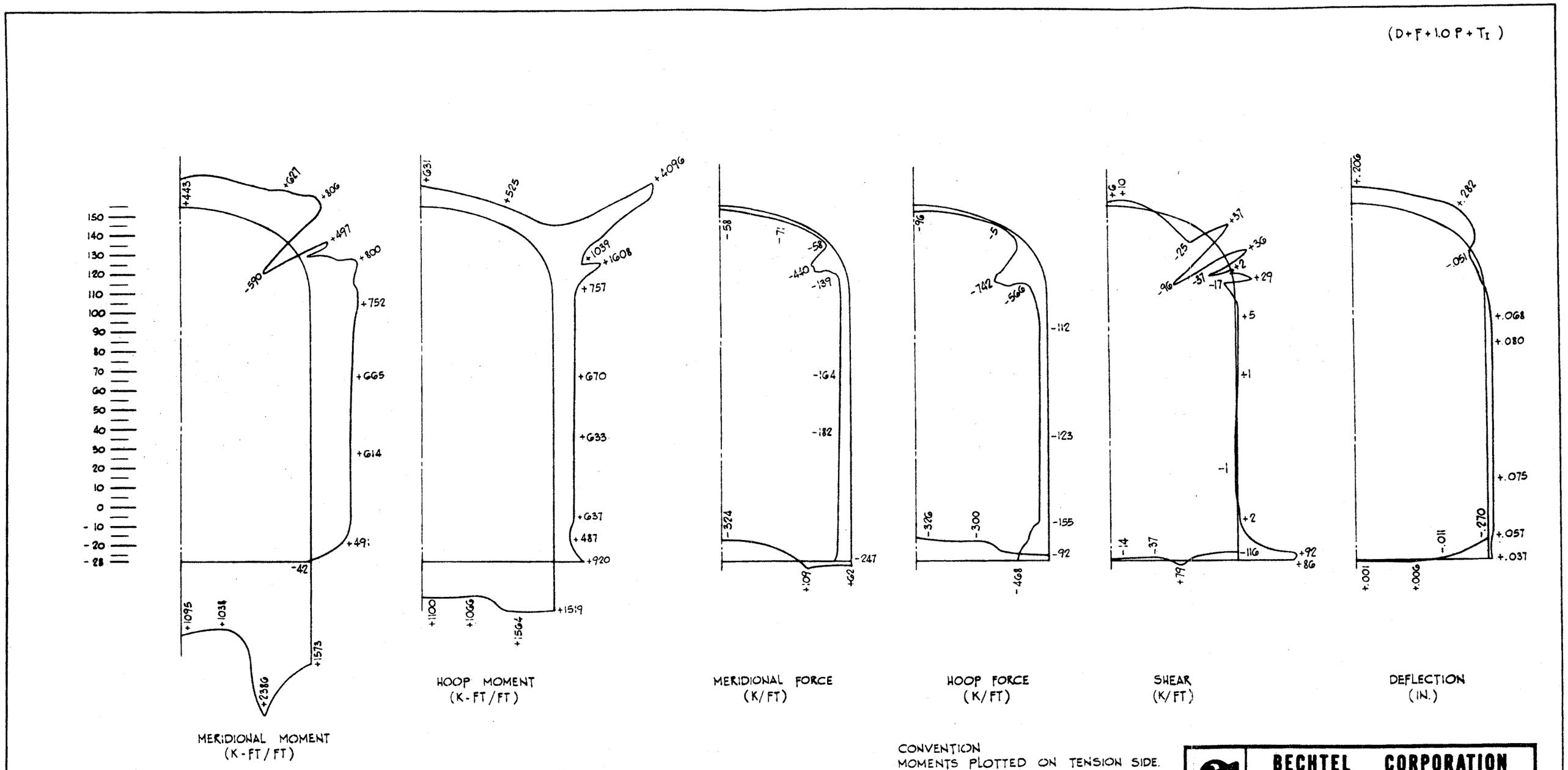
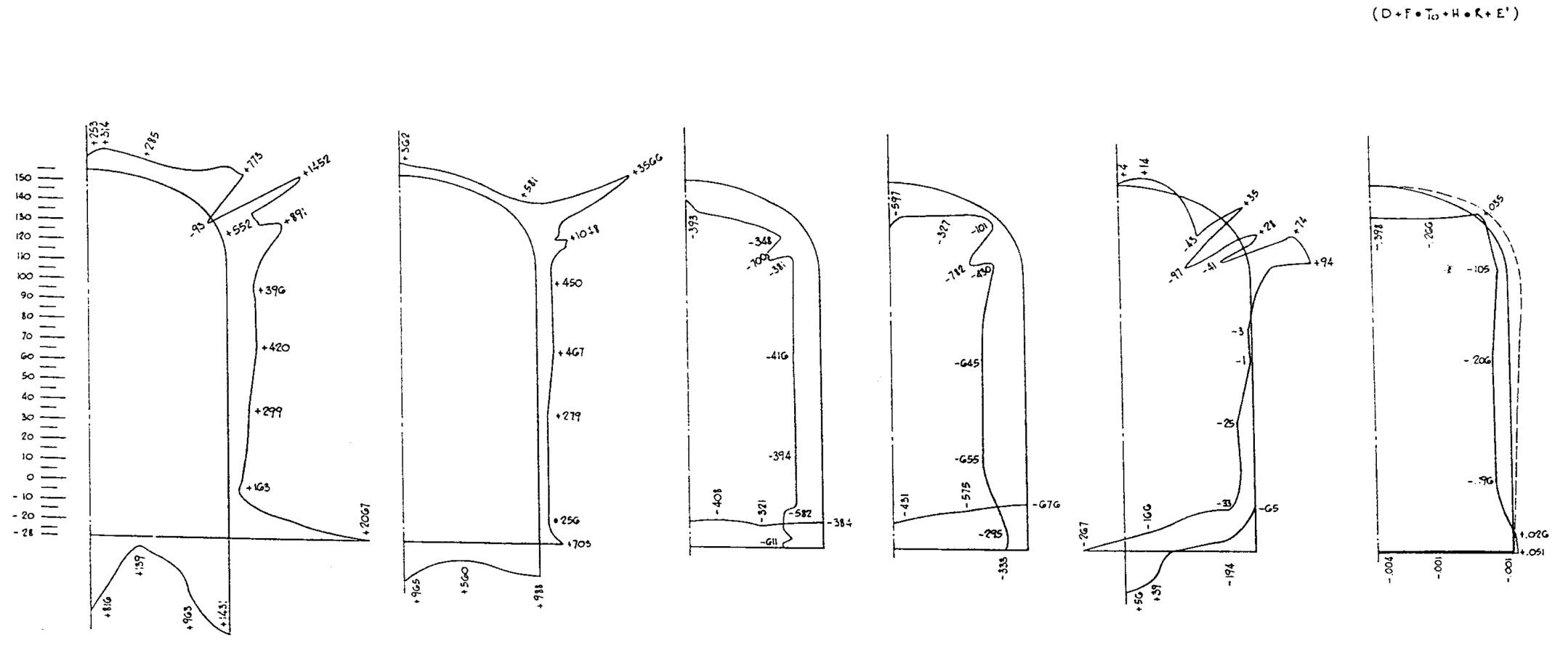


FIGURE 5.2–23 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, PRESTRESS, OPERATING TEMPERATURE, THERMAL EXPANSION FORCES OF PIPES, PIPE RUPTURE FORCES AND DBE

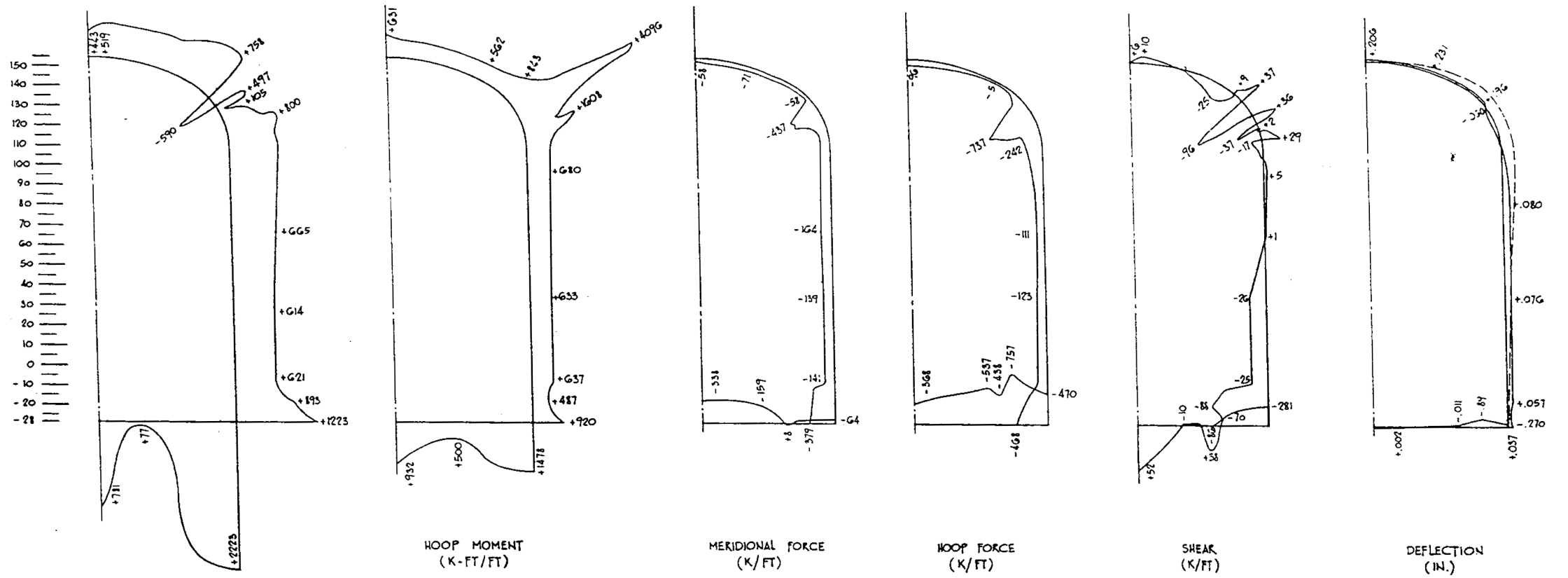


NOTE:
Changes resulting from the revised seismic analysis performed in 1998-99 have not been incorporated in this Figure, but the controlling stresses/strains resulting from the revision are shown in Table 5.2-6.

FIGURE 5.2-23
CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY

MARCH 1999

FIGURE 5.2–24 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, PRESTRESS, 100% ACCIDENT PRESSURE, THERMAL EXPANSION FORCES OF PIPES, ACCIDENT TEMPERATURE AND DBE



CONVENTION
MOMENTS PLOTTED ON TENSION SIDE.
FORCES ARE TENSILE, WHEN POSITIVE.

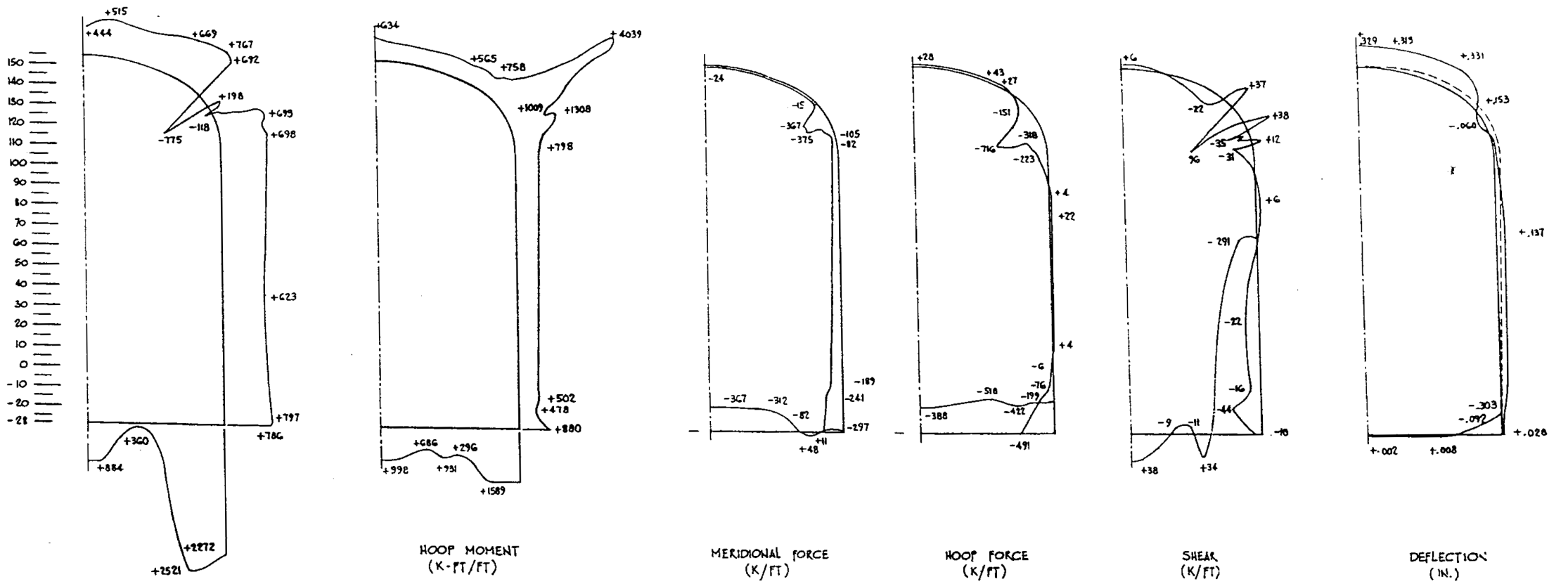
DEAD LOAD, PRESTRESS, 100% ACCIDENT PRESSURE, THERMAL EXPANSION FORCES OF PIPES, ACCIDENT TEMPERATURE AND DBE.
$(D+F+1.0P+H+T_i+E')$

BECHTEL CORPORATION	
GAITHERSBURG, MARYLAND	JOB 7604
THE MILLSTONE POINT COMPANY	

NOTE:
Changes resulting from the revised seismic analysis performed in 1998-99 have not been incorporated in this Figure, but the controlling stresses/strains resulting from the revision are shown in Table 5.2-7.

FIGURE 5.2-24
CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY
MARCH 1999

FIGURE 5.2–25 CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, PRESTRESS, 125% ACCIDENT PRESSURE, 125% THERMAL EXPANSION FORCES OF PIPES, ACCIDENT TEMPERATURE AND 125% OBE

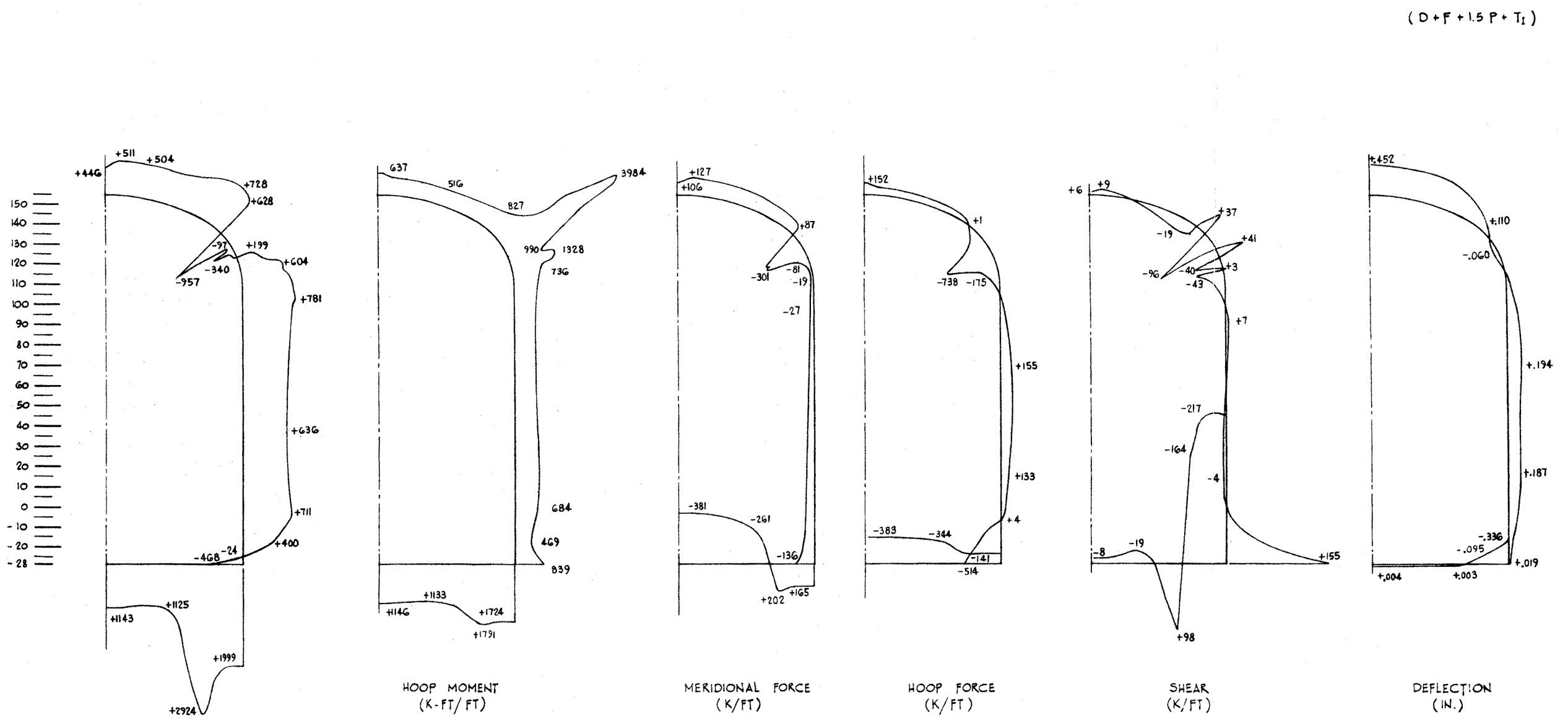


NOTE:
Changes resulting from the revised seismic analysis performed in 1998-99 have not been incorporated in this Figure, but the controlling stresses/strains resulting from the revision are shown in Table 5.2-8.

FIGURE 5.2-25
CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY

MARCH 1999

FIGURE 5.2–26 CONTAINMENT AND STRUCTURE STRESS ANALYSIS SUMMARY, DEAD LOAD, PRESTRESS, 150% ACCIDENT PRESSURE AND ACCIDENT TEMPERATURE



CONVENTION
MOMENTS PLOTTED ON TENSION SIDE.
FORCES ARE TENSILE, WHEN POSITIVE.

DEAD LOAD, PRESTRESS, 150 % ACCIDENT PRESSURE AND ACCIDENT TEMPERATURE
(D + F + 1.5 P + T_I)

BECHTEL CORPORATION	
GAITHERSBURG, MARYLAND	JOB 7604
THE MILLSTONE POINT COMPANY	
MILLSTONE NUCLEAR POWER STATION - UNIT NO. 2	
CONTAINMENT STRUCTURE STRESS ANALYSIS SUMMARY	
FIG. 5.2 -26	

JUN 1 0 1999

FIGURE 5.2–27 ISOLATION VALVE ARRANGEMENTS

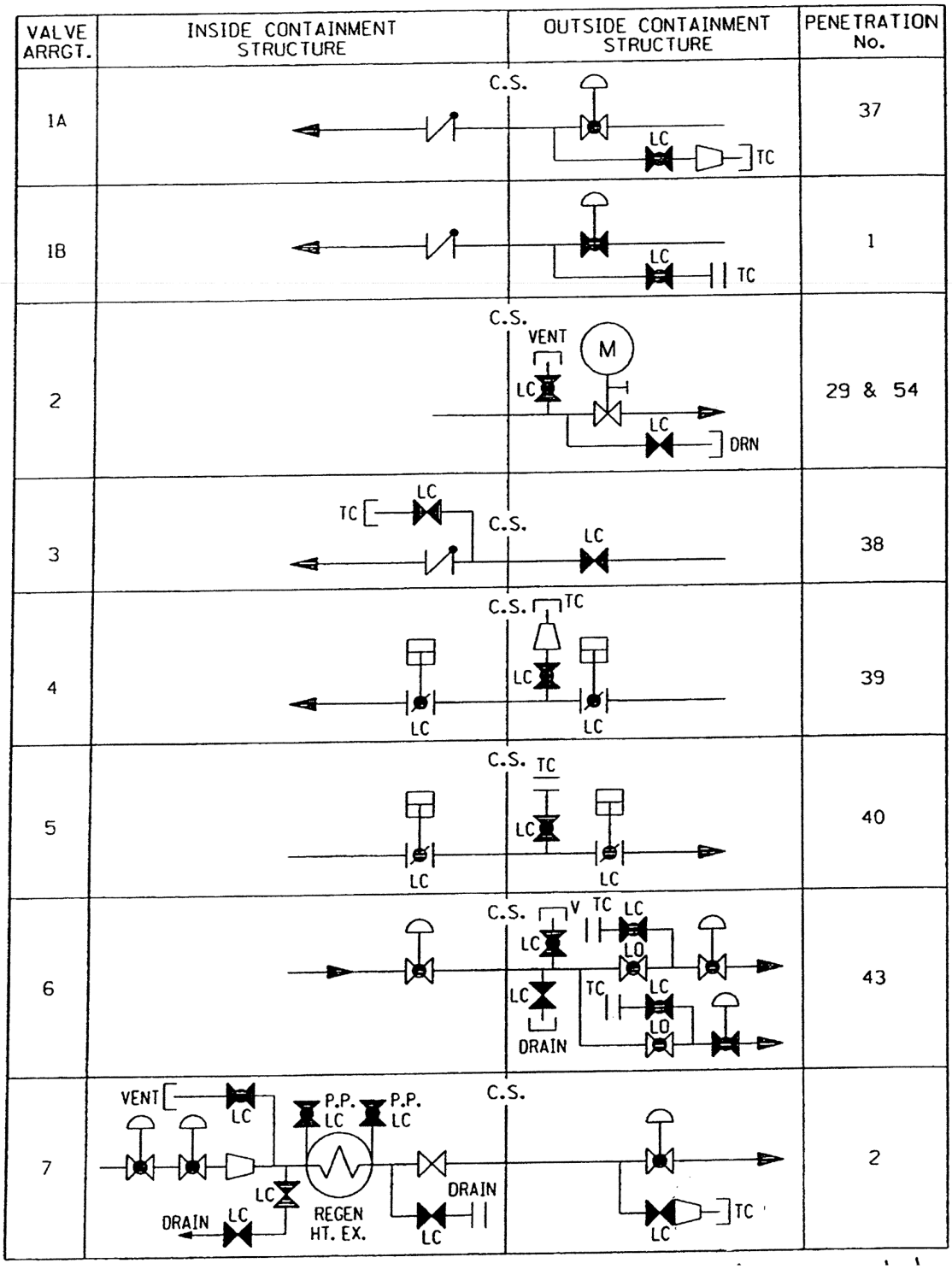


FIGURE 5.2–28 ISOLATION VALVE ARRANGEMENTS

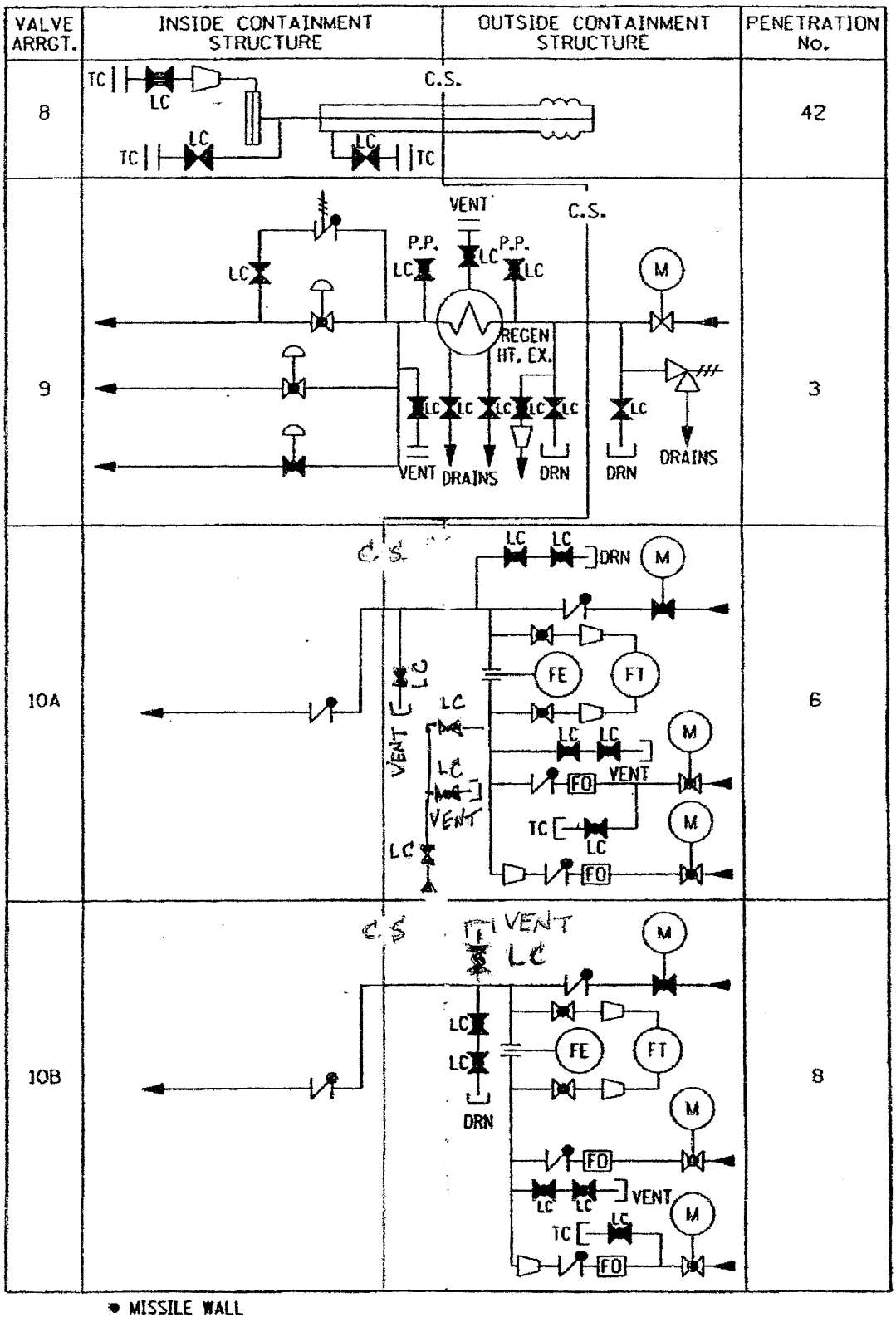


FIGURE 5.2–29 ISOLATION VALVE ARRANGEMENTS

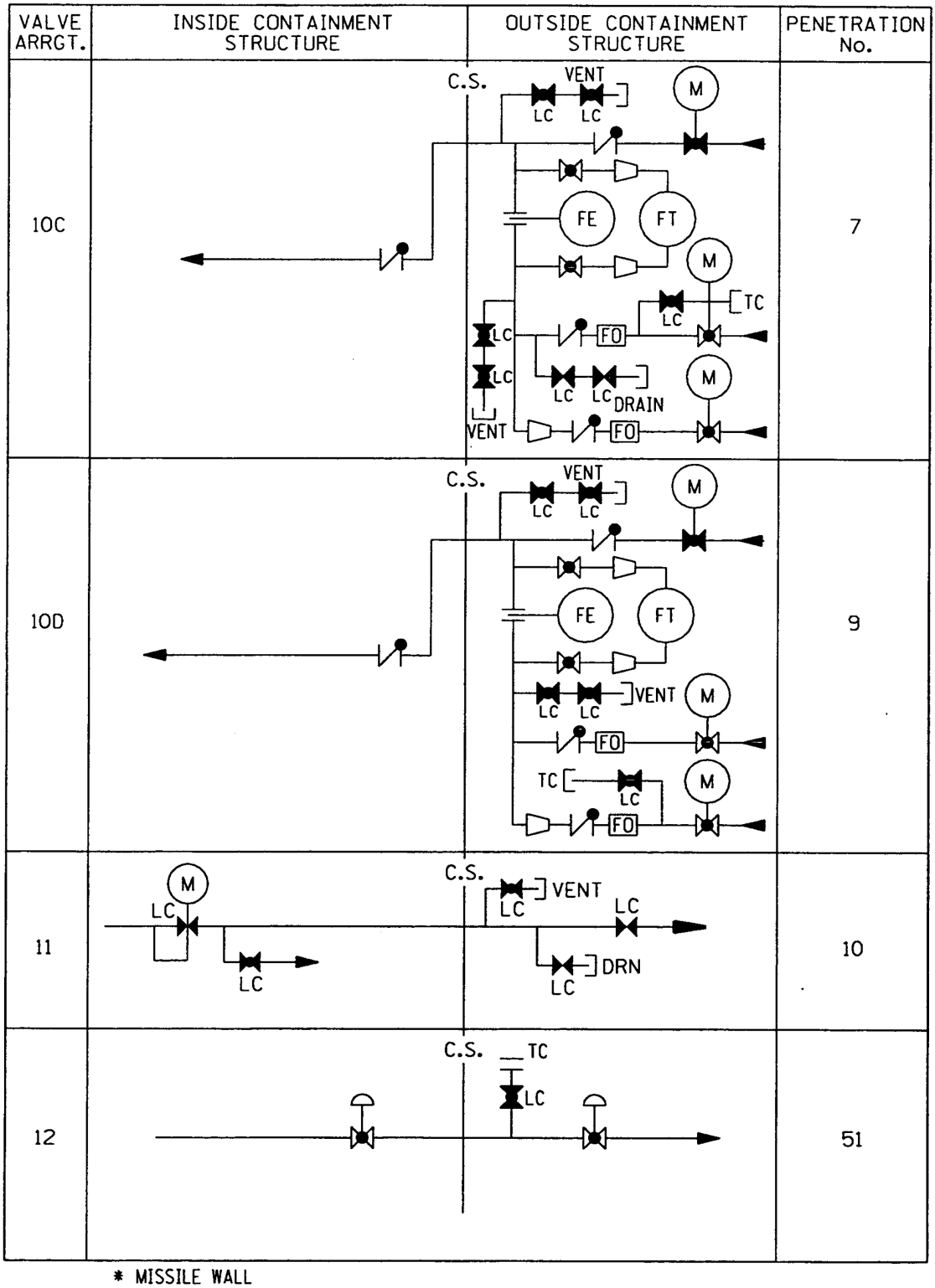


FIGURE 5.2–30 ISOLATION VALVE ARRANGEMENTS

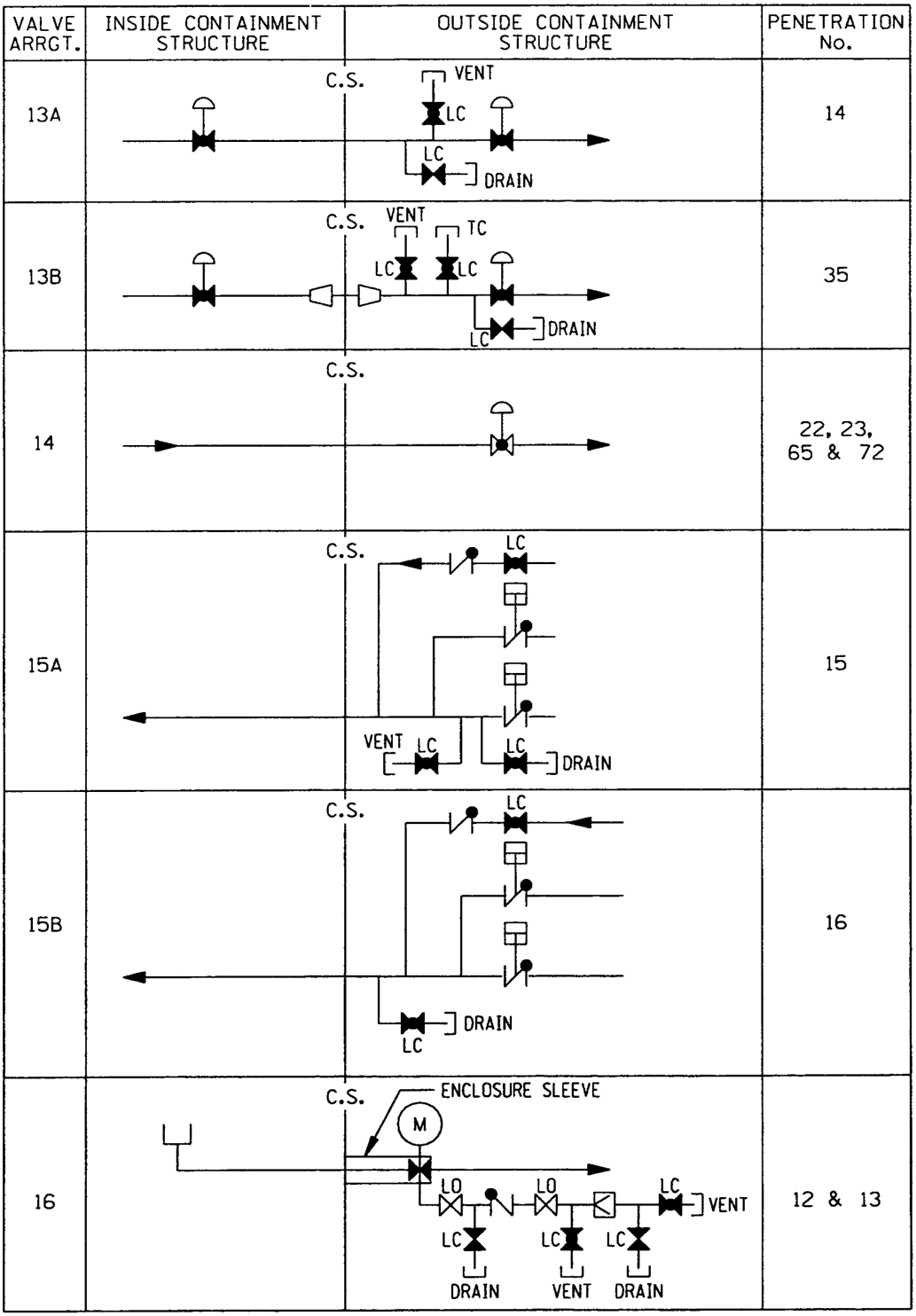


FIGURE 5.2–31 ISOLATION VALVE ARRANGEMENTS

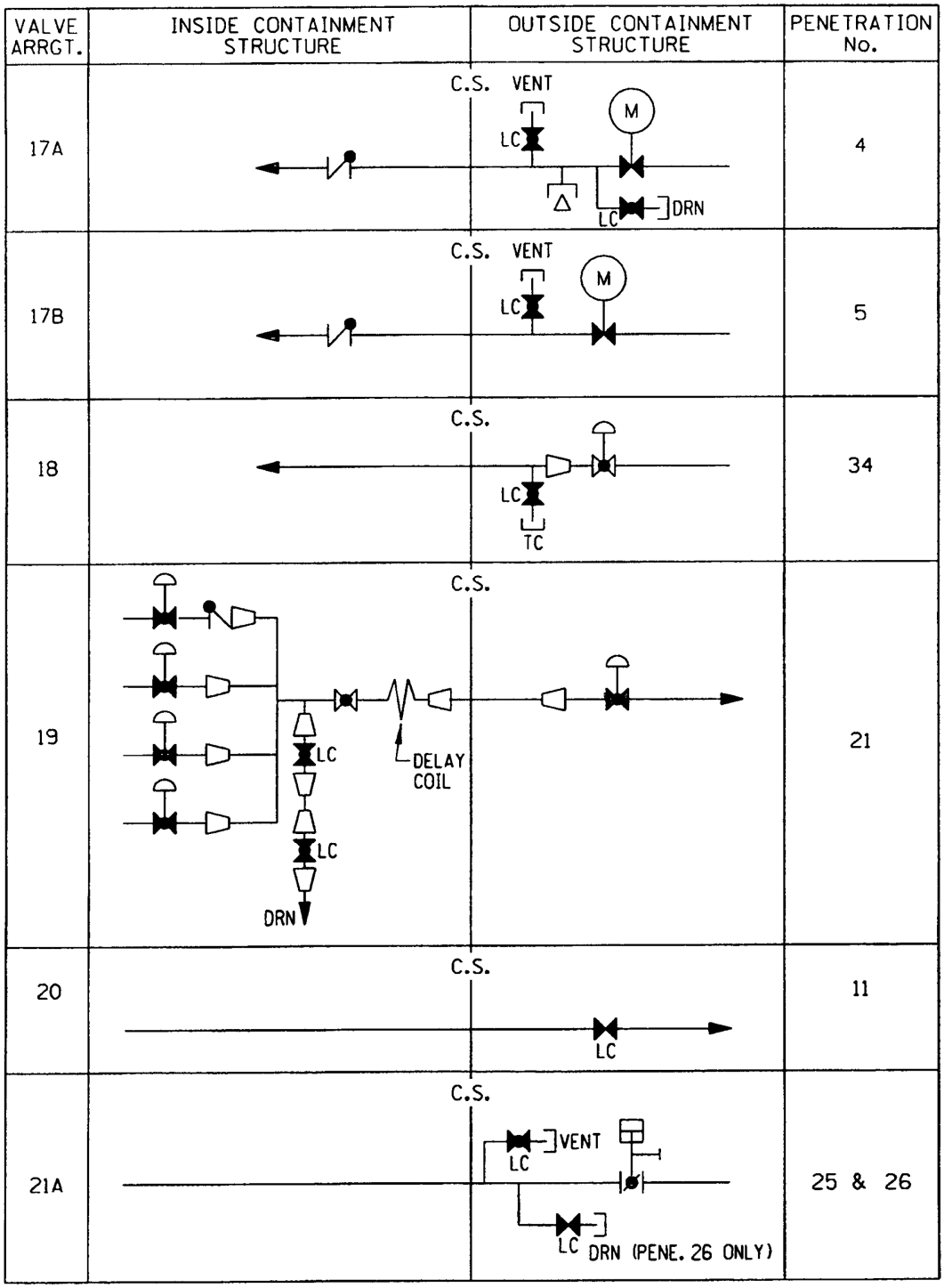


FIGURE 5.2–32 ISOLATION VALVE ARRANGEMENTS

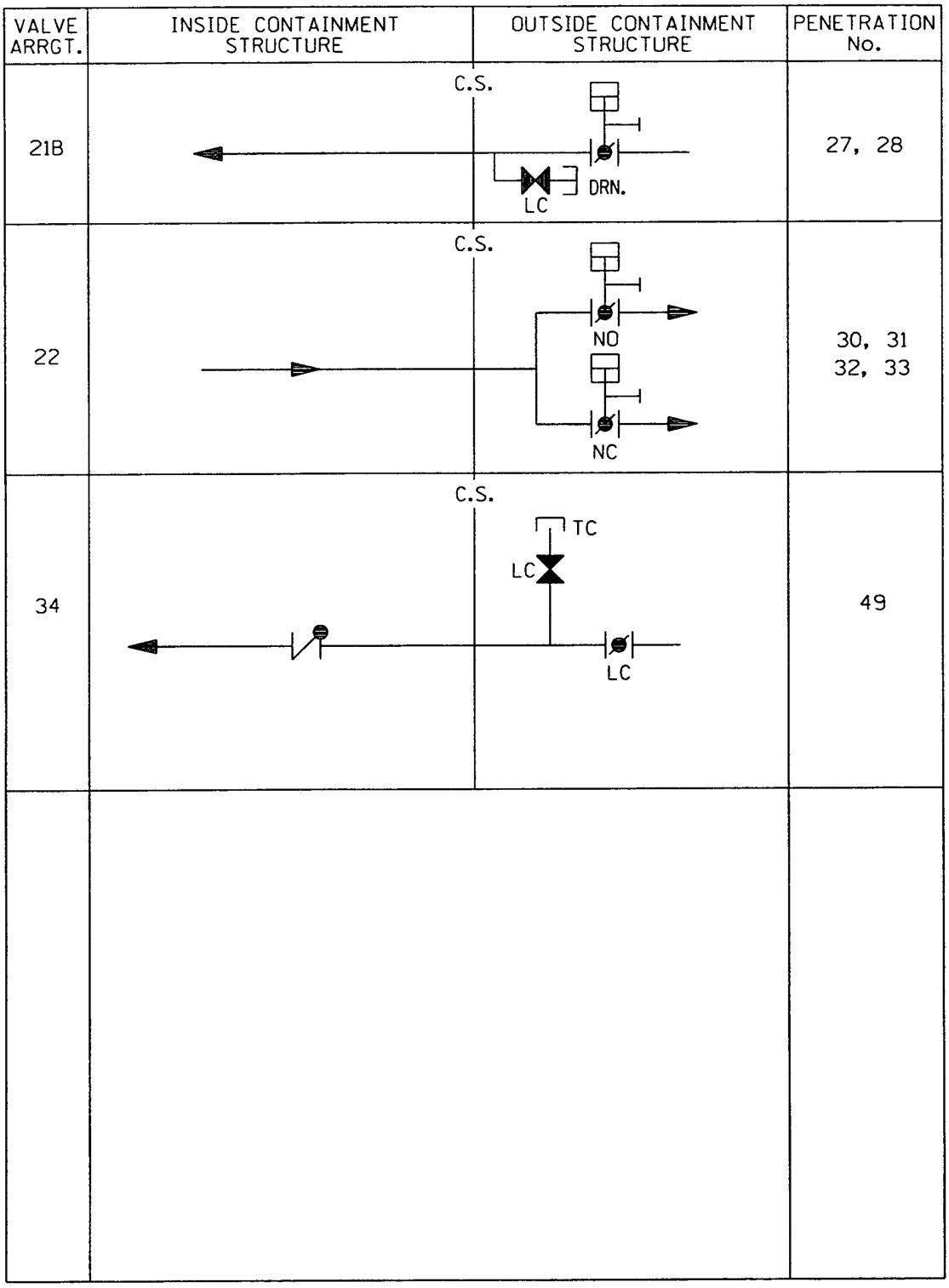


FIGURE 5.2–33 ISOLATION VALVE ARRANGEMENTS

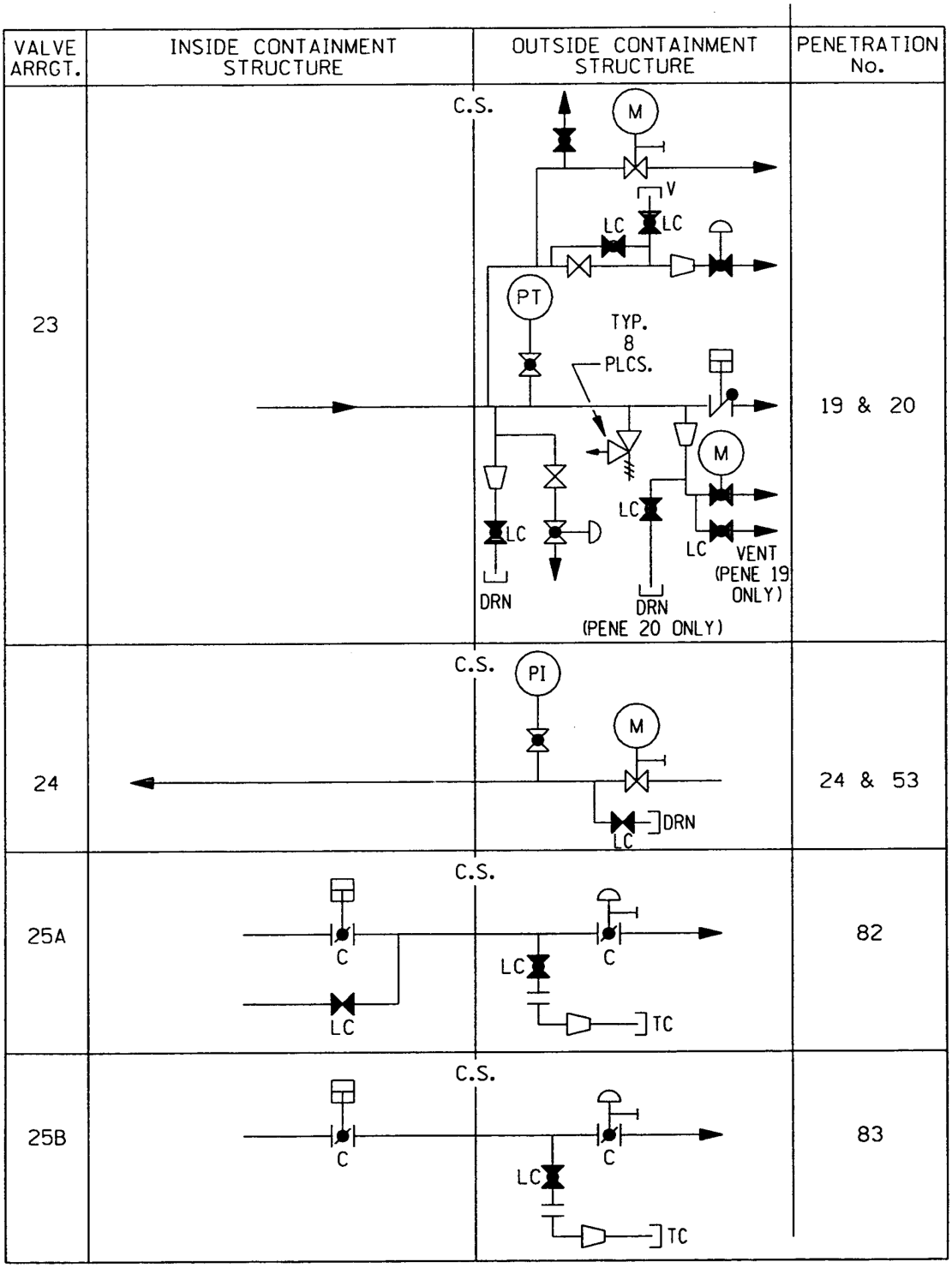


FIGURE 5.2–34 ISOLATION VALVE ARRANGEMENTS

MNPS-2 FSAR

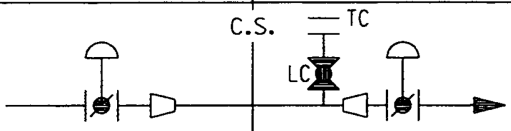
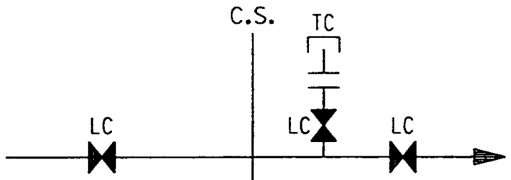
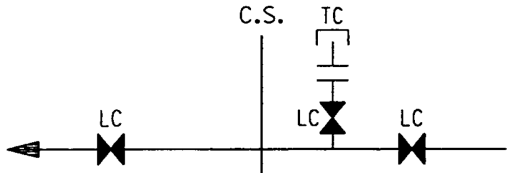
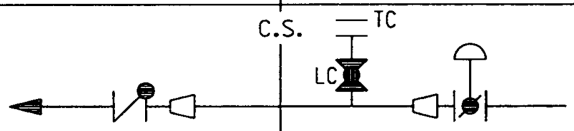
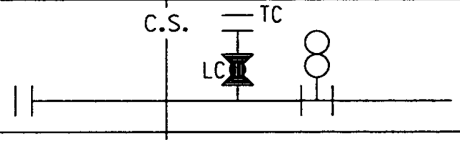
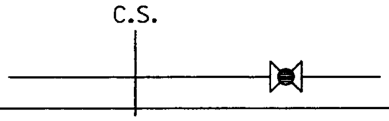
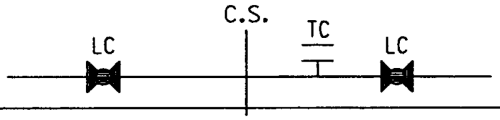
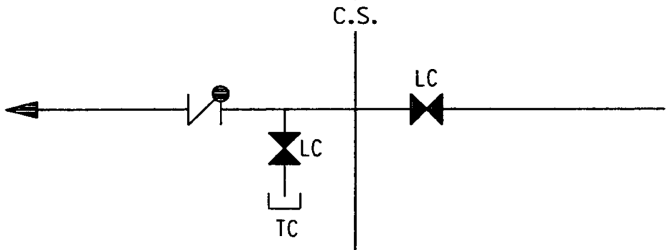
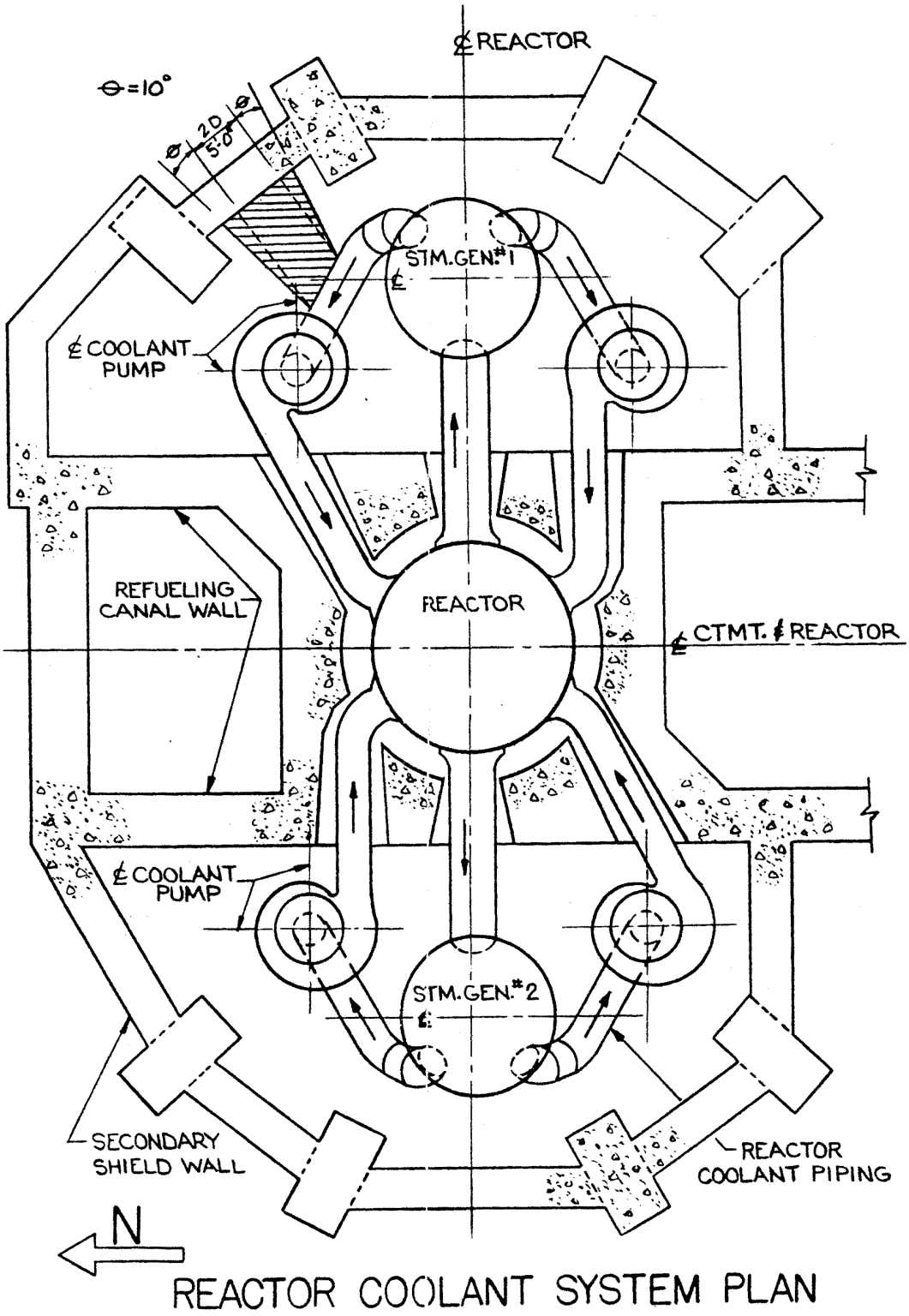
VALVE ARRGT.	INSIDE CONTAINMENT STRUCTURE / OUTSIDE CONTAINMENT STRUCTURE	PENETRATION No.
26		61, 86
27A		67
27B		68
28		62, 87
29		85
30		47, 69 70, 71
31		63, 64
33		36

FIGURE 5.2–35 REACTOR COOLANT SYSTEM PLAN



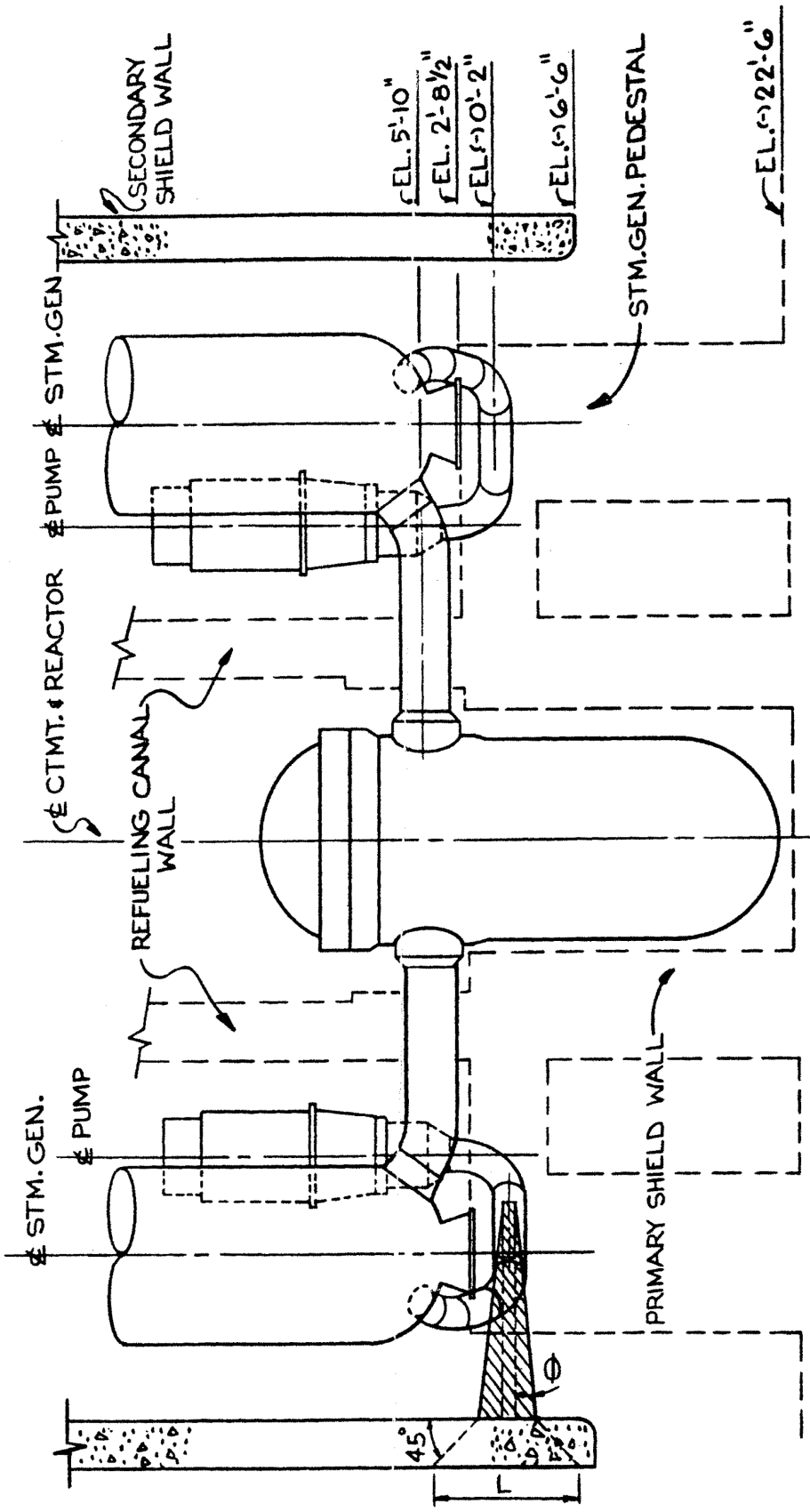


FIGURE 5.2–36 REACTOR COOLANT SYSTEM ELEVATION

MILLSTONE NUCLEAR POWER STATION
MILLSTONE UNIT 2
FIGURE 5.2-36

FIGURE 5.2–37 DELETED BY FSARCR 04-MP2-016

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.

Principal dimensions of the enclosure building are as follows:

Length (feet)	153.0
Width (feet)	147.0
Height (feet)	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 Millstone 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 3. 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 3, FSAR lists the

Millstone 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 one-quarter 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 feet 0 inches by 9 feet 6 inches. 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.

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.

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
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 Figures 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.

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 5.2.5.1.2 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.

The design of the enclosure building for tornado loads assumes that tornado wind is not coincident with a loss-of-coolant accident (LOCA) or earthquake.

5.3.3.2 Design Load Combination and Structural Analysis

The enclosure building is designed to meet the performance and strength requirements of the following loading combinations:

a. At design loads

b. At factored loads

The design of structural steel is in accordance with the AISC Manual of Steel Construction. The design of concrete is in accordance with the ACI Code 318-63.

5.3.3.2.1 At Design Loads

The enclosure building is analyzed for the following specific loading conditions:

a. D + L = Construction

b. D + l + E = Operating

c. D + L + W = Operating

Where

D = dead loads

L = live loads

E = operating basis earthquake loads (0.09 g)

W = wind or tornado loads

5.3.3.2.2 At Factored Loads

The enclosure building is analyzed for the factored load combination to ensure that its structural integrity is not affected.

$$C = \frac{1}{\varnothing}[(1.0)(D) + E']$$

Where

C = required capacity of the structure

∅ = 0.90 for fabricated structural steel

D = dead loads

E′ = design basis earthquake (0.17 g)

The stresses of the members of the structure at factored loads are limited to the yield stresses of the structural steels.

5.3.3.2.3 Seismic Analysis

The seismic analysis of the enclosure building is made on a mathematical model which consists of the lumped masses of the containment structure and the enclosure building. The seismic response of the combined model is obtained in accordance with the procedures outlined in Section 5.8.

5.3.4 THROUGH-LINE LEAKAGE EVALUATION

To evaluate the through-line leakage that can bypass the enclosure building filtration region (EBFR), the fluid systems penetrating containment are categorized as follows:

a. Piping Systems open to the containment post-accident atmosphere.

b. Piping Systems which are closed and therefore not exposed to the containment post-accident atmosphere.

The following assumptions are made to postulate the maximum hypothetical conditions:

a. There is either a seismic occurrence and all Seismic Class 2 lines are broken, or there is no seismic occurrence and all Seismic Class 2 lines are intact.

b. The single failure criterion applies to Seismic Class 1 components only.

The condition of a seismic occurrence is not considered. Should the pipe break within the EBFR, all potential containment leakage would be processed by the enclosure building filtration system (EBFS) as per design. The EBFS has ample capacity for this event.

From the basis formulated, systems which are not normally opened to the containment atmosphere, or normally closed systems, either do not leak (assuming no seismic event) or are vented to the EBFR (assuming a seismic event). Normally, closed systems which may be opened to the atmosphere during accident conditions, such as lines connected to the reactor coolant pressure boundary, are not considered. These systems are either operating at a higher pressure or form closed loops. Therefore, assuming a single failure, these lines either prevent leakage by the higher pressure or contain the leakage by the closed loop.

From this basis, an evaluation was performed to establish and document those containment penetrations that have the potential for providing leakage pathways from the reactor containment to areas beyond the EBFR. These leakage pathways could result in Post-Accident Containment Atmosphere bypassing the EBFR and discharging directly to the atmosphere thereby increasing on-site and off site doses under radiological accident conditions. Leakage through these pathways is referred to as "bypass leakage."

For a leakage pathway to viably result in bypass leakage, the pathway must be open to the containment atmosphere post-accident and provide a means of transporting the containment atmosphere beyond the EBFR as well as a means for the containment atmosphere to escape the piping or duct. For containment penetrations that are confirmed to contribute to bypass leakage, leakage rates may be based on measured values as opposed to the recommended or maximum allowable values used for testing.

In cases where measured values are used, steps are taken to ensure that degradation of the valve sealing capabilities are taken into account commensurate with the severity of service and the required time intervals between valve maintenance.

The evaluation concluded the following penetrations are considered to qualify as systems that are open both inside containment and outside containment, extend beyond the EBFR, and could contribute to bypass leakage:

Penetration	**System**
14	Containment Sump Pump Discharge
37	Instrument Air System
38	Station Air System
42	Fuel Transfer Tube
61	Hydrogen Monitoring System
62	Hydrogen Monitoring System
67	Refueling Cavity Drain
68	Refueling Cavity Skimmer
85	Containment Pressure Test Connection
86	Hydrogen Monitoring System
87	Hydrogen Monitoring System

The off-site and control room dose analyses are based on a calculated maximum bypass leakage (refer to Section 14.8.4). For each verified bypass leakage pathway, a recommended leak rate is provided based on the limits used to satisfy the leakage limits established for the testing required by 10 CFR 50, Appendix J. Total leakage from all verified bypass leakage pathways will be

summed and compared to the total limit. The control room and off-site radiological dose calculations establish the maximum limit for total bypass leakage. In the event that total bypass leakage exceeds this value, repairs will be performed to reduce bypass leakage to an acceptable level.

The provisions for initial and periodic leak testing of containment penetrations and maximum allowable leakage are specified in Table 5.2-11 of the FSAR and Section 3.6.1.2.c of the Technical Specifications respectively.

FIGURE 5.3–1 ENCLOSURE BUILDING

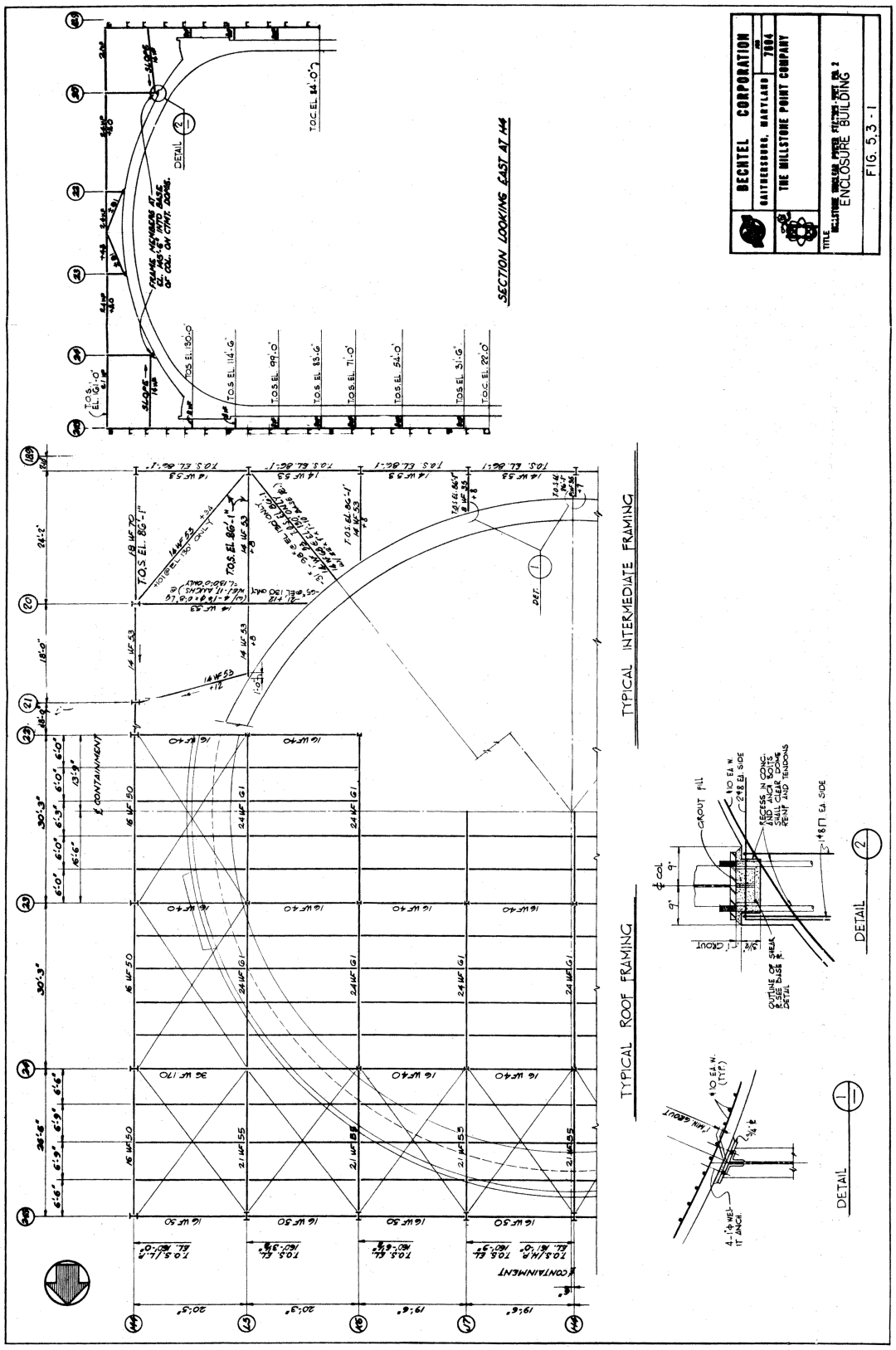


FIGURE 5.3–2 ENCLOSURE BUILDING LAYOUT

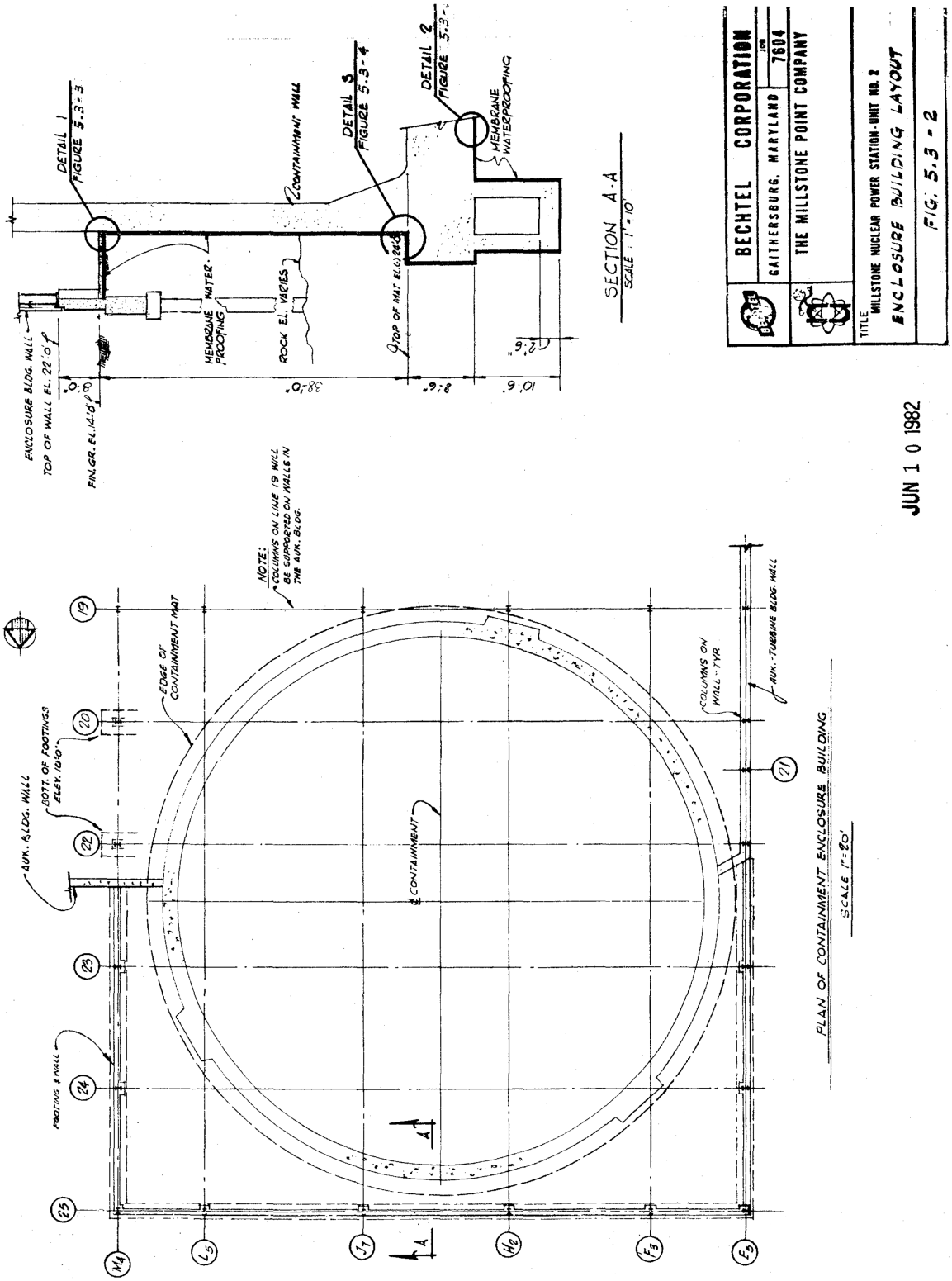
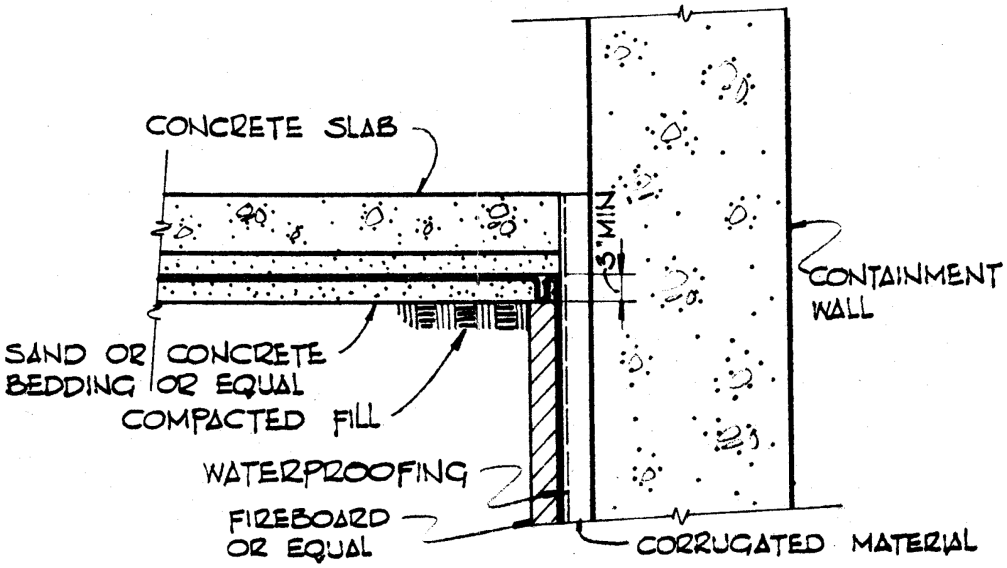
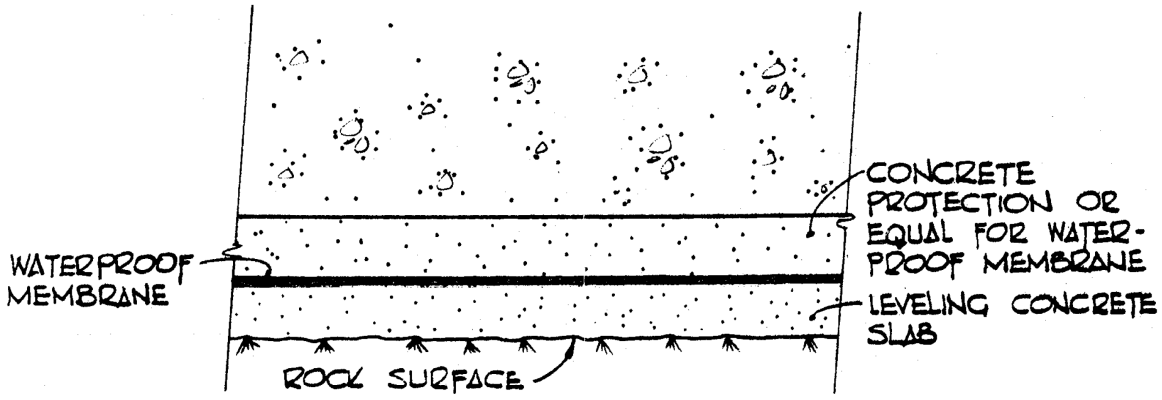


FIGURE 5.3–3 WATERPROOF MEMBRANE DETAILS



WATERPROOFING AT FLOOR SLAB AGAINST CONTAINMENT WALL

DETAIL 1



WATERPROOFING AT BOTTOM SIDE OF FOUNDATION MAT ON ROCK

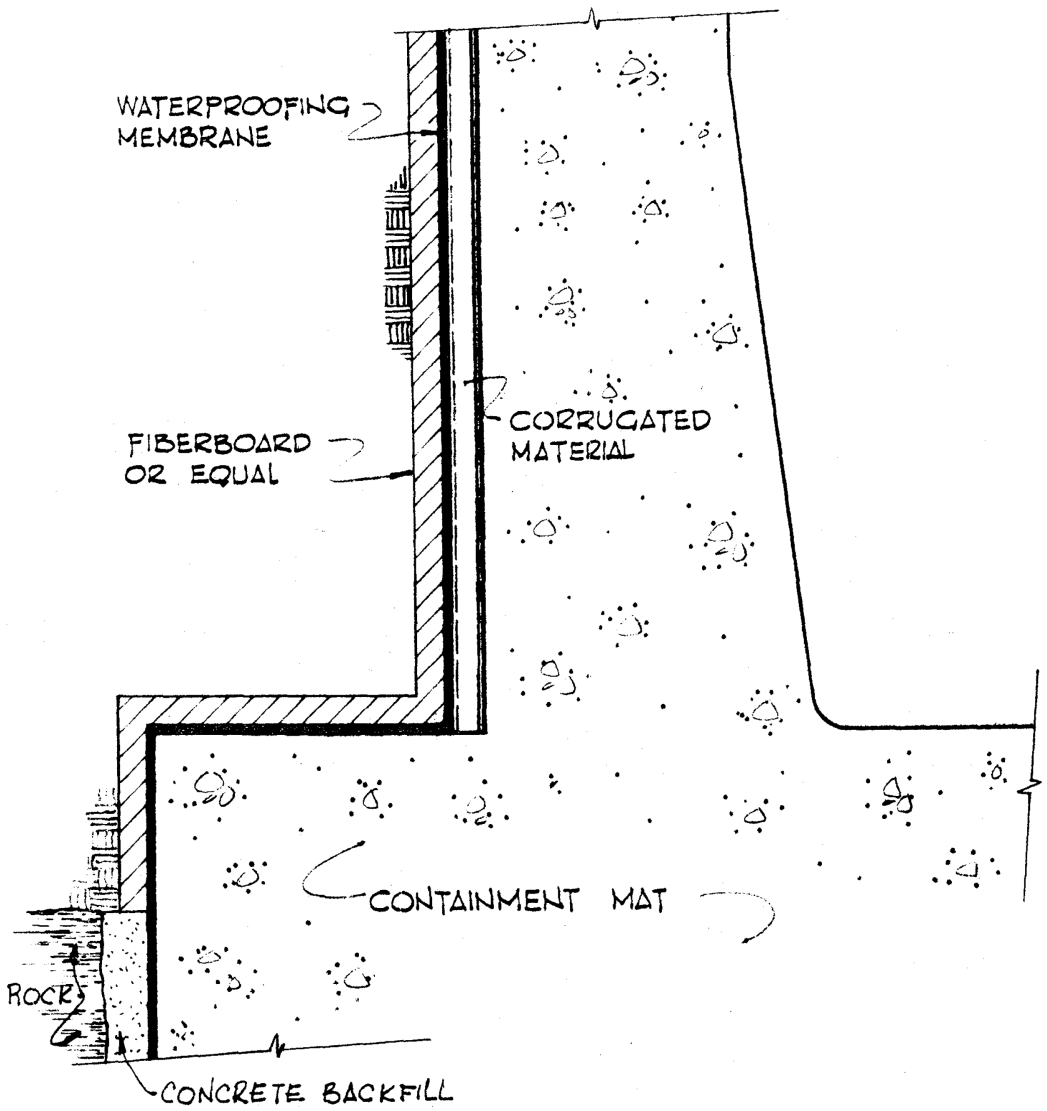
DETAIL 2

BECHTEL CORPORATION
GAITHERSBURG, MARYLAND JOB 7604
THE MILLSTONE POINT COMPANY
TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2
WATERPROOF MEMBRANE DETAILS

JUN 1 0 1982

FIG. 5.3 - 3

FIGURE 5.3–4 WATERPROOF MEMBRANE DETAILS



WATERPROOFING AT CONTAINMENT MAT AND WALL INTERSECTION

DETAIL 3

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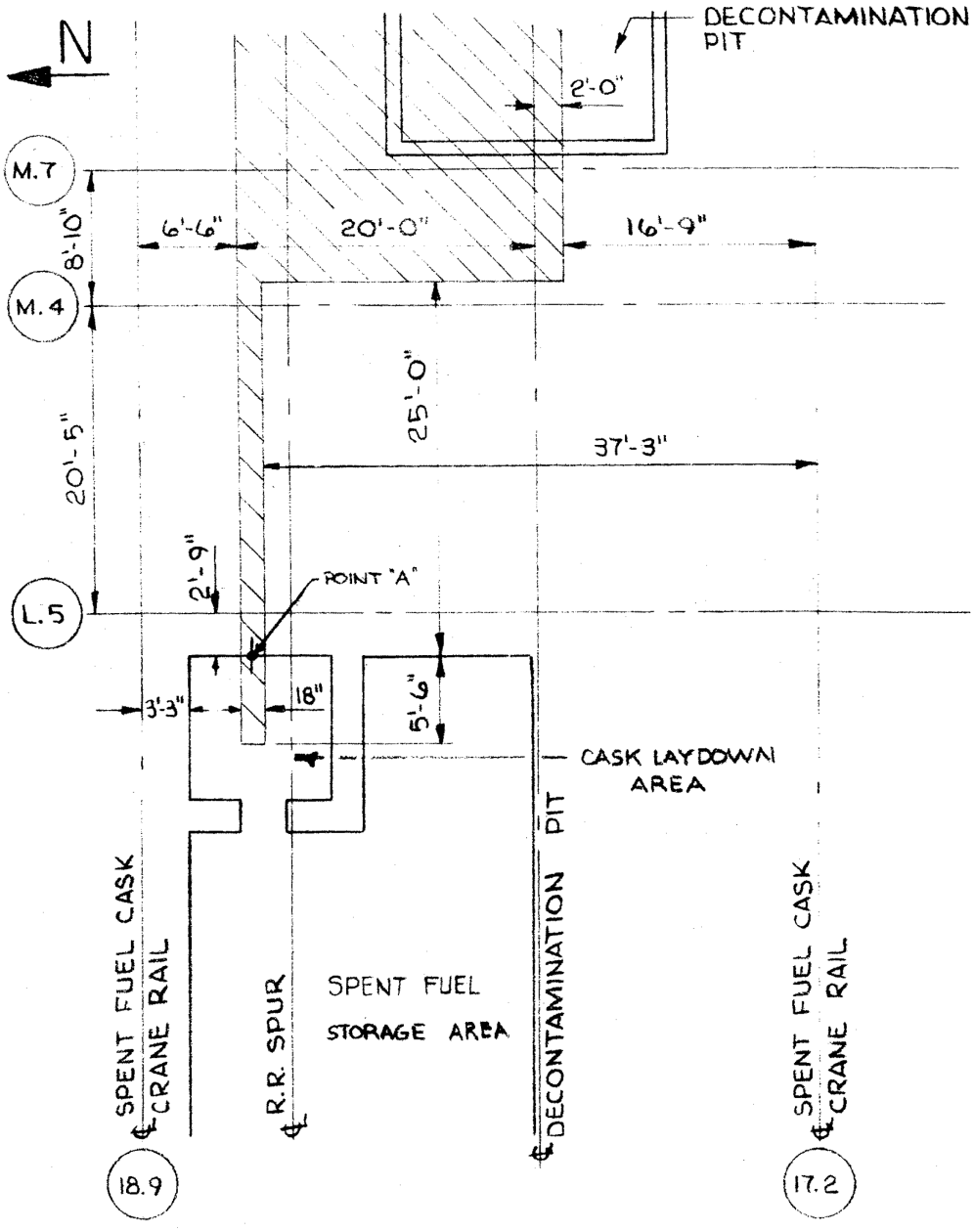
BECHTEL CORPORATION	
GAITHERSBURG, MARYLAND	JOB 7604
THE MILLSTONE POINT COMPANY	
TITLE MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2	
WATERPROOF MEMBRANE DETAILS	
FIG. 5.3-4	

FIGURE 5.3–5 SPENT FUEL CASK TRAVEL LIMITS



NOTE:
THE HATCHED AREA DEFINES THE LIMITS OF THE CENTERLINE OF THE SPENT FUEL CASK WHILE BEING HANDLED BY THE SPENT FUEL CASK CRANE IN THE CASK HANDLING MODE.

SPENT FUEL CASK TRAVEL LIMITS FIGURE 5.3-5

2 - 16 - 73

JUN 1 0 1982

5.4 AUXILIARY BUILDING

5.4.1 GENERAL DESCRIPTION

The auxiliary building is a multistory, reinforced concrete structure with flat slabs and shear walls. Some open areas of the building are supported by structural steel columns to preserve space and allow flexibility in the design. The portion of the building west of column line M.7 is founded on bedrock approximately 60 feet below the ground surface, while the eastern end of the building is supported by compacted structural backfill. These two portions of the building are separated from each other by an expansion joint at line M.7 to allow for differential movements.

The auxiliary building is separated from the containment, which is to the north, by an expansion joint and from the turbine building to the west by slotted connections. Although the control rooms of Units 1 and 2 are combined in one area, the buildings are separated by Teflon lined sliding bearings. These isolation joints provide the auxiliary building with structural independence from the surrounding buildings in the lateral direction.

General layouts at the various elevations and sections through the auxiliary building are shown on Figures 1.2–7 and 1.2–14.

5.4.1.1 Fuel Storage Facility

5.4.1.1.1 New Fuel Storage

The new fuel storage is bounded by column lines 17.2 and 18.9 and column lines S and N.3 at Elevation 38-6.

5.4.1.1.2 Spent Fuel Storage

Spent fuel storage is provided between column lines 17.2 and 18.9 and column lines H.4 and L.5 at Elevation (-)2–0. The storage area consists of a reinforced concrete pool lined with one-fourth inch thick stainless steel plate to Elevation 38-6. Normal water level is to Elevation 36-6. The spent fuel is protected from a main steam line rupture by a reinforced concrete wall which is located north of and parallel to the pool.

A leak chase system consisting of channels embedded behind the liner plate at all seams and connected to a collector system is used to monitor and control any possible leakage from the pool. A description of the monitoring system is provided in Section 5.4.3.3.4.

An overhead crane with the capacity of 125 tons is employed to handle the spent fuel cask. The crane is designed to meet the single failure requirements in accordance with NUREG-0612 and NUREG-0554. A system of interlocks is also provided on the crane that defines a safe load path. This safe load path is indicated on Figure 5.3–5.

5.4.1.1.3 Compliance with Safety Guide 13

The design of the fuel storage structures complies with the structural requirements of Safety Guide 13.

5.4.2 CONSTRUCTION MATERIALS

The following materials are used in the construction of the auxiliary building.

a. Structural and miscellaneous steel

Rolled shapes, plates, and bars	ASTM A-36
Crane rails	Bethlehem Steel
High strength bolts	ASTM A-325 or A-490
Stainless Steel	ASTM A-240 Type 304

b. Reinforcing steel

Deformed bars	ASTM A-615 Grade 60

c. Concrete, 28 day strength (psi)

Lean concrete backfill	2,000
Foundation mat slab	
Auxiliary Building	3,000
Warehouse portion of the Auxiliary Building	4,000
All other concrete	3,000

d. Interior coatings (Original Construction)

1. Concrete and masonry surfaces

Primer	Keeler & Long Number 7107 epoxy white primer
Finish coat (floor and wainscot)	Keeler & Long Number 7107 epoxy white enamel (tinted)
Finish coat (above wainscot)	Keeler & Long Number 7107 epoxy white enamel

2. Carbon steel

Primer (wainscot)	Keeler & Long Number. 7107 epoxy white primer
Finish coat (wainscot)	Keeler & Long Number. 7475 epoxy white enamel (tinted)

Primer (above wainscot)	Keeler & Long Tri-Polar Primer
Finish coat (above wainscot)	Keeler & Long Tri-Polar Enamel

e. Coating materials used in the auxiliary building represent current day technology and comply with current day environmental requirements regarding volatile organic compound (VOC) content and hazardous material constituents such as lead, asbestos, and hexavalent chrome. Specific products are selected to provide service performance comparable to or improved over that of the original coating materials. Coating colors are chosen to help maintain a safe working environment.

5.4.3 DESIGN BASES

The design of the auxiliary building provides the required features as outlined in General Design Criteria 1, 2, 3, 4, 5, 61, 62, 63, Appendix A of 10 CFR Part 50.

5.4.3.1 Bases for Design Loads

The auxiliary building is designed for all credible combinations of loading, including loads under normal operation, loads during a steam line rupture, and loads due to adverse environmental conditions. The following loadings are considered:

a. Dead loads

b. Live loads

c. Thermal loads

d. Earthquake loads

e. Lateral pressure loads

f. Wind and tornado loads

g. Pipe restraint loads

h. Pipe whipping loads

i. Cask drop loads

j. Fuel transfer tube bellows

5.4.3.1.1 Dead Loads

These loads consist of the weight of all structural materials, including all partitions, hangers, trays, pads and pedestals, and equipment dead loads. These are specified on the drawings supplied by the manufacturers of the equipment installed within the building.

5.4.3.1.2 Live Loads

Live loads consist of design floor loads, pool and tank liquid weights, piping loads, and equipment live loads as specified on the drawings supplied by the manufacturers of the equipment installed within the building. A snow load of 60 psf is applied on the roof.

5.4.3.1.3 Thermal Loads

Thermal loads are those induced in the spent fuel pool floor and walls due to the thermal gradients across these elements. Thermal gradients may be caused by an increase in water temperature during operating conditions or by an accident. The interior temperatures of the pool are assumed to be 150°F at operating conditions and 212°F during an accident. The ambient temperature exterior to the pool is assumed to be 55°F for computation of stresses.

5.4.3.1.4 Earthquake Loads

These loads are as defined in Sections 5.2.2.1.5 and 5.8.

5.4.3.1.5 Lateral Pressure Loads

The static lateral soil pressure loads are as follows:

Active -	above water table	55 (psf/ft)
	below water table	95 (psf/ft)
Passive -	above water table	370 (psf/ft)
	below water table	200 (psf/ft)

A surcharge of 200 psf, or an 8,000 pound wheel load, is considered in all cases.

Buoyant forces resulting from the displacement of ground water are supplied to the structure. The following water levels are considered:

Ground water	Elevation 5–0
Flood water	Elevation 18–1

The dynamic soil and hydrodynamic pressures are discussed in Section 5.8.2.2 and are considered to act on the structural elements below grade, where applicable.

5.4.3.1.6 Wind and Tornado Loads

Wind loads for the auxiliary building are determined on the basis of the American Society of Civil Engineers (ASCE) Paper 3269, “Wind Forces on Structures,” using the highest wind velocity at the site for a 100 year recurrence period. The ASCE Paper 3269 is used mainly to determine the shape factors. Based upon the site location and the structure classification, the design wind velocity is taken to be 115 mph with gusts up to 140 mph.

The auxiliary building has been analyzed for tornado loads (not coincident with an accident or earthquake) on the following basis:

a. Differential bursting pressure between the interior and exterior of the structure is assumed to be three psi pressure occurring in three seconds (1 psi/second), followed by a calm for two seconds and a re pressurization.

b. Lateral loads on the structure are based on a tornado funnel which is conservatively assumed to have a peripheral tangential velocity of 300 mph and a translational velocity of 60 mph. The applicable portions of the wind design methods described in the ASCE Paper 3269 are used, particularly for the shape factors. The provisions in the paper for gust factors and variation of wind velocity with respect to height are not applied. The wind velocity is assumed to be uniformly distributed over the height of the structure.

c. Tornado driven missiles as defined in Section 5.2.5.1.2.

With the exception of the missile impact area, the allowable stresses to resist the effects of tornadoes are 90 percent of the yield strength of the reinforcing steel and 85 percent of the ultimate strength of the concrete.

A discussion of the probability of tornado occurrence is presented in Section 2.3.

5.4.3.1.7 Pipe Restraint Loads

These are the loads imparted to the structure from the pipe restraints produced by either a postulated pipe rupture or an earthquake. See Section 6.1.4 for pipe rupture criteria.

5.4.3.1.8 Pipe Whipping Loads

These are the loads imposed on the structure due to whipping from a postulated pipe rupture. See Section 6.1.4 for pipe rupture criteria.

5.4.3.1.9 Cask Drop Loads

The upgrade of the MP2 Spent Fuel Cask Crane to single failure criteria has precluded the need to postulate a Spent Fuel Cask Drop accident. The crane upgrade will give positive control of the lifted loads even with the worst single failure. This section has been retained to provide the historical background for the design and analysis of the spent fuel pool.

The following design criteria were used in the analysis of the spent fuel pool in the event that a cask is accidentally dropped:

a. Weight of cask in air (lb.)	200,000	
b. Length of cask (feet)	19	
c. Diameter of cask (feet)	8	
d. Distance of drop (feet)		
In air		2.75
In water		35.5

As shown on Figure 5.3–5, the only area of the spent fuel pool into which the cask could be dropped directly is the cask laydown area. The cask laydown area is isolated from the spent fuel storage area by two-foot thick, permanent, reinforced concrete walls and a temporary gate placed in the fuel transfer slot. The base slab of the cask laydown area is composed of seven feet of reinforced concrete resting on a mass of monolithic concrete which, in turn, rests on bedrock. Therefore, a cask dropped in this area would travel vertically downward as restrained by the surrounding walls. Any damage would be limited to rupturing of the spent fuel pool liner and local superficial crushing of concrete in the area of impact of the end of the cask. Leakage through the ruptured liner would be detected in the control room and would be stopped by closing the valve that connects the leak collection channel for the ruptured zone(s) to the leak detection instrumentation.

If during handling, the cask is dropped on or near point “A,” as shown on Figure 5.3–5, there exists a possibility that the cask could fall or tumble into the spent fuel storage area. The fall would provide some local concrete crushing in the spent fuel pool and laydown area walls at elevation (+) 38 feet 6 inches. The cask would then slide into the spent fuel storage area of the pool. The cask would crush the spent fuel rack module(s) that it landed on, but the buoyant effect of the water combined with the crushing of the rack would dissipate most of the kinetic energy of the falling cask. Therefore, the probable damage would be limited to rupture of the spent fuel pool liner and local crushing of concrete where the cask impacted. The dose impact of damage to spent fuel stored in the pool (both intact and consolidated) would be mitigated by administratively controlling the age of the stored fuel in the affected area around the cask laydown area of the spent fuel pool. Whenever a shielded cask is on the refueling floor, all stored fuel within a specified distance of the cask laydown area shall have decayed a minimum amount of time from subcritical reactor operation in accordance with the Technical Specifications. The seven foot thick base and six foot thick walls, of reinforced concrete, would remain intact. Leakage would be detected and stopped as described above.

Makeup water would be available as discussed in Section 9.5.

5.4.3.1.10 Fuel Transfer Tube Bellows

The following loads were used in the design of the fuel transfer tube and bellows:

Design pressure, internal (psi)	60
Design temperature (°F)	290

Lateral movement (inches)	0.14
Axial movement, expansion or contraction (in.)	0.5

Displacements are selected to accommodate an assumed differential settlement of one-eighth inch between the buildings. Since both the containment and auxiliary buildings are founded on rock, this motion is minimal.

5.4.3.2 Design Load Combinations

To ensure the structural integrity of the auxiliary building, the working stress method of design is used for the various loading combinations. For the operating conditions, normal allowable stresses given in the American Institute of Steel Construction (AISC) Manual of Steel Construction 1963, and the American Concrete Institute (ACI)-318-63, "Building Code Requirements for Reinforced Concrete" are used. These allowable stresses are increased by 33-1⁄3 percent for the 115 mph base wind loads and the operating basis earthquake (OBE) loads. For the tornado wind and the design basis earthquake (DBE), the allowable stresses are 90 percent of the yield strength of the reinforcing, and 85 percent of the ultimate strength of concrete.

The load combinations are listed:

a. $D + L$

b. $D + L + W_w$

c. $D + L + W_t$

d. $D + L + E$

e. $D + L + E'$

f. $D + L + P_e + W_t + H_w$

g. $D + L + T + E$

h. $D + L + T + E'$

i. $D + L + P_e + E$

j. $D + L + P_e + E'$

k. $D + L + F_p + E'$

l. $D + L + F_c$

m. $D + L + F_r + E'$

where:

D = dead loads

L = live loads

W_w = wind loads (115 mph base)

W_t = tornado loads (360 mph base

E = OBE

E′ = DBE

P_e = soil pressure

F_p = pipe whipping loads

F_c = cask drop loads

F_r = pipe restraint loads

H_w = hydrostatic pressure

T = thermal loads

These load combinations are applied to the portions of the structure housing or associated with the various systems as follows:

Item		
a.	Cask crane structure	a, b, c, d, e
b.	Chemical addition and sampling system	a, c, d, e, f, i, j
c.	Chemical and volume control system (CVCS)	a, c, d, e, f, i, j
d.	Containment spray pumps	a, c, d, e, f, i, j
e.	Control room	a, b, c, d, e
f.	Diesel generator room and day tanks	a, c, d, e, f
g.	Electrical distribution system	a, b, c, d, e
h.	New fuel storage	a, b, c, d, e
i.	Reactor building closed cooling water (RBCCW) system	a, c, d, e, f, i, j
j.	Safety injection systems (SIS)	a, c, d, e, f, i, j
k.	Spent fuel cooling system	a, c, d, e, f, i, j
l.	Spent fuel pool	a, b, c, d, e, g, h, k, l
m.	Waste processing systems	a, c, d, e, f, i, j

The design loads and stresses for all concrete walls and columns are within the allowable values for each wall or column. Design stresses are not applicable for the columns inasmuch as the interaction diagrams are used to proportion reinforcing and concrete required to support a given load with its corresponding eccentricity. Exterior walls having vertical and lateral loads are designed for bending and axial loadings, and the resulting combined stresses are kept within the code allowables. There are numerous walls, slabs, columns, and beams within the buildings, each element of which was designed for the pertinent loading. Design stresses in the various components are recorded in the design calculations. The allowable loading, or combined loading, depends on the reinforcing which was added and varies considerably depending upon the applied conditions.

The maximum combined stress ratio in the steel cask crane frame is 0.904, resulting from an axial stress of 3.53 ksi and a bending stress of 31.8 ksi. This stress occurs under loading combination e., above.

In addition to the various load combinations included, all Category I structures outside the containment that could be pressurized in the event of a postulated pipe rupture are designed to satisfy the following load combinations:

(1) $U = D + L + T + R_a + 1.5P_a$

(2) $U = D + L + T + R_a + 1.25P_a + F_r + F_c + F_j + 1.25E$

(3) $U = D + L + T + R_a + P_a + F_r + F_c + F_j + E'$

where:

U = total design load

D = dead loads

L = live loads

R_a = pipe reactions under thermal condition generated by a postulated break

F_r = equivalent static pipe restraint loads

F_c = equivalent static pipe whipping loads, including the effects of missiles

F_j = equivalent static jet impingement loads

P_a = equivalent static differential pressure load generated by a postulated pipe break

E = OBE loads

E′ = DBE loads

T = thermal loads under thermal conditions generated by a postulated break

For the above loading conditions, the allowable stresses are as follows:

Concrete Construction:

90 percent of the yield strength for reinforcing steel

85 percent of the ultimate strength of concrete

Steel Construction:

Allowable stresses specified in Part 2 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Building," April, 1963.

5.4.3.3 Structural Analysis

5.4.3.3.1 Seismic Analysis

Seismic analysis is performed in accordance with Section 5.8.

5.4.3.3.2 Wind and Tornado Analysis

The design wind loads on the auxiliary building are a function of the kinetic energy per volume of the moving air mass. The product of one-half of the air density and the square of the resultant design velocity results in a pressure corresponding to the design wind.

Determination of the design wind pressure on the structure is in accordance with the ASCE Paper 3269, "Wind Forces on Structures."

The pressure corresponding to the standard air at 0.07651 pcf at 15°C and 760 mm of mercury in terms of the velocity at the appropriate height zone is given by:

$$q = 0.002558V^2$$

Similarly, the design pressure, including the effect of the shape coefficient (C_d), is given by:

$$p = q \times C_d = 0.002558V^2C_d$$

Using these equations and the wind velocities given in Section 5.4.3.1.6, the wind forces are calculated for the various parts of the auxiliary building and are then applied to the structure.

The spent fuel pool is protected from the tornado missile, which is described in Section 5.4.3.1.6, by a concrete roof over the cask crane frame and missile proof metal siding, as per Section 5.4.3.3.6.

5.4.3.3.3 Cask Drop in Spent Fuel Pool

The design criteria stated in Section 5.4.3.1.9 is applied using the following assumptions:

Water density at 120°F (lb/cubic foot)	61.7 borated water density
Strength of concrete (psi)	3,000
Modulus of concrete (E) (ksf)	4.78×10^5

The striking velocity at collision is determined by the use of the formula as defined in the paper "Tornado Protection for the Spent Fuel Storage Pool" by Miller and Williams, APED-5696 Class I, November 1968. The friction factor is ignored. A dynamic pressure factor (C_q) of unity is assumed in the analysis.

To determine the effects of accidentally dropping the cask on the bottom of the spent fuel pool in the cast laydown area, the kinetic energy was computed. This energy is considered to be dissipated as elastic strain energy within the bounds of the lean concrete mass supporting the laydown area. The contact area between the cask and the concrete is small.

However, the actual contact stress is calculated and found to be less than 1,000 psi. The impact of this contact may cause some local damage to the liner plate and/or concrete. However, the extent of the damage is small and will not result in any significant structural damage to the floor. The maximum stress in the concrete occurs at Elevation (-)2-0 and diminishes rapidly as the stress profile extends downward.

5.4.3.3.4 Stainless Steel Liner Plate for Spent Fuel Pool

[Note: Section 5.4.3.3.4 describes preoperational testing and repair of the spent fuel pool liner. It is retained, without change, for a historical record. The spent fuel pool leak monitoring and detection system is described in Section 9.5.2.1.]

Provision is made for ensuring the leak tightness of the spent fuel pool and refueling canal liner plate.

The test consists of two parts. In the first part of the test, a halogenated hydrocarbon gas is forced through the leak monitoring channels and a halogenated hydrocarbon detector is used to locate leaks in the liner plate weld seams inside the spent fuel pool. All leak indications are marked and repaired after the halogenated hydrocarbon gas is removed from the leak monitoring channels. All weld repairs are checked by a liquid penetrant test.

Upon completion of the repair, the pool is filled with water to the design level and monitored for 48 hours.

If no water is detected in the leak monitoring system, the pool is considered acceptable.

5.4.3.3.5 Fuel Transfer Tube

One expansion joint is installed in the fuel transfer tube. It is a bellows located in the fuel transfer canal in the auxiliary building.

The loads used in its design are given in Section 5.4.3.1.10.

The outside of the expansion joint in the transfer canal may be visually inspected by draining the transfer canal or by remote means. A test connection on the bellows provides a means of testing for bellows integrity. Repair would require draining the transfer canal.

A detail of the fuel transfer tube is shown in Figure 5.2–10.

5.4.3.3.6 Spent Fuel Pool Missile Protection

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

Although a large portion of the missile energy would be spent in penetrating the conventional siding (if, indeed, penetration would occur), there is no possible trajectory that would allow the missile to directly impact the fuel assembly elements located at the bottom of the spent fuel pool.

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

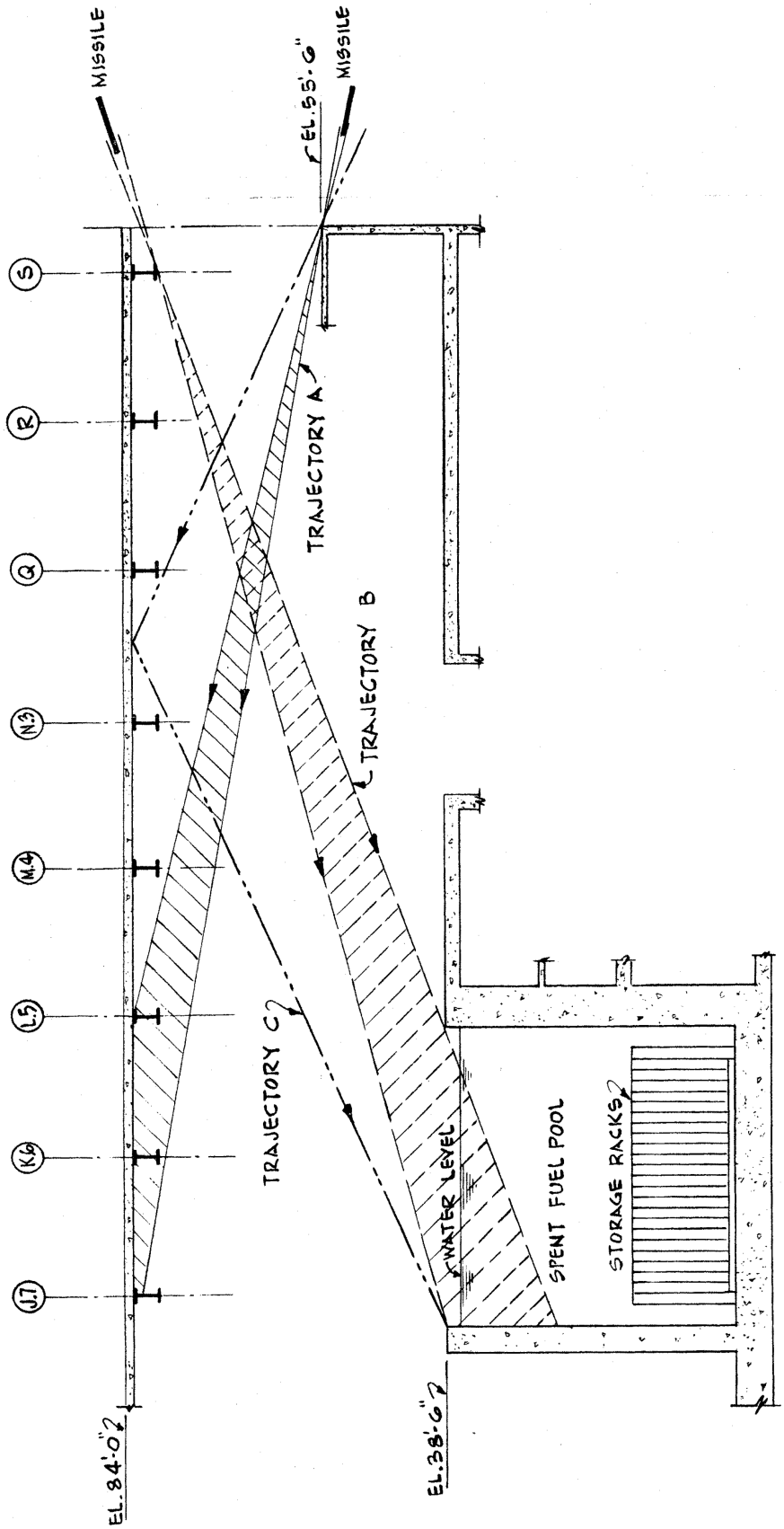


FIGURE 5.4–1 PROBABLE MISSILE TRAJECTORIES INSIDE AUXILIARY BUILDING

FIGURE 5.4–2 MISSILE RESISTANT SIDING DETAIL

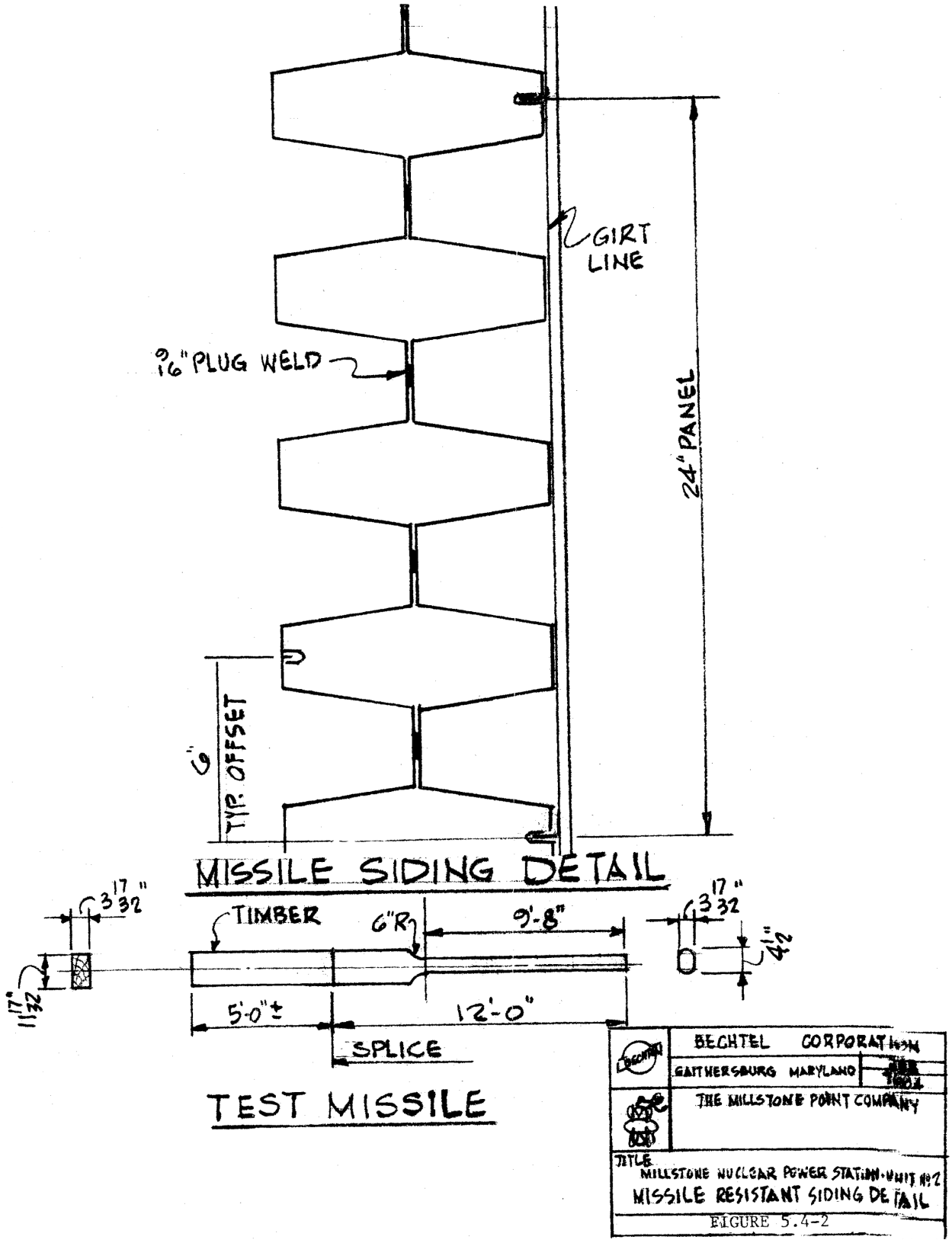
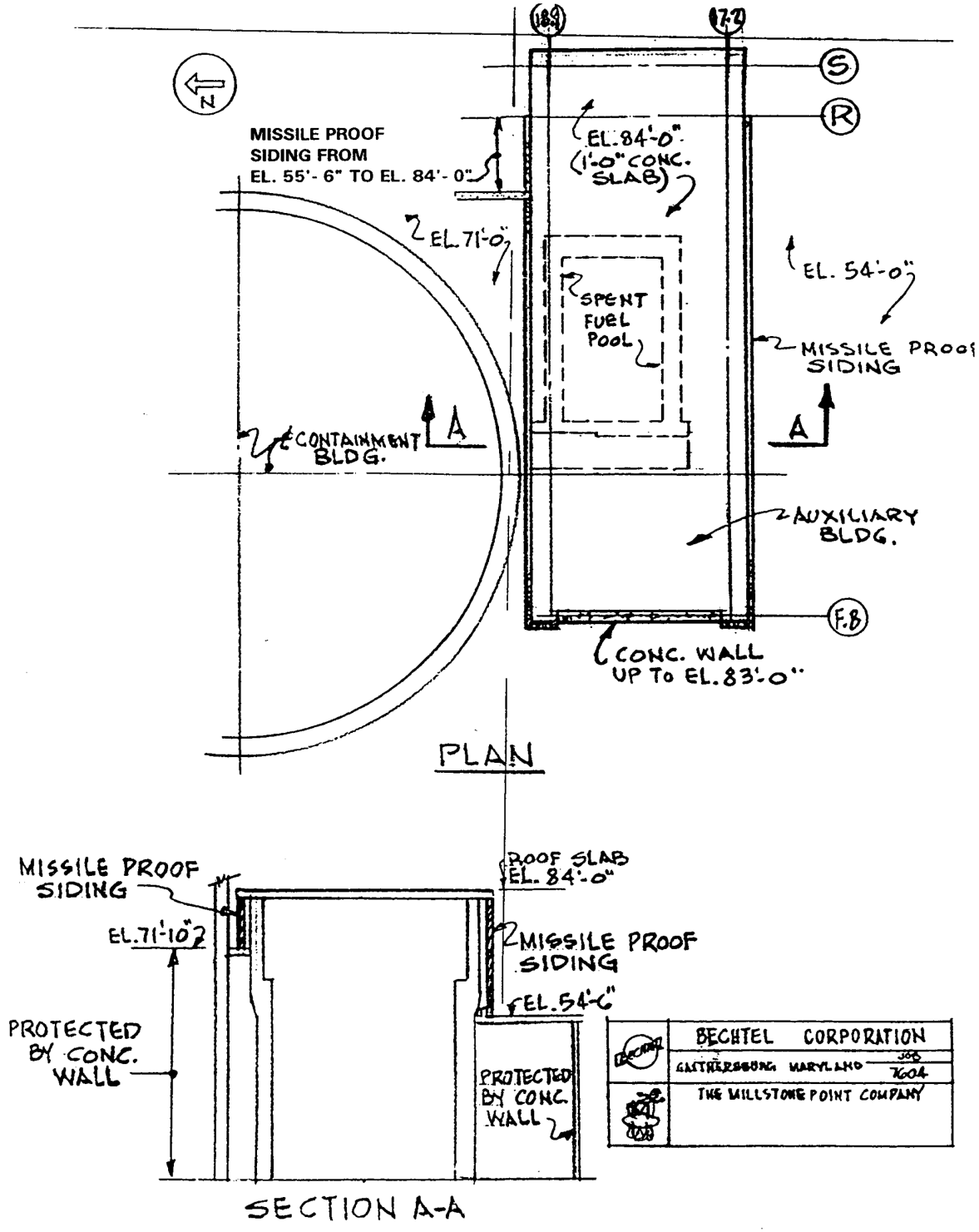


FIGURE 5.4–3 LOCATION OF MISSILE RESISTANT SIDING



5.5 TURBINE BUILDING

5.5.1 GENERAL DESCRIPTION

The turbine building is a rigid framed steel structure with metal siding and precast concrete panels on the exterior. Blowout panels are located on the east wall, column line “E” on the upper portion of the metal siding. The foundations for the frame are spread footings bearing on lean concrete backfill which extends to rock. The turbine-generator pedestal is a low-tuned mass concrete structure which is also founded on lean concrete backfill which extends to rock.

The pedestal is separated from the surrounding floor slabs by teflon lined sliding bearings as indicated on Figure 5.5–1. The turbine building main frame is connected to the Unit 1 turbine building by sliding connections so that it is an independent structure.

As shown on Figures 1.2–3, 1.2–4, and 1.2–5, the heater bay running between lines E and E.5, separates the turbine building from the auxiliary building. This bay is connected to the auxiliary building for lateral support, but is separated from the turbine building main frame by sliding connections as indicated on Figure 5.5–2. Sections through the turbine building are shown on Figures 1.2–15 and 1.2–16.

The Unit 1 and Unit 2 turbine buildings are on the same centerline so that the crane runway system is continuous through both units. A 195 ton overhead crane operates on the runway system. The crane is designed to meet the loading requirements of the applicable portions of Crane Manufacturers Associated of America (CMAA) Specification 70.

5.5.2 CONSTRUCTION MATERIALS

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

a. Structural and Miscellaneous Steel

Rolled shapes, plates and bars	ASTM A-36
Crane rails	Bethlehem Steel
High strength bolts	ASTM A-325 or A-490

b. Reinforcing Steel

Column ties in turbine pedestals	ASTM A-615 Grade 40
All other deformed bars	ASTM A-615 Grade 60

c. Concrete, 28 day strength (psi)

Turbine pedestals, operating floor slabs, and column footings	4000
All other concrete	3000

d. Interior coatings (Original Construction)

Concrete and masonry surfaces in switchgear room

Floor	Keeler & Long Number 7107 Epoxy Grey

e. Interior maintenance coatings

Coating materials used in the turbine building represent current day technology and comply with current day environmental requirements regarding volatile organic compound (VOC) content and hazardous material constituents such as lead, asbestos, and hexavalent chrome. Specific products are selected to provide service performance comparable to or improved over that of the original coating materials. Coating colors are chosen to help maintain a safe working environment.

5.5.3 DESIGN BASES

The design of the turbine building provides the required features as outlined in Criteria 2, 3, 4, 5, Appendix A of 10 CFR Part 50.

5.5.3.1 Bases for Design Loads

Although only a portion of the turbine building houses Class I equipment and components, the entire structural system is designed for a Seismic Class I loading.

The design loads imposed on the structure are as follows:

a. Dead loads

b. Live loads

c. Thermal loads

d. Earthquake loads

e. Wind and tornado loads

f. Crane loads

5.5.3.1.1 Dead Loads

These loads consist of the weight of all structural materials, including all partitions, hangers, trays, pads, and pedestals. Equipment dead loads are those specified on the drawings supplied by the manufacturers of the various equipment installed within the building.

5.5.3.1.2 Live Loads

These loads consist of design floor loads, tank liquid weights, piping loads, and equipment live loads specified on the drawings supplied by the manufacturers of the various equipment installed within the building.

A snow load of 60 psf is applied to the exposed roof over the area housing Class I equipment or components, and 40 psf for all other exposed roof area.

5.5.3.1.3 Thermal Loads

Expansion and contraction in structural members due to changes in temperature are considered. Provisions for normal expansion and contraction are made by the use of slotted connections as required.

5.5.3.1.4 Earthquake Loads

Seismic analysis is as defined in Sections 5.2.2.1.5 and 5.8.

5.5.3.1.5 Wind and Tornado Loads

Wind loads for the turbine building are determined on the basis of the ASCE Paper 3269, “Wind Forces on Structures,” using the highest wind velocity at the site for a 100 year recurrence period. The ASCE paper is used mainly to determine the shape factors. Based upon the site location and the structure classification, the design wind and velocity is taken to be 115 mph with gusts up to 140 mph.

The turbine building is analyzed for tornado loads on the following basis:

a. Differential bursting pressure between the interior and exterior of the structure is assumed to be 3 psi pressure occurring in three seconds (1 psi/second), followed by a calm for two seconds and a re pressurization.

b. Lateral loads on the structure are based on a tornado funnel which is conservatively assumed to have a peripheral tangential velocity of 300 mph and a translational velocity of 60 mph. These velocities are added together, resulting in a design basis tornado wind velocity of 360 mph. The applicable portions of the wind design methods described in the ASCE paper are used, particularly for the shape factors. The provisions in the paper for gust factors and variation of wind velocity with respect to height are not applied. The wind velocity is assumed to be uniformly distributed over the height of the structure.

c. A tornado driven missile as defined in Section 5.2.5.1.2.

With the exception of the missile impact area, the allowable stresses to resist the effects of tornadoes are 90 percent of the yield strength of the reinforcing steel and 85 percent of the ultimate strength of the concrete.

A discussion of the probability of tornado occurrence is presented in Section 2.3.

5.5.3.1.6 Crane Loads

These loads include the dead and live loads of the turbine building crane.

5.5.3.2 Design Load Combinations

To ensure the structural integrity of the turbine building, the allowable stresses specified in the 1963 AISC "Manual of Steel Construction," and the ACI-318-63 "Building Code Requirements for Reinforced Concrete," are used. These allowable stresses are increased by 33-1/3 percent for the 115 mph base wind loads and the operating basis earthquake loads. For the tornado wind and the design base earthquake, the allowable stresses are 90 percent of the yield strength of the reinforcing, and 85 percent of the ultimate strength of concrete.

The following load combinations are considered:

a. $D + L + C_D + C_L$

b. $D + L + C_D + W_w$

c. $D + L + C_D + W_T$

d. $D + C_D + L + E$

e. $D + C_D + L + E'$

Where

D = dead loads

L = live loads

C_D = crane dead loads

C_L = crane live loads

W_w= wind loads (115 mph base)

W_t = tornado loads (360 mph base)

E = operating basis earthquake

E′ = design basis earthquake

The design loads and stresses for all concrete walls and columns are within the allowable values for each wall or column. Design stresses are not applicable for the columns inasmuch as the interaction diagrams are used to proportion reinforcing and concrete required to support a given load with its corresponding eccentricity. Exterior walls having vertical and lateral loads are designed for bending and axial loadings and the resulting combined stresses are kept within the

code allowables. There are numerous walls, slabs, columns, and beams within the building, each element of which was designed for the pertinent loading. Design stresses for the various components are recorded in the design calculations. The allowable loading, or combined loading, depends on the reinforcing which was added and varies considerably depending upon the applied conditions.

The following loading combinations and reinforcing ratios (of moment/axial force and shear) shown in Table 5.5-1 governed the design.

5.5.3.3 Structural Analysis

The main frame of the turbine building and the turbine generator pedestal are designed for the load combinations stated in Section 5.5.3.2 using working stress methods.

5.5.3.3.1 Seismic Analysis

Analysis of the turbine building for the effects of an earthquake is performed in accordance with Section 5.8.

These earthquake loads are superimposed on the other structural loads to obtain the loading combinations as stated in Section 5.5.3.2.

5.5.3.3.2 Wind and Tornado Analysis

The design wind loads on the turbine building are a function of the kinetic energy per volume of the moving air mass. The product of one-half of the air density and the square of the resultant design velocity results in a pressure corresponding to the design wind.

Determination of the design wind pressure on the structure is in accordance with the ASCE Paper 3269, “Wind Forces on Structures.”

The pressure corresponding to the standard air at 0.07651 pcf at 15°C and 760 mm of mercury in terms of the velocity at the appropriate height zone is given by:

$$q = 0.002558\ V^2$$

Similarly, the design pressure, including the effect of the shape coefficient (C_d), is given by:

$$p = q \times C_d = 0.002558 V^2 C_d$$

Using these equations and the wind velocities given in Section 5.5.3.1.5, the wind forces are calculated for the main frame and are then applied to the structure.

Class I systems and equipment are protected from a tornado missile by structural walls and slabs.

TABLE 5.5-1 MAXIMUM ACTUAL STRESSES - TURBINE BUILDING

Design Component	Loading Combination	Moment/Axial Force As required ≤ 1.0 As provided	Shear Actual ≤ 1.0 Allowable
Footing slab	C	0.92	0.98
Substructure walls	C	1.00	0.84
Operating floor	C	0.94	0.80
Superstructure walls	D	0.81	0.79
Roof	Tornado Missile Protection Requirements		

FIGURE 5.5–1 JOINT DETAIL AT TURBINE PEDESTAL

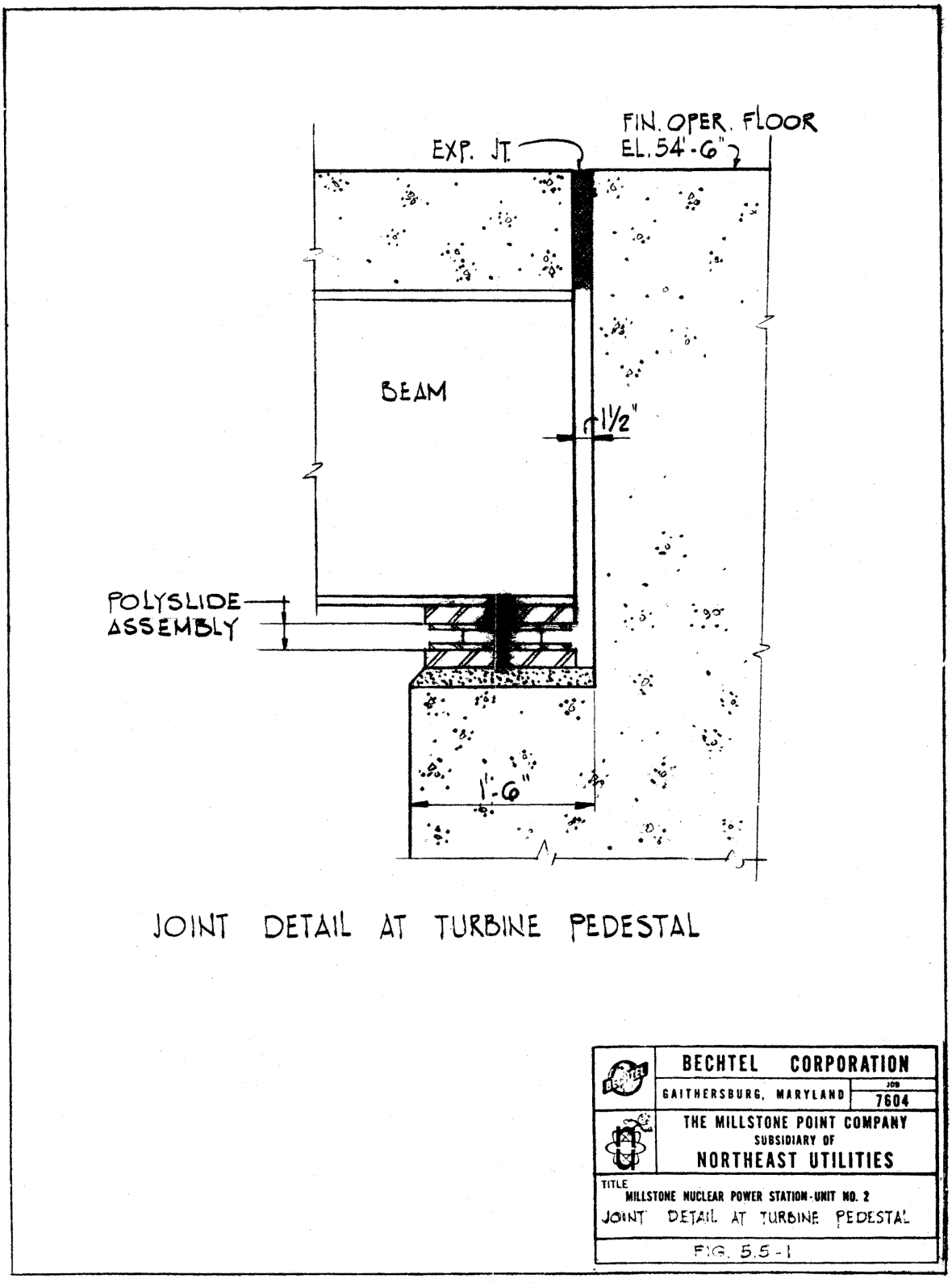
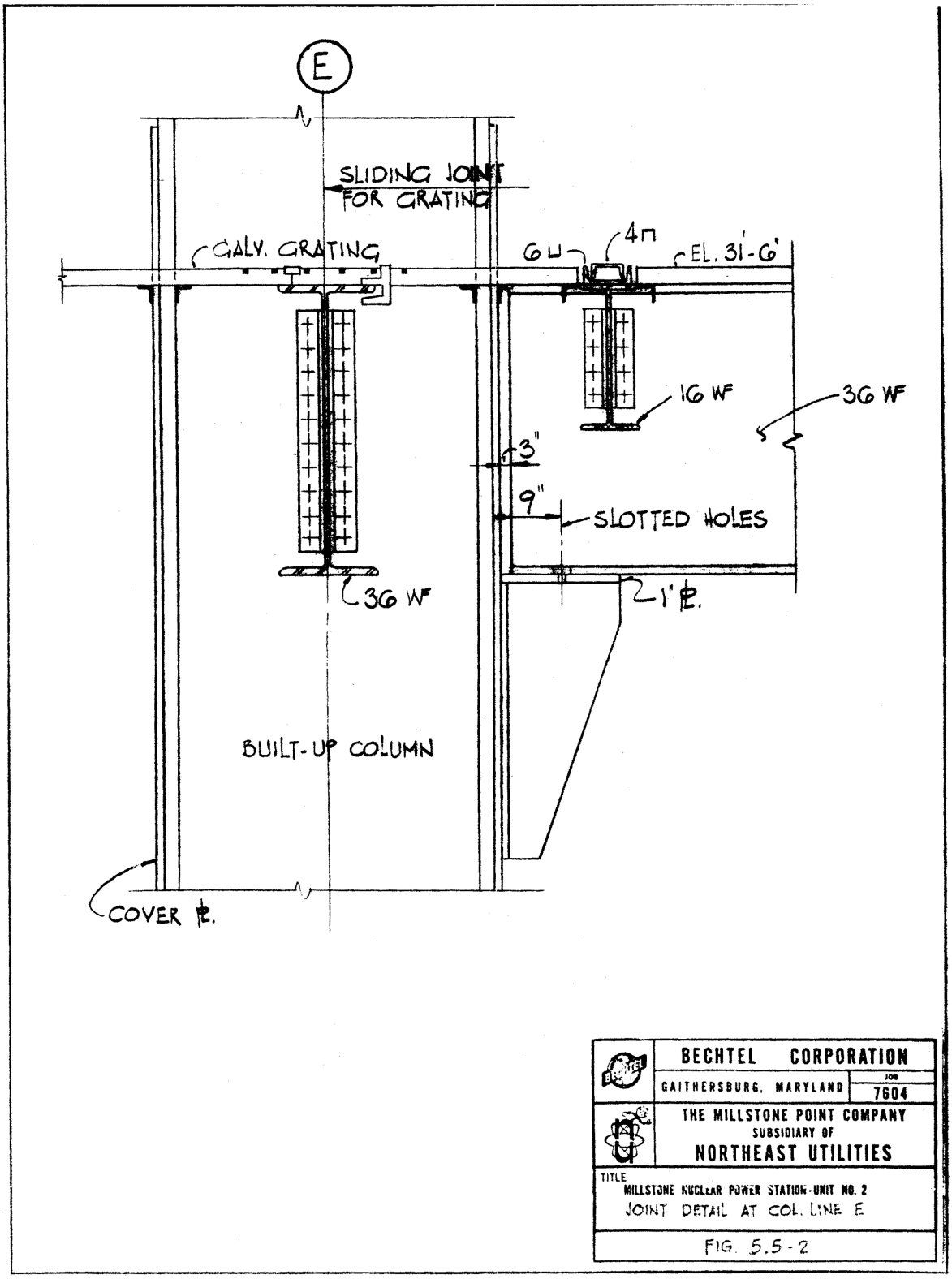


FIGURE 5.5–2 JOINT DETAIL AT COL. LINE E



5.6 INTAKE STRUCTURE

5.6.1 GENERAL DESCRIPTION

The intake structure, located west of the main plant, is a reinforced concrete structure founded on bedrock. It houses four circulating water pumps which supply water from Niantic Bay to the condensers positioned under the turbine-generator. Also located in the structure are three service cooling water pumps for the closed cooling water system. Access to these pumps is provided through hatches with removable covers.

The design of the intake structure incorporates several features which will ensure the safe, continuous operation of the circulating water system. These items are:

a. Trash racks and traveling screens which protect the pumps and condensers from foreign bodies present in the water supply.

b. A cutoff wall which extends 10 feet below the minimum water level to prevent the ecologically rich surface water from entering the system.

c. Passages provided as exit routes for fish which enter below the cutoff wall.

d. Vertical guides in the sides of each intake channel to receive stop-logs so that individual channels may be drained.

e. A monorail and trolley provided to service the interior of the intake structure.

An adjacent Class II building, which houses the chlorination equipment, is isolated from the intake structure by a joint filled with compressible material.

General layouts of the intake structure and circulating water system are shown on Figures 5.6–1 and 5.6–2, respectively.

5.6.2 CONSTRUCTION MATERIALS

The following materials are used in the construction of the intake structure:

a. Structural and miscellaneous steel		
	Rolled shapes, plates and bars	ASTM A-36
	High strength bolts	ASTM A-325 or A-490
b. Reinforcing steel	ASTM A-615 Grade 60	
c. Concrete, 28 day strength (psi)	4000	
d. Interior coatings (Original Construction)		
	Surfaces below elevation 14–0	Woolsey antifouling paint

Carbon steel	Epoxy

e. Interior maintenance coatings

Coating materials used in the intake structure represent current day technology and comply with current day environmental requirements regarding volatile organic compound (VOC) content and hazardous material constituents such as lead, asbestos, and hexavalent chrome. Specific products are selected to provide service performance comparable to or improved over that of the original coating materials. Coating colors are chosen to help maintain a safe working environment.

5.6.3 DESIGN BASES

The design of the intake structure provides the required features as outlined in Criteria 1, 2, 3, 4, 5, 44, 46, Appendix A of 10 CFR Part 50.

5.6.3.1 Bases for Design Loads

The intake structure is designed for all credible conditions of loadings including loads from normal operation and those due to adverse environmental conditions. The following loadings are considered.

a. Dead loads

b. Live loads

c. Earthquake loads

d. Lateral pressure loads

e. Wind and tornado loads

f. Hurricane wave loads

5.6.3.1.1 Dead Loads

These loads consist of the weight of all structural materials including partitions, hangers, trays, and pads. Equipment dead loads are those specified on the drawings supplied by the manufacturers of the various types of equipment installed within the building.

5.6.3.1.2 Live Loads

These loads consist of the design floor loads, tank liquid weights, piping loads, and equipment live loads specified on the drawings supplied by the manufacturers of the various types of equipment installed within the building.

A snow load of 60 psf is applied to the exposed roof.

5.6.3.1.3 Earthquake Loads

These loads are as defined in Section 5.8.

5.6.3.1.4 Lateral Pressure Loads

The lateral pressure loads include the active and the passive soil pressures where applicable. The buoyant and lateral forces of the displaced water are also considered.

5.6.3.1.5 Wind and Tornado Loads

Wind loads on the intake structure are determined on the basis of the ASCE Paper 3269, "Wind Forces on Structures," using the highest wind velocity at the site for a 100 year recurrence period. The ASCE Paper is used mainly to determine the shape factors. Based upon the site location and the structure classification, the design wind velocity is taken to be 115 mph with gusts up to 140 mph.

The intake structure is analyzed for tornado loads on the following basis:

a. Differential bursting pressure between the interior and exterior of the structure is assumed to be 3 psi pressure occurring in three seconds (1 psi/second), followed by a calm for two seconds and a re pressurization.

b. Lateral loads on the intake structure are based on a tornado funnel which is conservatively assumed to have a peripheral tangential velocity of 300 mph and a translational velocity of 60 mph. These velocities are added together, resulting in a design basis tornado wind velocity of 360 mph. The applicable portions of the wind design methods described in the ASCE Paper 3269 are used, particularly for the shape factors. The provisions in the paper for gust factors and variation of wind velocity with respect to height are not applied. The wind velocity is assumed to be uniformly distributed over the height of the structure.

c. A tornado-driven missile as defined in Section 5.2.5.1.2.

With the exception of the missile impact area, the allowable stresses to resist the effects of tornadoes are 90 percent of the yield strength of the reinforcing steel and 85 percent of the ultimate strength of the concrete.

A discussion of the probability of tornado occurrence is presented in Section 2.3.

5.6.3.1.6 Hurricane Wave Loads

These loads, resulting from a hurricane wave striking the front of the intake structure, are considered.

5.6.3.2 Design Load Combinations

The structural integrity of the intake structure is ensured by using the allowable stresses as specified in the 1963 AISC Manual of Steel Construction, and the ACI-318-63, "Building Code Requirements for Reinforced Concrete," for the various loading combinations. A 33-1/3 percent increase in the allowable stresses is permitted for the combined stresses involving wind, earthquake, or tornado.

The following load combinations are used in the design:

a. $D + L$

b. $D + L + W_t$

c. $D + L + E'$

d. $D + L + W + H$

where:

D = dead loads

L = live loads

W_t = tornado loads

W = wind loads

E′ = design basis earthquake

H = hurricane wave loads

The design loads and stresses for all concrete walls and columns are within the allowable values for each wall or column. Design stresses are not applicable for the columns inasmuch as the interaction diagrams are used to proportion reinforcing and concrete required to support a given load with its corresponding eccentricity. Exterior walls having vertical and lateral loads are designed for bending and axial loadings and the resulting combined stresses are kept within the code allowables. There are numerous walls, slabs, columns, and beams within the buildings, each element of which was designed for the pertinent loading. Design stresses for the various components are recorded in the design calculations. The allowable loading, or combined loading, depends on the reinforcing which was added and varies considerably depending upon the applied conditions.

The maximum combined stress ratio in the building frame is 0.98, resulting from an axial stress of 6.75 ksi and a bending stress of 25.1 ksi. This stress occurs under loading combination c above.

5.6.3.3 Structural Analysis

The intake structure is designed and analyzed using the working stress method.

5.6.3.3.1 Seismic Analysis

The analysis of the intake structure subjected to seismic loads is performed in accordance with the method described in Section 5.8.

These earthquake loads are superimposed on the other structural loads to obtain the loading combination as stated in Section 5.6.3.2.

5.6.3.3.2 Wind and Tornado Analysis

The design wind loads on the intake structure are a function of the kinetic energy per volume of the moving air mass. The product of one-half of the air density and the square of the resultant design velocity results in a pressure corresponding to the design wind.

Determination of the design wind pressure on the intake structure is in accordance with the ASCE Paper 3269, “Wind Forces on Structures.”

The pressure corresponding to the standard air at 0.07651 pcf at 15°C and 760 mm of mercury in terms of the velocity at the appropriate height zone is given by:

$$q = 0.002558\ V^2$$

Similarly, the design pressure, including the effect of the shape coefficient (Cd), is given by:

$$p = q \times C_d = 0.002558 V^2 C_d$$

Using the equations given and the wind velocities noted in Section 5.6.3.1.6, the wind forces are calculated and then applied to the structure.

For the tornado loading condition, the hatches over the circulating water pumps and the traveling screens serve as blowout panels to relieve the pressure differential. Additional vents are also placed in the west wall and roof.

5.6.3.3.3 Hurricane Wave Analysis

The maximum hurricane wave is determined by the use of the U. S. Coastal Engineering Research Center Paper, “Shore Protection Planning and Design,” 1966. The static and dynamic forces from this wave are applied to the front of the intake structure.

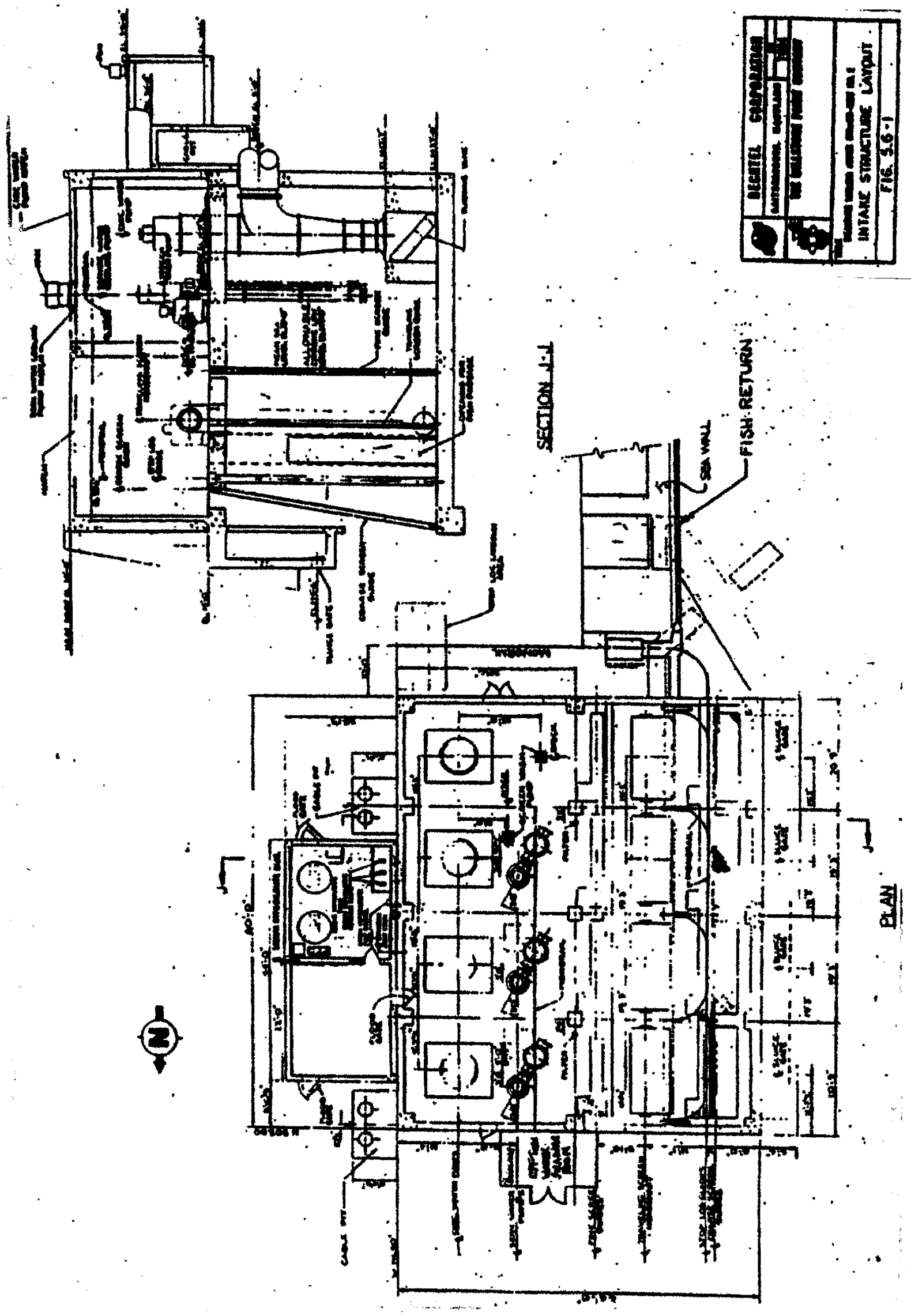


FIGURE 5.6–1 INTAKE STRUCTURE LAYOUT

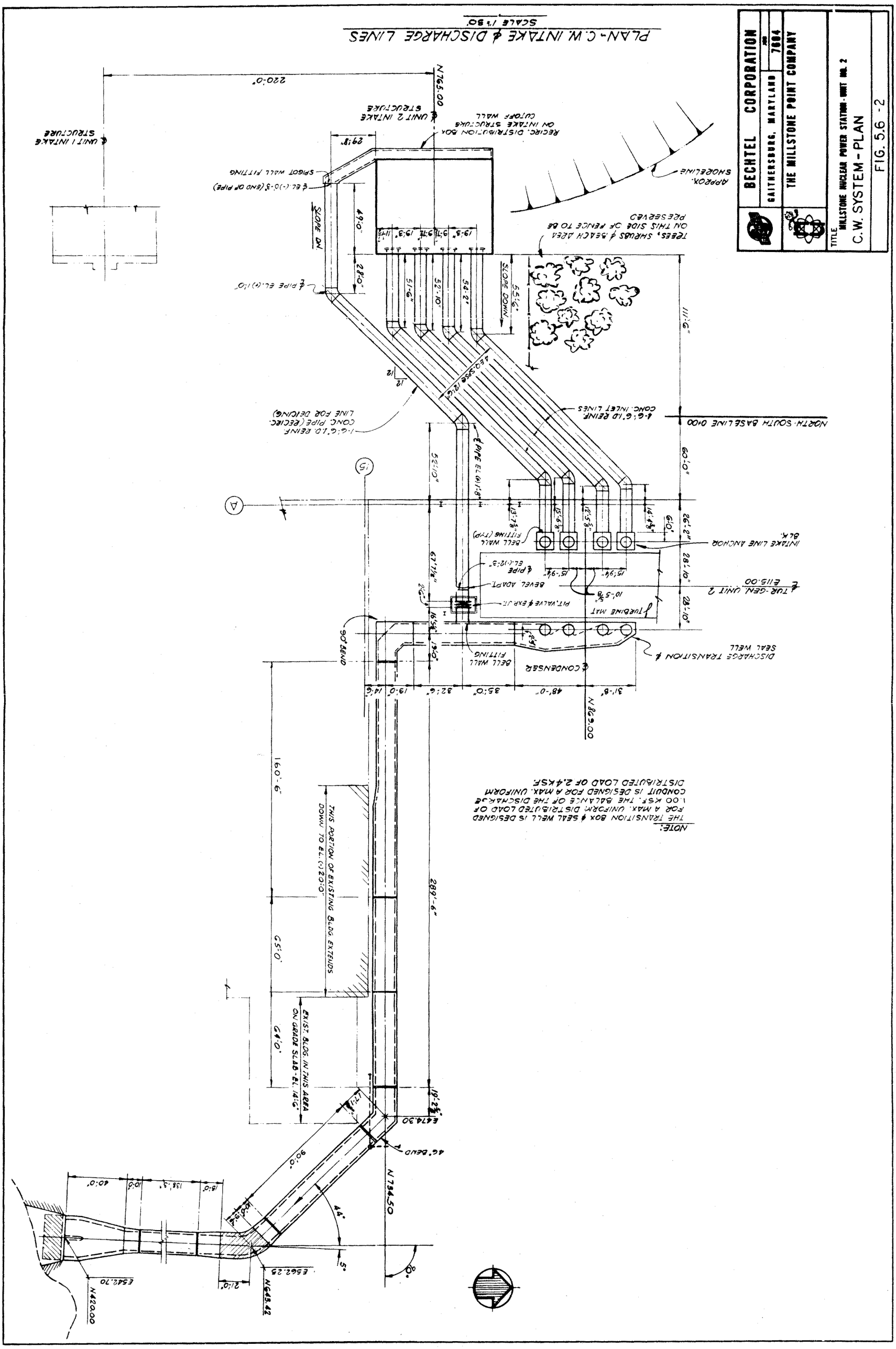


FIGURE 5.6–2 C. W. SYSTEM - PLAN

5.7 EXTERNAL CLASS I TANKS

5.7.1 GENERAL DESCRIPTION

The following tanks in the yard are classified as Seismic Class I structures:

a. Refueling water storage tank

b. Condensate storage tank

The condensate storage tank is located to the northwest, and the refueling water storage tank is located to the northeast of the containment, respectively, as shown in Figure 1.2–2. These tanks are supported on concrete foundations which rest on compacted structural backfill. The backfill is compacted to 95 percent of the maximum density as obtained by the Modified Proctor test in accordance with ASTM D-1557. The structural concrete foundations are proportioned such that the applied contact pressure, from dead loads and live loads in each design loading combination which result in uniform long-term settlement, does not exceed 3000 psf. The underlying supporting material for the condensate storage tank and refueling water storage tank is undisturbed glacial till and compacted glacial till, respectively. Allowable soil bearing pressures of 12000 psf for undisturbed glacial till and 5000 psf for compacted glacial till are utilized in the foundation design. These allowable values may be increased by one-third for wind and seismic conditions.

5.7.2 CONSTRUCTION MATERIALS

Structural concrete for the foundations conforms to the requirements of Section 5.9.3.1 and has a design strength of 3,000 psi at 28 days. Reinforcing steel conforms to Specification ASTM A-615, Grade 60.

The refueling water storage tank is fabricated from stainless steel conforming to ASTM A-240, Type 304. Design and fabrication are in accordance with Section III of the ASME Code.

The condensate storage tank was originally designed in accordance with the American Water Works Association Standard, AWWA D100. It is fabricated from carbon steel which conforms to Specification ASTM A-285, Grade C. The applicable portions of API 650, “Welded Steel Tanks for Oil Storage,” API 620, “Recommended Rules for Design and Construction of Large, Welded, Low Pressure Storage Tanks,” ACI 318-89, “Building Code Requirements for Reinforced Concrete,” ACI 349-80, “Code Requirements for Nuclear Safety Related Concrete Structures,” and 1980 ASME Section III, Division I, Subsection NE for Class MC were used to design the structural modifications necessary to support the addition of a nitrogen blanketing system.

5.7.3 DESIGN BASES

The design of the external Class I tanks provides the required features as outlined in Criteria 1, 37, Appendix A of 10 CFR Part 50.

5.7.3.1 Bases for Design Loads

The external Class I tanks are designed for all credible combinations of loading including loads under normal operation and loads due to adverse environmental conditions. The following loadings are considered:

a. Dead loads

b. Live loads

c. Earthquake loads

d. Wind and tornado loads

5.7.3.1.1 Dead Loads

These loads consist of the weight of all structural materials.

5.7.3.1.2 Live Loads

Live loads consist of tank liquid weights and a snow load of 60 psf applied to the roofs for the condensate storage tank, the live load included the internal operating pressure band of +1.0 psig to -0.34 psig which resulted from the addition of a nitrogen blanketing system.

5.7.3.1.3 Earthquake Loads

These loads are defined in Section 5.8.

5.7.3.1.4 Wind and Tornado Loads

Wind loads for the external Class I tanks are determined on the basis of the ASCE Paper 3269, “Wind Forces on Structures,” based upon a design wind velocity of 115 mph with gusts up to 140 mph.

The condensate storage tank is analyzed for tornado loads on the following basis:

a. Differential bursting pressure between the interior and exterior of the structure is assumed to be 3 psi pressure occurring in three seconds (1 psi/sec), followed by a calm for two seconds and a re pressurization.

b. Lateral loads on the structure are based on a tornado funnel which is conservatively assumed to have a peripheral tangential velocity of 300 mph and a translational velocity of 60 mph. These velocities are added together, resulting in a design basis tornado wind velocity of 360 mph. The applicable portions of the wind design methods described in the ASCE paper are used, particularly for the shape factors. The provisions in the paper for gust factors and variation of wind

velocity with respect to height are not applied. The wind velocity is assumed to be uniformly distributed over the height of the structure.

c. Provision is made for protection of the condensate storage tank against tornado driven missiles as defined in Sections 5.2.5.1.2 and 10.4.5.3 so as to provide sufficient water for a safe shutdown.

5.7.3.2 Design Load Combinations

The load combinations considered in the design of these tanks are listed below:

a. $D + L$

b. $D + L + W_w$

c. $D + L + W_t$ (applicable to CST only)

d. $D + L + E$

e. $D + L + E'$

Where

D = dead loads

L = live loads

W_w = wind loads (115 mph base)

W_t = tornado loads (360 mph base)

E = operating basis earthquake

E' = design basis earthquake

Stresses from load combination a, b, c, and d do not exceed the allowable permitted by the ASME Code. Standard Review Plan criteria were used for the condensate storage tank modifications which resulted from the addition of a nitrogen blanketing system. Stresses from load combination e do not exceed the allowable stresses of 90 percent of the steel yield strength and 85 percent of the concrete ultimate strength.

5.8 SEISMIC DESIGN

5.8.1 INPUT CRITERIA

The response spectrum technique is used to analyze Class I structures, systems and equipment when they are subjected to seismic motion. The response spectrum technique assumes a constant damping factor for each mode of the model of the major structural elements. The input ground motion is expressed in terms of a smooth design spectrum curve associated with the damping factor. Small equipment, as well as piping and cables located within the structure, are neglected in the model due to their relatively insignificant masses.

For the design of the piping systems and equipment, design spectrum curves at each of their supports are generated from a synthetic time history ground motion.

5.8.1.1 Design Response Spectra

The operating basis earthquake (OBE) used in the design of this plant is based on a ground motion having a maximum horizontal ground acceleration of 0.09 g and a vertical ground acceleration of 0.06 g, acting simultaneously. For the safe shutdown earthquake (SSE), a maximum horizontal ground acceleration of 0.17 g and a vertical ground acceleration of 0.11 g are used. The design response spectrum curves for structures supported on rock are shown on Figures 5.8–1 and 5.8–2, and the design response spectrum curves for structures supported on compacted structural backfill are shown on Figures 5.8–3 and 5.8–4.

A synthetic time history whose response spectrum curve corresponds to the design response spectrum curve is used to generate the response spectrum curves at different elevations within the structure. These are used in analyzing Class I equipment and piping at the respective locations. Comparisons of the response spectra derived from the time history and site seismic design response spectra for the damping values of 0.5, 1.0, 2.0, and 5.0 in percent of critical damping are shown in attached Figures 5.8–5, 5.8–6, 5.8–7 and 5.8–8.

The system period intervals at which the spectra values are calculated are as follows:

Frequency Range (cps)	**Frequency Increment (cps)**
0.2 to 1	0.05
1 to 10	0.1
10 to 33	1.0

The frequency increment shown above is always less than 10 percent apart between two consecutive frequencies, except that for the first frequency range of 0.2 to 1.0 cps. To verify that the 0.05 is a sufficiently small increment for this frequency range, the 2 and 5 percent damping response spectra are computed at the smaller increment of 0.0125 cps, which is one-quarter of the original one. Figure 5.8–9 shows these spectra in comparison with those based on 0.05 cps frequency increment, and with the design spectra, indicating that the 0.05 cps increment is indeed

sufficient for engineering purposes. Comparison of the response spectrum curve generated by the synthetic time history and the design response spectrum with two percent of critical damping is shown on Figure 5.8–5. A composite comparison of the design response spectrum curve with the N69W components of the 1952 Taft and the N-S components of the 1940 El Centro recorded earthquakes, normalized to the same ground acceleration of 0.09 g with two percent of critical damping, is shown on Figure 5.8–10.

5.8.1.2 Synthetic Time History

The synthetic time history is generated as follows:

The response of a set of linear single degree-of-freedom systems to seismic action is governed by the equation:

$$\ddot{X} + 2\beta\omega_n\dot{X} + \omega_n^2 X = -\ddot{u}(t) \qquad \text{Eq. (1)}$$

where

$\ddot{X}$ = relative acceleration

$\dot{X}$ = relative velocity

X = relative displacement

β = percent of critical damping

ω_n = natural frequency

$\ddot{u}(t)$ = forcing acceleration time history as a function of time t

The acceleration spectra is defined by:

$$S_a = (X+u)_{max} = [\omega_d \int_o^t u(\tau) e^{-\beta w_n (t-\tau)} \sin\omega_d (t-\tau) d\tau]_{max} \qquad \text{Eq. (2)}$$

where $\omega_d = \omega_n (1-\beta^2)^{1/2}$

T = time variable

If S_a - is specified as an arbitrary function of ß and ω, it is desirable to determine the $\ddot{u}(t)$ that will produce the acceleration spectra.

Assume that the acceleration has the form:

$$\ddot{u}(t) = \sum_{i=0}^{M} \ddot{u}_i \cos(\omega_i t + \phi_i) \qquad \text{Eq. (3)}$$

where M = number of Fourier Series terms

The absolute magnitude of the response of a single degree-of-freedom system to a sinusoidal acceleration is represented by:

$$\left|\frac{X}{\ddot{u}_i}\right|_{max} = \frac{1/\omega_n^2}{\left\{\left[1 - \frac{\left(\omega_i^2\right]^2}{\omega_n^2}\right] + \left[2\beta\frac{\omega_i}{\omega_n}\right]^2\right\}^{1/2}} \qquad \text{Eq. (4)}$$

Assuming that:

$$\ddot{X} = -X\omega_n^2 \cos\,\omega_n t \qquad \text{Eq. (5)}$$

The following is obtained:

$$S_{\ddot{a}} = \left(\frac{\ddot{x}}{\ddot{u}} + 1\right)\ddot{u} = \sum_{i=0}^{M}\left\{1 + \frac{1}{\left\{\left(1 - \frac{\omega_i^2}{\omega_n}\right)^2 + \left(2\beta\frac{\omega_i}{\omega_n}\right)^2\right\}^{1/2}}\right\}\ddot{u}_i \qquad \text{Eq. (6)}$$

Equation (6) will be satisfied by solving $\ddot{u}_i$ for the unknown coefficients. It is assumed that the maxima will occur at the same time. A matrix equation is set for frequencies:

$$\omega_{i+1} = 1.05\omega_i \qquad i = 1, 2, ...143 \qquad \text{Eq. (7)}$$

where $\omega_1 = 0.1$ cps

The damping values are as stated in Table 5.8-1.

Once a solution vector has been determined from:

$$\ddot{u} = A^{-1}S_a \qquad \text{Eq. (8)}$$

and substituted into Equation (3), the maximum acceleration spectra is formed by numerically integrating Equation (1). The maximum acceleration spectra, Sa_i, is then compared to the originally assumed spectra, $S\ddot{a}_o$. Weighting functions are developed from the relationship:

$$\varepsilon_i = f(\omega_i) = \frac{S\ddot{a}_o}{S\ddot{a}_i} \qquad \text{Eq. (9)}$$

Thus, from the nth iteration the following is obtained:

$$\ddot{u}_{n+1,\dot{c}} = \ddot{u}_{n,i}\ \varepsilon_{n,i} \qquad \text{Eq. (10)}$$

An acceleration time history has been developed using this procedure. It has been determined that for all frequencies considered, the bounds of Sa predicted by Equation (2) exceeded a 10 percent variation at certain frequencies.

5.8.2 SOIL-STRUCTURES INTERACTION

5.8.2.1 Soil-Foundation Interaction

The outlines of the foundations for Millstone Unit 2 structures are shown on Figures 5.8–11 to 5.8–12. These structures are supported by different materials.

The structures which are supported on bedrock are:

a. Containment

b. Enclosure building

c. Auxiliary building (except as noted below)

d. Intake structure

e. Turbine building (except as noted below)

The following structures are supported on compacted structural backfill:

a. Warehouse portion of the auxiliary building.

b. Auxiliary feedwater pump foundations located in the auxiliary bay of the turbine building.

Soil-foundation interaction is considered by introducing equivalent springs and viscous dashpots for the supporting mediums while the foundations are assumed to be rigid. The horizontal translational and rocking effects on structures are represented by equivalent spring stiffness. The

horizontal translational and rocking modes have been checked for structures founded on rock (i.e., with dynamic soil modulus greater than 500 x 10^3 ksf). The rigidity of the rock is so much greater than the structures it supports that rocking does not occur. The deflection patterns are such that there is no slope at the bases of the structures. Hence, for structures supported on rock, fixed-base assumption is used in the mathematic models formulated for the structural analysis.

For structures resting on compacted structural backfill, the equivalent spring stiffnesses for soil are evaluated using the formulas developed by Richart, Hall and Woods (Reference 5.2-61).

Isolation joints are provided between the foundations of the main structures. The differential movement of adjacent structures due to seismic motion is evaluated, and the size of the isolation joint is based on the anticipated horizontal movement of the foundations during the operating basis earthquake and the safe shutdown earthquake. The isolation joint is filled with a compressible material to minimize the influence of the foundations of the main structures on each other.

5.8.2.2 Dynamic Soil Pressure on Structures

The horizontal earth pressures for the walls are evaluated for both the static and the dynamic conditions. The rigidity of the walls and the backfill that is placed after the walls are constructed and framed at the top do not allow sufficient movement for the development of the active earth pressure case. Therefore, the at-rest condition is developed.

An equivalent fluid pressure is derived for the soil subject to seismic motion. The earth pressures are determined based on the characteristics of the materials to be used for backfill from the site, grading, and pit excavation. The backfill used is sand and silty sand. The equivalent fluid unit weight above the water table is 55 lb/cubic feet. Below the water table equivalent fluid unit weight is 95 lb/cubic feet which includes both the water pressure and the lateral at-rest earth pressure. Pressure distribution is assumed to be hydrostatic.

The dynamic earth pressures are considered for this plant. The analysis is based on work by Newmark, Ishii, Terzaghi, and the U.S. Army Corps of Engineers. These references provide the pressure coefficients which depend upon the magnitude of the acceleration factor of the earthquake. Although there is uncertainty concerning the behavior of backfill during earthquakes, the dynamic earth pressures can be approximated by the methods outlined in these references. The horizontal earthquake acceleration is combined with the static earth pressure acting on the wall. The values of the dynamic pressures are dependent on the types of backfills. These references show that, for typical sandy or silty sand backfill materials, the dynamic earth pressures are equivalent to the static earth pressures plus the static earth pressures times 2 a, where “a” is the ratio between acceleration produced by an earthquake shock and gravitational acceleration. Based on this information, the dynamic earth pressure is found to be equal to 1.34 times the static earth pressure for the safe shutdown earthquake.

The maximum soil reaction includes stresses due to the structural weight, the maximum overturning moment from the lateral analysis and the maximum inertia force from the vertical component of earthquake. The absolute sum of stresses due to the above effects will not exceed

the allowable bearing pressure for the soil. The overturning effect of the structure in a seismic event is considered negligible.

5.8.2.3 Underground Structures

Seismic analyses are performed on the following underground structures.

a. One 36 inch carbon steel off-gas pipe from the auxiliary building to the Millstone stack.

b. Two 24 inch cast iron headers from the Service Water System are routed from the intake structure to the auxiliary building.

c. One 10 inch carbon steel condensate water pipe from the condensate storage tank to the condenser.

d. Two electrical ducts encased in reinforced concrete from the intake structure to the turbine building.

The Class I underground ducts subjected to earthquake motion are analyzed biaxially, i.e., along the duct run and perpendicular to the duct run. Assuming that the ducts displace with the adjacent soil, the relative movements of the ducts to their supports will be determined. With these displacements, soil duct support models are formulated to determine the induced stresses in the ducts.

a. Due to wave propagation

b. At the supports due to differential movements of buildings and soil

c. At bends

5.8.3 SEISMIC STRUCTURAL ANALYSIS

5.8.3.1 Methods of Analysis

For seismic analysis of Class I structures, the response spectrum technique, using the design response spectrum curves, is employed. For Class I equipment analysis, response spectrum curves at the equipment bases are generated by the time history technique.

The procedure used to account for the number of earthquake cycles during one seismic event includes consideration of the number of significant motion peaks expected to occur during the event. The number of significant motion peaks during one seismic event would be expected to be equivalent in severity to no more than 40 full load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire event. Based upon this consideration and the assumption that seismic events equivalent to 5 Operating Basis Earthquakes

will occur during the life of the plant, Class I systems, components and equipment are designed for a total of 200 full load cycles.

The seismic analyses performed for Class I structures are based on elastic and linear behavior of all components involved, and as such, do not include any gradual or accidental deterioration of the structure. The blowdown forces associated with a concurrent loss-of-coolant accident (LOCA) are computed separately and combined with the seismic loads.

There are removable concrete slabs located in the containment building and auxiliary building. These slabs are placed over low pressure radwaste equipment, such as filters and demineralizers, and weigh approximately 4,000 pounds. The slabs will not receive a seismic acceleration in the upward direction sufficient to cause the slab to become a missile.

Removable blocks are self-locking and contain staggered horizontal and vertical joints. These block panels are designed to remain in place by use of retainers during a safe shutdown earthquake.

5.8.3.2 Procedure for Analysis

5.8.3.2.1 Structural Responses

5.8.3.2.1.1 Response Spectrum Method

The seismic loads on the containment are determined from a dynamic analysis of the structure. The dynamic analysis is made on a mathematical model consisting of lumped masses and weightless elastic columns acting as spring restraints. It is performed in the following two parts:

a. Determination of the natural frequencies of the structure and its mode shapes.

b. Determination of the model responses of the modes to the earthquake motions by the response spectrum method.

The natural frequencies and mode shapes are computed from the equations of motion of the lumped masses established by a stiffness or displacement method. They are solved by the interaction techniques through the use of a computer program. The form of the equation is:

$$[K][\Delta]_n = \omega^2[M][\Delta]_n$$

where:

[K] = matrix of stiffness coefficients including the combined effects of shear, flexure, rotation and horizontal translation.

[M] = matrix of concentrated masses

[Δ] = matrix of mode shape

The computation results in several values of w_n and mode shapes $(\Delta)_n$ for n = 1, 2, 3---n, where n is the number of degrees of freedom (i.e., lumped masses) assumed in the idealized structure.

The response of each mode of vibration to the safe shutdown earthquake is then computed by the response spectrum technique, as follows:

a. The base shear contribution of the nth mode

$$V_n = W_n \delta_{an}(W_n \beta_n)$$

where

W_n = effective weight of the structure in the nth mode

$$W_n = \frac{(\Sigma_x \Delta_{xn} W_x)^2}{\Sigma_x \Delta_{xn}^2 W_x}$$

where the subscript, x, refers to the levels throughout the height of the structure, and

Δ_{xn} = mode shape for the mode under consideration

δ_{an} (ω_n, β_n) = spectral acceleration of a single degree-of-freedom system with a damping coefficient of β_n' obtained from the response spectrum.

ω_n = angular frequency of the nth mode.

b. The horizontal load distribution for the nth mode is then computed as:

$$Fx = \frac{V_n(\Delta_{xn} W_x)}{\Sigma_x \Delta_{xn} W_x}$$

Then, using the modal inertia force Fx, the shears V_x at each point for each mode are calculated. The moments for each mode are obtained by integrating the shear diagram of the structure from the top down if a cantilever model is used. For coupled systems, with interconnecting members between cantilever members, the modal inertia forces are applied at the mass points, and analysis is made to obtained shears and moments at each point. The design shears and moments for the structure are the absolute sums of modal shears and moments. The overturning moment of the structure is the moment obtained at the base of the model.

For structure founded on structural backfill, a rotational, a vertical and a horizontal spring are used to represent the soil properties. Formulas for computing the equivalent spring stiffnesses for the case of rectangular base mat are based on Richart, Hall and Woods. Reference for the above article is given in Section 5.8.2.1.

The smoothed ground response spectrum curves used as input for the response spectrum method are derived from the report of J.A. Blume and Associates. The report was presented in Appendix F of the Design and Analysis Report (AEC Docket Number 50 - 245) which is a supplement to the Millstone Unit 1 FSAR. The spectrum curves were derived based on a careful examination of available historical records in the vicinity of the site and the underlying soil condition. The spectrum curves provide maximum responses over the frequency range from 2 cps to 10 cps in which the natural frequencies of the structures fall. Therefore, modal period variation in the mathematical models for Class I structures due to variations in material properties would not result in any significant increase in the resultant seismic loads.

5.8.3.2.1.2 Time History Method

The floor response spectrum curves are generated using time history modal analysis. Consider a viscosity damped, multi-degrees of freedom system subjected to the base acceleration ü(t); the equation of motion is given by:

$$[M]\ddot{x} + [\beta]\dot{x} + [K]x = -[M]\ddot{u}(t)\Delta^T \qquad \text{Eq. (11)}$$

where

Δ^T is the unit vector

This equation can be uncoupled to a set of independent equations analogous to the equation for a single degree of freedom system. The multi-degrees of freedom system can then be defined simply in terms of its mode shapes, frequencies and mass distribution as follows:

$$M_1 \ddot{x}_i + 2\beta_1 \omega_1 M_1 \dot{x}_1 + \omega_1^2 M_1 x_1 = -[M]\ddot{u}(t)\Delta_1^T$$

$$M_n \ddot{x}_n + 2\beta_n \omega_n M_n \dot{x}_n + \omega_n^2 M_n x_n = -[M]\ddot{u}(t)\Delta_n^T$$

Any of these equations can be rearranged as:

$$\ddot{x}_n + 2\beta_n W_n \dot{x}_n + \omega_n^2 \chi_n = \frac{[M]\Delta^{T_n}\ddot{u}(t)}{M_n} \qquad \text{Eq. (12)}$$

This set of equations can then be integrated numerically and independently. The spectrum values at any mass point can be obtained by direct application of Equation 12 to the digitized earthquake record at equal time intervals of 0.01 second.

The following tabulation shows a comparison of maximum seismic accelerations at selected critical locations in Class I structures as computed by the response spectrum and time history methods. The results based on the methods of “sum of absolute values” and “square root of sum of squares” by response spectrum technique are both shown for comparison. From the table as

shown below, it can be concluded that the results obtained by response spectrum and time history methods are consistent with each other and that the Class I structures are conservatively designed since the responses by the "sum of absolute values" were used.

STRUCTURE	MASS POINT	ACCELERATION (g's)		
		Response Spectrum: ABS	Response Spectrum: SRSS	TME HISTORY
Containment	7	0.265	0.170	0.192
Structure	14	0.425	0.306	0.393
Containment	4(N-S)	0.337	0.289	0.275
Internals	6(N-S)	0.457	0.356	0.329
Auxiliary	5(N-S)	0.274	0.250	0.296
Building	5(E-W)	0.285	0.254	0.325

5.8.3.2.2 Combination of Vertical and Horizontal Responses

The vertical ground design spectrum curves are derived as two-thirds of the horizontal values. This two-third value is considered to be conservative based on the strong motion records from both the United States and foreign countries.

Analyses for both the horizontal and vertical directions are performed using the ground design spectrum curves. The forces, moments and resulting stresses are combined directly, assuming a simultaneous occurrence of the vertical and horizontal motions. Vertical structural elements are considered vertically rigid. Horizontal structural elements of the Class I structures were further investigated for vertical responses and were found to be rigid.

The vertical ground response spectrum curve is used for equipment design. The equipment is attached to the rigid portions of the structure which have high natural frequencies. The ground motion would not be appreciably altered. The forces, moments and stresses on the equipment from both the vertical and horizontal motions are considered as acting simultaneously.

5.8.3.2.3 Torsional Effect Considerations

The torsional effect induced by the unsymmetric nature of the building was compensated for by considering a static torsional moment acting at the elevation under consideration. The magnitude of this moment is equal to the sum of the individual products of the inertia force and the eccentricity between the center of rigidity and center of gravity on and above that elevation.

To justify the above procedure, a torsional analysis was made on auxiliary building which is the least symmetric structure in this plant. A natural frequency of 10.4 CPS was obtained for the

torsional mode compared to the natural frequencies of 4.1 CPS and 7.5 CPS obtained for the translational mode in two major axes. The torsional natural frequencies for the other buildings are expected to be higher than the translational natural frequencies. The justification of combining the uncoupled results is presented in Appendix C of the topical report BC-TOP-4, Seismic Analysis of Structures and Equipment for Nuclear Power Plants, Rev. 1, dated September, 1972, Bechtel Corporation.

5.8.3.2.4 Natural Frequencies and Response Loads

The natural frequencies, loads in the form of mode shapes and total response loads, and the response spectra at critical plant equipment elevations for the following structures are presented in the forms of graphs. (See Figures 5.8–14 through 5.8–61.)

1. Containment and Intervals
2. Auxiliary Building
3. Turbine Building
4. Warehouse
5. Intake Structure

5.8.3.3 Damping Values

Material damping values used in the seismic analyses of structures and systems are shown in Table 5.8-1.

For structures made of a single material, the damping values are selected from values listed in Table 5.8-1 for the material under consideration. For structures made up of composite materials, the damping values can be considered as a function of both the mass and the particular mode shape value to calculate the composite damping values. The following Mass Mode Weighting method is used:

$$\beta_c = \frac{\sum_{i=1}^{i=n} M_i |\phi_i| \beta_i}{\sum_{i=1}^{i=n} M_i |\phi_i|}$$

β_c composite damping

β_i damping associated with mass point

$|\phi_i|$ absolute value of the mode shape at mass point

M_i mass at mass point

The only structure composed of major subsystems that are made of different materials is the warehouse area of the auxiliary building. Results based on the energy method and Mass Mode Weighting method were compared and negligible difference was obtained in this case.

For a structure whose motion is primarily composed of translational (flexural) displacement and foundation rotation (rocking), the mode shape must be broken down into its translational and rotational components, denoted as ϕ_γ and ϕ_f, respectively. Since the rotation is due to the fact that the structure is supported on a flexible foundation, the foundation damping, denoted as β_γ will influence the total damping value. Denoting the damping of the structures material by β_f, the composite damping can be computed by the following equation:

$$\beta_c = \frac{\beta_\gamma \phi_\gamma + \beta_f \phi_f}{\phi_\gamma + \phi_f}$$

Since the only Class I structure supported on a flexible foundation is the warehouse portion of the auxiliary building, the above procedure is not used. A constant damping value of 2% is used for all the modes and is a conservative approach.

5.8.4 SEISMIC SYSTEM ANALYSIS

To determine the seismic response of equipment, a time history analysis is performed on the structural model, using an earthquake as the input ground motion. This analysis generates the floor acceleration time- histories at the various mass points at which the equipment is located. The equipment response spectrum curve is then generated for each of the floor acceleration time-histories at various damping values and is used in the design of the equipment.

The equipment response spectrum curves are broadened by a smooth curve extended 10 percent each way at the peak response associated with the natural frequencies of the structure. This measure reflects the expected variations in the natural frequencies of the structure due to variations in structural material properties.

To determine the piping and instrumentation responses to an earthquake, Class I seismic piping systems are analyzed dynamically by means of a three dimensional model using two-thirds horizontal ground response spectra for the vertical spectra. The valves are included in the model by means of lumped masses and eccentric moments arms to account for the torsional effects of valves in the seismic piping analysis. The locations of seismic supports and restraints for the piping system are determined so that the piping system will not be in resonance with the supporting structures. The induced seismic effects of Class II piping on Class I piping systems are

also considered by including the Class II piping in the model. For each piping system, a horizontal response spectrum analysis was performed for the north-south and for the east-west directions. Modal responses are combined using the square-root-of-the-sum-of-the-squares method (except for RCS piping and components discussed in Appendix 4.A). The results of each analysis are combined with the results of excitation in the vertical direction. The design internal force or moment, or displacement is the larger number obtained from either of these analyses. The possible combined vertical and horizontal amplified response loads for the design of piping and instrumentation include the effects of the responses of building, floors, supports, equipment, and components.

The piping system is analyzed for the relative seismic displacements between piping supports, i.e., floors and components, at different elevations within a building and between buildings. Stresses in the piping system due to the most unfavorable directions of movements of supports are combined with thermal, seismic and operating stresses and used for piping design.

The locations of seismic supports and restraints for seismic Category I piping, piping system components, and equipment, including placement of snubbers and dampers are determined so that the resulting seismic stresses when combined with operating stresses will not exceed the allowable stresses given by governing codes.

A field surveillance is conducted to assure that the supports, restraints, etc. have been installed in the designated locations. Any change in location due to interference or other factors must be approved by engineers. For 2 inch and smaller Category I piping, a field installation manual is provided so that field engineers can properly design and locate pipe supports and restraints. Upon completion the design is reviewed by the engineers.

For seismic Category I buried piping, the pipe was assumed fixed at the end entering a structure and extending infinitely into the soil. The horizontal and vertical movements at the entry point, resulting from the seismic analysis of the structure, was then taken as end displacement in computing the stresses.

For seismic Category I piping outside the containment structure, extending from one structure to another, the differential movements at support points of the two structures were assumed to be out of phase. The resulting stresses when combined with thermal stresses are within allowable stresses.

5.8.5 SEISMIC EQUIPMENT ANALYSIS

For all purchased Class I equipment, the vendors are required to submit seismic calculations made in compliance with the equipment specification to demonstrate the capability of the equipment to satisfy the functional requirements under specified seismic conditions. Equipment is not released to operations without engineering approval of the calculations.

The supports for all Class I equipment are designed for the induced seismic forces. There are no significant gaps between the equipment and their supports, and, hence, they are not considered in the seismic analysis of the equipment.

The equipment hatch and personnel lock of the containment are seismic Class I equipment and are designed for the following accelerations (OBE):

Lock	Accelerations: Vertical (g)	Accelerations: Horizontal (g)
Equipment Lock	0.06	0.20
Personnel Lock	0.06	0.27

The acceleration values are multiplied by the normal operating weight of the hatch lock, or parts of the hatch lock, to obtain the horizontal and vertical components of the earthquake forces. Both horizontal and vertical earthquake components are considered acting simultaneously with normal operating loads, without exceeding code allowable stresses at a temperature of 120°F of the materials.

The earthquake forces due to the safe shutdown earthquake are obtained by multiplying the aforementioned accelerations by 1.95. The equipment hatch and personnel lock are designed to withstand the simultaneous action of design basis earthquake components and the accident loads, as stated in Section 5.2.2.3.4, at a temperature of 289°F, without exceeding material yield stresses and without loss of function.

For certain Class I systems and equipment, where analytical models and normal theory do not produce results of a significant confidence level, dynamic testing of prototypes or similar equipment is substituted to ensure functional integrity. Test data conform to one of the following:

a. Performance data of equipment which, under the specified conditions, have been subjected to equal or greater dynamic loads than those to be experienced under the specified seismic conditions.

b. Test data from previously testing comparable equipment which, under similar conditions, have been subjected to equal or greater dynamic loads than those specified.

c. Actual testing of equipment in accordance with one of the following methods:

    1. The equipment is subjected to an artificial time history response at the elevation of interest.

    2. The equipment is subjected to a sinusoidal excitation, sweeping through the desired range of significant frequencies, using input acceleration amplitudes for the forcing function which simulates the specified seismic conditions.

    3. The equipment is subjected to a transient sinusoidal motion synthesized by a pulse exciting a group of octave filters such that the response of the shaking table and the duration of loading simulates the artificial response spectrum curve at the elevation of interest.

The certified test data and results are required to be submitted for engineering approval.

All Class II components and equipment are sufficiently separated from Class I components and equipment so that the Class II components and equipment will not damage the Class I components and equipment under seismic conditions.

5.8.5.1 Static Tests

Supports for lightly loaded safety related components rely on friction to resist vertical and seismic force. These components are not subject to thermal cycling or mechanical vibration.

Reliance on friction as the sole means of restraining vertical and seismic forces has been verified by testing performed on safety related accumulator tanks.

Reliance on friction is considered appropriate, provided that the frictional forces include a margin of safety consistent with the appropriate design criteria for the structure.

5.8.5.2 STERI Evaluations

The STERI process, Seismic Technical Evaluation of Replacement Items, EPRI TR-104871 (Reference 5.8-1) may be used to demonstrate that seismically rugged or insensitive replacement items exhibit seismic performance equivalent to original items. STERI evaluations document that the seismic qualification status of the original items and host equipment is maintained by the replacement items.

5.8.5.3 GIP NARE Evaluations

As an alternative to the methods described in preceding paragraphs, the Seismic Qualification Utilities Group Generic Implementation Procedure Revision 3, “GIP-3” (Reference 5.8-2), as modified and supplemented by the U. S. Nuclear Regulatory Commission Supplemental Safety Evaluation Reports SSER Number 2 (Reference 5.8-3) and SSER Number 3 (Reference 5.8-4), may be used as an alternative to existing methods for the seismic design and verification of modified, new and replacement equipment classified as Seismic Class I (NARE).

Only those portions of GIP-3 which apply to the seismic design and verification of mechanical and electrical equipment, electrical relays, tanks and heat exchangers, and cable and conduit raceway systems shall be used. The other portions of the GIP are not applicable since they contain administrative, licensing, and documentation information which is applicable only to the USI A-46 program. Plant procedures provide detailed GIP-3 implementation guidance.

The GIP method should not be used on equipment systems for which seismic qualifications have been imposed or committed to IEEE 344-1975 as listed below.

- Reactor Coolant Pump Speed Sensing System (RCPSSS)

- Feedwater Regulating System and Feedwater Pump Speed Control System, specifically the SPEC 200 control equipment
- Auxiliary Feedwater Automatic Initiation System and Component Parts, except logic power supplies and in-containment mounted sensors
- Auxiliary Steam Line Break Detection/Isolation System (ASDI)
- Alternating Current Instrumentation and Control Components, specifically the inverter and static switches

5.8.6 SEISMIC INSTRUMENTATION PROGRAM

5.8.6.1 Conformance with NRC Requirements

Seismic instrumentation for Millstone Point Unit 2 is provided on the basis of the existing NRC requirements specified in Regulatory Guide 1.12, Revision 1, (Instrumentation for Earthquakes), Appendix A of 10 CFR 100 and upon the best available information on the ability of seismic instrumentation to predict plant responses to seismic motion.

5.8.6.2 Description of Program

The following instrumentation is used to measure plant response to earthquake motion:

a. Five triaxial time history accelerographs.

b. Four peak accelerographs.

c. One triaxial response spectrum recorder.

Number	Accelerograph Location
1	Elevation (-)24 feet 0 inches containment base slab at 215 degrees outside of the containment.
2	Elevation 75 feet 0 inches containment structure at 215 degrees outside of the containment.
3	Elevation 14 feet 6 inches warehouse area of the auxiliary building.
4	Elevation 18 feet 0 inches intake structure south wall.
5	Elevation 14 feet 6 inches ground level on pad 139 degrees southeast of condenser storage tank.

Accelerographs Numbers 1 and 2 represent two key locations in the model used in the containment seismic analysis. They are located outside the containment structure such that they are accessible for periodic servicing. Accelerograph Number 3 is located on the warehouse base

slab founded on compacted backfill. Accelerograph Number 4 is located in the intake structure which has more than half of its structure below grade. Accelerograph Number 5 represents the free field accelerograph.

The five triaxial time history accelerographs are combined within the basic accelerograph system. It features central recording on magnetic tape cassettes with remote transducer unit and a separate triggering unit. Both the transducer and the triggering unit which normally remain dormant are connected to a recording and playback system in the control room. Upon a seismic event, the system is activated within 0.1 second. Once the trigger is activated by a seismic event, the operators are alerted by visual and audible alarms. Signals from the transducer unit are fed into a multi-channel tape recording unit and recorded on a time/history basis with a common time signal. The recording system, once triggered, will continue to operate for at least five seconds beyond the last detection of a seismic signal of triggering intensity. At the end of the seismic event, the recorded tapes are transferred to a strip chart through a playback system. The entire event, from seismic trigger to visual accelerograph can be accomplished within a few minutes after an earthquake.

The transducer unit contains three accelerographs mounted in a triaxial orthogonal array 90 degrees apart. Sensitivity of the accelerograph is 0.001g to 1g with a natural frequency of at least 33 Hz.

The triggering unit which is located in the same area as the Number 1 accelerograph will activate all triaxial time history accelerograph and the recording system. The triggering unit generates a triggering signal by either an omni-directional horizontal or vertical component of a seismic acceleration. Trigger sensitivity is adjustable from 0.005g to 0.02g for the vertical trigger and the horizontal trigger has an adjustable gap from 0.005 inches to 0.06 inches. The trigger unit is engineered to discriminate against false starts from other operating inputs such as from traffic, elevators, people, and rotating equipment. The triggering unit is initially set to trigger at 0.01g vertical acceleration and 0.02 inch gap for the horizontal displacement. At this level no damage can be done in the plant and no spurious triggering of the recording system is expected. Triggering levels are determined on the basis of plant operating experience to ensure proper operation of this system.

Number	Recorder Location
1	Elevation (-)24 feet 0 inches containment base slab. (Outside Containment).
2	Elevation (-)0 feet 7 inches Steam generator Number 1 support.
3	Elevation 14 feet 6 inches Pressurizer support.
4	Elevation 38 feet 6 inches Safety injection tank support.

The peak accelerographs are all located on the major Class I equipment which has designed response spectrum readily available for comparison with the recorded acceleration level in the event of an earthquake.

The peak accelerographs are the Teledyne Geotech model PRA-103. These are mechanical devices which detect and record peak amplitudes of acceleration caused by seismic disturbances.

Number	Spectral Recorder Location
1	Elevation (-)24 feet 0 inches containment base slab. (Outside Containment)

This location corresponds to the pertinent input vibratory motion assumed in the containment seismic analysis.

The triaxial response recorder is an Engdahl Model PSR 1200 Peak Shock Recorder. This is a mechanical device which records the peak acceleration experienced at each of twelve frequencies for three mutually orthogonal directions. This device contains metal reeds which vibrate when excited at their natural frequencies. The maximum excursion of each reed is scribed onto a plate by a stylus mounted at the reed end. This plate is removed after a seismic occurrence and the length of the scribed line is measured and converted to its corresponding peak acceleration. The twelve reeds are resonant at the following frequencies:

Reed Number	Frequency (Hertz)
1	2.02
2	2.54
3	3.20
4	4.02
5	4.92
6	6.02
7	8.08
8	10.2
9	12.7
10	16.2
11	20.6
12	26.1

The above values should be considered nominal values since the resonant frequency of these reeds may vary from calibration to calibration.

5.8.6.3 Action Following an Earthquake

Activation of the seismic trigger on the accelerograph system will be annunciated in the control room.

The operator will verify the operational status of the plant by means of Control Room instrumentation and will initiate a visual inspection of the plant.

If the operator identifies abnormal or emergency conditions, he carries out appropriate procedures.

Following an earthquake, the operator will retrieve the recordings made by the peak recording spectral accelerograph for comparison of this data with the OBE for the plant site. The recordings from the strong motion accelerographs will also be analyzed to determine accurately the actual seismic acceleration spectrum experienced. The results of this analysis will be considered accurate, will override any preliminary indication by the peak shock spectral recorder, and by comparison with the OBE will determine whether the plant should continue to operate, be shutdown, or resume operations, as required by Appendix A of 10 CFR 100.

Following an earthquake of sufficient magnitude to shut the plant down, an extensive program will then be performed to evaluate the adequacy of all safety related structures, systems and equipment. The data on the earthquake's frequency and amplitude recorded by the strong motion accelerographs will be translated into computer codes best available at that time. For structure and system analyses using response spectrum techniques, the designed spectrum will be compared with that developed by the recorded time history. If the measured responses are less than the values used in the design for the SSE, the structure and system are considered adequate for future operations. Otherwise, the structure and system will be analyzed to check their adequacies. For system analysis such as the NSSS system using the time history technique, the recorded time history will be used as a direct input. Time histories at different elevations will be generated using the structure model. The fundamental frequency of the containment will be verified by the two accelerographs installed on the containment structure. Results will be evaluated to check the adequacy of the system.

5.8.7 REFERENCES

5.8-1 Seismic Technical Evaluation of Replacement Items, Ref: EPRI TR-104871

5.8-2 Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3, Seismic Qualification Utilities Group, May 16, 1997.

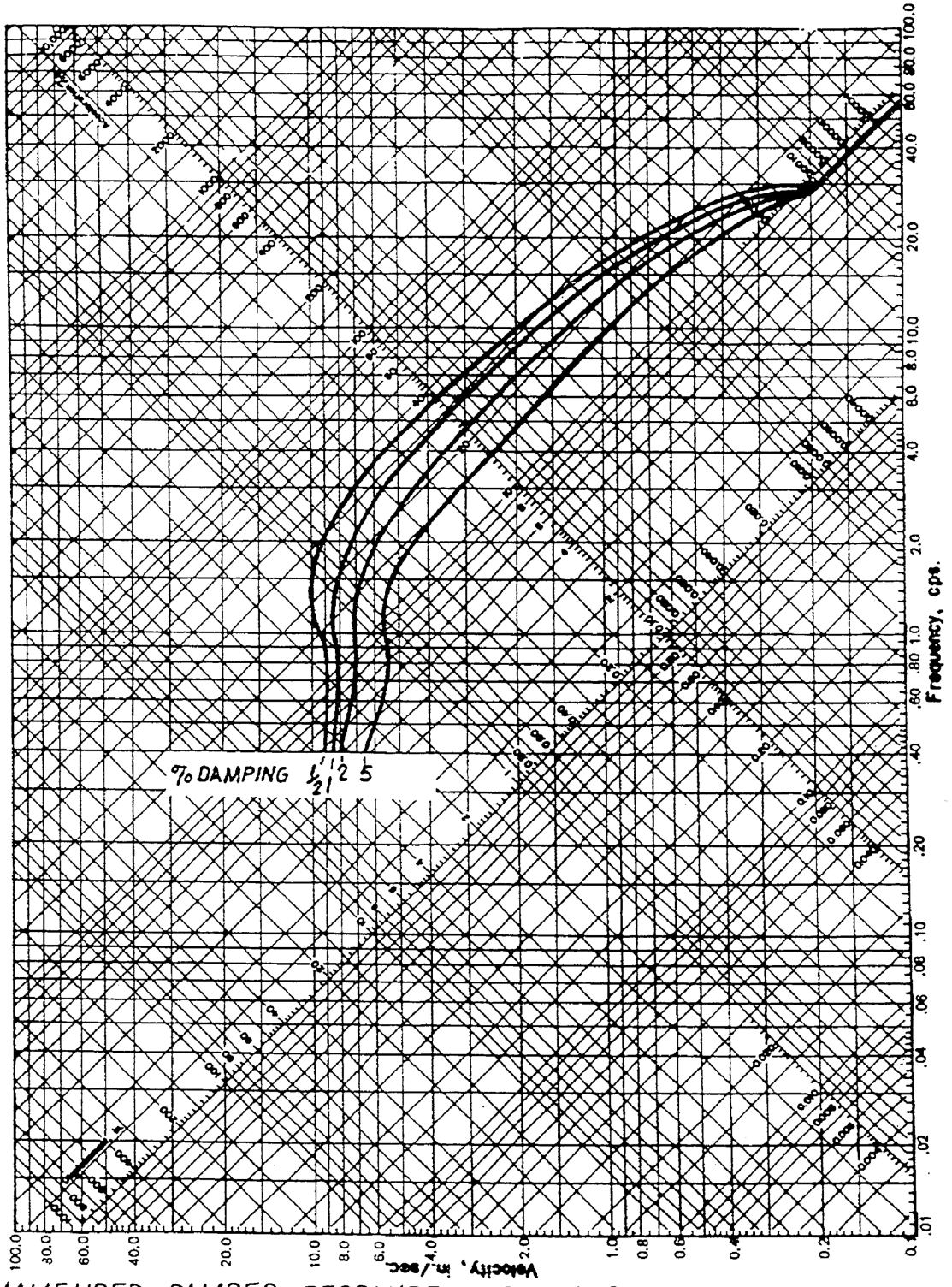
5.8-3 U.S. NRC Supplemental Safety Evaluation Report Number 2 (SSER Number 2) on SQUG Generic Implementation Procedure, Revision 2, as corrected on February 14, 1992 (GIP-2). May 22, 1992.

5.8-4 U.S. NRC Supplemental Safety Evaluation Report Number 3 (SSER Number 3) on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment Updated May 16, 1997 (TAC Number M93624), December 4, 1997.

TABLE 5.8-1 MATERIAL DAMPING VALUES

	Critical Damping	
	OBE (0.09 g ground acceleration)	**DBE (0.17 g ground acceleration)**
Welded steel plate assemblies	1	1
Welded steel framed structures	2	2
Bolted or riveted steel framed structures	2.5	2.5
Reinforced concrete equipment Supports	2	3
Reinforced concrete frames and buildings	3	5
Prestressed concrete structures	2	5
Steel piping	0.5	0.5
Soil (foundation)	2	5

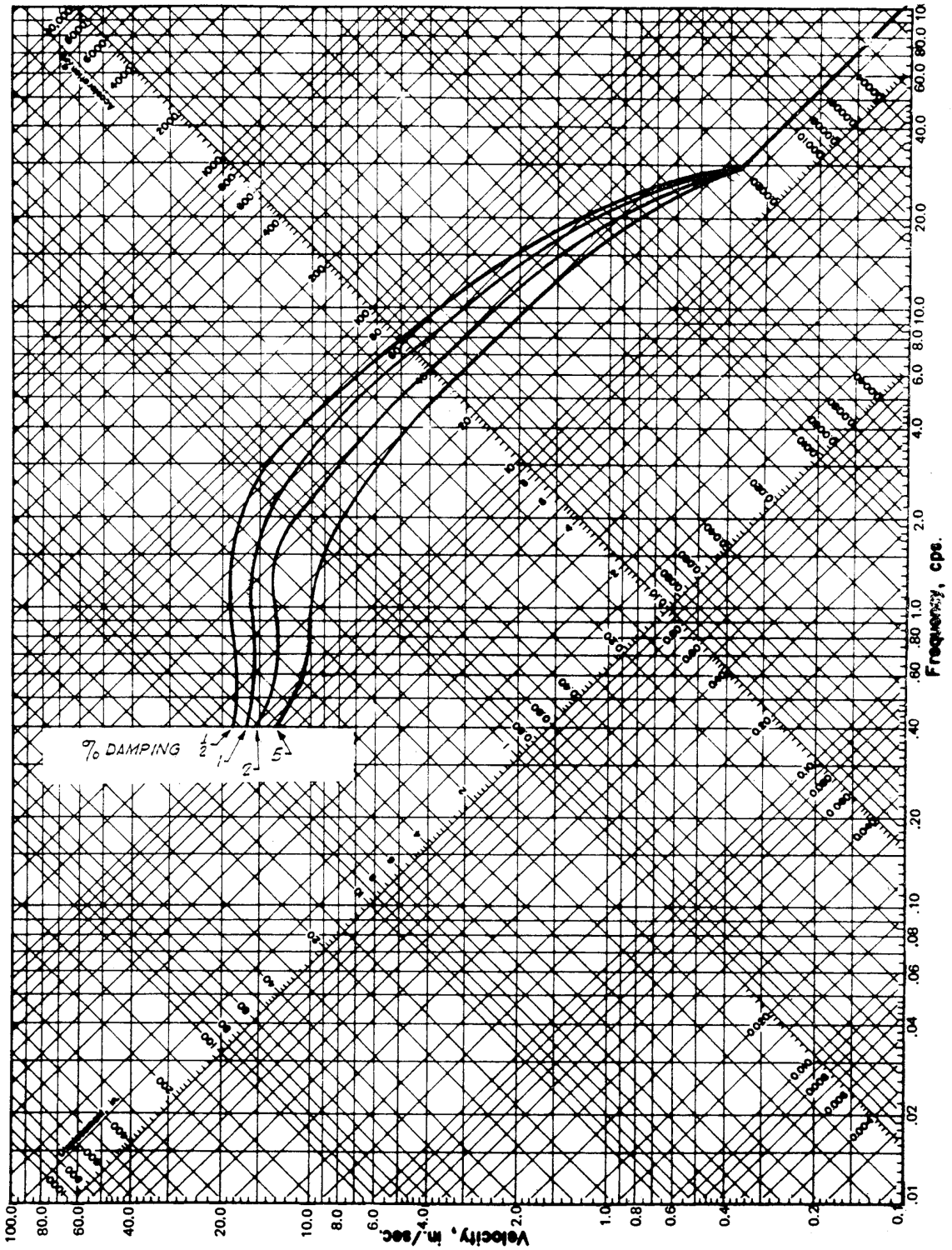
FIGURE 5.8–1 RECOMMENDED DAMPED RESPONSE SPECTRA 9%G ACCELERATION, OPERATING BASIS EARTHQUAKE



RECOMMENDED DAMPED RESPONSE SPECTRA 9 % g ACCELERATION
OPERATING BASIS EARTHQUAKE

FIGURE 5.8 -1

FIGURE 5.8–2 RECOMMENDED DAMPED RESPONSE SPECTRA 17%G ACCELERATION, DESIGN BASIS EARTHQUAKE



RECOMMENDED DAMPED RESPONSE SPECTRA 17% g ACCELERATION
DESIGN BASIS EARTHQUAKE

FIGURE 5.8 -2

FIGURE 5.8–3 RECOMMENDED DAMPED RESPONSE SPECTRA 9%G ACCELERATION, OPERATING BASIS EARTHQUAKE (SOIL SURFACE)

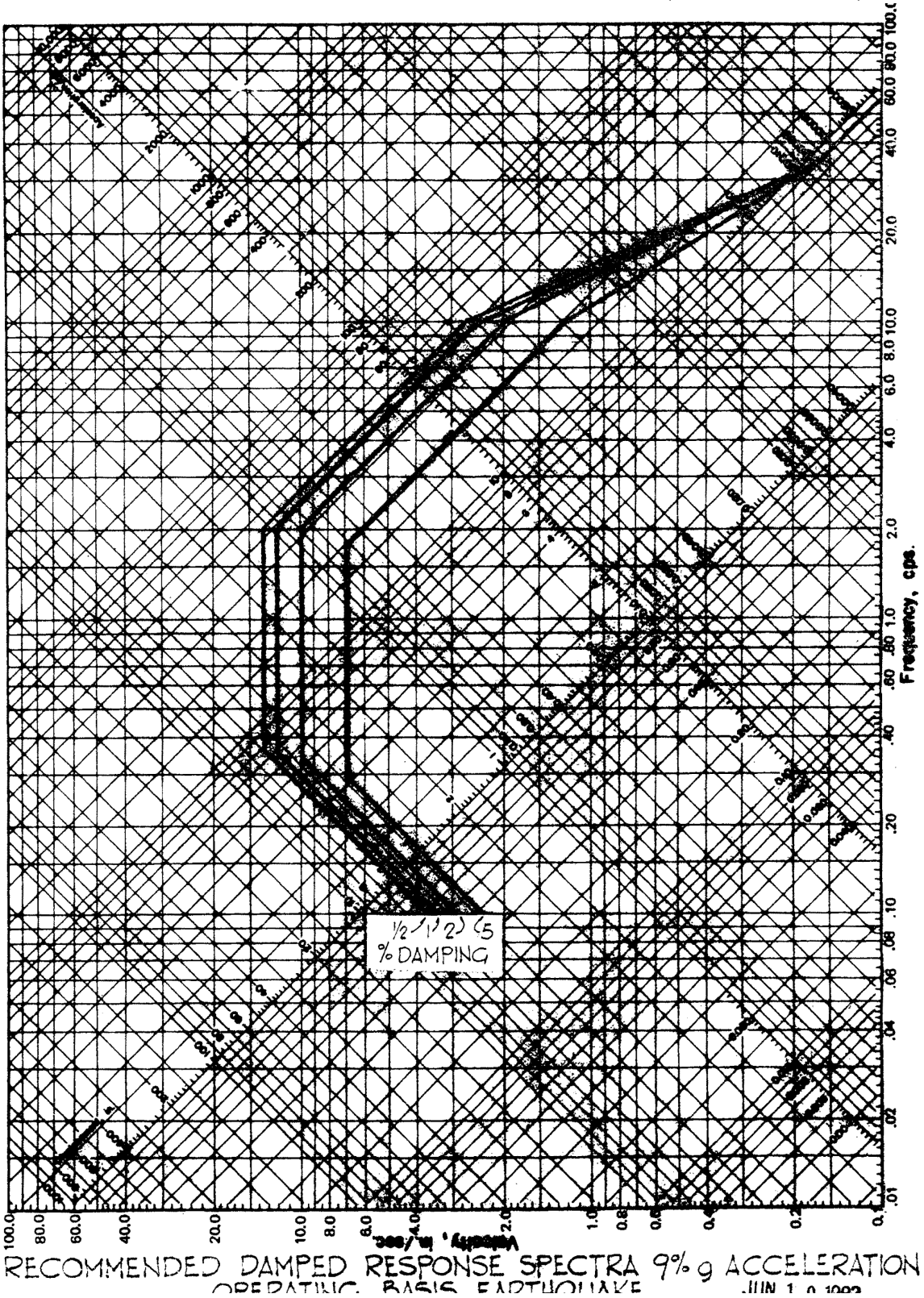
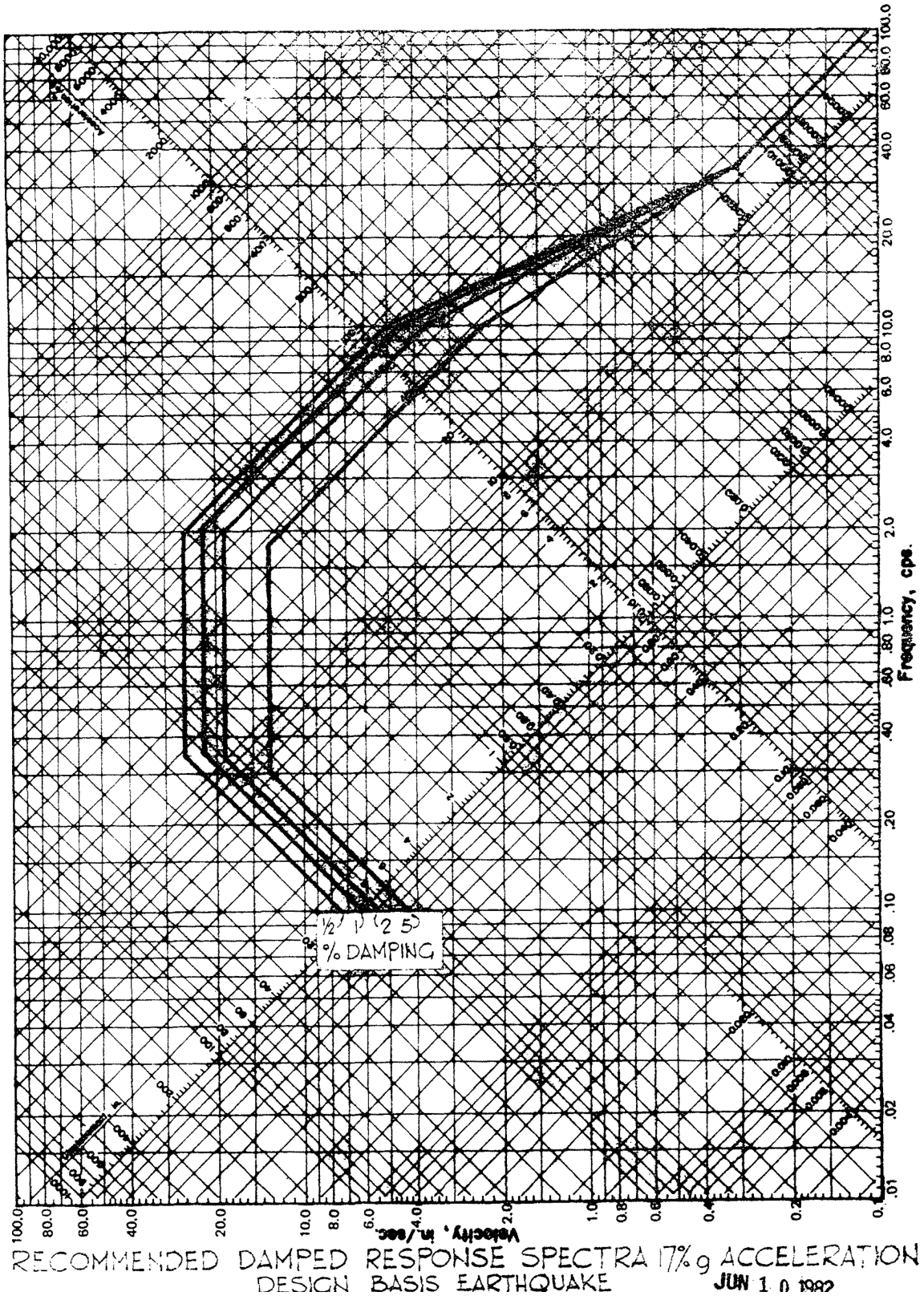


FIGURE 5.8–4 RECOMMENDED DAMPED RESPONSE SPECTRA, 17%G ACCELERATION, DESIGN BASIS EARTHQUAKE (SOIL SURFACE)



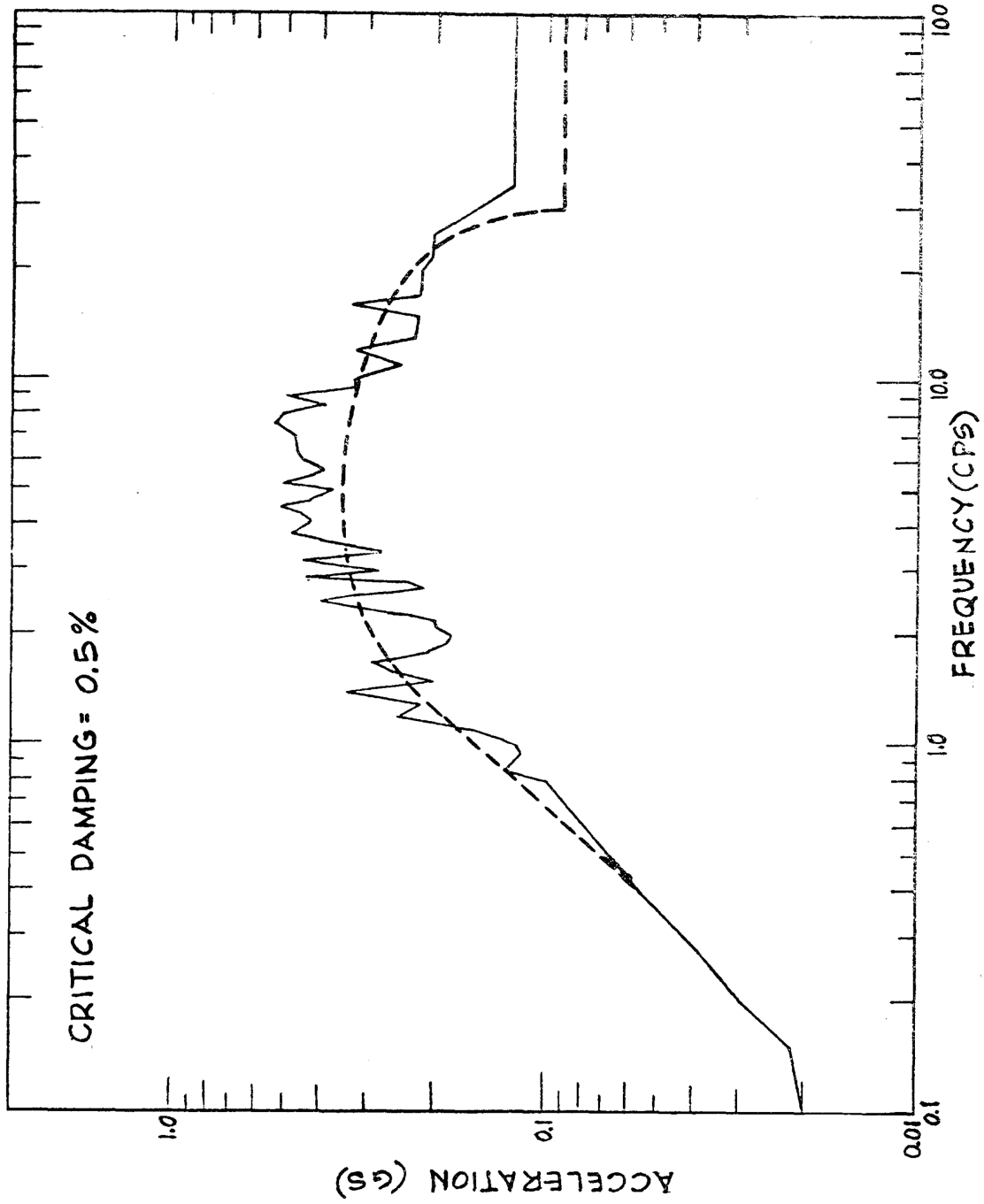


FIGURE 5.8-5 COMPARISON OF SMOOTH RESPONSE SPECTRA VS. RESPONSE SPECTRA FROM THE ANAMET TIME HISTORY DESIGN EARTHQUAKE (CRITICAL DAMPING = 0.5%)

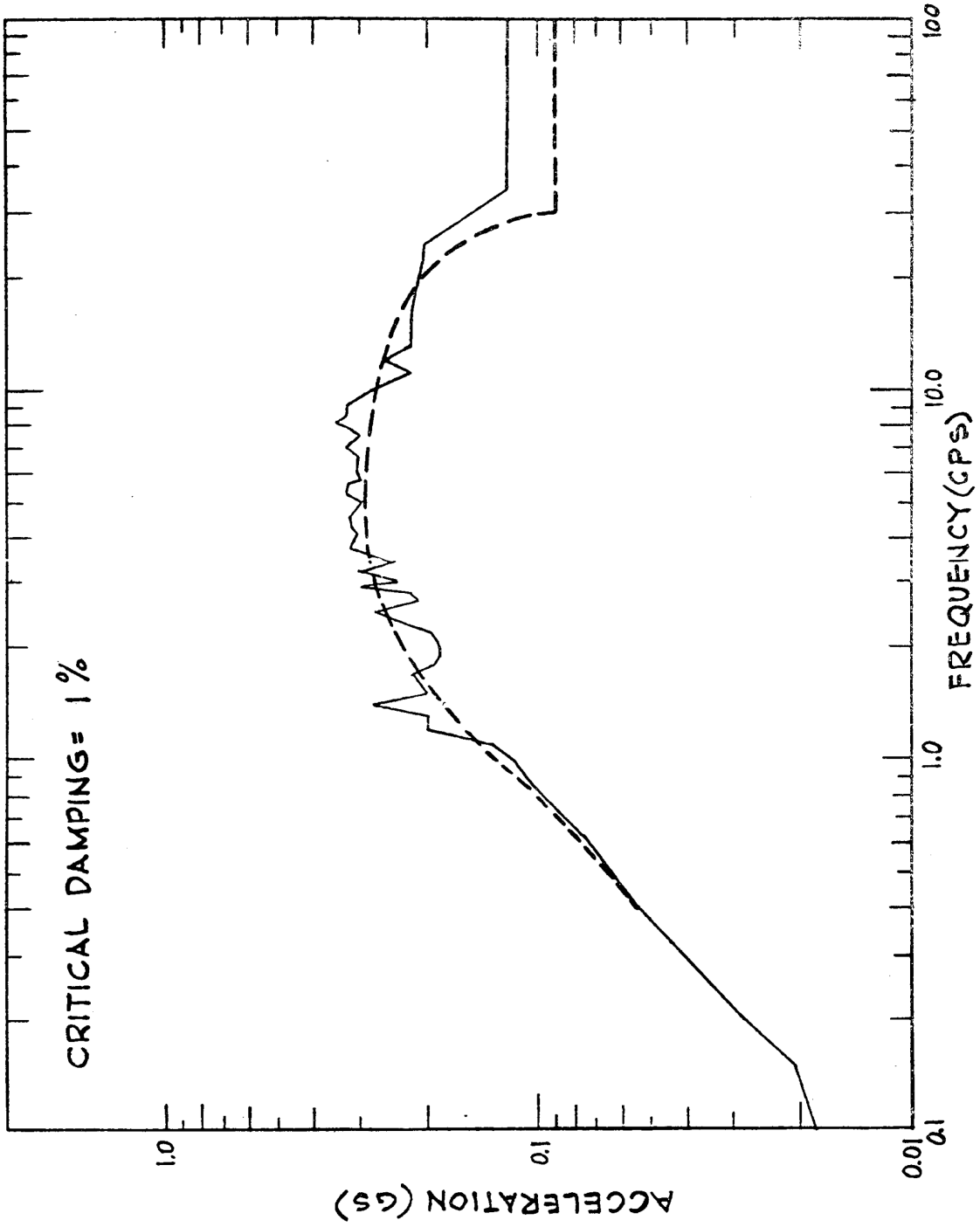


FIGURE 5.8–6 COMPARISON OF SMOOTH RESPONSE SPECTRA VS. RESPONSE SPECTRA FROM THE ANAMET TIME HISTORY DESIGN EARTHQUAKE (CRITICAL DAMPING = 1%)

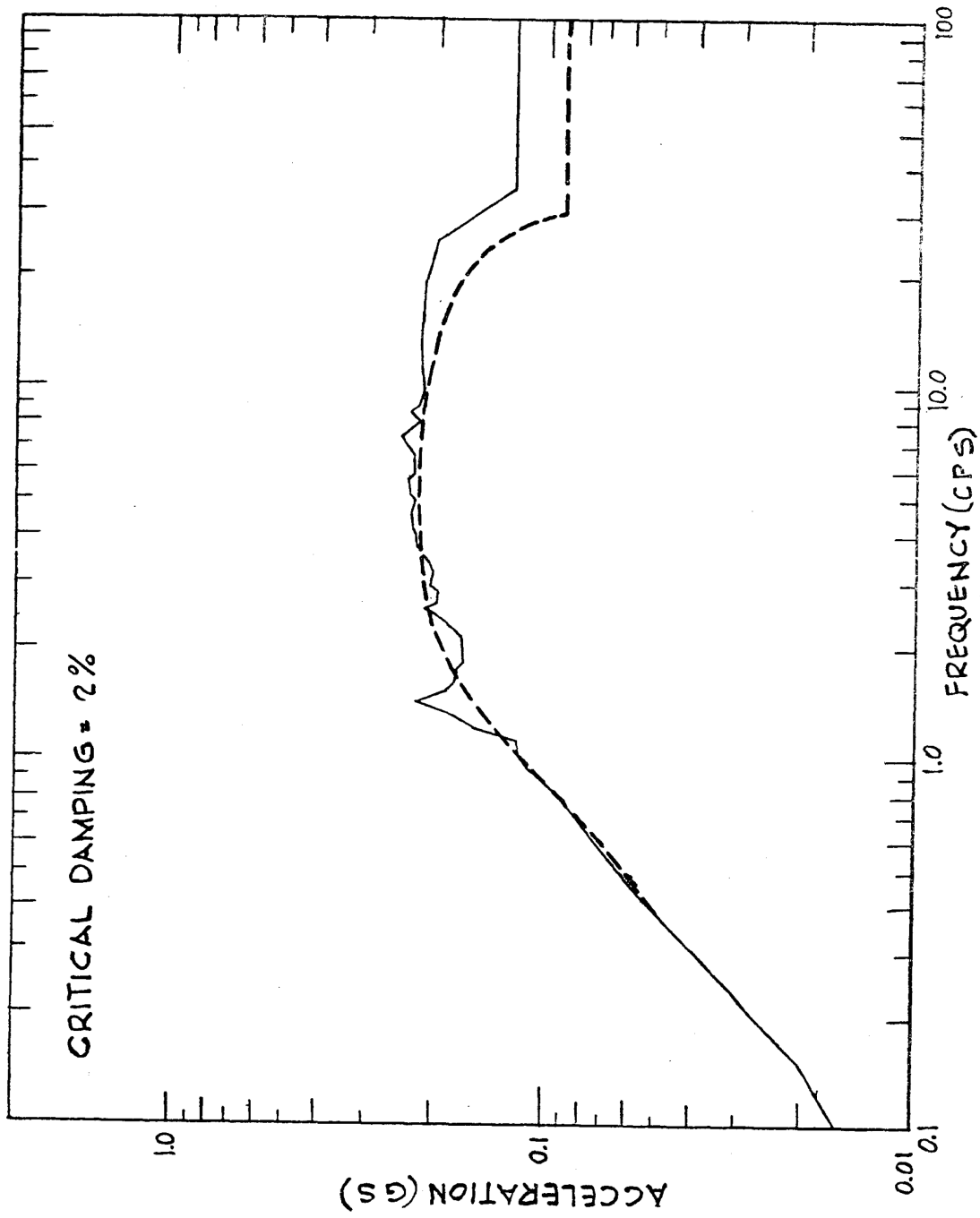
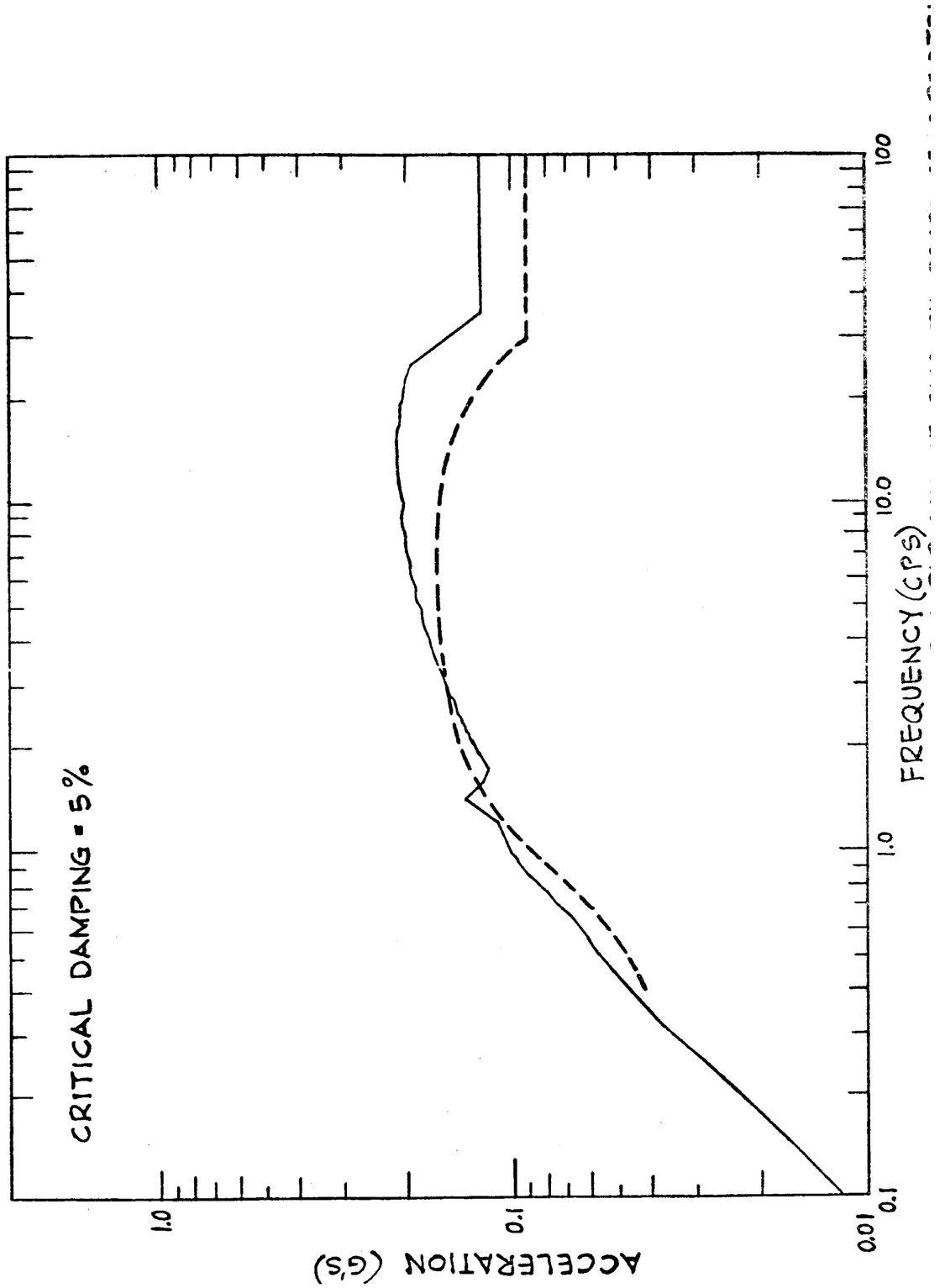


FIGURE 5.8–7 COMPARISON OF SMOOTH RESPONSE SPECTRA VS. RESPONSE SPECTRA FROM THE ANAMET TIME HISTORY DESIGN EARTHQUAKE (CRITICAL DAMPING = 2%)

FIGURE 5.8–8 COMPARISON OF SMOOTH RESPONSE SPECTRA VS. RESPONSE SPECTRA FROM THE ANAMET TIME HISTORY DESIGN EARTHQUAKE (CRITICAL DAMPING = 5%)



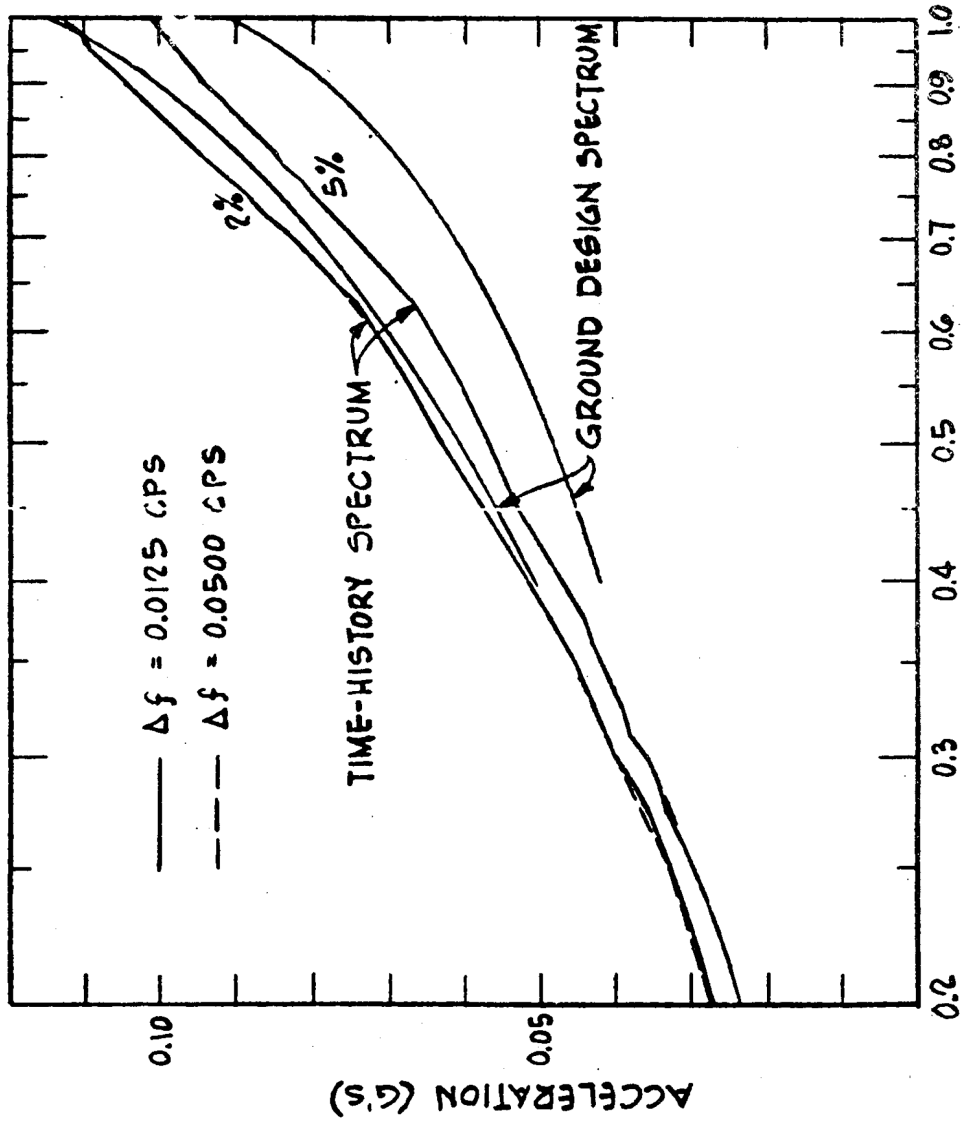
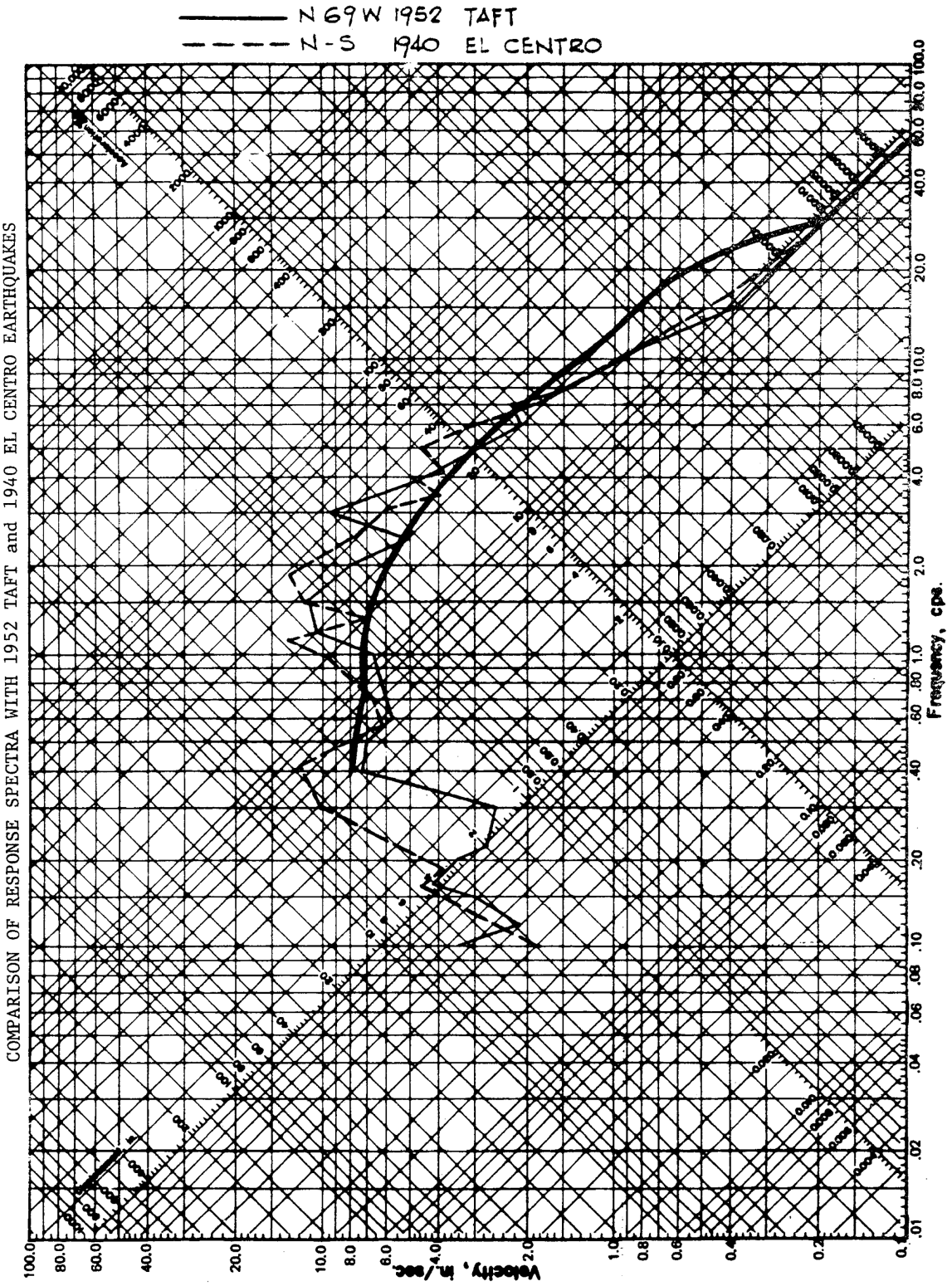


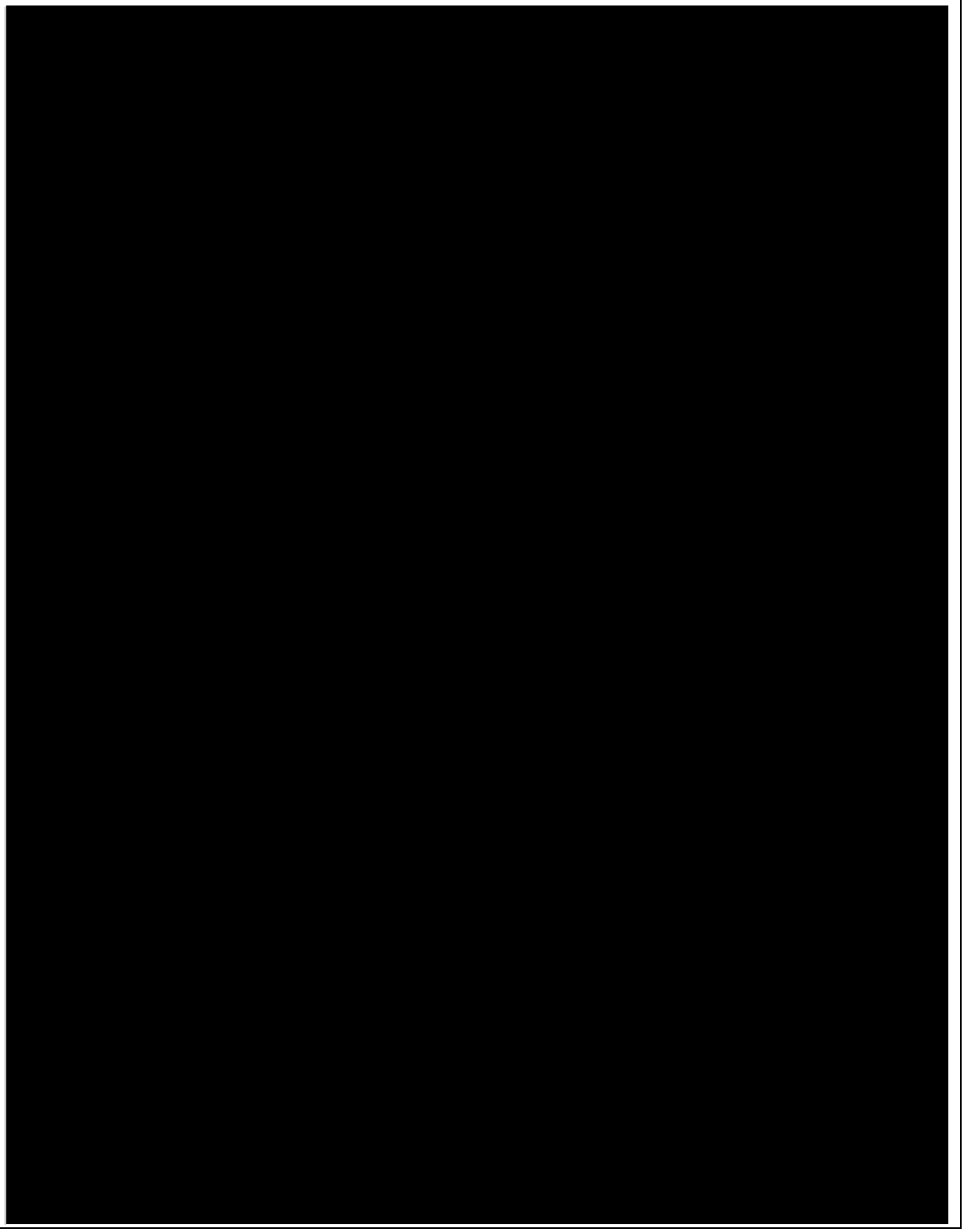
FIGURE 5.8–9 RESPONSE SPECTRA FROM THE TIME HISTORY DESIGN EARTHQUAKE WITH VARIOUS FREQUENCY INTERVALS

FIGURE 5.8–10 COMPARISON OF RESPONSE SPECTRA WITH 1952 TAFT AND 1940 EL CENTRO EARTHQUAKE



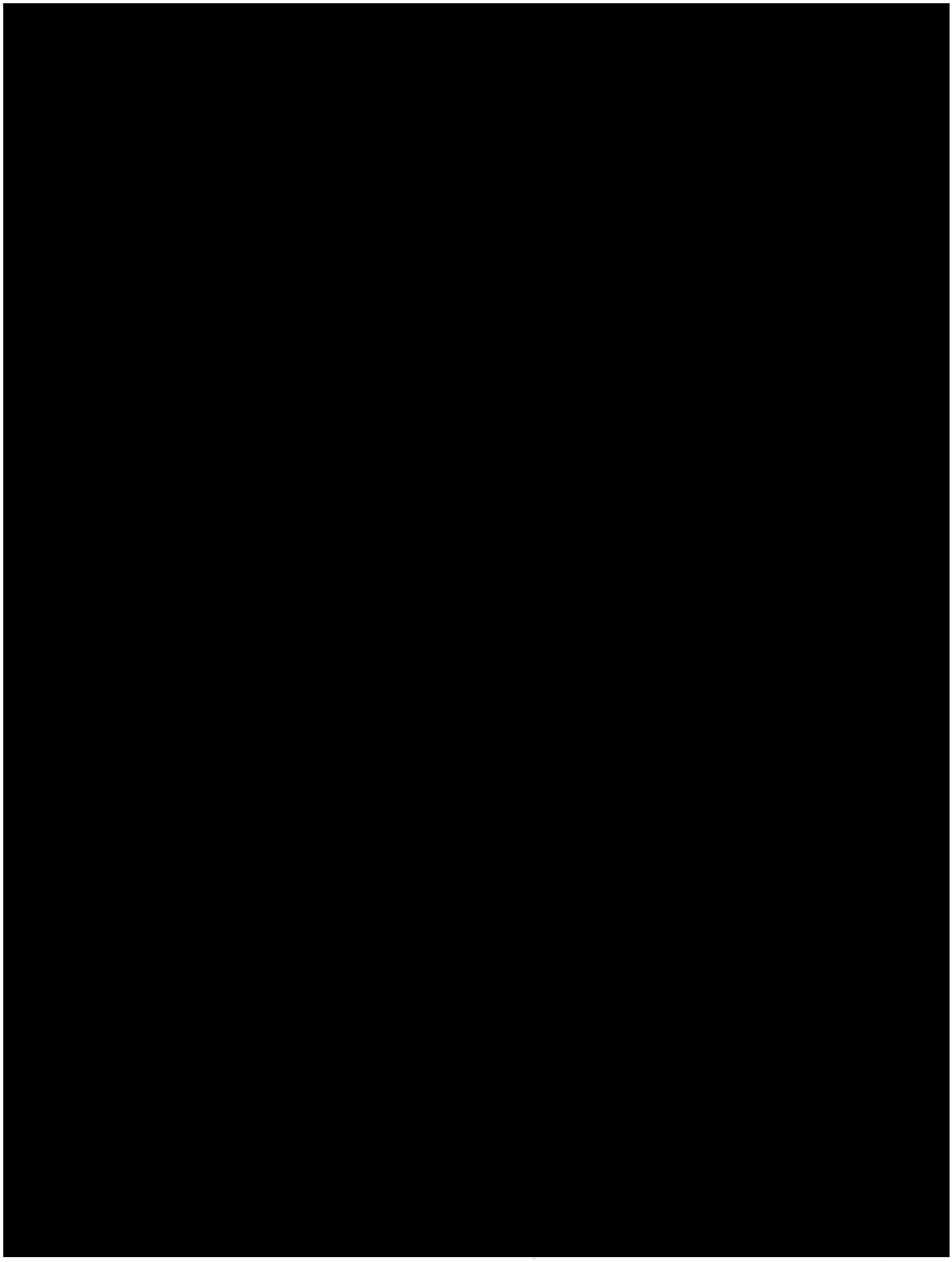
Withhold under 10 CFR 2.390 (d) (1)

FIGURE 5.8–11 FOUNDATION OUTLINE



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 5.8–12 SECTIONS A-A & B-B



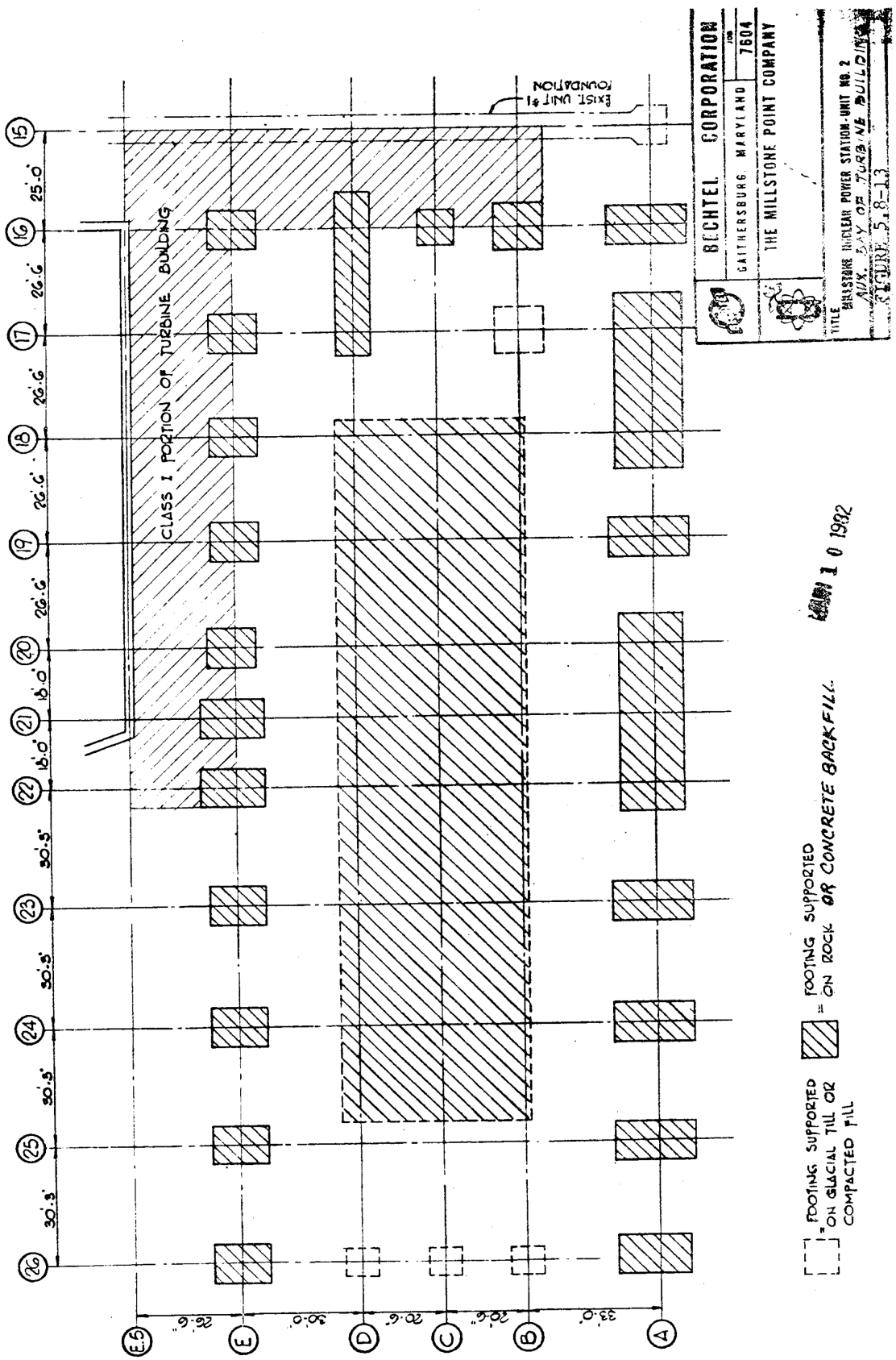
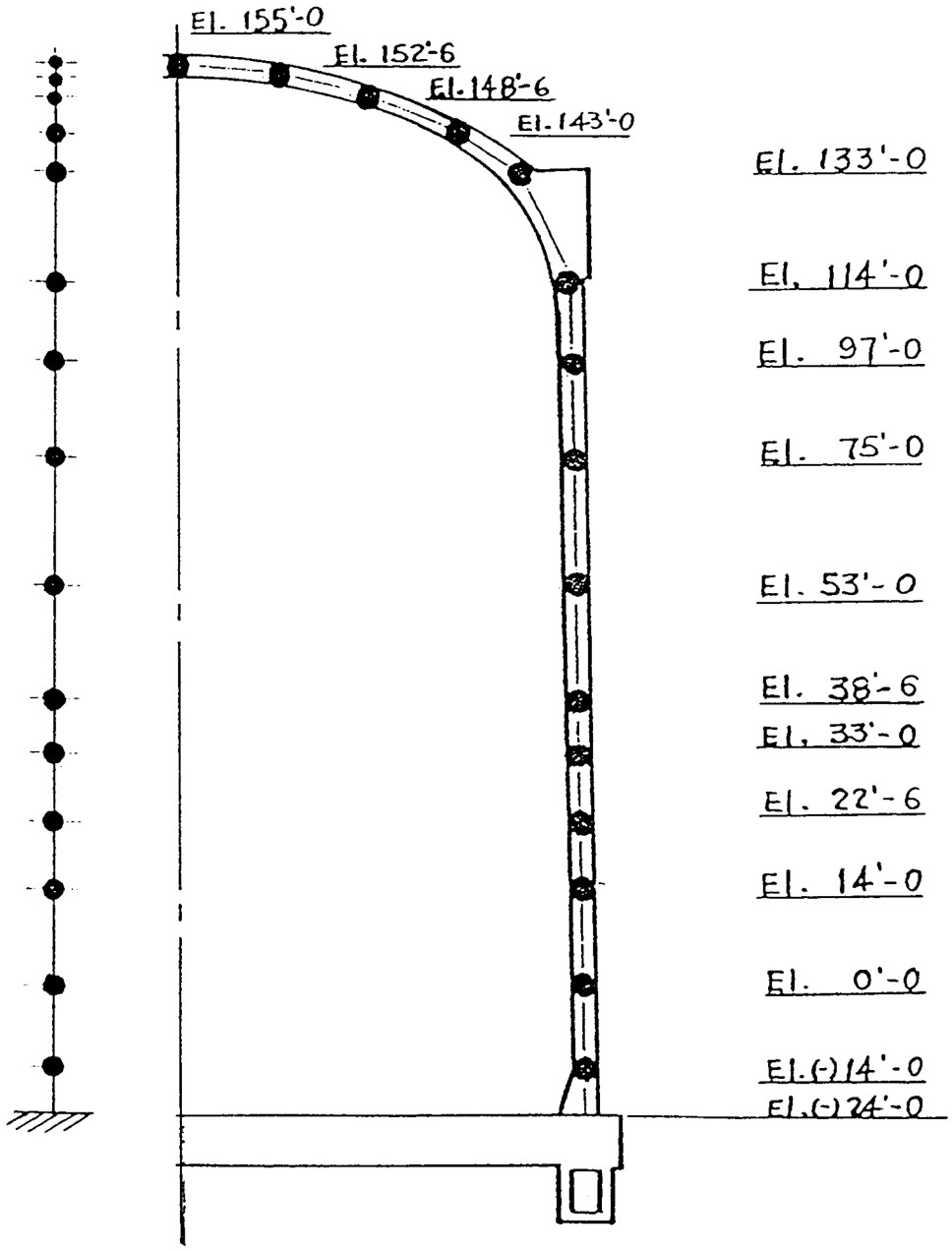


FIGURE 5.8–13 AUXILIARY BAY OF TURBINE BUILDING

FIGURE 5.8–14 CONTAINMENT MASS MODEL



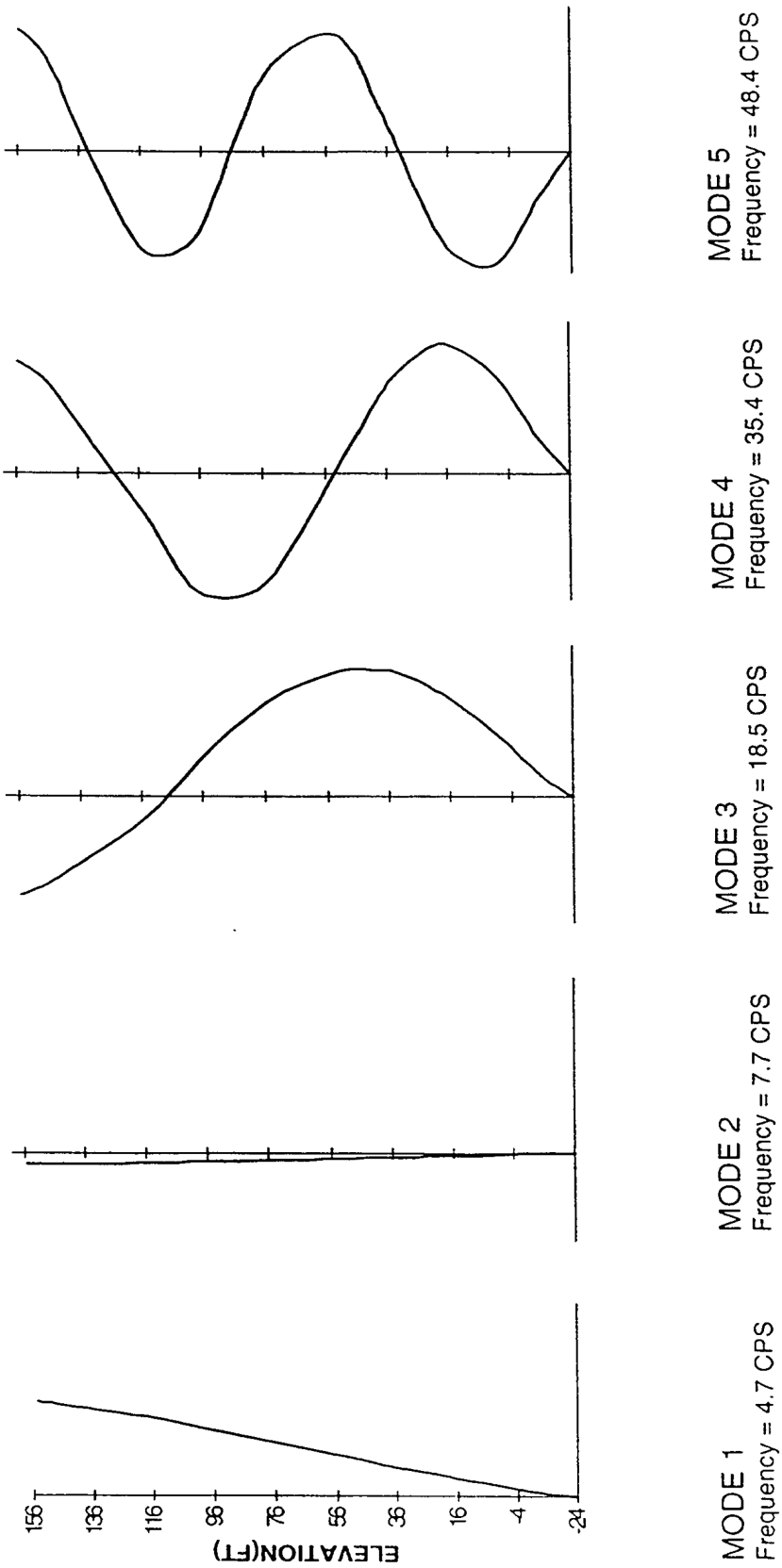


FIGURE 5.8–15 MODE SHAPES FOR THE CONTAINMENT BUILDING

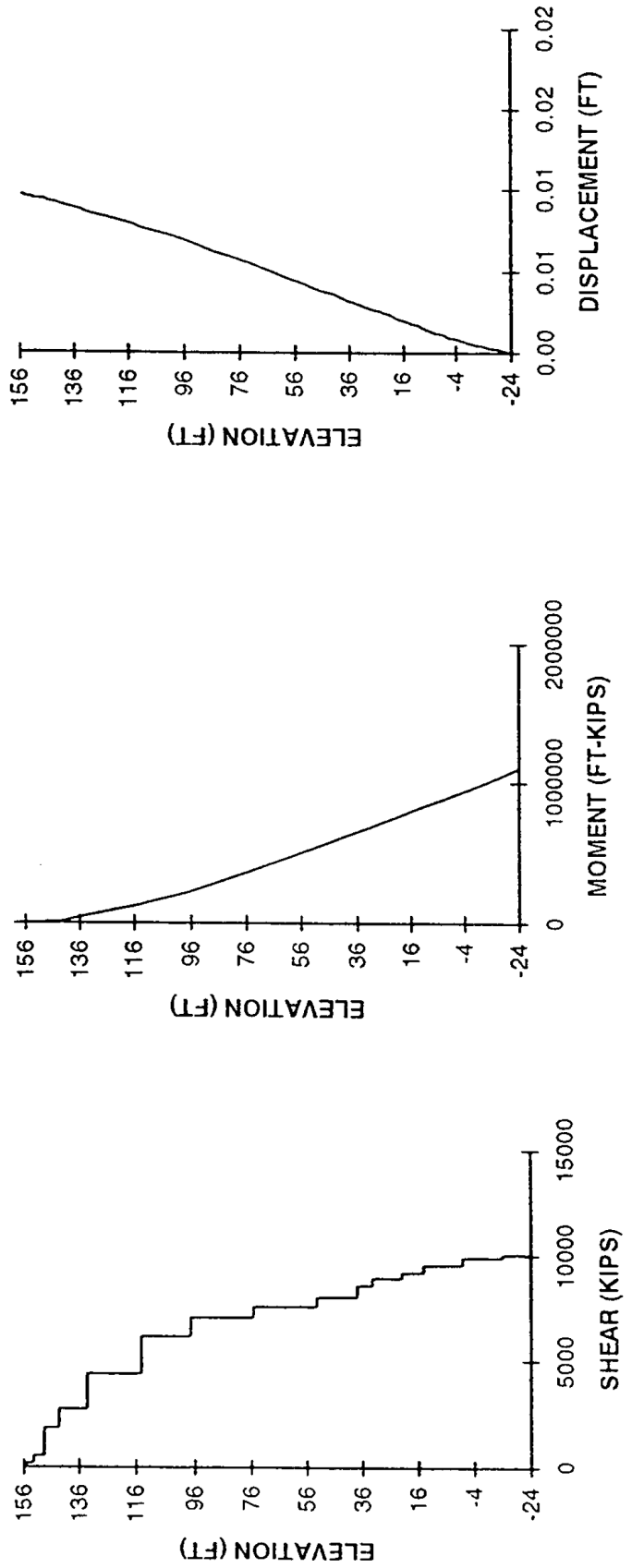


FIGURE 5.8–16 CONTAINMENT BUILDING DESIGN EARTHQUAKE 9% GROUND ACCELERATION

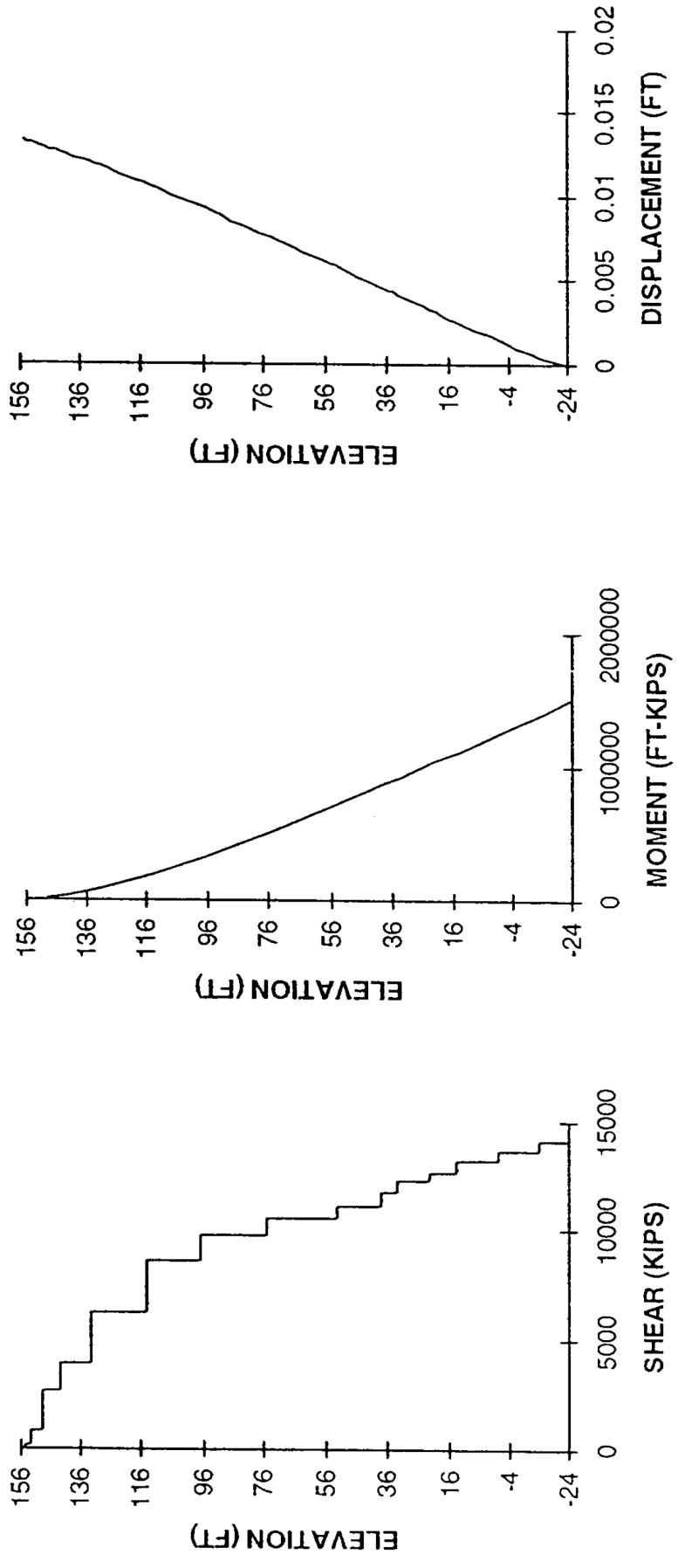


FIGURE 5.8–17 CONTAINMENT BUILDING DESIGN EARTHQUAKE 17% GROUND ACCELERATION

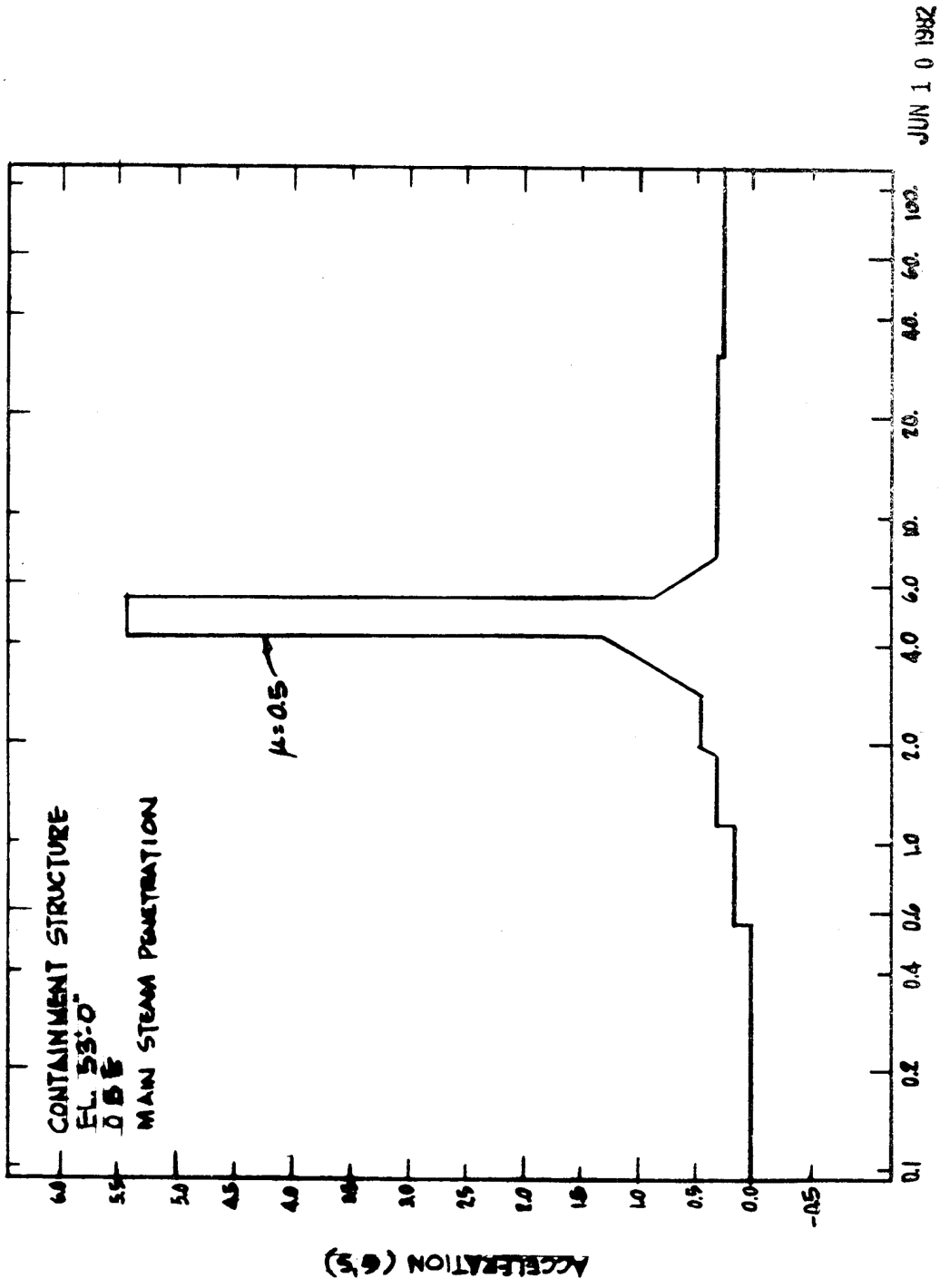


FIGURE 5.8–18 CONTAINMENT STRUCTURE ELEVATION 53 FEET 0 INCHES

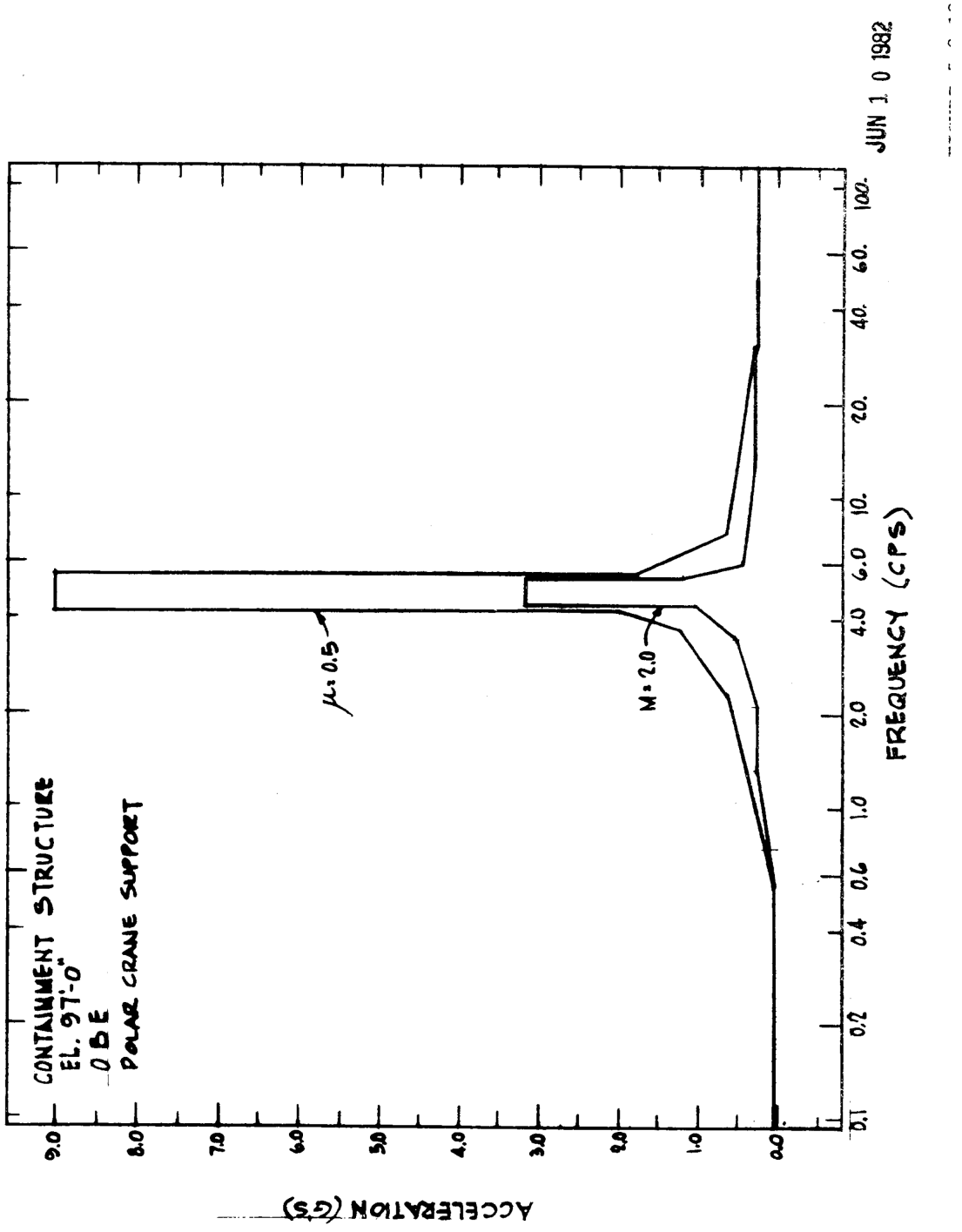


FIGURE 5.8–19 CONTAINMENT STRUCTURE ELEVATION 97 FEET 0 INCHES

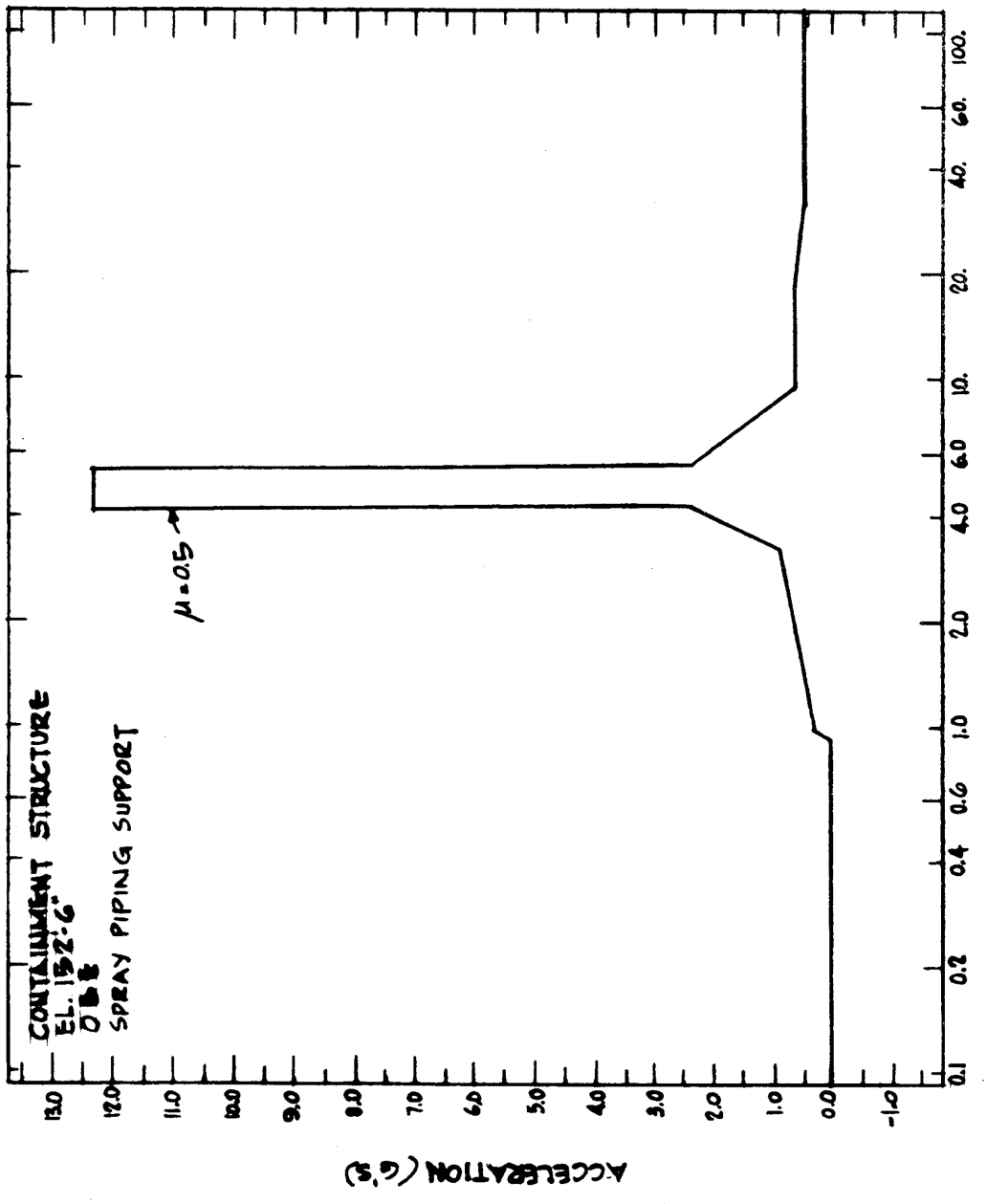


FIGURE 5.8–20 CONTAINMENT STRUCTURE ELEVATION 152 FEET 6 INCHES

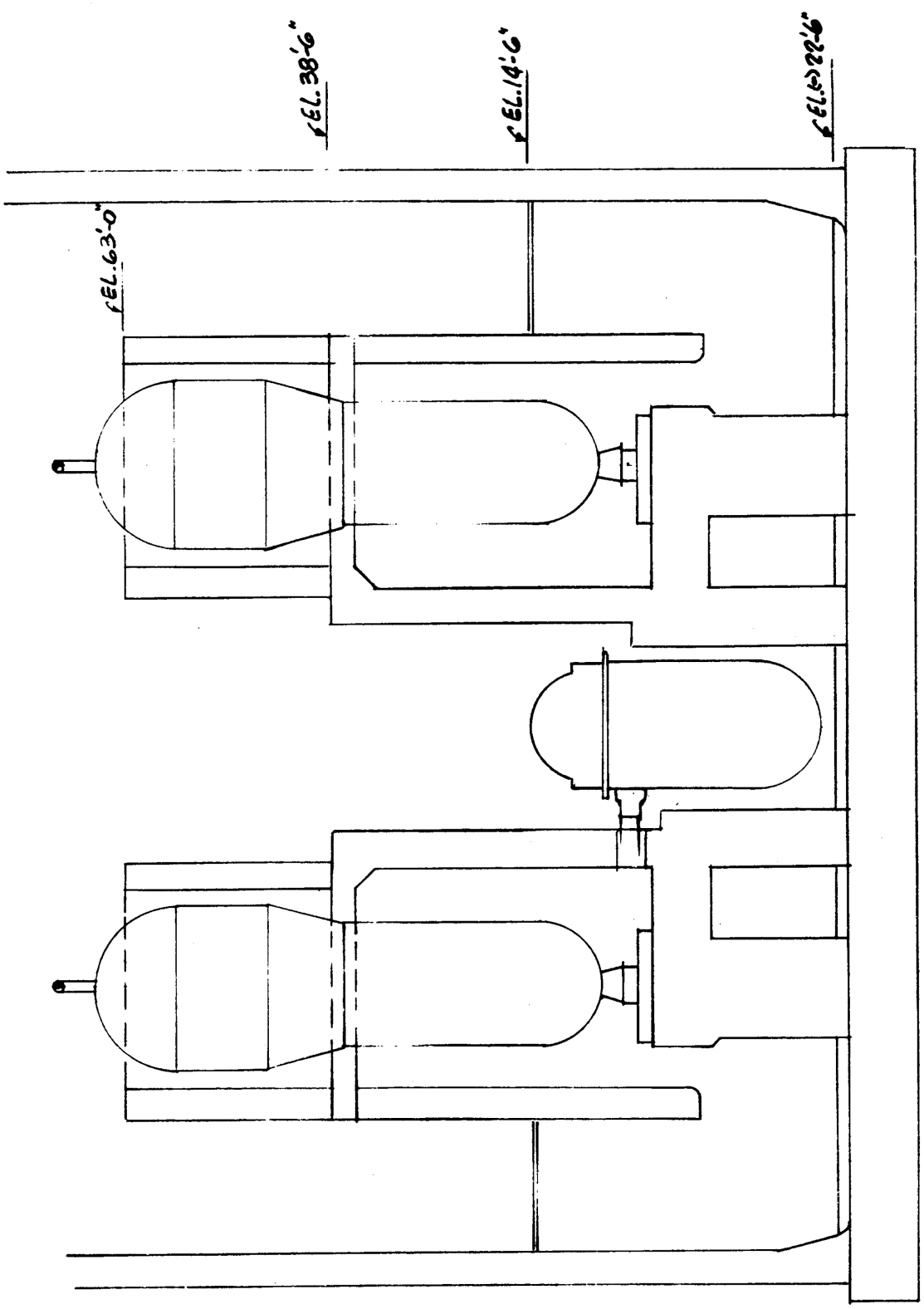


FIGURE 5.8–21 CONTAINMENT INTERNALS MODEL

FIGURE 5.8–22 MODE SHAPES & FREQUENCIES CONTAINMENT INTERNALS - NORTH-SOUTH

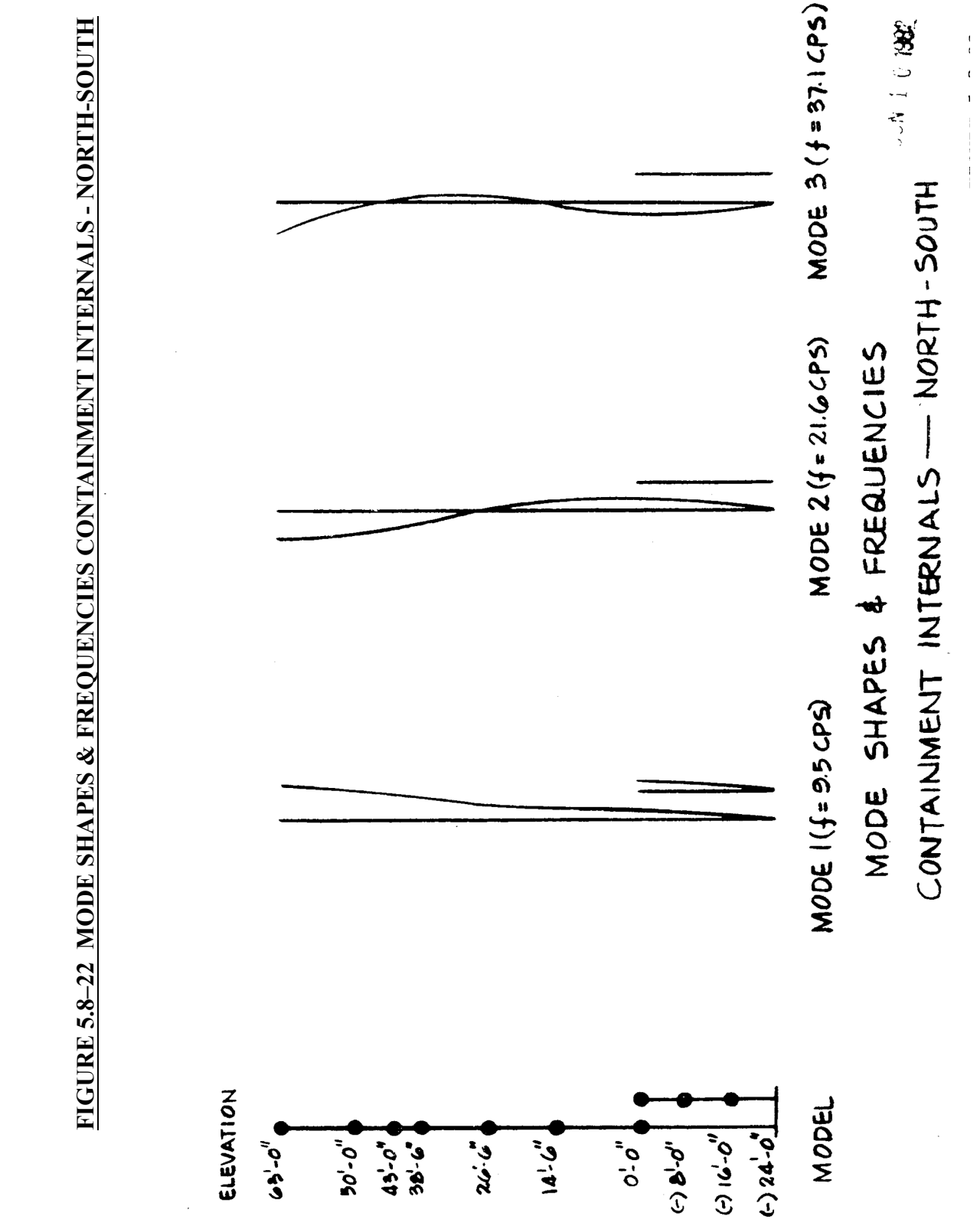


FIGURE 5.8–23 MODE SHAPES & FREQUENCIES CONTAINMENT INTERNALS - NORTH-SOUTH OBE (DBE)

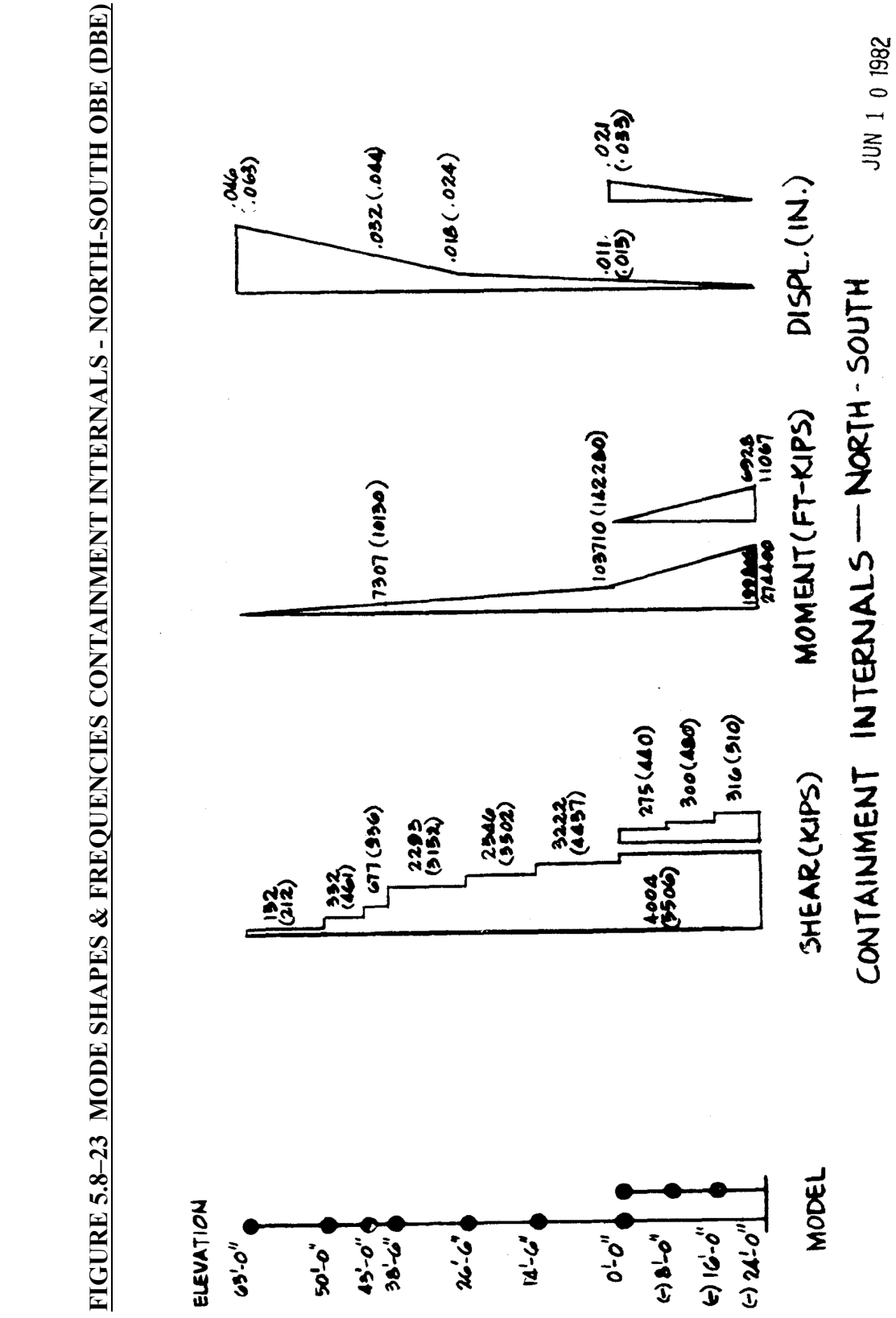


FIGURE 5.8-23

FIGURE 5.8–24 MODE SHAPES & FREQUENCIES CONTAINMENT INTERNALS - EAST-WEST

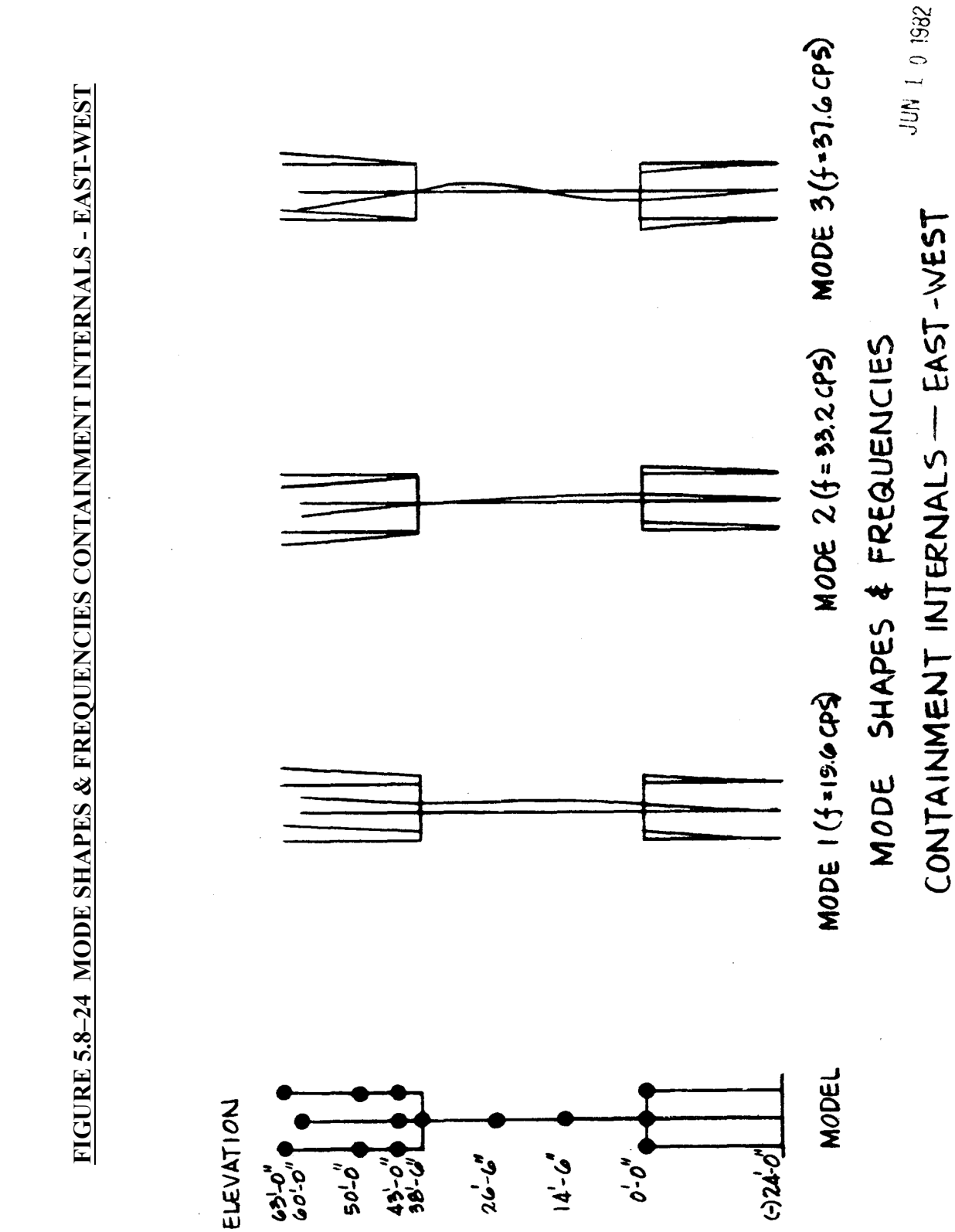


FIGURE 5.8–25 CONTAINMENT INTERNALS - EAST-WEST OBE (DBE)

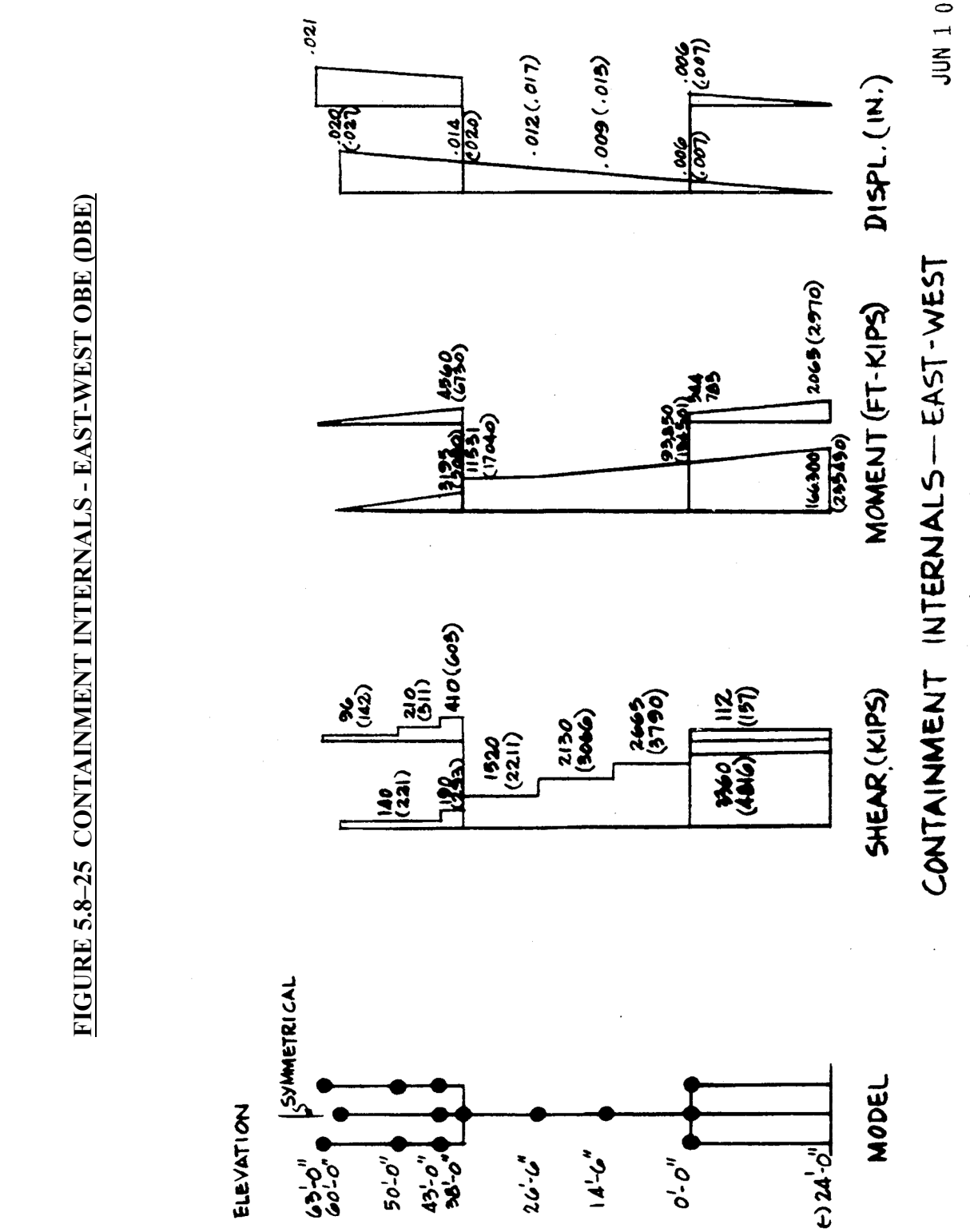
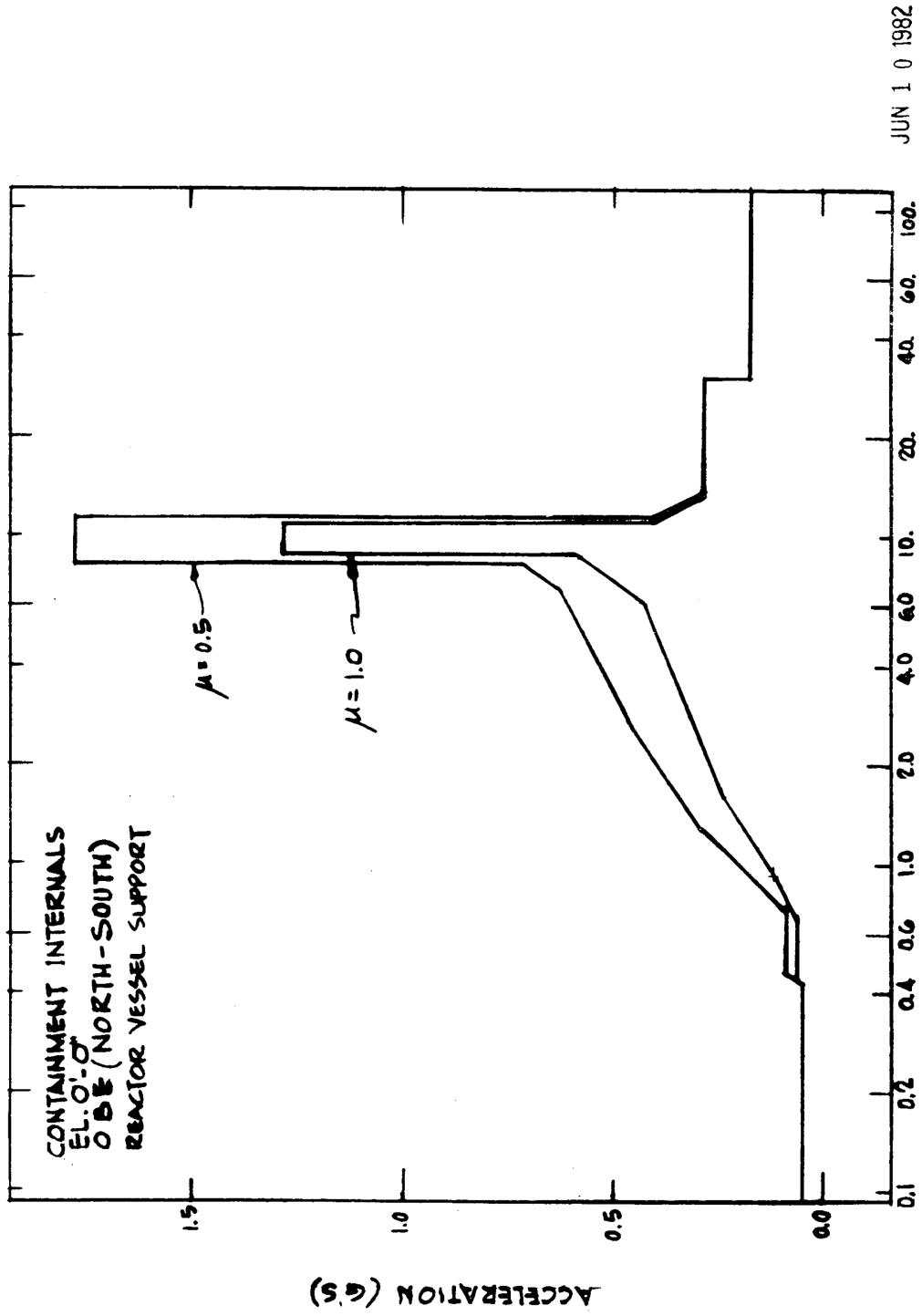


FIGURE 5.8–26 CONTAINMENT INTERNALS ELEVATION 0 FEET 0 INCHES - OBE (NORTH-SOUTH) REACTOR VESSEL SUPPORT



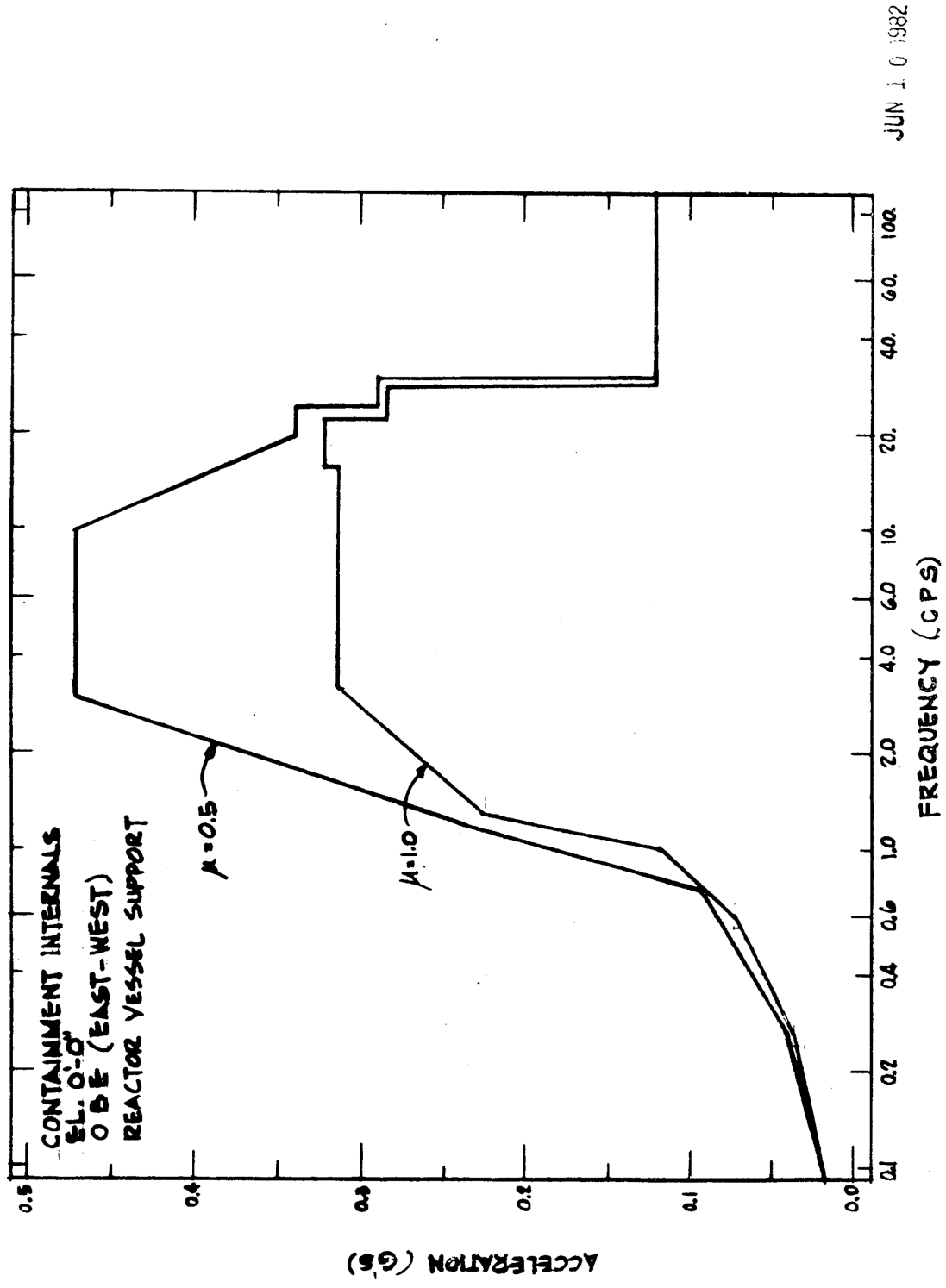
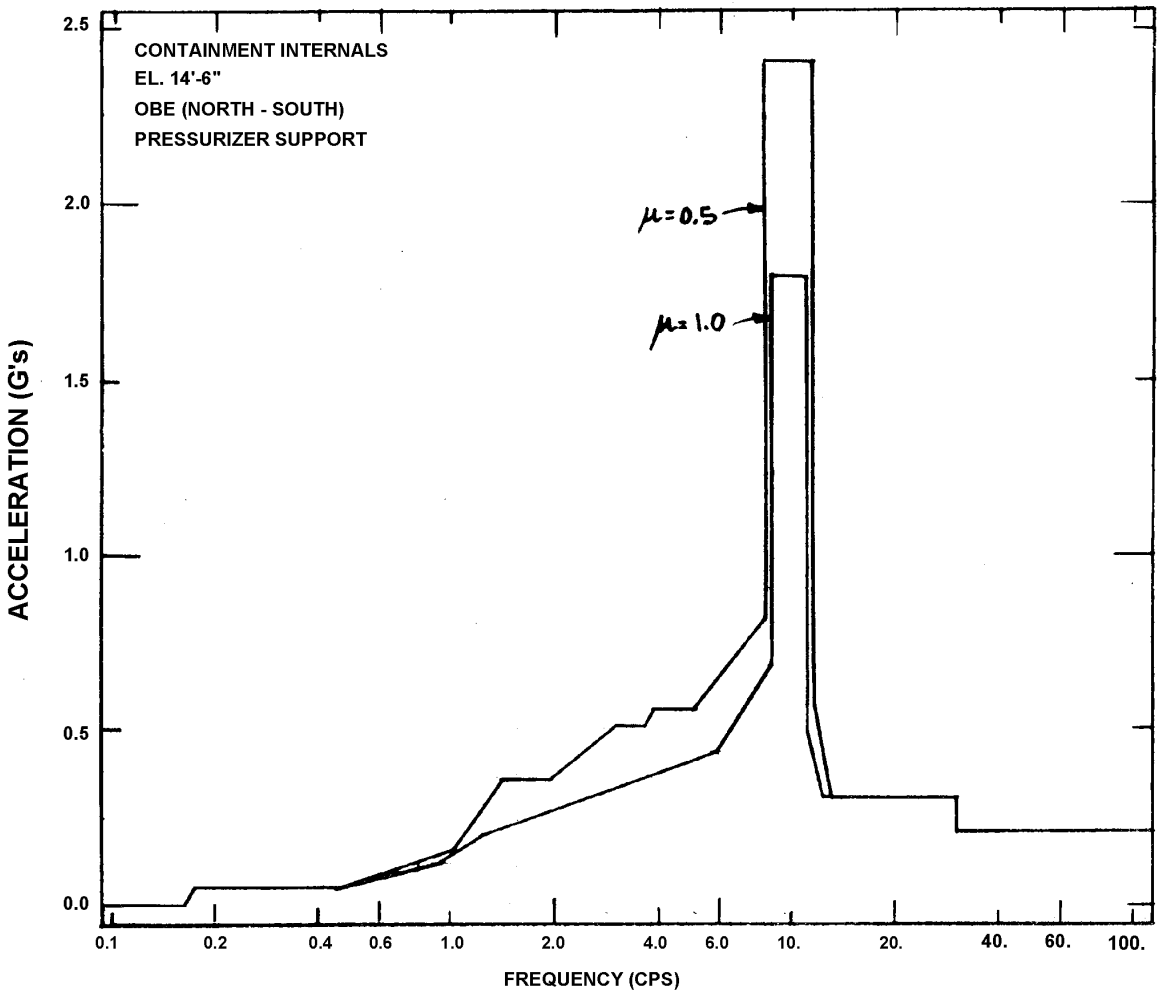


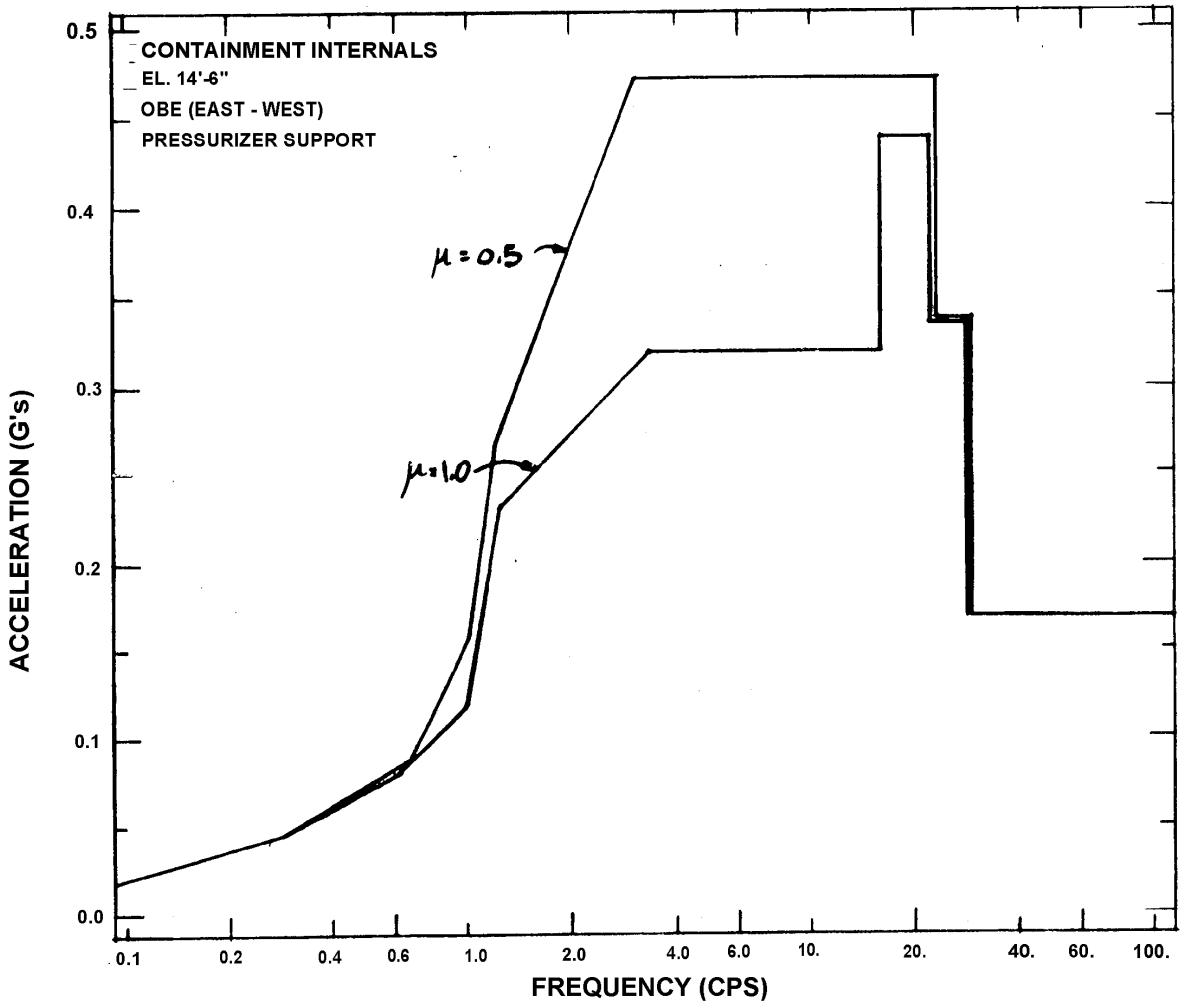
FIGURE 5.8–27 CONTAINMENT INTERNALS ELEVATION 0 FEET 0 INCHES - OBE (EAST-WEST) REACTOR VESSEL SUPPORT

FIGURE 5.8–28 CONTAINMENT INTERNALS ELEVATION 14 FEET 6 INCHES - OBE (NORTH-SOUTH) PRESSURIZER SUPPORT



Note that this ARS is retained in here for historical purpose only. The ARS for the modified pressurizer cubicle are contained in Specification SP-M2-ME-368.

FIGURE 5.8–29 CONTAINMENT INTERNALS ELEVATION 14 FEET 6 INCHES - OBE (EAST-WEST) PRESSURIZER SUPPORT



Note that this ARS is retained in here for historical purpose only. The ARS for the modified pressurizer cubicle are contained in Specification SP-M2-ME-368.

FIGURE 5.8–30 CONTAINMENT INTERNALS ELEVATION 38 FEET 6 INCHES - OBE (NORTH-SOUTH) SAFETY INJECTION TANK SUPPORT

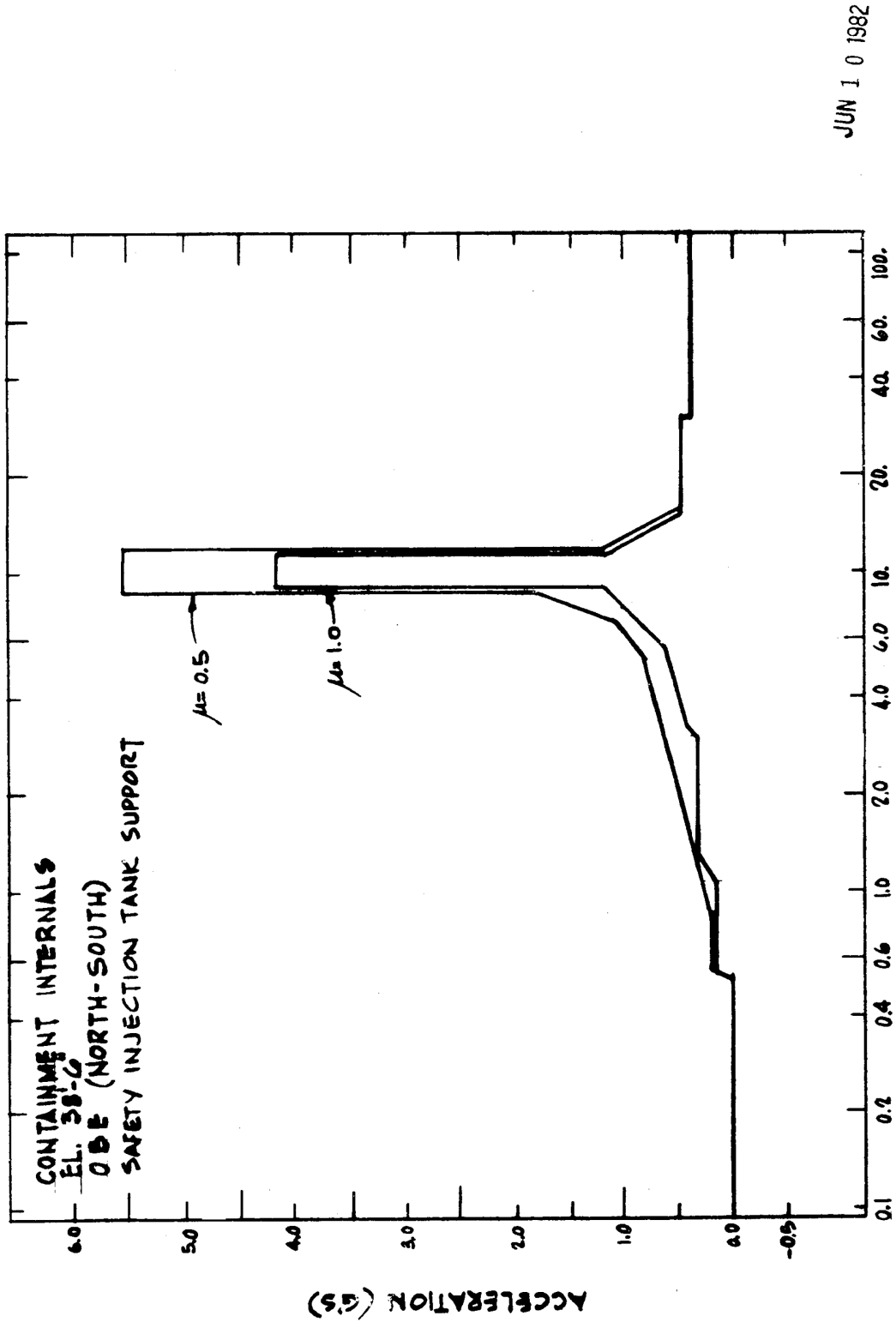


FIGURE 5.8-30

FIGURE 5.8–31 CONTAINMENT INTERNALS ELEVATION 38 FEET 6 INCHES - OBE (EAST-WEST) SAFETY INJECTION TANK SUPPORT

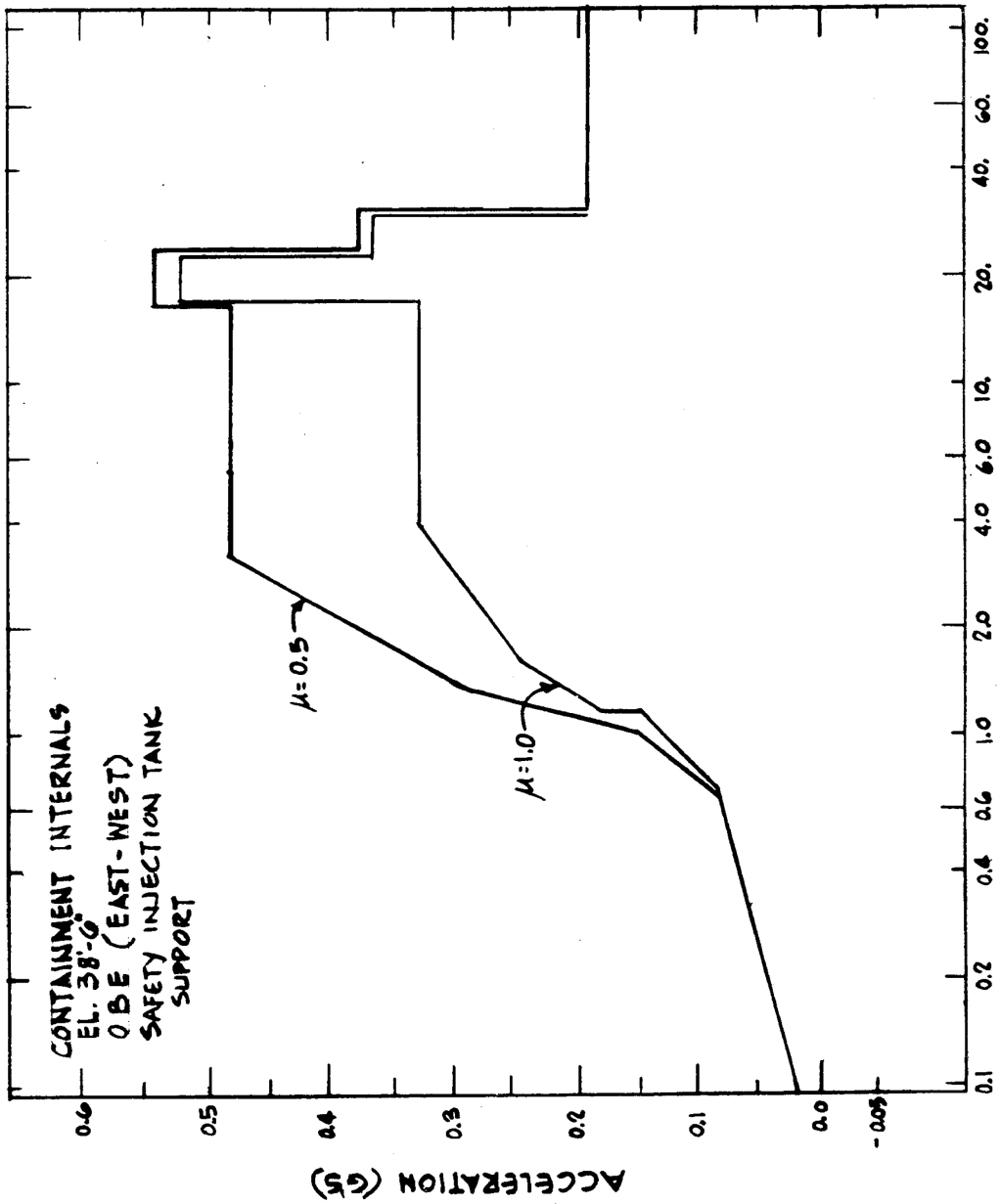


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FIGURE 5.8-31

FIGURE 5.8–32 CONTAINMENT INTERNALS ELEVATION 43 FEET 0 INCHES - OBE (NORTH-SOUTH) STEAM GENERATOR UPPER SUPPORT (SNUBBERS)

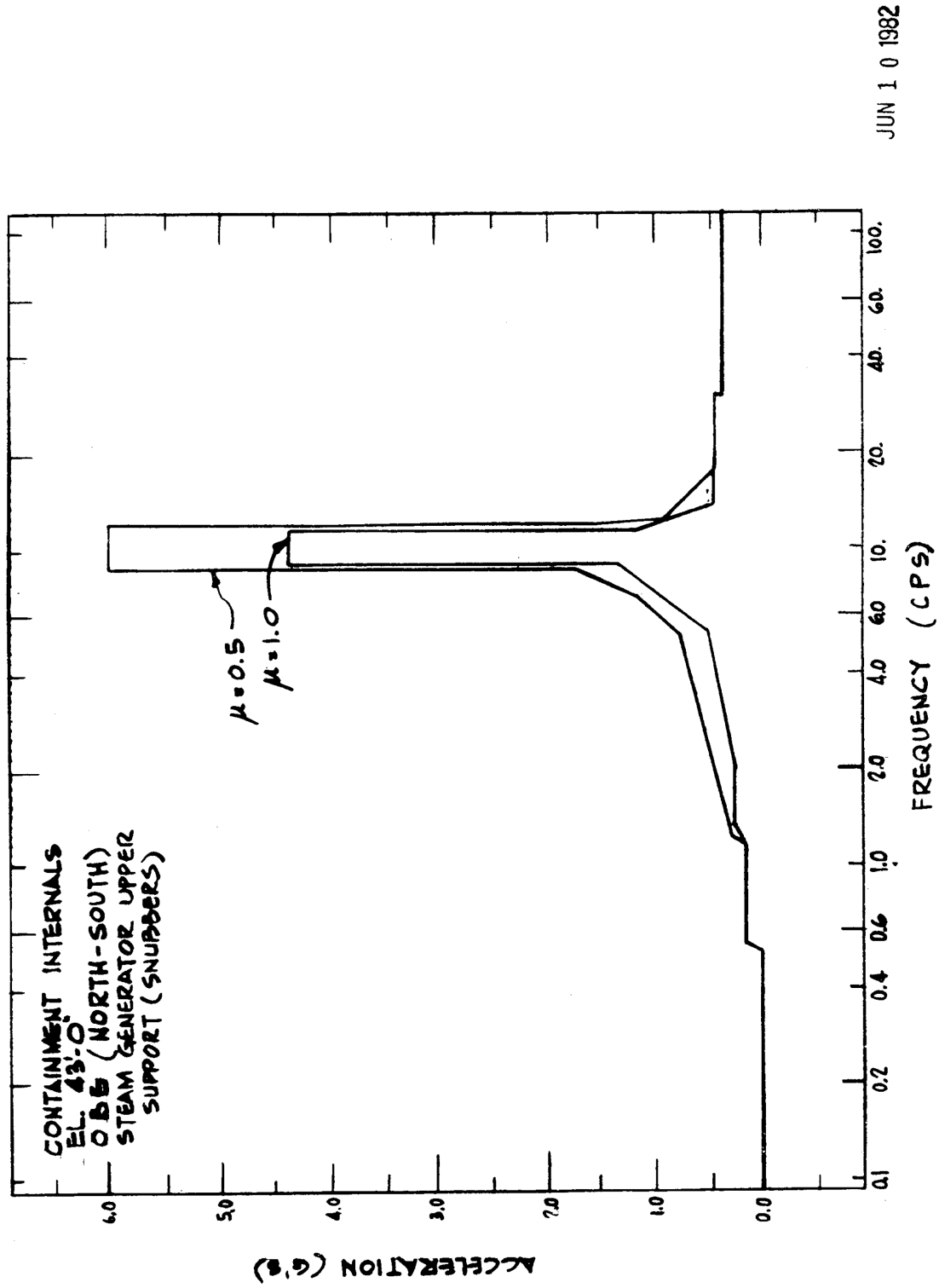


FIGURE 5.8–33 CONTAINMENT INTERNALS ELEVATION 43 FEET 0 INCHES - OBE (EAST-WEST) STEAM GENERATOR UPPER SUPPORT (SNUBBERS)

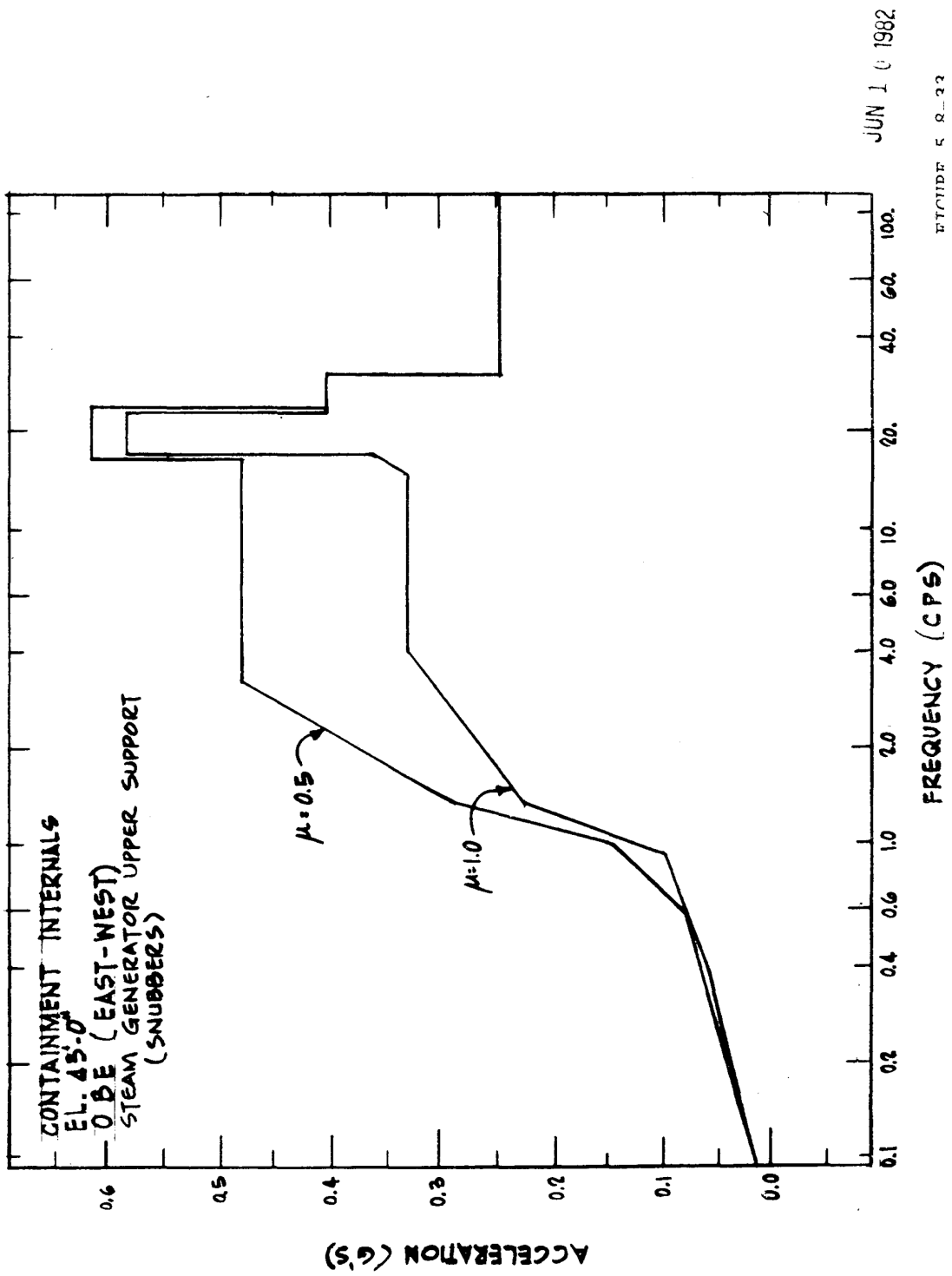


FIGURE 5.8–34 CONTAINMENT INTERNALS ELEVATION 50 FEET 0 INCHES - OBE (NORTH-SOUTH) STEAM GENERATOR UPPER SUPPORT (SHEAR KEYS)

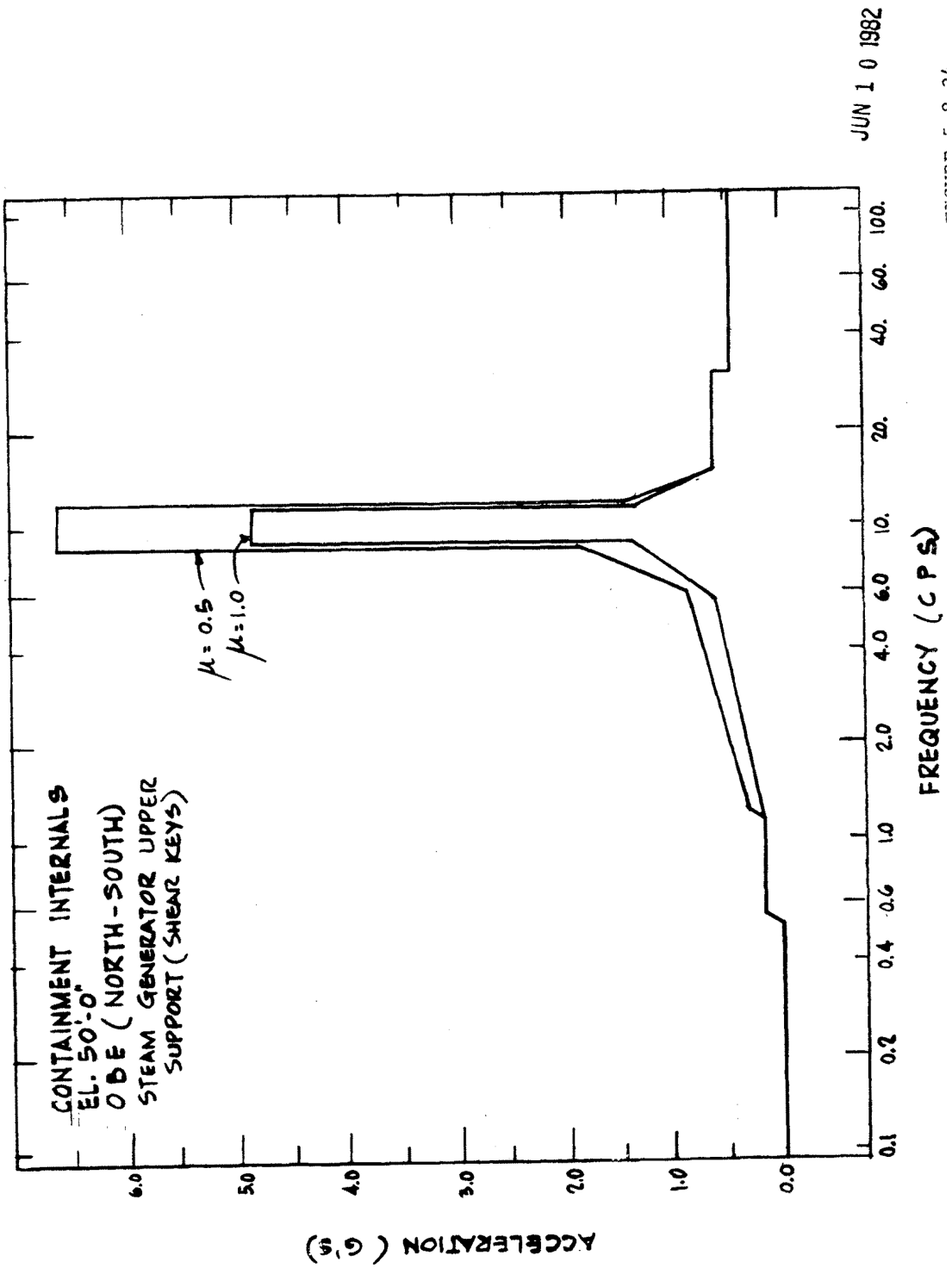


FIGURE 5.8–35 CONTAINMENT INTERNALS ELEVATION 50 FEET 0 INCHES - OBE (EAST-WEST) STEAM GENERATOR UPPER SUPPORT (SHEAR KEYS)

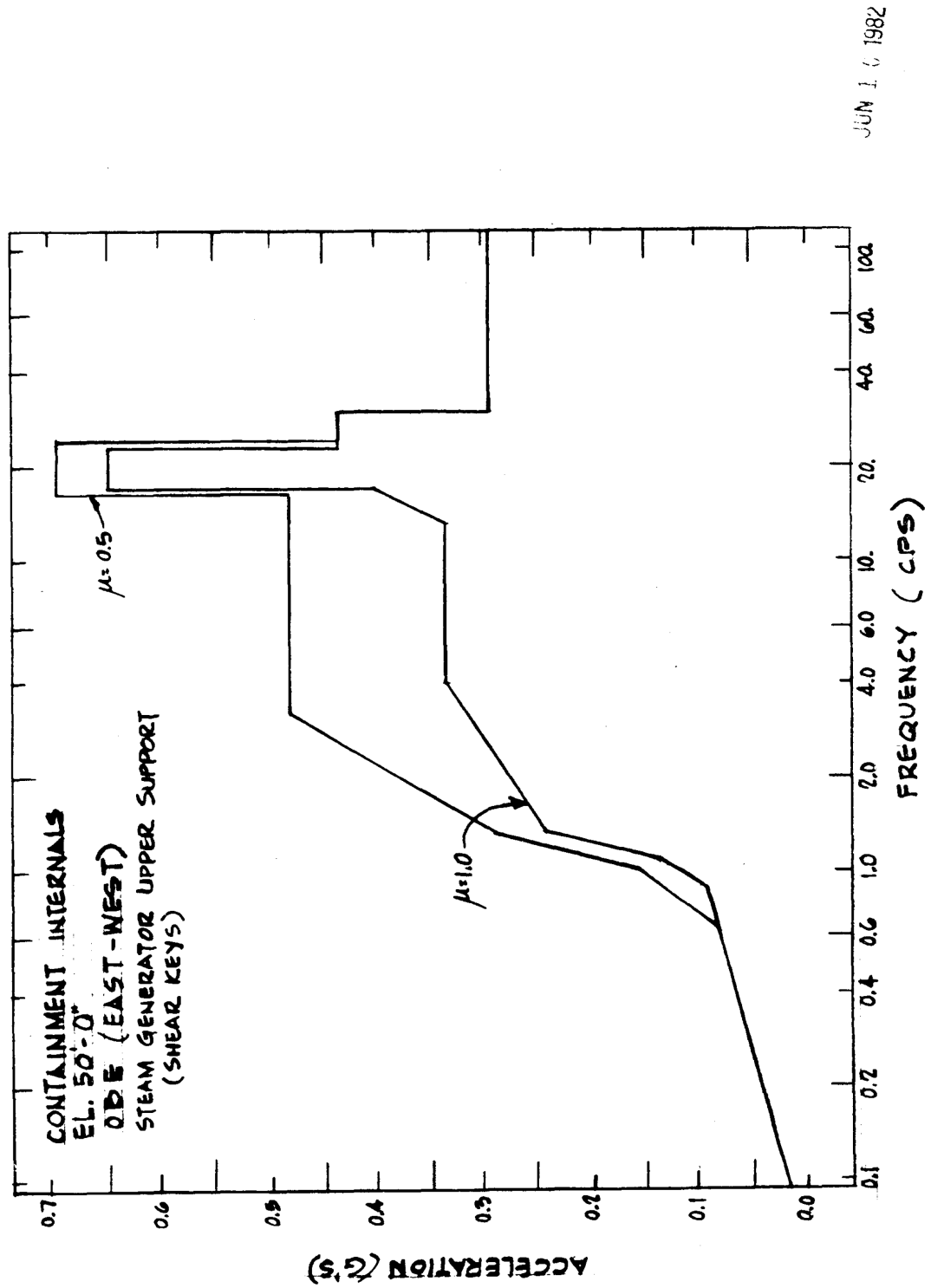


FIGURE 5.8–36 CONTAINMENT INTERNALS ELEVATION0 FEET 0 INCHES - OBE (NORTH-SOUTH) STEAM GENERATOR LOWER SUPPORT

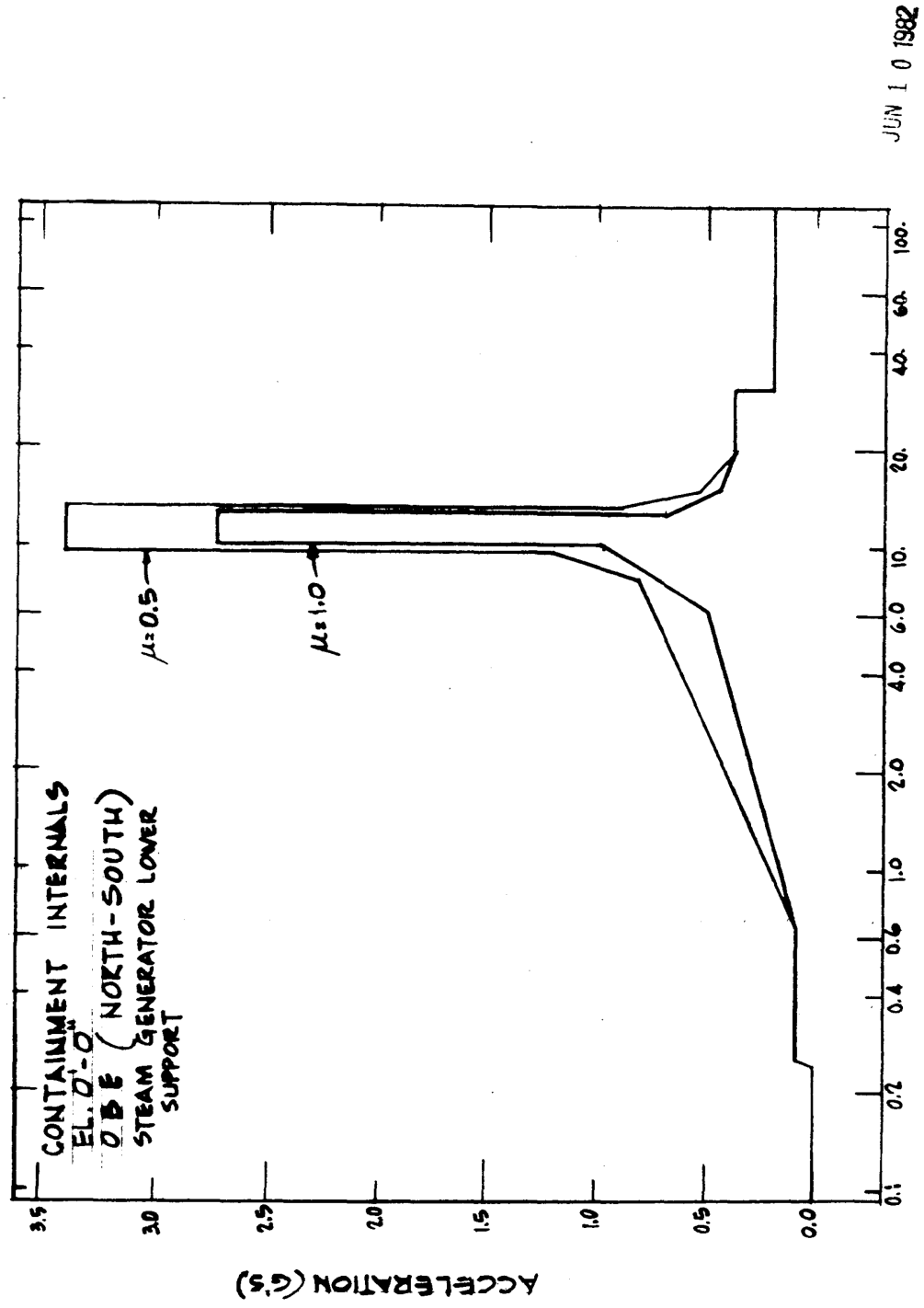


FIGURE 5.8-36

FIGURE 5.8–37 CONTAINMENT INTERNALS ELEVATION 0 FEET 0 INCHES - OBE (EAST-WEST) STEAM GENERATOR LOWER SUPPORT

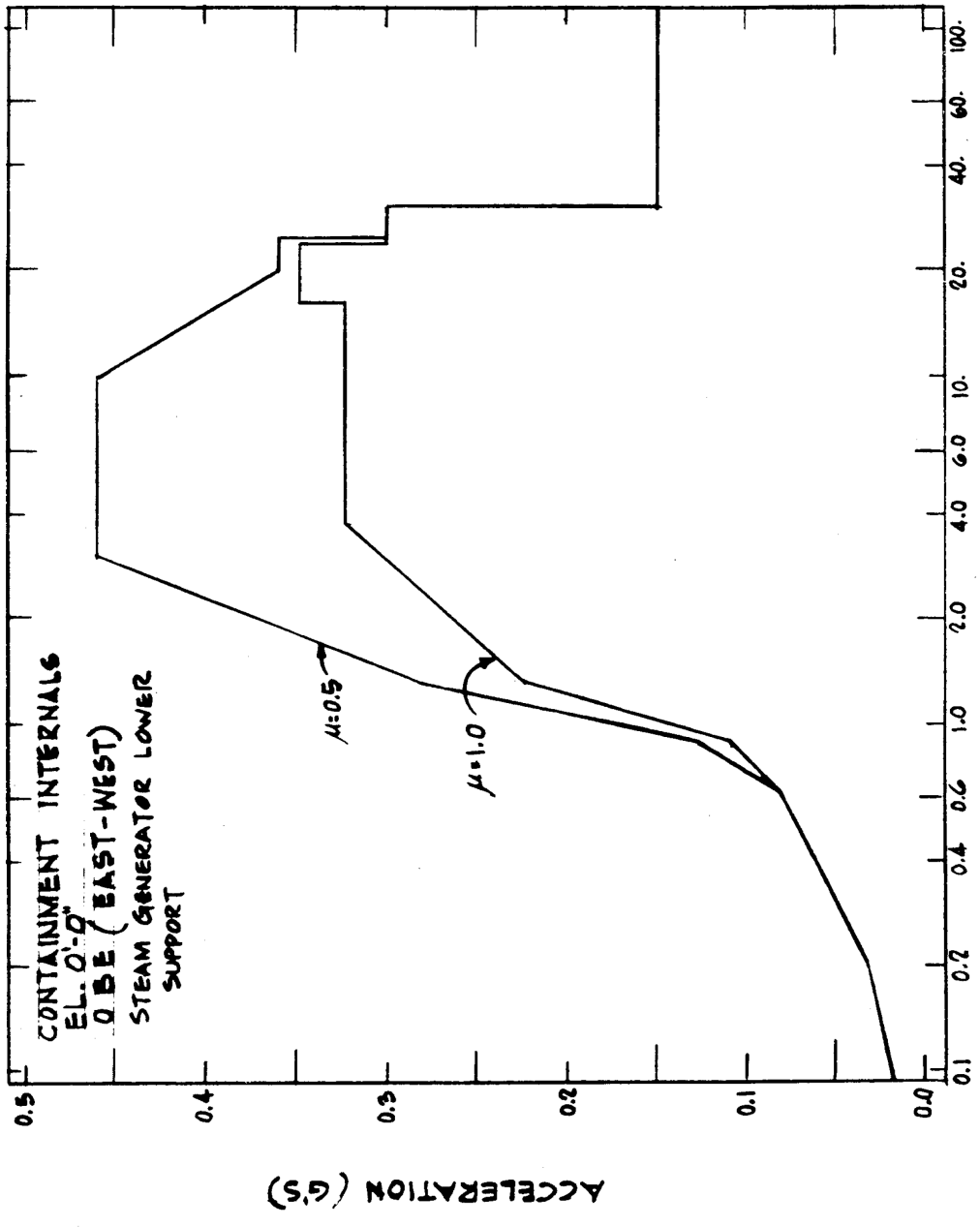


FIGURE 5.8–37

FIGURE 5.8–38 AUXILIARY BUILDING MODEL

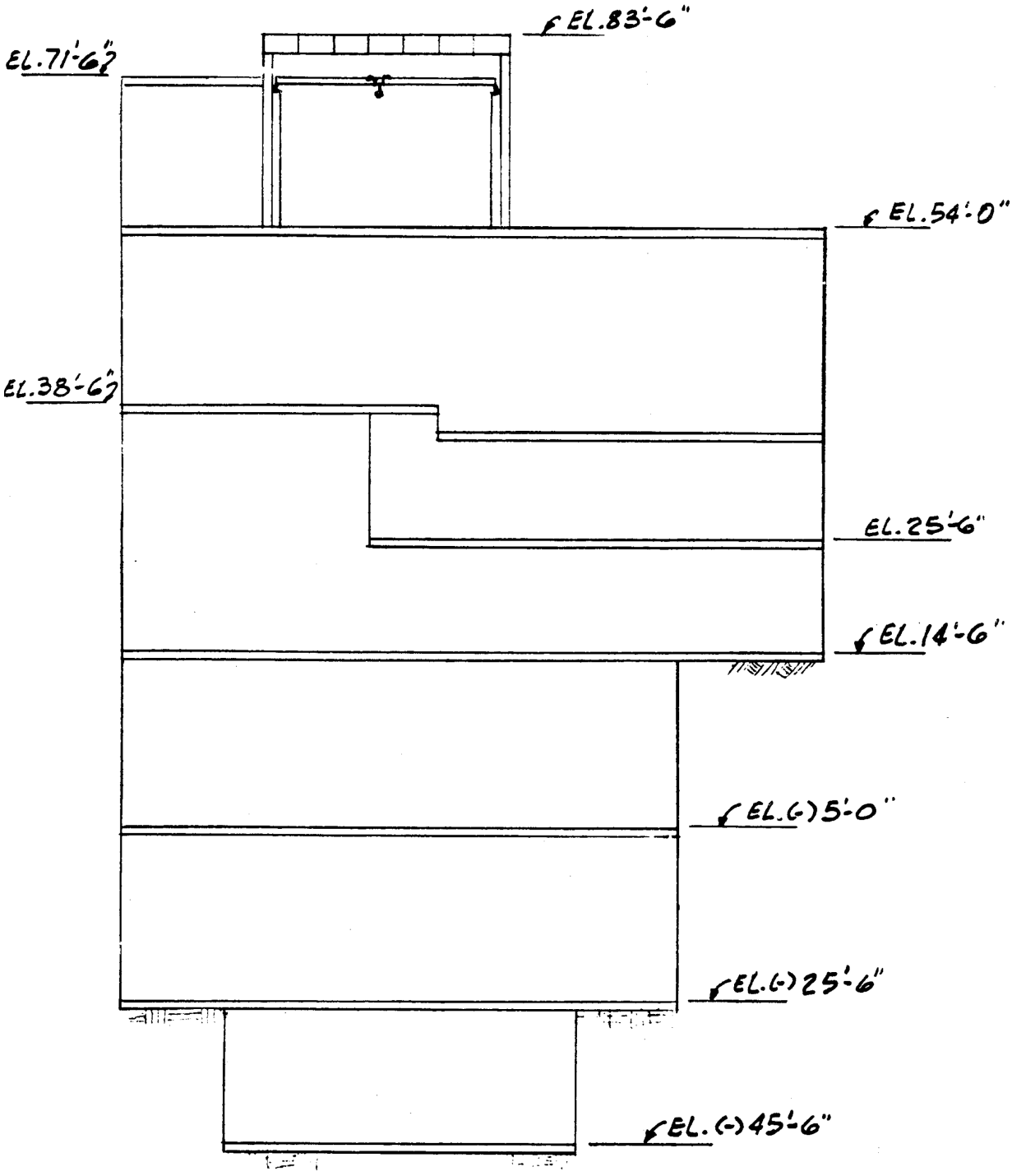


FIGURE 5.8–39 MODE SHAPES & FREQUENCIES AUXILIARY BUILDING - NORTH-SOUTH

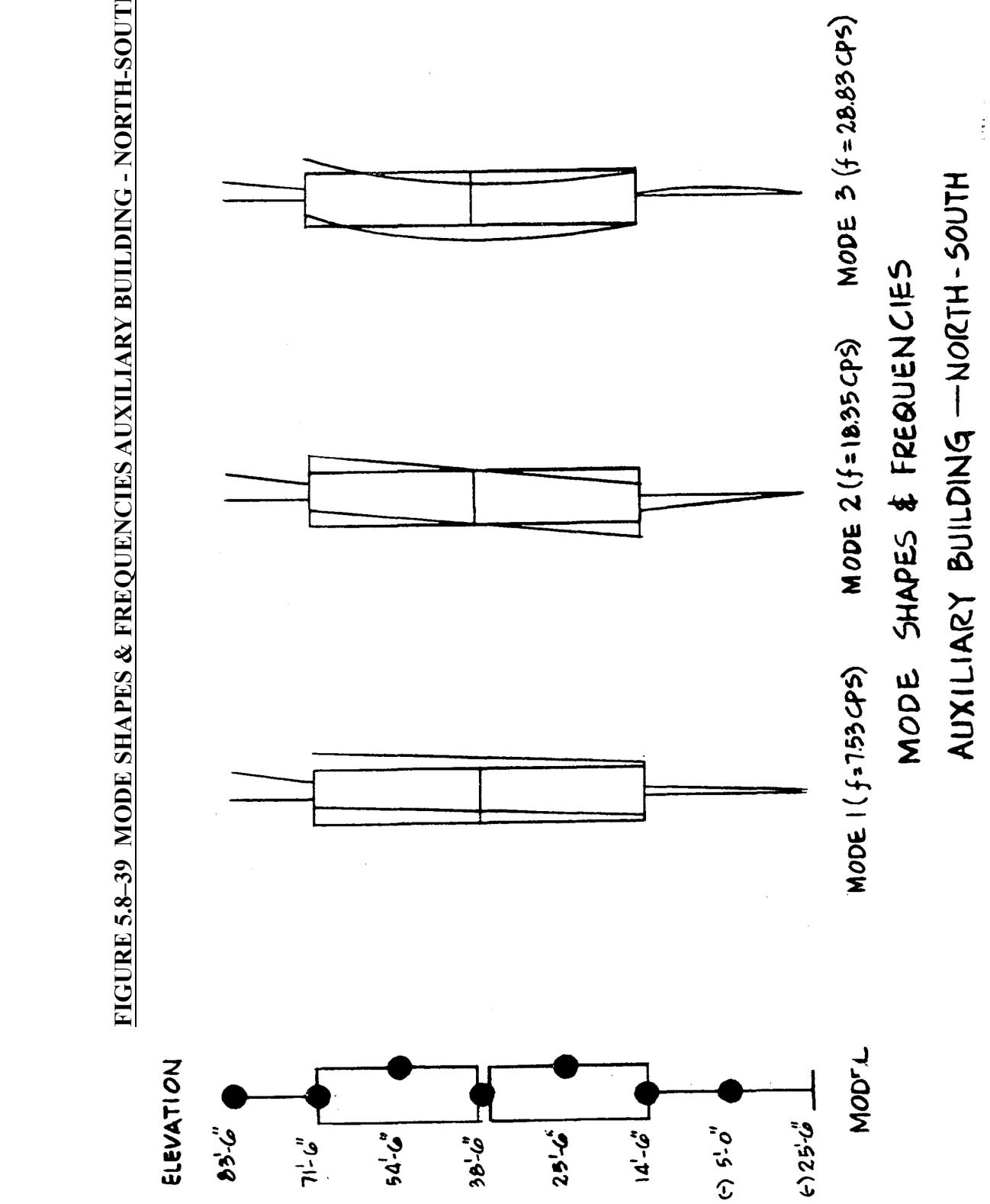


FIGURE 5.8–40 AUXILIARY BUILDING - NORTH-SOUTH OBE (DBE)

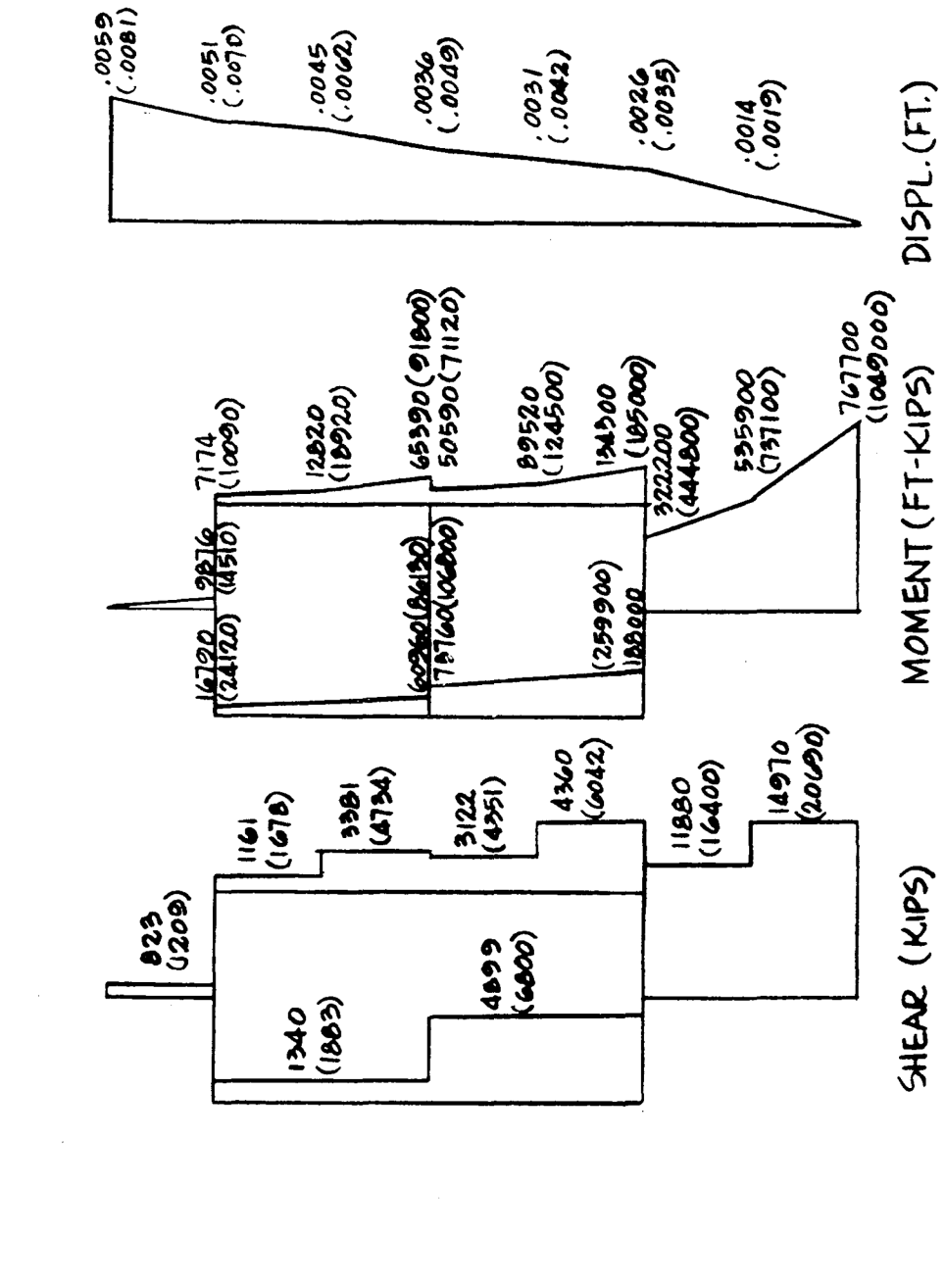
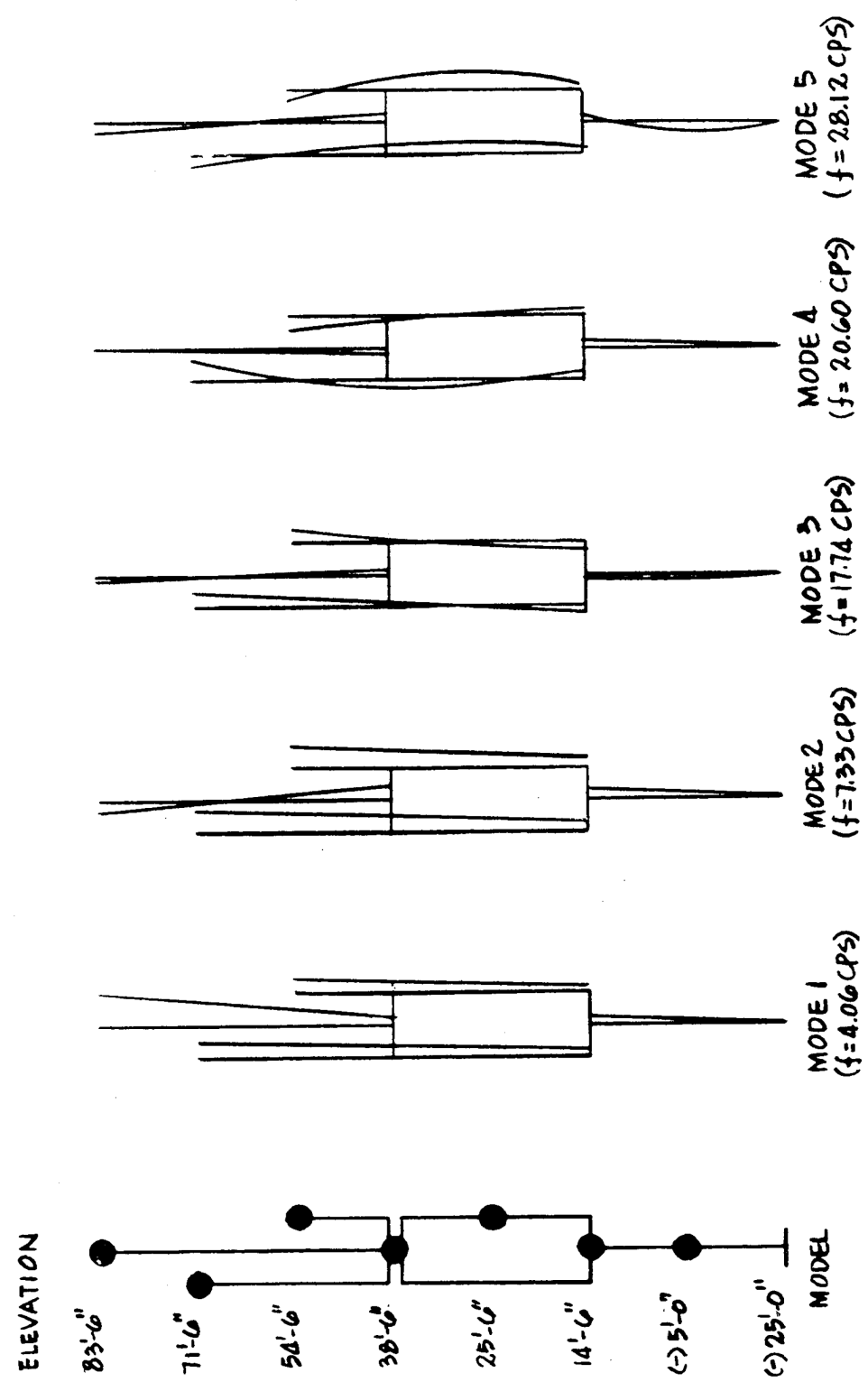


FIGURE 5.8–41 MODE SHAPES & FREQUENCIES AUXILIARY BUILDING - EAST-WEST



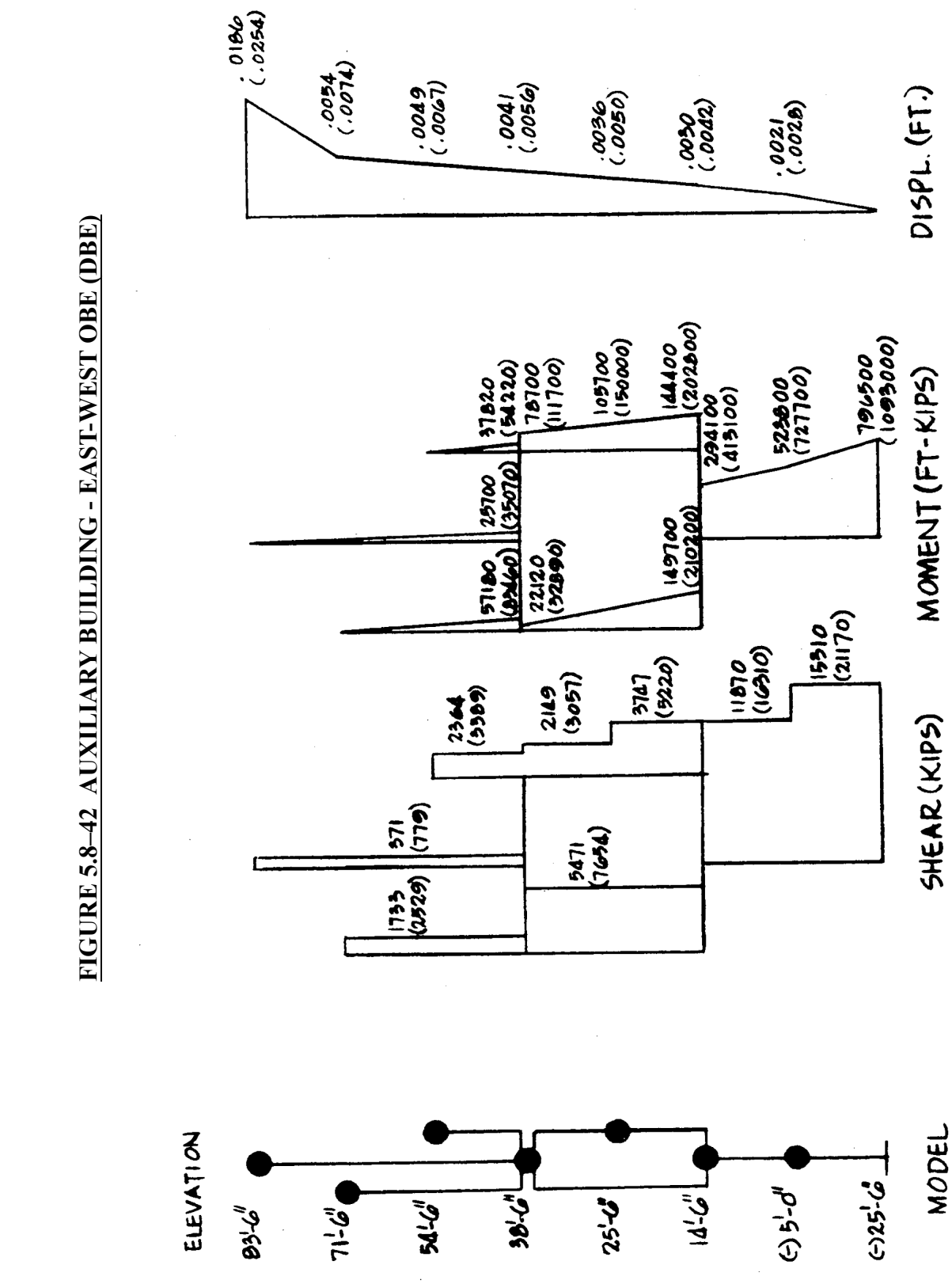
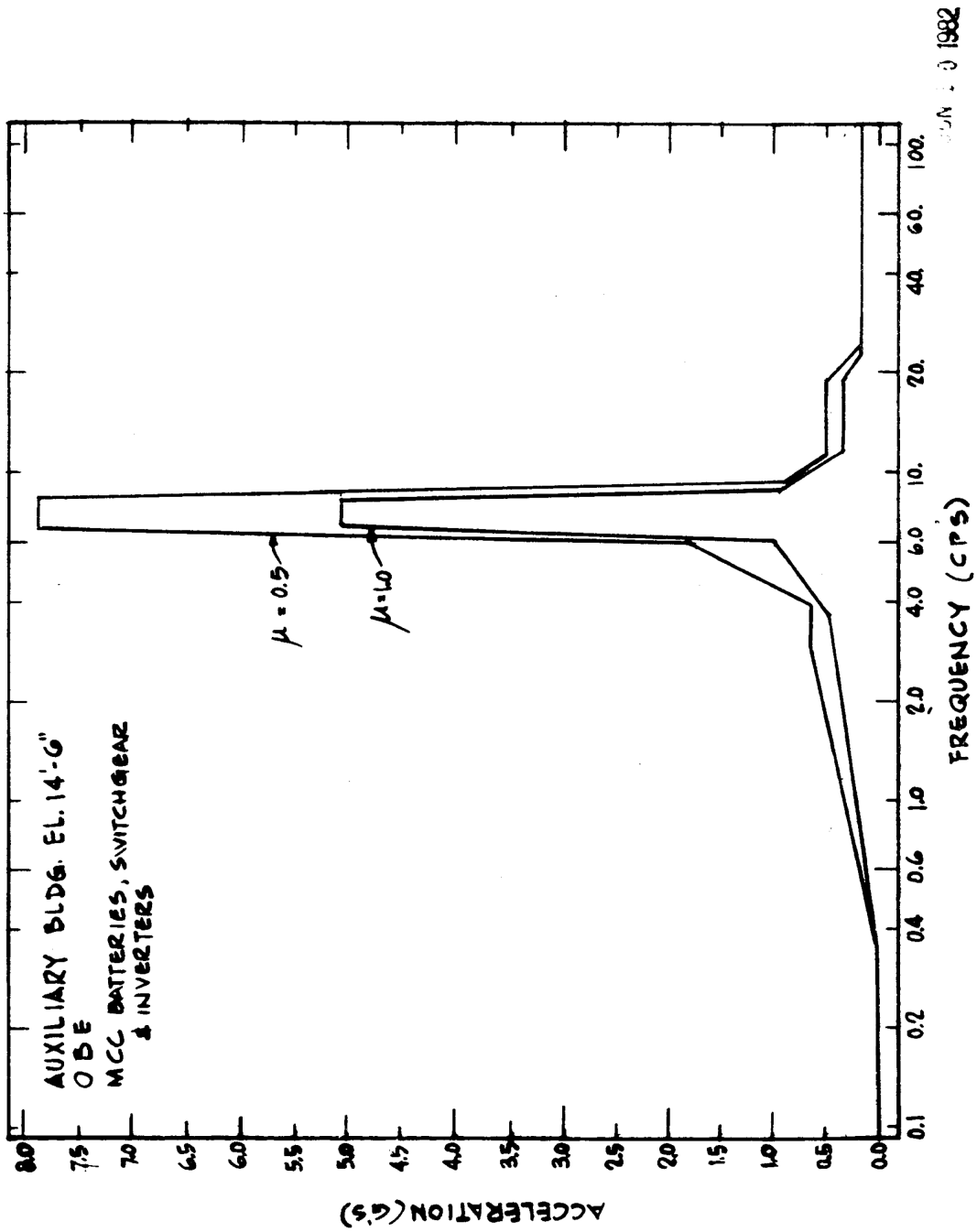


FIGURE 5.8–42 AUXILIARY BUILDING - EAST-WEST OBE (DBE)

FIGURE 5.8–43 AUXILIARY BUILDING ELEVATION 14 FEET 6 INCHES



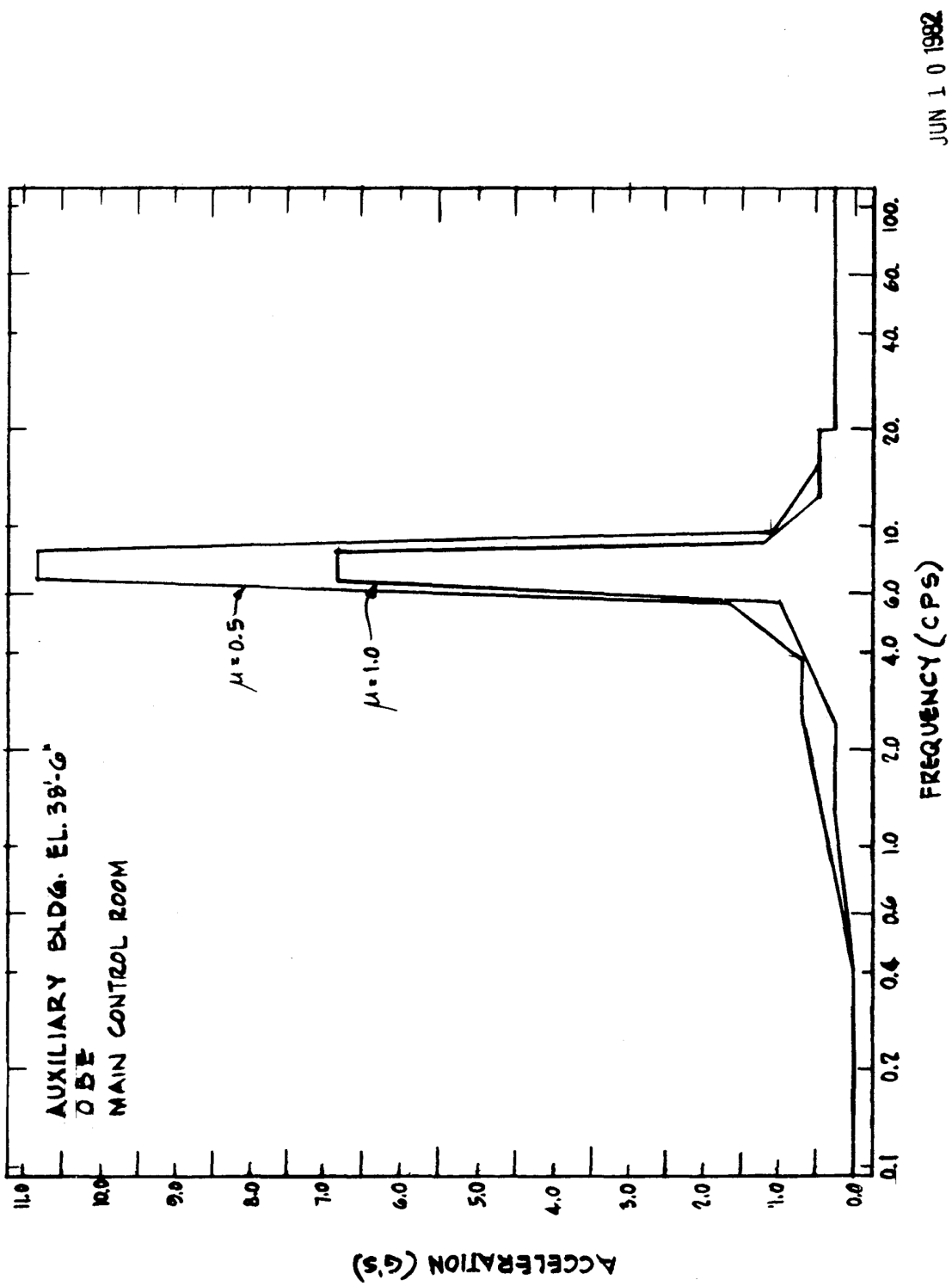


FIGURE 5.8-44 AUXILIARY BUILDING ELEVATION 38 FEET 6 INCHES

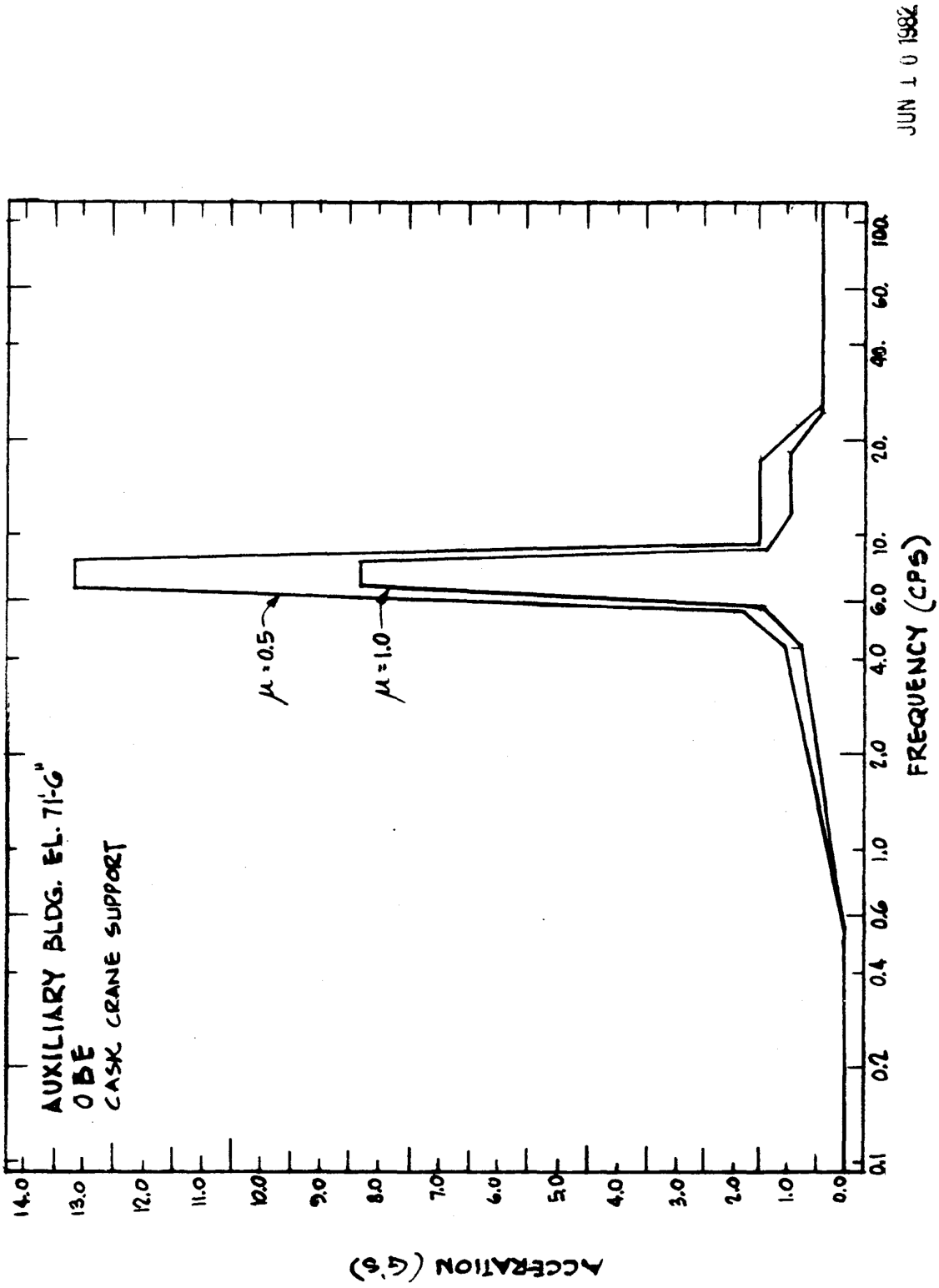


FIGURE 5.8-45 AUXILIARY BUILDING ELEVATION 71 FEET 6 INCHES

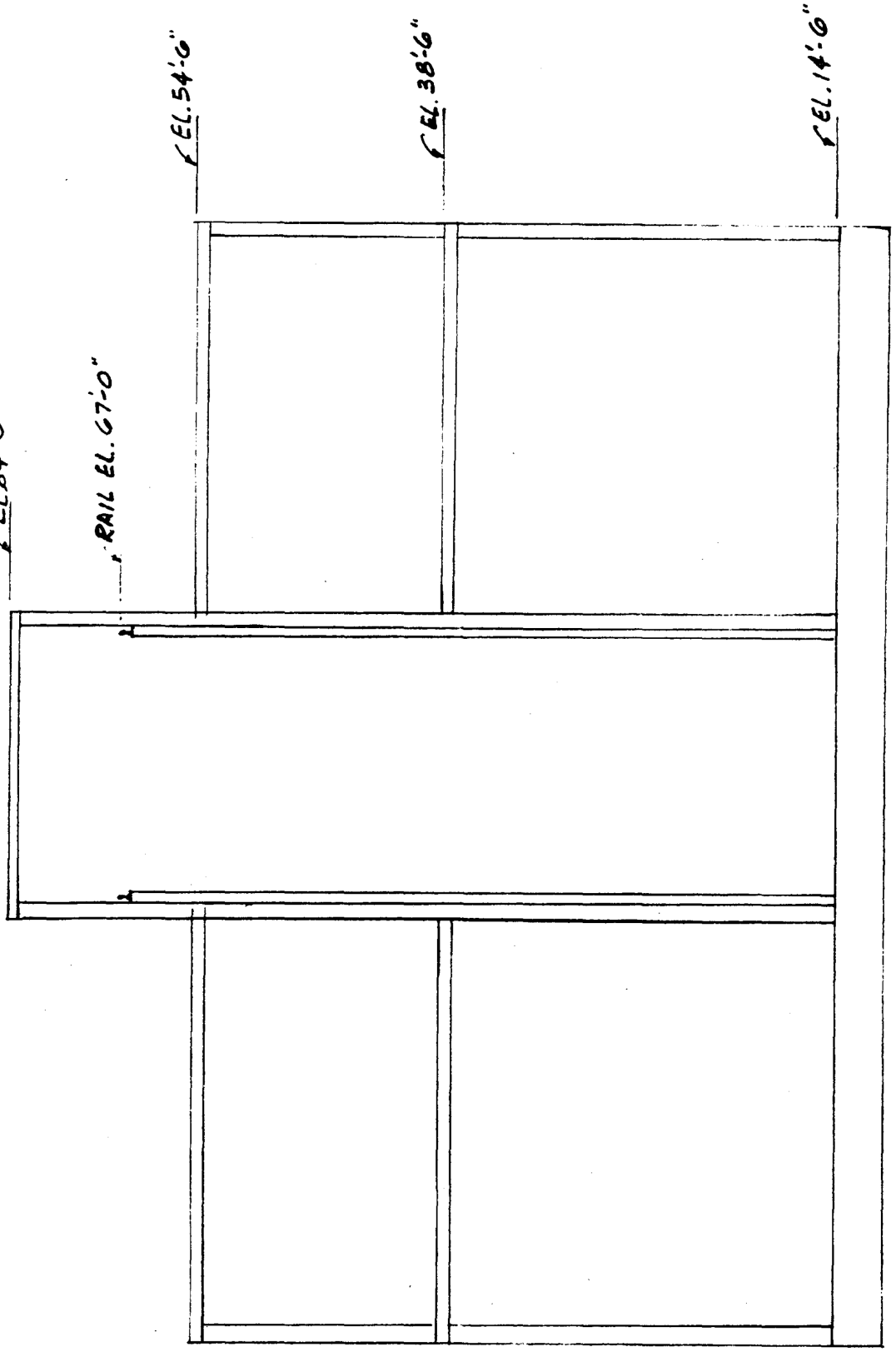


FIGURE 5.8–46 WAREHOUSE MODEL

FIGURE 5.8–47 MODE SHAPES & FREQUENCIES WAREHOUSE BLDG. - EAST-WEST

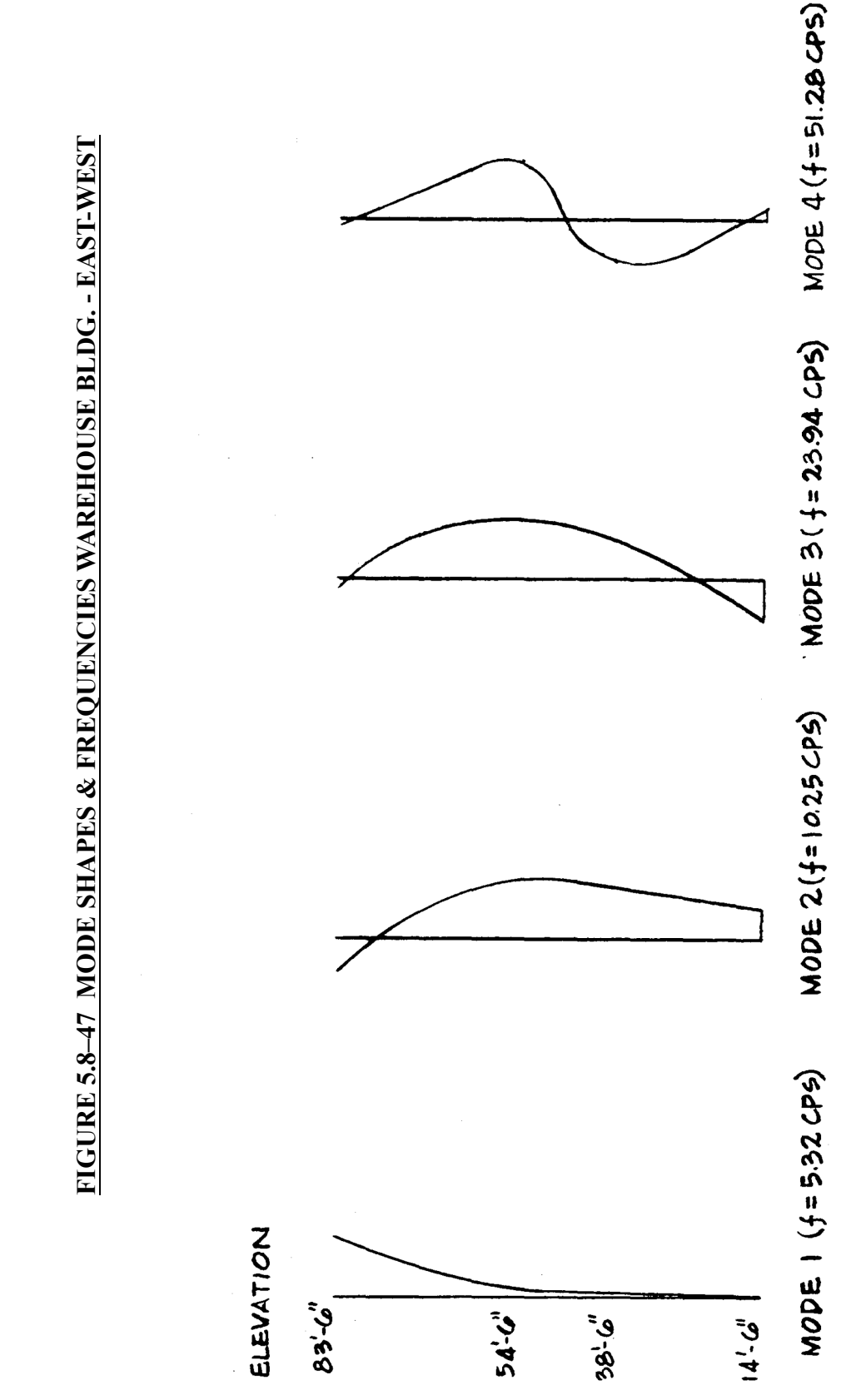
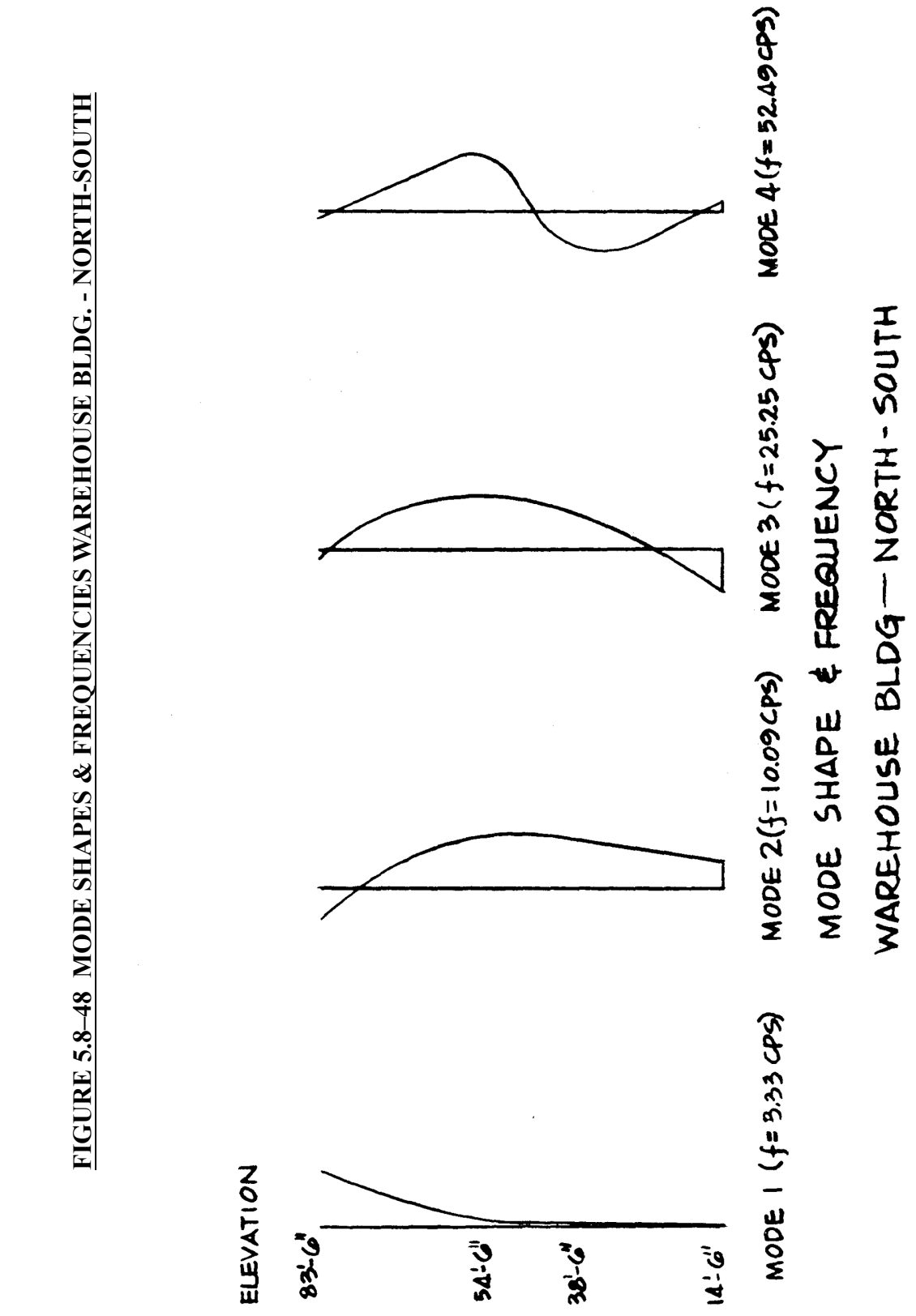


FIGURE 5.8–48 MODE SHAPES & FREQUENCIES WAREHOUSE BLDG. - NORTH-SOUTH



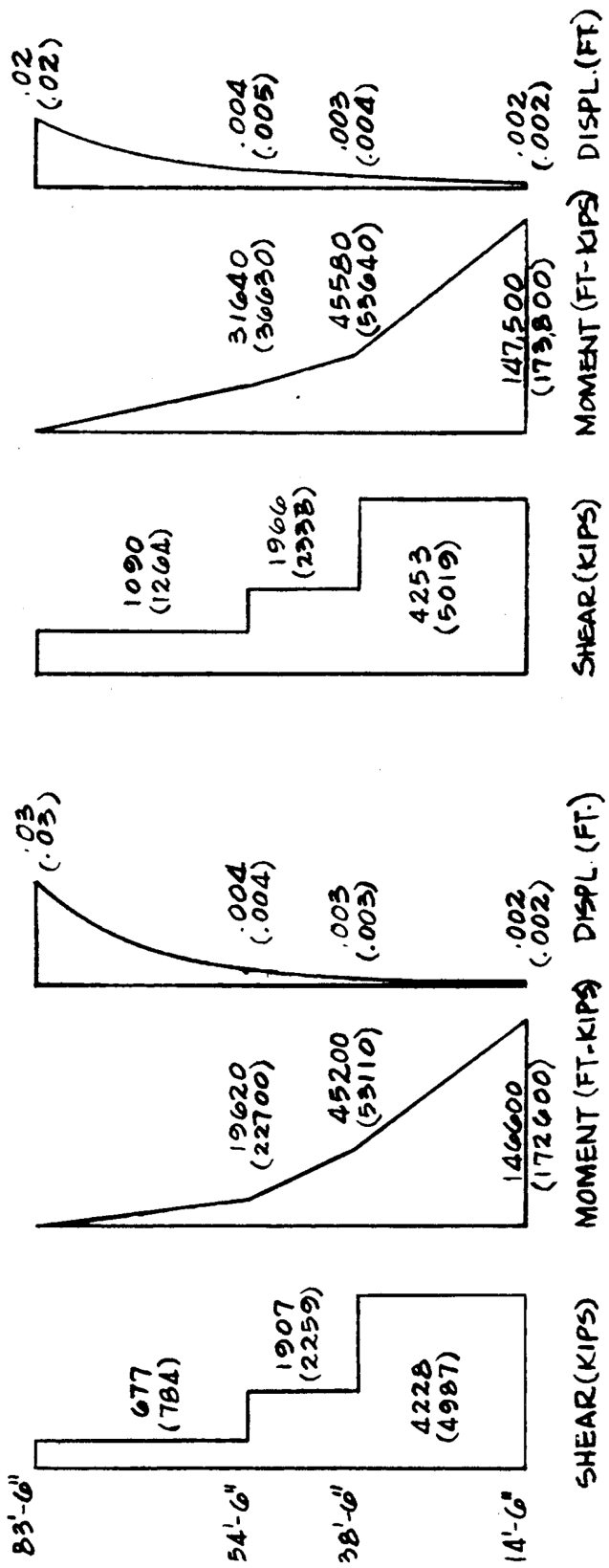


FIGURE 5.8–49 WAREHOUSE BLDG. - OBE (DBE)

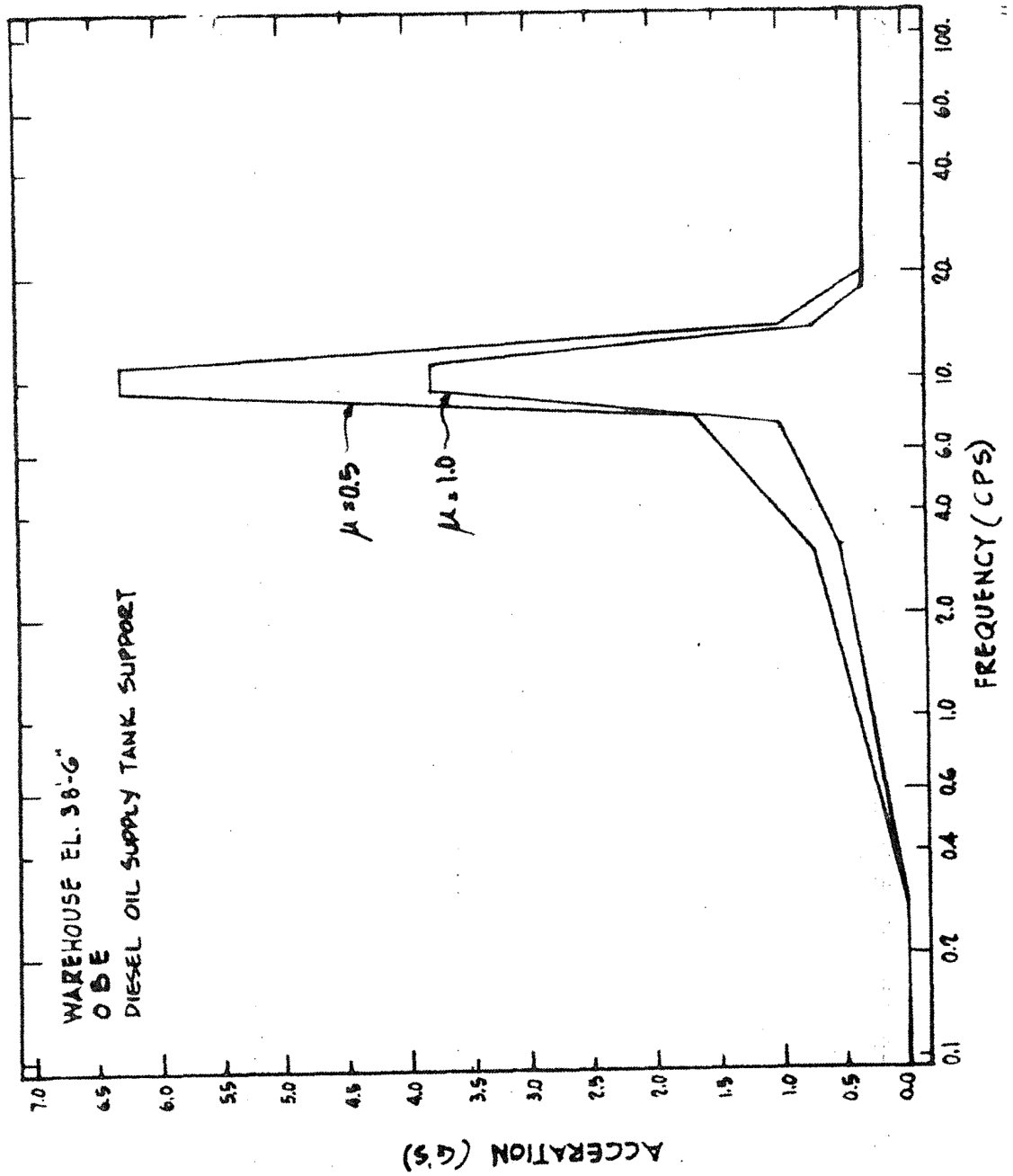


FIGURE 5.8–50 WAREHOUSE ELEVATION 38 FEET 6 INCHES

FIGURE 5.8–51 TURBINE BUILDING MODEL

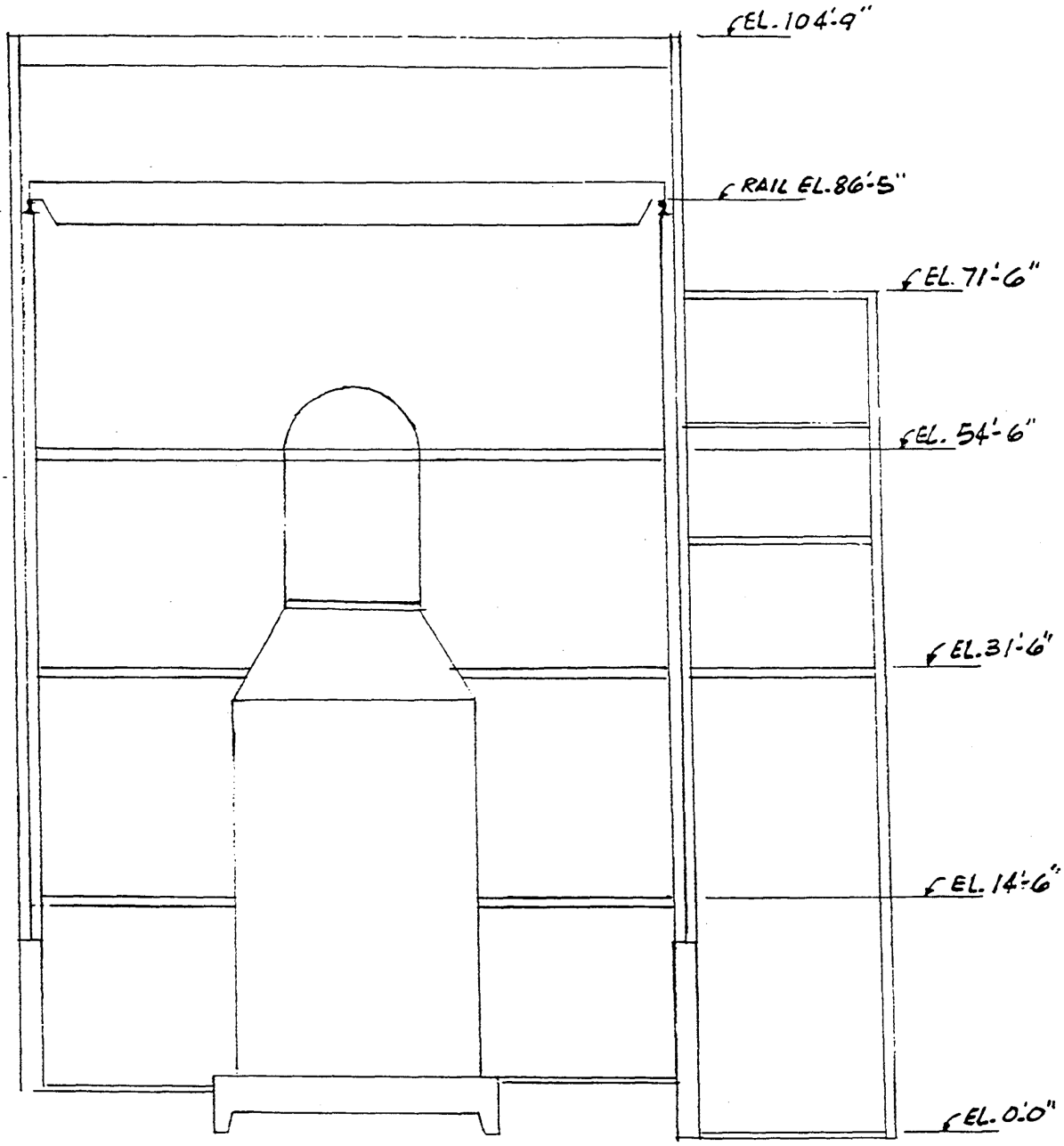


FIGURE 5.8–52 MODE SHAPES & FREQUENCIES TURBINE BLDG. - NORTH-SOUTH

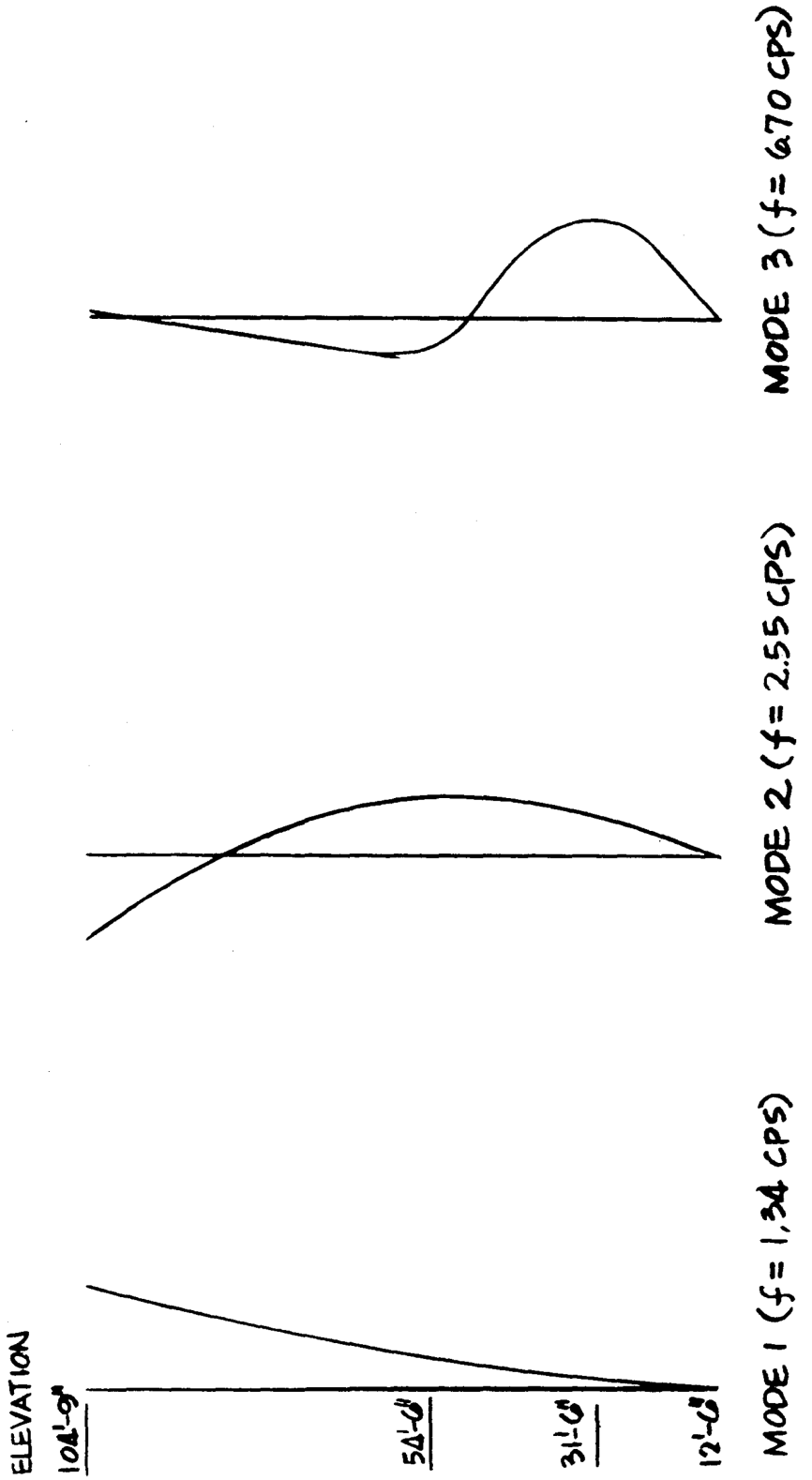


FIGURE 5.8–53 TURBINE BLDG. OBE (DBE) EAST-WEST

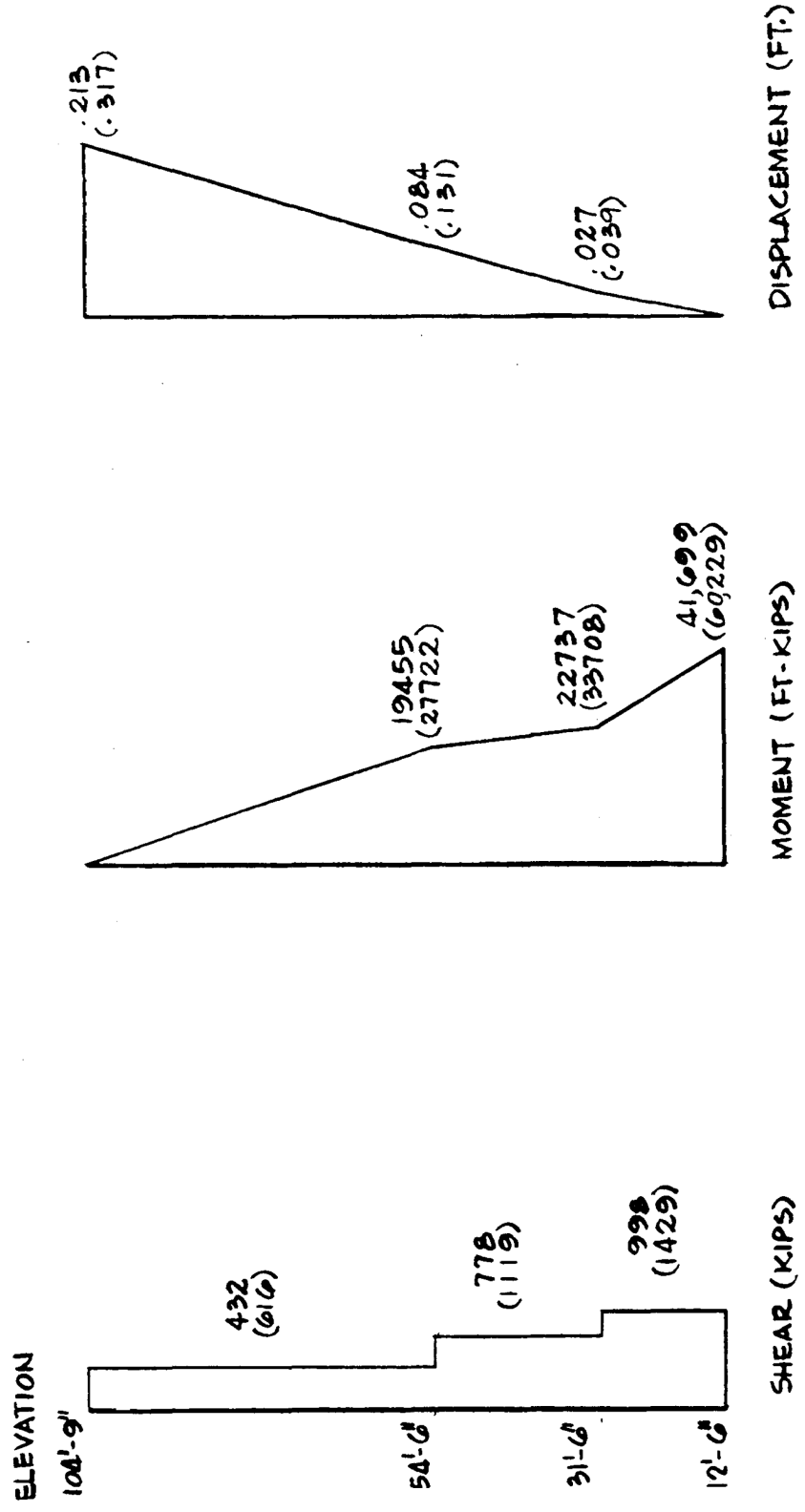


FIGURE 5.8–54 MODE SHAPES & FREQUENCIES TURBINE BLDG. - EAST-WEST

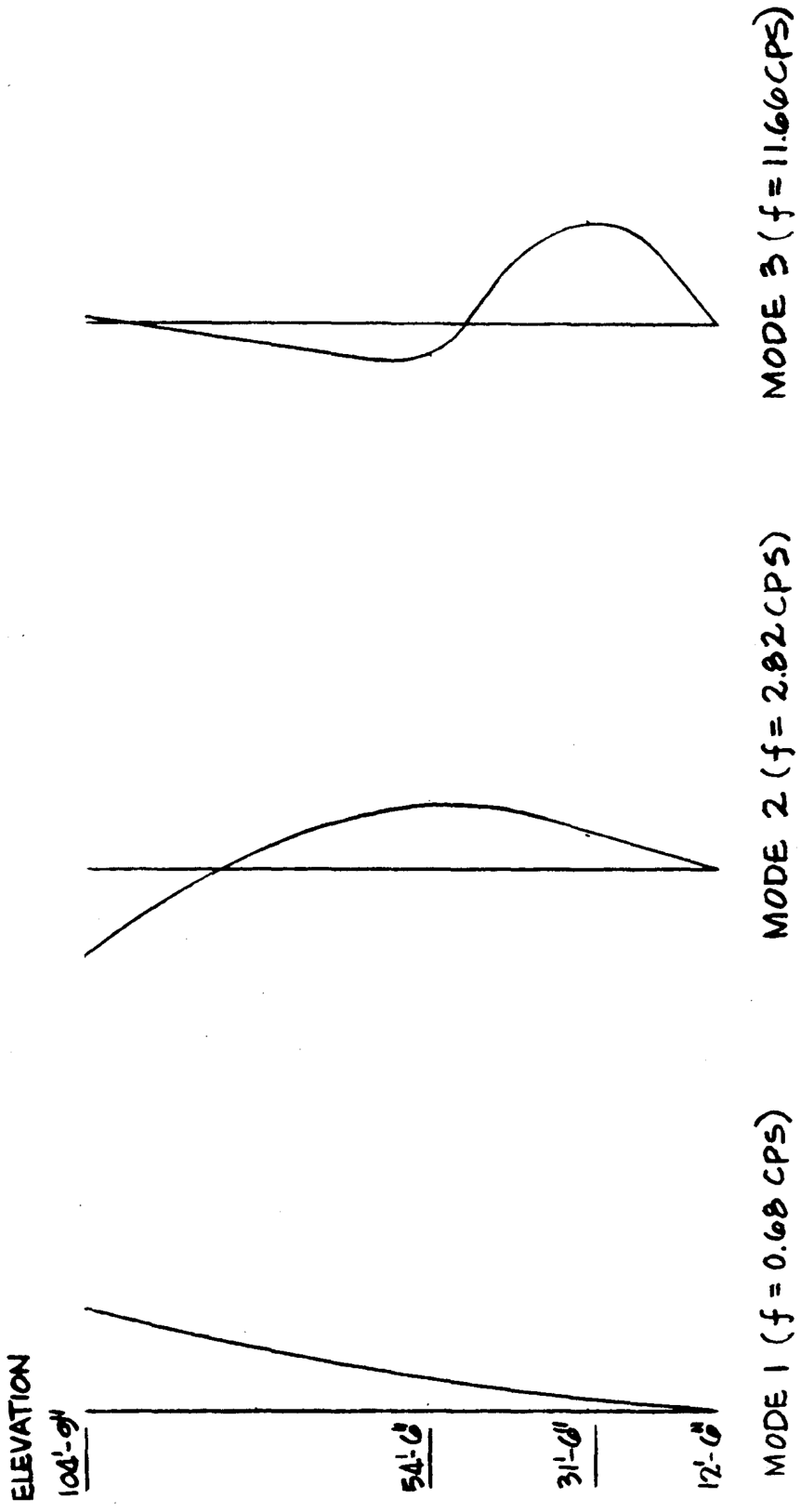
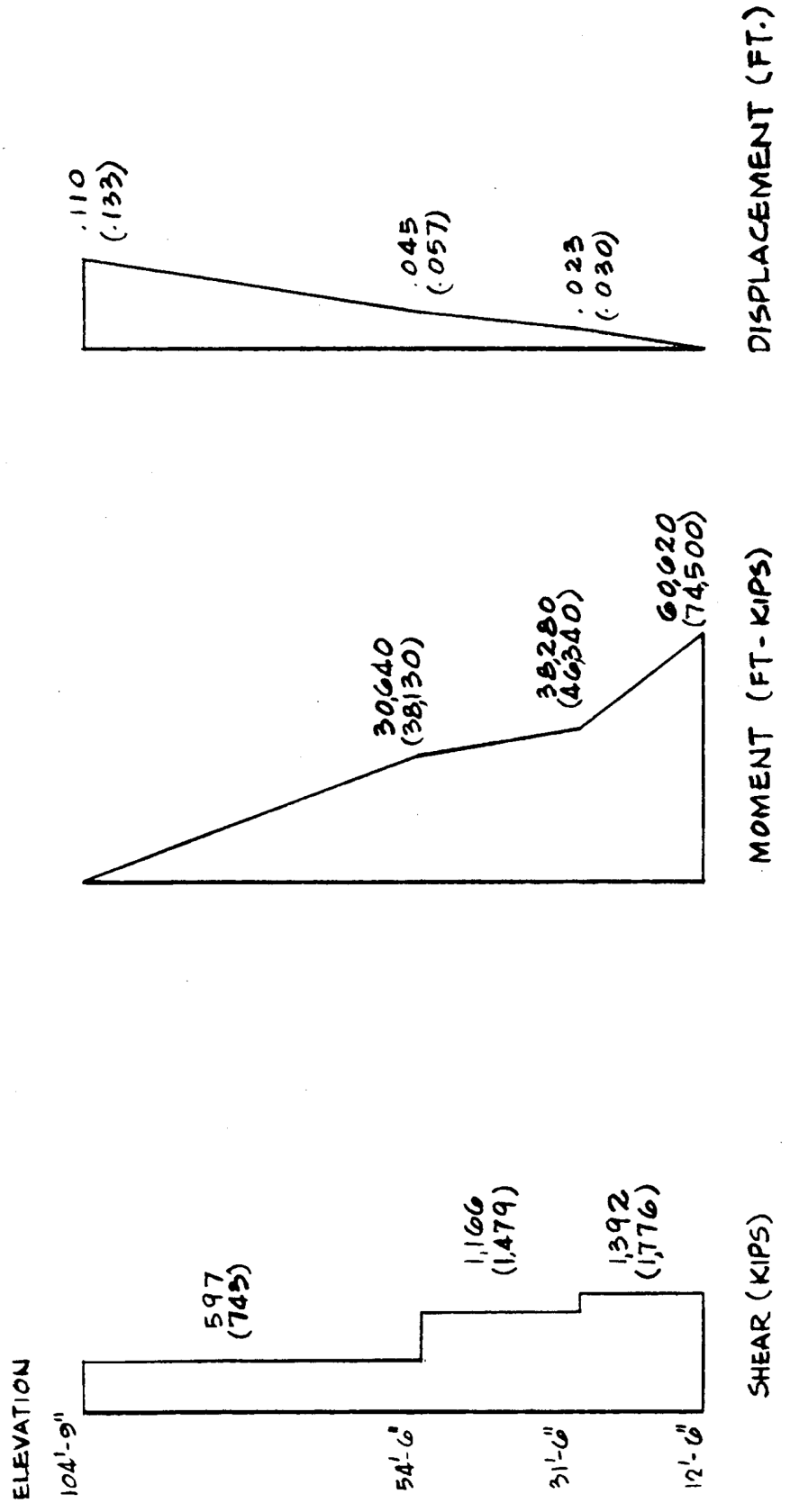


FIGURE 5.8–55 TURBINE BLDG. OBE (DBE) NORTH-SOUTH



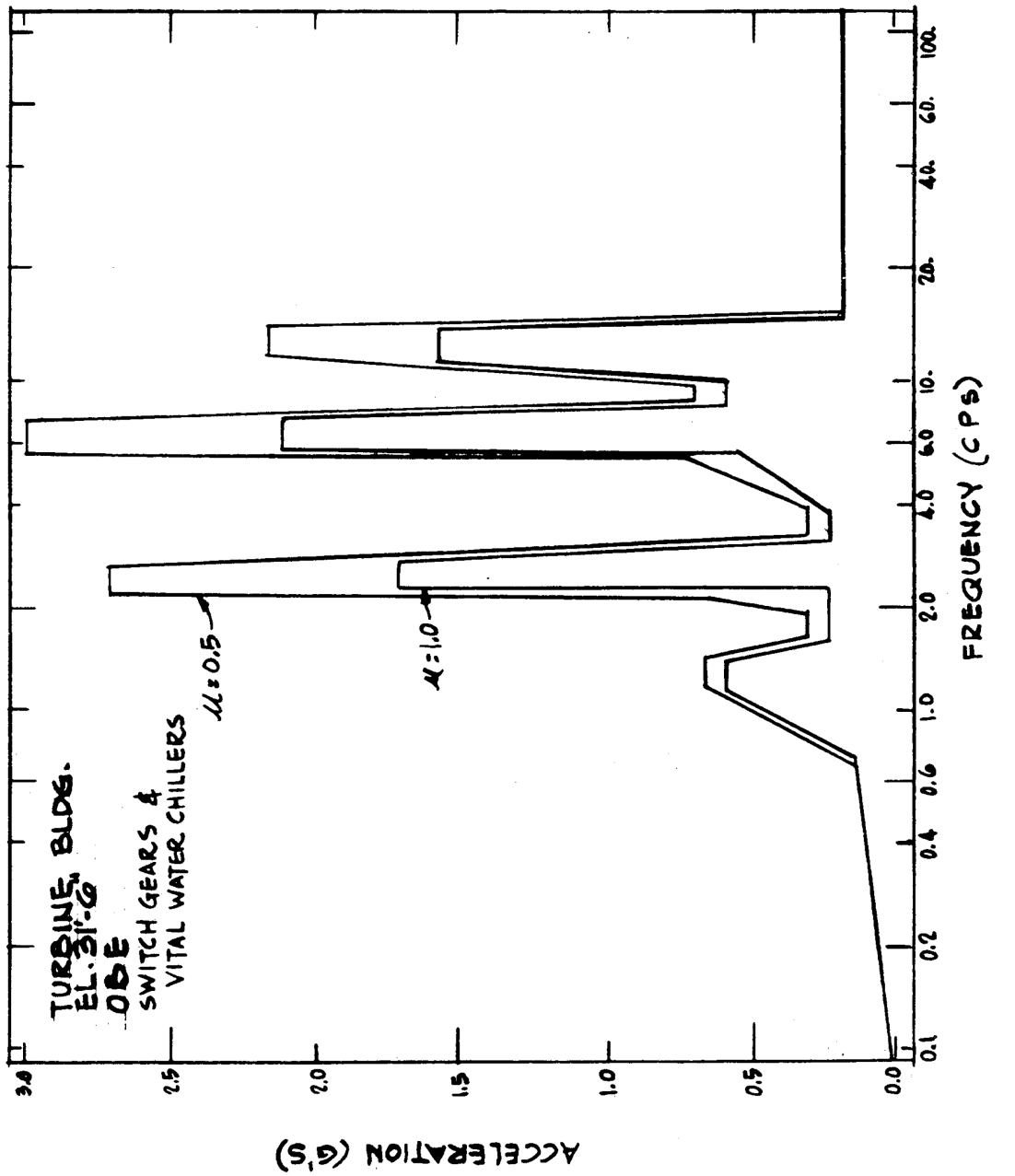


FIGURE 5.8–56 TURBINE BUILDING ELEVATION 31 FEET 6 INCHES

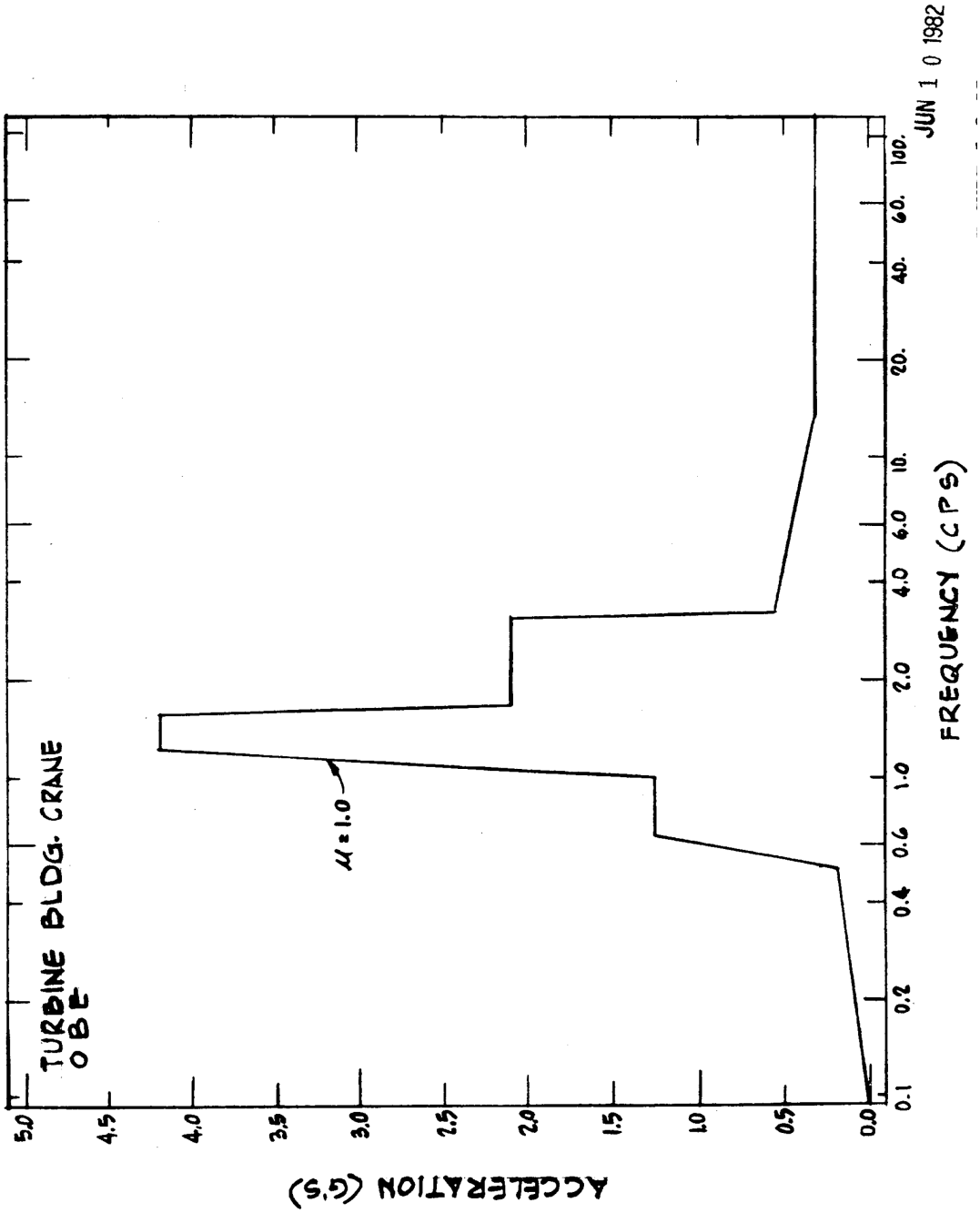


FIGURE 5.8–57 TURBINE BUILDING CRANE - OBE

FIGURE 5.8–58 INTAKE STRUCTURE MODEL

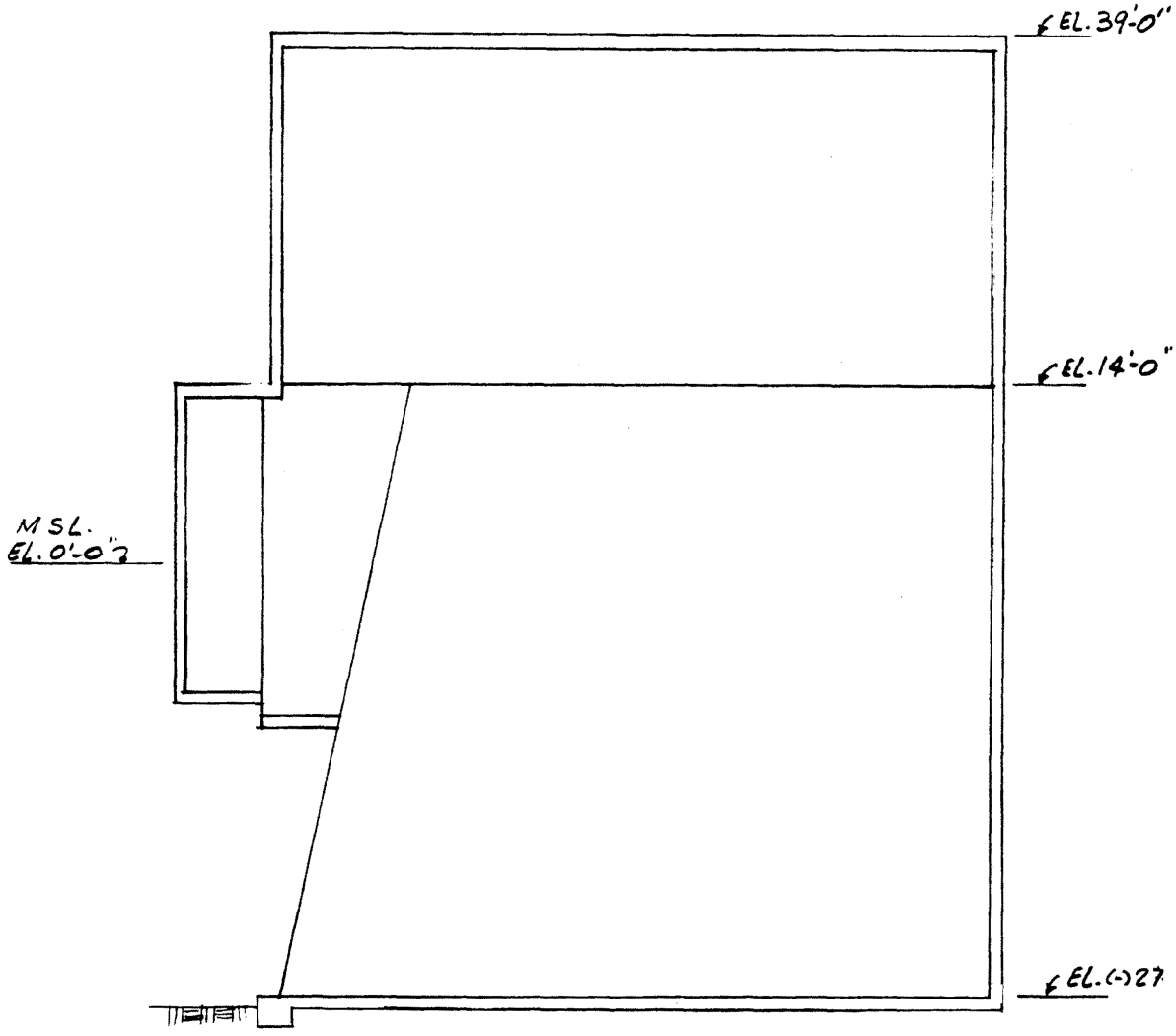


FIGURE 5.8–59 INTAKE STRUCTURE OBE (DBE)

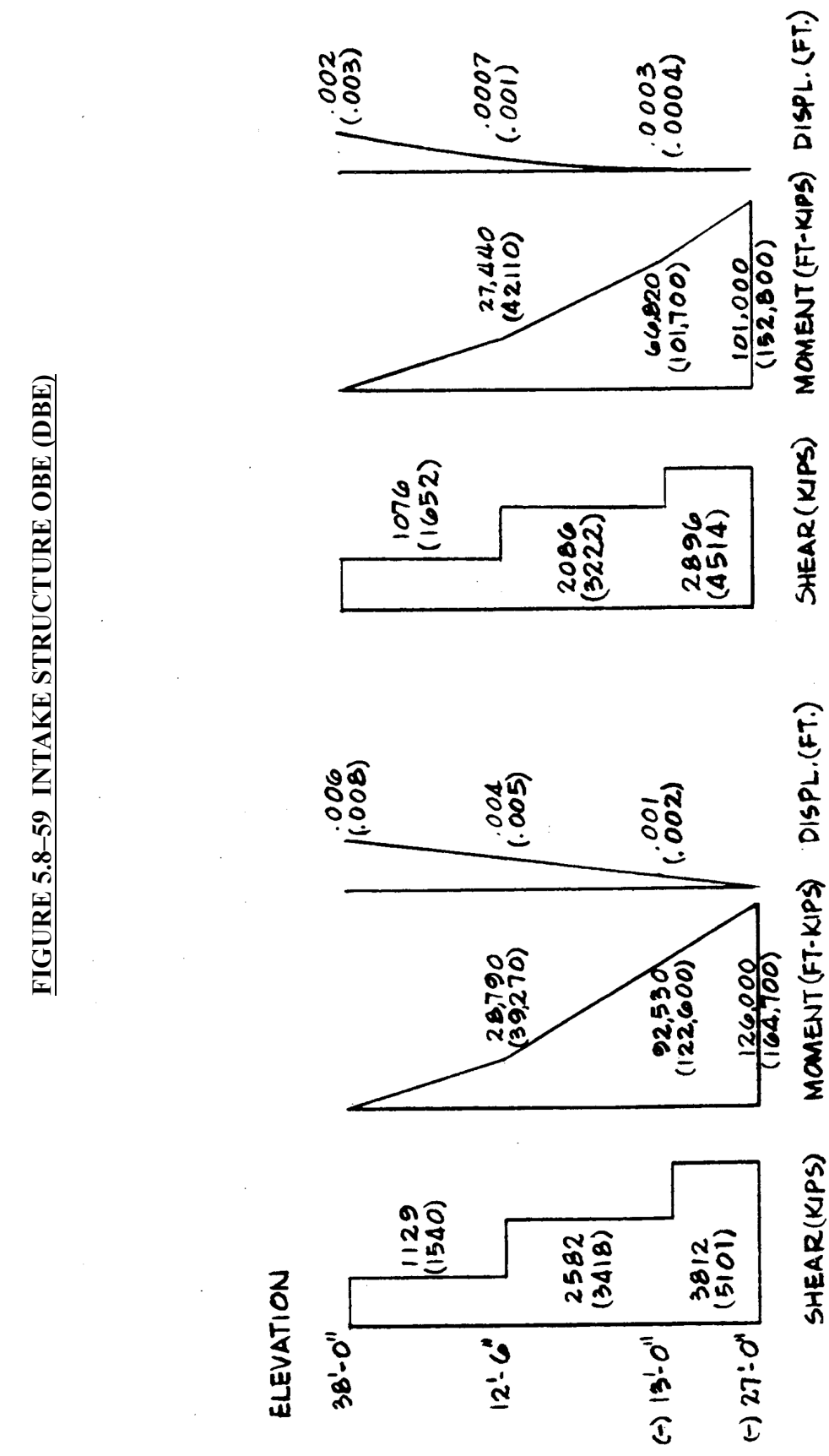


FIGURE 5.8–60 MODE SHAPES & FREQUENCIES INTAKE STRUCTURE

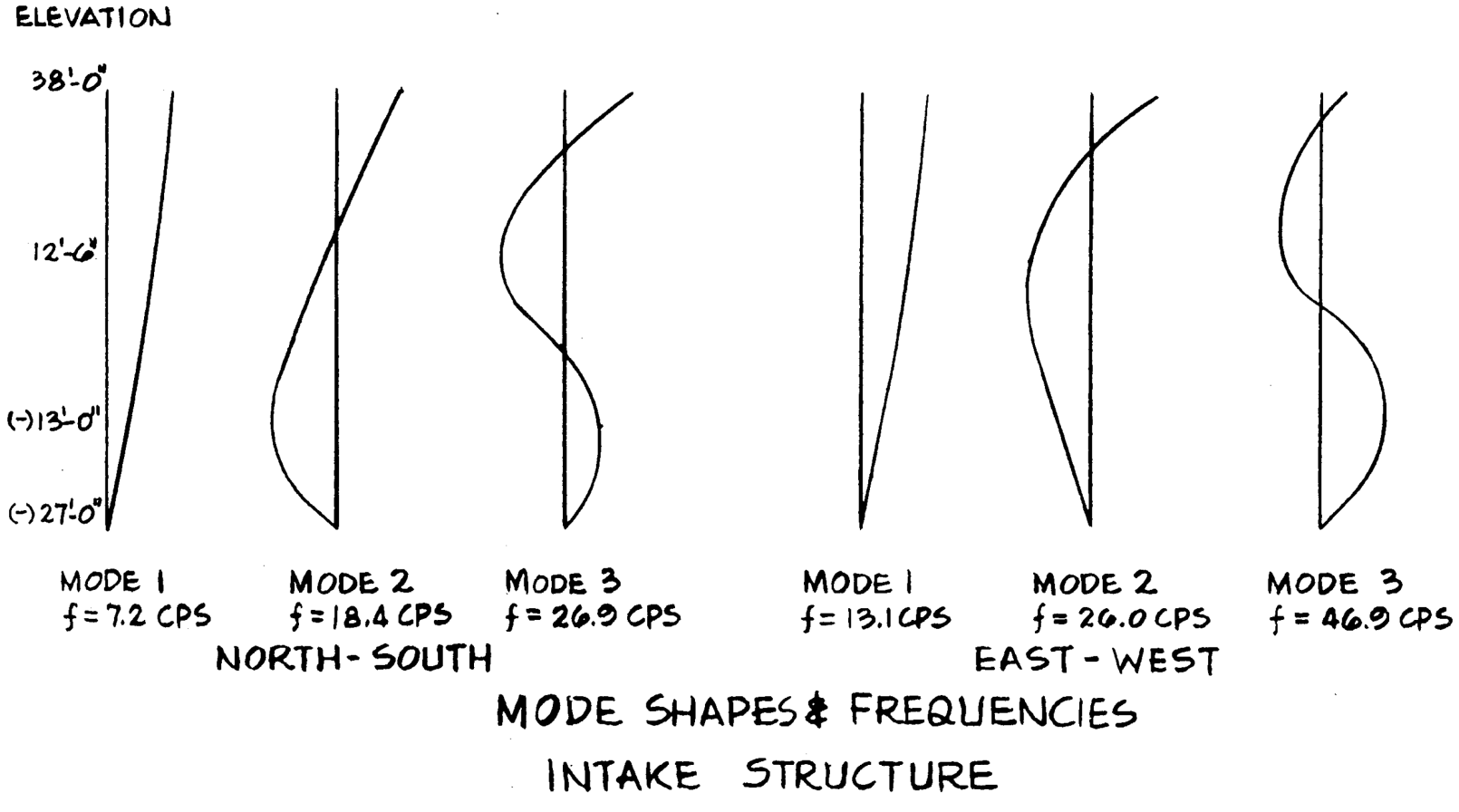


FIGURE 5.8–61 INTAKE STRUCTURE ELEVATION 14 FEET 0 INCHES OBE

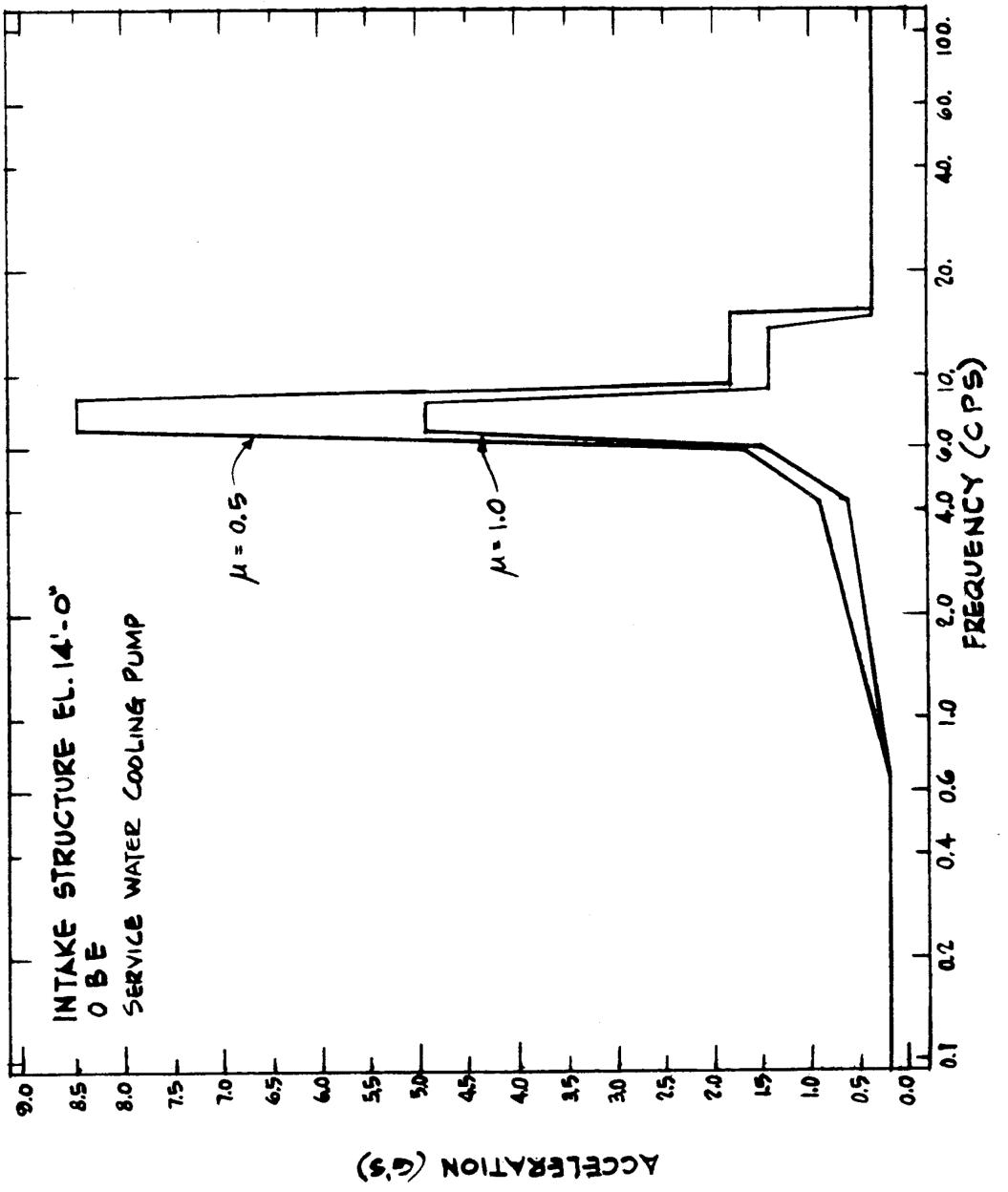


FIGURE 5.8-61

5.9 CONSTRUCTION PRACTICE AND QUALITY ASSURANCE

5.9.1 APPLICABLE CONSTRUCTION CODES

The following codes of practice are used to establish standards for construction procedures:

ACI 214-1	Recommended Practice for Evaluation of Compression Test Results of Field Concrete (ACI 214-65)
ACI 301-1	Specification for Structural Concrete for Buildings (ACI 301-66)
ACI 306-1	Recommended Practice for Cold Weather Concreting (AC 306-66)
ACI 315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI 318-1	Building Code Requirements for Reinforced Concrete (ACI 318-63)
ACI 347-1	Recommended Practice for Concrete Formwork (ACI 347-68)
ACI 305-1	Recommended Practice for Hot Weather concreting (ACI 605-59)
ACI 211-1	Recommended Practice for Selecting Proportions for Concrete (ACI6B-54)
ACI 304-1	Recommended Practice for Measuring, Mixing and Placing Concrete (ACI 614-59)
ACS	Manual of Concrete Inspection
PCI	Inspection Manual
AISC	Manual of Steel Construction
AWS	Code for Welding in Building Construction (D1.0-69)
AWS	Specifications for Welded Highway and Railroad Bridges (D2.0-69)
ASME	Boiler and Pressure Vessel Code, Section VIII, Part UW – Requirements for Unfired Pressure Vessels. Fabricated by Welding.
CRD	U.S. Army Corps of Engineers Waterways Experiment Station

Dimensional tolerances for construction, unless otherwise stated in design drawings, are in compliance with the ACI 301-66 and ACI 318-63 for placing reinforcing bars and concrete, and with the AISC Code of Standard Practice for erection of steel.

5.9.2 QUALITY ASSURANCE PROGRAM

A Quality Assurance Program has been developed and implemented to assure conformance to regulatory requirements and accepted industry standards. This is generally explained in FSAR Section 12.8.

5.9.3 CONSTRUCTION MATERIALS INSPECTION AND INSTALLATION

Basically, materials used in the construction of the structures are as follows:

a. Concrete

b. Reinforcing steel

c. Structural and miscellaneous steel

d. Prestressing steel tendons, anchorages and sheaths

e. Steel liner plate

f. Interior coatings

The basic specifications for material inspection and installation are discussed in the following sections.

5.9.3.1 Concrete

All concrete work is done in accordance with ACI 318-63, “Building Code Requirements for Reinforced Concrete,” and to ACI 301-66, “Specifications for Structural Concrete for Buildings,” except as otherwise stated herein or in the appropriate job specifications or design drawings.

The concrete is a dense, durable mixture of sound coarse aggregates, fine aggregates, cement, and water. In some areas, fly ash is substituted for portions of cement used in the concrete. Admixtures are added to improve the quality and workability of the plastic concrete during placement and to retard the set of the concrete. The sizes of aggregates, water-reducing additives, and slumps are selected to maintain low limits on shrinkage and creep.

The concrete is placed in a manner which assures sound concrete, free of cold joints and defects. Careful attention is given to the placing of concrete around tendon anchorage bearing plates in the containment so that high quality concrete is obtained at these critical locations.

5.9.3.1.1 Aggregates

Aggregates comply with ASTM C-33, “Specifications for Concrete Aggregates.” Acceptability of the aggregates is based on the User Tests listed in Table 5.9-1.

The initial tests in Table 5.9-1 are performed by the supplier. The user tests are performed by an independent laboratory for every 5000 tons delivered to the jobsite.

The following aggregate testing are performed at least once per shift, when concrete is being placed, and more frequently when required at the direction of the Contractor.

Samples are taken from weighing hoppers or the belt as directed by Contractor in the batch plant:

a. Two sand samples for gradation.

b. One coarse aggregate sample from each nominal size group for gradation.

a. Test for flat and elongated particles as directed by Contractor.

b. One sand organic test.

5.9.3.1.2 Cement

Cement is Type II, low-alkali cement as specified in “Standard Specification for Portland Cement,” ASTM C-150, and is tested to comply with the requirements of ASTM C-114. The inspection and testing of cement, in addition to the test performed by the cement manufacturers, is performed in accordance with Table 5.9-2.

The purpose of these tests is to ascertain conformance with ASTM C-150.

The initial tests are performed by the supplier. The user tests are performed by an independent laboratory for every 5000 cubic yards of concrete produced. During construction, the periodic tests are made to check storage environmental effects on cement characteristics. These tests are in addition to visual inspection of material storage procedures.

5.9.3.1.3 Fly Ash

Fly ash conforms to ASTM C-618 Class F, “Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete,” and is tested to comply with the requirements of ASTM C-311, “Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete.” All tests performed on the fly ash are listed in Table 5.9-3.

The user tests in Table 5.9-3 are performed by an independent laboratory for every 100 tons delivered to the job site. During construction, tests are made to check storage environmental effects on properties of fly ash. These tests are in addition to visual inspection of material storage procedures.

A typical chemical analysis from each source of fly ash is presented in Table 5.9-4. Fly ash from NUSCO Devon Plant is used in the concrete work, while the fly ash from the RG&E Company Russell plant is a standby and has not been used to date. The analyses on both fly ashes indicate

that sulfates and chlorines are not present and that sulfur trioxide (SO_3) content is well within the limit set by the ASTM C-618.

5.9.3.1.4 Water and Ice

Water and ice used in mixing concrete are free from injurious amount of acid, alkali, organic matter, and other deleterious substances as determined by AASHO-T-26. Water does not contain impurities in amounts that will cause either a change in the time of setting of Portland cement of more than 25 percent or a reduction in the compressive strength of mortar of more than 10 percent compared with results obtained with distilled water. In addition, mixing water (including ice for cooling) complies with the following criteria:

(*Criteria*)	Percent
Alkalinity in terms of calcium carbonate	0.025 maximum
Total organic solids	0.025 maximum
Total inorganic solids	0.05 maximum
Total chlorides	0.025 maximum

These tests are performed quarterly.

5.9.3.1.5 Admixtures

The selected water-reducing agent MBHC, manufactured by the Masters Builders Company, possesses a shrinkage reduction effect similar to the type prescribed by ASTM C-494, "Specifications for Chemical Admixtures for Concrete."

An air entraining agent, Vinsol Resin, manufactured by the Masters Builders Company, is added to the concrete mix to increase workability.

Admixtures containing chlorides are not used.

5.9.3.1.6 Concrete Mix Design

Concrete mixes are designed in accordance with ACI-211-1, "Recommended Practice for Selecting Proportions for Concrete," using materials qualified and accepted for this work. Only concrete mixes meeting the design requirements specified for the structures are used.

Mixes are tested in accordance with the applicable ASTM Specifications as indicated:

ASTM	Test
C-39	Compressive strength tests
C-143	Slump
C-192	Making and curing cylinder in laboratory
C-231	Air content
C-232	Bleeding

For the containment, concrete test cylinders are cast from the basic mix designed for the structure. The following properties were determined by Professor David Purtz at the University of California, Berkeley.

a. Compressive strength (ASTM C-39)

b. Thermal diffusivity (ASTM C-342 and CRD C-36)

c. Autogeneous shrinkage (ASTM C-342)

d. Thermal coefficient of expansion (ASTM C-342 and CRD C-124)

e. Modulus of elasticity and Poisson's ratio (ASTM C-469)

f. Uniaxial creep (ASTM C-512)

g. Tensile strength (ASTM C-496)

Concrete design compressive strength for the elements of the structures are defined in the respective sections under Construction Materials.

5.9.3.1.7 Concrete Production and Testing

The concrete batch plant is located on the site and operates in a fully automatic mode. The rated capacity of the plant is 136 cubic yards per hour. A full time inspector from an independent testing laboratory is assigned to the plant to continually monitor the concrete batching operation.

Concrete samples are taken from the mix as prescribed in ASTM C-172, “Sampling Fresh Concrete.” Cylinders for compression tests are prepared from these samples which are cured in accordance with ASTM C-31, “Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field.” Slump, air content, temperature, and unit weight are determined and recorded when the compression cylinders are cast.

Slump tests are performed in accordance with ASTM C-143, “Test for Slump of Portland Cement Concrete.” In addition to the performance of a slump test when compressive cylinders are cast, slump is measured at the batch plant for every 50 cubic yards of concrete mixed for delivery.

Air content tests are performed in accordance with ASTM C-231, “Test for Air Content of Freshly Mixed Concrete by the Pressure Method.”

Compressive strength tests are performed in accordance with ASTM C-39, “Test for Compressive Strength of Molded concrete Cylinders.” Evaluating of compressive strength tests is done in accordance with ACI 214, with the standard of control that which is required for “excellent” concrete.

Six cylinders, three sets of two each, are prepared for each placement of concrete as shown in the following tabulation:

Placing	Class I (cubic yards)	Class II (cubic yards)
Conventional, Plant	100	200
Conventional, Field	300	300
Pumping, Plant	100	200
Pumping, Field	100	300

Two cylinders are tested for compressive strength at each time internal of 7, 28, and 90 days, except that when correlation test data have been established for each design mix, test cylinders for the 90 day interval are disregarded with the exception of prestressed concrete.

To provide for accurate testing and concrete production, the equipment is calibrated using the following schedule:

Equipment Calibration Schedule

Testing Agency

Items	Calibration Interval
Platform Scales	6 months
Laboratory Scales	3 months
Meters, Air	3 months

Testing Agency

Items	Calibration Interval
Cylinder Compression Machine	12 months
Thermometers	6 months
Slump Cone	Examine for wear and replace as necessary

Concrete Supplier

Items	Calibration Interval
Pen Recorders	Daily
Batching System, Low-Limit and High-Limit Setpoints for Cement Aggregate and Flyash	1 month

5.9.3.2 Reinforcing Steel

5.9.3.2.1 Reinforcing Steel Materials

All reinforcing steel, except column ties and beam stirrups for some areas of the structures, is deformed billet steel bars conforming to ASTM A-615, Grade 60. Spiral reinforcing steel conforms to ASTM A-82.

Mill test reports are obtained from the reinforcing steel supplier for each heat of steel to ensure that the physical and chemical properties of the steel are in compliance with the applicable ASTM specifications. User tests to determine the strength and ductility of the reinforcing steel are used to supplement the standard mill tests. These are witnessed by an independent testing company. Procedures for obtaining samples for the reinforcing steel user test are defined in Section 5.9.3.2.2.

Procedures for splicing reinforcing bars using the Cadweld process is defined in Section 5.9.3.2.3.

5.9.3.2.2 Reinforcing Steel User Test Sampling

5.9.3.2.2.1 Procedures

The following procedure is used for sampling and testing reinforcing bars at the supplier's plant at Steelton, Pennsylvania. All tests are performed in accordance with ASTM A-615 except as otherwise noted herein. Only full size specimens are utilized in the user test sampling.

a. Upon receipt of a notice from the supplier that the stock reinforcing bars are available for sampling and testing, an inspector from an independent testing laboratory is sent to the plant to select random specimens and witness testing.

b. The inspector obtains the information regarding the heat numbers and bar sizes which require sampling and testing, and verifies the information by the rolling marks on the bars, the sources, grades and sizes of bars being supplied.

c. Specimens for tensile strength and cold bent tests are taken from the bars which have been selected at random by the inspector, in accordance with Sections 11.1 and 11.2 of ASTM A-615. The test specimens are then properly tagged and delivered to the supplier's metallurgical laboratory.

d. After test specimens are taken, the stockpiled materials are marked by tags and held until the results of the user tests are available.

e. When the results of the test specimens verify that the requirements of the specification have been met, the stockpiled materials are released for fabrication.

f. If during the tests, a specimen develops flaws, it may be discarded and another substituted.

g. If the test results of a specimen indicate that the tensile strength does not meet the minimum requirements, a second full size specimen from the same heat is taken and tested. If the second set of test results meets the minimum strength requirements, the tensile strengths from these two tests and the mill test are combined and averaged. If the average result meets the minimum strength requirements, the heat is accepted.

   If the averaged results of all three tests do not meet the minimum strength requirements, the heat is rejected.

h. Prior to shipping the fabricated materials to the site, the inspector returns to the plant for verification of the materials. Certified copies of the mill test reports, showing the physical properties and the ladle chemical analysis, are reviewed for compliance with the applicable ASTM Specifications.

The mill test reports, which accompany the materials to the site, and the shipping notices are initialed, dated and stamped by the inspector.

5.9.3.2.2.2 Deviation from Safety Guide 15

The procedure deviates from the Safety Guide 15 in that it allows a second tensile test to be conducted in the event that the first tensile test does not meet the minimum requirements. However, no tensile test failure was encountered during the user test and, consequently, the second tensile test was never required.

5.9.3.2.3 Splicing Reinforcing Bars

Reinforcing bars Number 11 and smaller are generally lap-spliced in accordance with ACI 318-63 except in congested areas where some bars are Cadweld-spliced. Reinforcing bar greater than Number 11 are Cadweld-spliced exclusively.

Reinforcing bars are not spliced by welding.

5.9.3.2.3.1 Scope

These procedures cover the mechanical splicing of deformed reinforcing bars using the Cadweld process, for full tension loadings.

All splices are made in accordance with the manufacturer's instructions as described in Erico Products Bulletin RBIS-10M269 “Cadweld Rebar Splicing,” except as modified hereinafter.

A manufacturer's representative, experienced in Cadweld splicing of reinforcing bars, is present at the site at the outset of the work to demonstrate the equipment and techniques used for making quality splices. He is also present for the first 50 production splices to observe and verify that the equipment is being used correctly and that quality splices are being obtained.

5.9.3.2.3.2 Materials

Cadweld T-Series materials are used for full tension, reinforcing-to-reinforcing splices. These are used for full tension, reinforcing to structural plates or shapes, where indicated on design drawings. C-Series splice materials are not used.

5.9.3.2.3.3 Qualifications of Operators

Prior to production splicing reinforcing bars, each operator or crew, including the foreman or supervisor for that crew, prepares and tests a splice for each of the positions used in the production work. These splices are made and tested in strict accordance with these procedures, using ASTM A-615, Grade 60 and the largest size bar spliced during production work. To qualify, the completed splices shall meet the acceptance standards of Section 5.9.3.2.3.7 for workmanship, visual quality and minimum tensile strength. A list of qualified operators and their qualification test results are maintained at the jobsite.

5.9.3.2.3.4 Care and Handling of Splice Kits and Equipment

a. The splice sleeves, cartridges, asbestos wicking, ceramic inserts and graphite parts are stored in a clean, dry, temperature controlled area with adequate protection from the elements to prevent absorption of moisture.

b. Splice sleeves are wrapped in a special rust inhibiting paper and are not unwrapped until ready to be used in the joining procedure.

c. Each splice sleeve is visually examined immediately prior to its being used to ensure that there is no rust or other foreign material on the inside surface.

d. The graphite pouring basins and crucibles are preheated with an oxyacetylene or propane torch to burn off moisture at the beginning of each shift when the molds are cold, when a new mold is put to use, or when the mold temperature is below ambient.

e. All graphite parts (except crucible covers) are cleaned with a whisk broom, rag, coarse brush or rolled up newspaper before reusing. A wire brush is not used on graphite parts.

5.9.3.2.3.5 Reinforcing Bar End Preparation

a. The bar ends to be spliced must be in good condition with full size, undamaged deformations.

b. The reinforcing bar deformations, except longitudinal ribs, which become engaged in the Cadweld splice are not ground, flame cut or altered in any way. Oversize longitudinal ribs are ground down to match the diameter of the bar deformations, but are not ground to a diameter less than the bar deformations.

c. For a minimum distance of two inches beyond where the ends of the sleeve are located, the bar ends are heated with an oxyacetylene “rosebud” torch to 300°F minimum to remove all moisture and burn off any organic foreign material.

d. After the heating operation, the bar ends are thoroughly cleaned by power brushing or sandblasting to remove all loose mill scale, concrete, dirt and other foreign material not removed by burning.

e. A reference line is painted 12 inches from the end of each bar that is to be spliced to confirm that the bar ends are properly centered in the sleeve.

f. Immediately before the splice sleeve is placed in final position, the previously cleaned bar ends are again surface preheated to 200° to 300°F minimum with an oxyacetylene rosebud torch to ensure complete removal of moisture.

g. Special attention is given to maintaining the alignment of sleeve and guide tube to ensure a proper fill.

h. When the temperature is below freezing or the relative humidity is above 65 percent, the splice sleeves are externally preheated with an oxyacetylene or propane torch after all materials and equipment are in position. The Cadweld operation is suspended during any form of precipitation.

i. A hairpin piece of soft twisted wire may be inserted at the top of the horizontal splices between the bar and the sleeve. This provides an escape route for the gases generated during the casting of the filler material.

j. The packing material at the ends of the horizontal splices and at the top of the vertical splices are not hard packed. Although the material is held firmly in place, it has to be loose enough to allow the escape of gases.

5.9.3.2.3.6 Splice Tensile Testing

Selected splices are tensile tested for each position, bar size and grade of bars. The test splices consist of sister splices, each three feet long, spliced in sequence in position and adjacent to the production works. Production splices, cut from in-place reinforcement material, are included. The following schedule is used:

a. Test splices, reinforcing-to-reinforcing:

1. Production splice from the first 10 production splices.

2. One production and three sister splices from the next 90 production splices.

3. One production and two sister splices from the next and subsequent units of 100 splices.

b. The splices to be used in reinforcing structural steel shapes and plates consist of four sister splices from each unit of 100 production splices or fraction thereof.

5.9.3.2.3.7 Splice Acceptance Standards

a. All completed splices are visually inspected at both ends of the splice sleeves and at the tap hold in the center of the splice sleeves.

b. Sound, nonporous filler metal should be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed one-quarter inch from the ends of the sleeve due to the packing material, and is not considered a poor fill.

c. Splices that contain slag or porous metal in the riser, tap hole or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble, present below the riser, is not considered demineralized and should be distinguishable from the general porosity described in Item b.

d. There must be evidence of filler material between the sleeve and the bar for the full 360 degrees; however, the splice sleeves need not be exactly concentric or axially aligned with the bars.

e. Both horizontal and vertical Cadweld splices may contain voids at either or both ends of the Cadweld splice sleeve. The allowable limits for end voids are as shown on Table 5.9-5. The area of the voids is assumed to be the maximum depth of the wire probe minus 3/16 inch multiplied by the width at the inside surface of the sleeve.

f. The average tensile strength of the Cadweld spliced joints are equal to or greater than the minimum tensile strength for the particular grade of reinforcing steel as specified in ASTM A-615. The minimum acceptable tensile strength of any splice is 125 percent of the specified minimum yield strength for the particular bar size.

5.9.3.2.3.8 Deviation from the Safety Guide 10

This procedure deviates from the Safety Guide 10 in that only one qualification splice is required for each crew member, including the foreman. The Safety Guide recommends that two qualification splices be required.

On the first 245 sister and production splices tested, only one failed below the specified minimum strength of 125 percent of the yield strength of the reinforcing bar. This failure was attributed to a niche in the bar which was used for identification purposes, and was not caused by the failure of the splice itself.

The exceptional performance of the test splices more than justifies the omission of a second qualification splice.

5.9.3.3 Post-Tensioning System

5.9.3.3.1 Tendons

The tendons are composed of stabilized, low relaxation wires of one-quarter inch diameter with a tensile strength of 240,000 psi in accordance with ASTM A-421. The pertinent features of the tendons are as follows:

Number of wires	186
Ultimate tensile capacity (kips)	2190
Design tensile capacity (kips)	1315

End anchorages Buttonheads

Sampling and testing of the tendon material conform to ASTM A-421. The following procedure is used.

a. One buttonhead test on each end of each reel of wire to establish the suitability and acceptance of the wire for buttonheading.

b. One buttonhead rupture test from each reel of wire.

c. Random samples test of each lot of wire in accordance with ASTM-421. With each sample of wire tested, a certificate was submitted stating the manufacturer's minimum guaranteed ultimate tensile strength of the wire sampled. Stress-strain curves were plotted for each of these tests and the yield and tensile strength of the wire was verified.

5.9.3.3.2 Anchorages

The basic performance requirements for the end anchors of the tendons are stated qualitatively by the Seismic Committee of the Prestressed Concrete Institute and published in their Journal of June 1966 as follows:

> "All anchors of unbonded tendons should develop at least 100 percent of the guaranteed ultimate strength of the tendon. The anchorage gripping should function in such a way that no harmful notching would occur on the tendon. Any such anchorage system used in earthquake areas must be capable of maintaining the prestressing force under sustained and fluctuating loads and the effect of shock. Anchors should also possess adequate reserve strength to withstand any overstress to which they may be subjected during the most severe probable earthquake. Particular care should be directed to accurate positioning and alignment of end anchors."

The end anchors used are capable of developing 100 percent of the minimum tensile strength of the prestressing steel as defined by ASTM A-421.

The end anchors are capable of maintaining integrity for 500 cycles of loads corresponding to an average axial stress variation between 0.7 and 0.75 f'_s at a repetition rate of one cycle in 0.1 second. This requirement sets minimum acceptable limits on fatigue effects due to notching by the end anchor and tendon performance in response to earthquake loads.

The number of cycles was set by increasing to 500 from the 100 predicted. The stress variation was increased from a conservatively predicted 0.6 to 0.64 f'_s to the 0.7 to 0.75 f'_s. Further, the number of cycles caused by the earthquake loads was predicted as only 30 of the total of 100 resulting from using all the strong ground motions which exceed one/half of the peak ground motion of the earthquake.

The stress variations due to the earthquake motion alone were predicted as being 10 percent of the total of the predicted stress variations of 0.04 f'_s. The predicted 0.04 f'_s stress variation, in turn, resulted from the combinations of earthquake, wind, and incident loadings. Analyses made during the investigation included consideration of tendon excitation, both parallel and perpendicular to the tendon axis.

The anchorage assemblies, including the bearing plates, are capable of transmitting the ultimate loads of the tendons into the structure without brittle fracture at an anticipated lowest service temperature of -30°F.

The anchorage assemblies used are capable to prelude brittle fracture at a design temperature of -50°F.

5.9.3.3.3 Sheathing

Sheaths for the tendons are classified as concrete forms and are not subjected to any standard codes. They provide a void in the concrete wherein the tendons were installed, stressed and greased after the concrete was placed.

The sheaths are made from 22 gauge, galvanized ferrous metal, and have an internal diameter of five inches clear of corrugations. Couplers are provided at all field splices and sealed by tape.

After sheathing installation, and prior to concrete placement, the sheathing is surveyed to assure accurate alignment. An inspection is also performed to ascertain that all sheaths are continuous and unblocked by obstructions.

Before installation of the tendons, the sheathing is carefully cleaned to remove all water and debris.

Vent tubing and temporary valves were provided to permit drainage at all low points.

Splash caps at the ends of all sheaths, to prevent concrete and laitance from entering into the sheaths during construction, were provided.

5.9.3.3.4 Corrosion Protection

Suitable atmospheric corrosion protection was maintained for the tendons from the point of manufacture to the installed locations. The atmospheric corrosion protection provided assurance that the tendon integrity was not impaired due to exposure to the environment.

Prior to shipment, they were all coated with a thin film of petroleum that contained rust inhibitors. After the tendons were installed in the sheaths and stressed, the interior of the sheathing was pumped full of a modified, thixotropic, refined petroleum oil-based product to provide corrosion protection. The tendons and end anchors were also surrounded with the corrosion-protection material which was encapsulated in the sheathing and gasketed end caps that were sealed against the bearing plates. When the sheaths and end caps are filled, the corrosion-protection material

displaces air and water vapor before thickening to a soft gel. Once the filler material had cooled, contracted, and gelled, the vertical tendons were topped-off by injecting additional filler material through the upper grease cap fittings.

As a result of the Millstone Unit Number 2 tendon surveillance program, sixteen horizontal tendons have been identified as subject to ground water intrusion. To prevent ground water intrusion, the corrosion protection material is continuously supplied to the subject tendons at a pressure slightly above hydrostatic pressure of the ground water. The tendons so pressurized are horizontal tendons 12H01 through 12H06, 12H08 through 12H10, 31H01 through 31H04, 31H01, 32H02, and 32H03.

Testing of the permanent corrosion-protection material indicates that there were no significant amounts of chlorides, sulfides, or nitrates present. However, to further verify the chemical composition of the filler material, test samples are taken from each shipment with at least one sample per factory batch. The samples were analyzed as follows:

a. Water-soluble chlorides (C1) are determined in accordance with ASTM D512-67 with a limit of accuracy of 0.5 ppm.

b. Water-soluble nitrates (NO_3) are determined by the Water and Sewage Analysis Procedure of the Hach Chemical Company, Ames, Iowa.

c. Water-soluble sulfides (S) are determined in accordance with American Public Health Association Standards (APHA) with a limit of accuracy of 1 ppm. The APHA Standards methods may be modified to use standard reagents and procedures such as those available from Hach Chemical Company.

No significant traces of the impurities are allowed. The chemical composition of the filler material, being about 98 percent petroleum jelly, indicates that it possesses the normal stability of linear hydrocarbons for the site temperature ranges.

NOTE: For description of water intrusion into the tendon gallery during construction and methods of repair, see Appendix 5.F.

5.9.3.4 Structural and Miscellaneous Steels

All structural and miscellaneous steels conform to the following ASTM specifications:

Rolled shapes, plates, tubing and bars	A-36
Crane rails	A-1
High strength bolts	A-325 or A-490
Anchor bolts (nonwelding)	A-575, Grade 1020
Stainless steel	A-240, Type 304

Mill test reports are obtained for all materials used with the exceptions of hand rails, toe plates, kick plates, stairs and ladders.

Detailing, fabrication, and erection of the structural and miscellaneous steels are in accordance with AISC Manual of Steel Construction.

Welding is accomplished in accordance with AWS D1.0, “Code for Welding in Building Construction,” and where applicable, AWS D2.0, “Specifications for Welded Highway and Railroad Bridges.”

Quality control procedures for field welding are defined in Section 5.9.4.

5.9.3.5 Steel Liner Plate and Penetration Sleeves

5.9.3.5.1 General

The containment is lined with a one-quarter inch welded steel plate to ensure leak tightness. The design, construction, inspection, and testing of the liner plate are not covered by an recognized code or specification, since it is not a pressure vessel and serves only as a leak-tight membrane. However, components of the liner which must resist the full containment design pressure, such as the penetration sleeves, are designed, fabricated, constructed, and tested to meet the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, 1968 Edition through the summer 1969 addenda of the ASME Code, except where otherwise noted herein.

The liner is designed to function only as a leak-tight membrane. It is not designed to serve as a structural element to resist the tensile loads from an internally applied pressure such as might result from a loss-of-coolant incident. Structural integrity of the containment is maintained by the pre-stressed, post-tensioned concrete. Since the principal stresses of the liner due to thermal expansion are in compression, and no significant tensile stresses are expected from the internal pressure loading, special nil ductility transition temperature requirements are not applied to the liner plate materials. However, all materials for the liner components which must resist tensile stresses resulting from internally applied pressure, such as the penetration sleeves are impact tested in accordance with the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, 1968 Edition through the summer 1969 addenda of the ASME Code.

Materials used in the construction of the steel liner plate and penetration sleeves are defined in Section 5.2.3.1.

5.9.3.5.2 Fabrication and Erection

A basic requirement for the fabrication and erection of the steel liner plate is that all welding procedures and welding operators be qualified by tests as specified in Section IX of the 1968 ASME Code.

Penetration sleeves are shop fabricated in accordance with the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, 1968 Edition through the summer 1969 addenda of the ASME

Code, including welding and NDT procedures and third party inspection, but excluding pressure testing. A modified code data sheet is prepared for each sleeve. The sleeves are not code stamped because they are not included as part of the vessel under the ASME Code.

Quality control procedures for field welding and nondestructive examination are defined in Section 5.9.4.

5.9.3.5.3 Inspection and Testing

The following inspection and testing are performed on the steel liner plate.

5.9.3.5.3.1 Radiography

For quality control purposes, completed liner plate weld seams are spot radiographed by the subcontractor in accordance with the following schedule.

a. One 12 inch film is taken during the first 50 feet of each welder's work, in each welding position.

b. Thereafter, a minimum of 10 percent of the welding is progressively spot examined as welding is performed, using a 12 inch film. Locations for spot radiograph are designated by the engineer in such a manner that an approximately equal number of spot radiographs is taken from each welder's work.

c. If a radiograph discloses welding which does not meet the acceptance criteria, two additional spot radiographs, each 12 inches in length are taken in the same weld seam, at locations away from the original spot designated by the engineer.

   If the two additional radiographs show welding which meets the acceptance criteria, the entire weld seam represented by the three radiographs is considered acceptable except at the area of defect. This area is removed and repaired.

d. If either of the two additional radiographs show welding which does not meet the acceptance criteria, the entire portion of the weld seam represented by these radiographs is rejected and the defective welding removed and repaired.

e. The repaired weld seams are completely reradiographed to ensure compliance with the acceptance criteria.

The techniques used for radiographic examinations are in accord with Paragraph UW-51 of Section VIII of the 1968 ASME Boiler and Pressure Vessel Code, using X-ray and fine grain films. The double film, single viewing technique is used for all radiography.

The acceptance criteria for examinations are in accordance with Paragraph UW-52, Section VIII, of the 1968 ASME Boiler and Pressure Vessel Code. All radiographic films are submitted by the subcontractor to the engineer for review, interpretation and record.

The Safety Guide 19 was not conformed with where weldments could be radiographed. In these instances, 10 percent of the weldments are inspected by radiography. This exceeds the Safety Guide suggested minimum of two percent. The first 50 feet of welding performed by each welder is inspected instead of the first 10 feet, as suggested by the Safety Guide.

When radiographic inspection is not feasible, magnetic particle inspection is substituted.

Leak chase channel testing is performed in accordance with the suggested Safety Guide requirements with the exception that the system is pressurized for a minimum of 25 minutes and a maximum of 30 minutes, instead of two hours as stated therein. This testing may be repeated at any time during the plant life.

The limited deviations from the Safety Guide do not in any way affect the functional integrity of the steel liner plate.

5.9.3.5.3.2 Visual Examination and Dye Penetrant Testing

All weld seams are 100 percent visually examined in accordance with Section 5.9.4.5.3. Weldments which on the basis of visual examination are judged to be of questionable quality by either the subcontractor or engineer, are also inspected by dye penetrant testing.

All dye penetrant inspection is in accordance with Section VIII, of the 1968 ASME Boiler and Pressure Vessel Code.

5.9.3.5.3.3 Magnetic Particle Testing

Where nonradiographable welds are used, magnetic particle testing is substituted for radiography. A minimum of 10 percent of such welding, including splices with welded backing strips, is examined as the welding is performed.

All magnetic particle testing is performed in accordance with Appendix VI of Section VIII of the 1968 ASME Code, Dry Particle, Direct Current Production Method.

5.9.3.5.3.4 Vacuum Box Testing

All liner welds which must maintain leak-tightness integrity, including plates, shell plates, and dome plates are tested by the subcontractor as the work proceeds, using a vacuum box that can be placed over the test area and evacuated.

A 5 psi minimum pressure differential with respect to the atmospheric pressure is maintained for a minimum of 20 seconds, and verified by a gauge. The soap suds solution is continuously observed for bubbles which indicate leaks, from the time evacuation of the box is started until 20 seconds after the required vacuum has been obtained.

All leaks, regardless of size, are repaired by completely removing the defects and rewelding. The area needing repair and a minimum of two inches either side is reinspected by vacuum box testing.

Welds which cannot be vacuum box tested due to configuration and space limitations are dye penetrant tested in accordance with Section VIII, of the 1968 ASME Boiler and Pressure Vessel Code.

5.9.3.5.3.5 Halogen Testing

The leak chase system over the floor liner plate is Freon tested. A standard high sensitivity industrial halogen leak detector capable of detecting leakage in the order of 1×10^{-9} scc/second was used.

The leak chase system is initially charged with a tracer gas until a pressure of 15 psig is attained. It is then pressurized with air until a test pressure of 60 psig is reached.

The pressurized leak chase system is allowed to stand for 25 minutes minimum, before starting the probe test. Pressure measurements are taken at the beginning and the end of the holding period. Any pressure decay greater than 2 psig on a 0 to 100 psig, 4.5 inch test gauge, is rejected.

Prior to each test, the leak detector is calibrated in accordance with the manufacturer's instruction against a standard leak of 1.0×10^{-5} scc/second With the control unit of the leak detector set on automatic balance, the tip of the probe is placed on the weld seam to be tested and scanned at the rate of 1 ips. All leaks larger than 1.0×10^{-5} scc/second are located, removed, repaired and retested.

5.9.3.5.4 Quality Control of Field Welding Electrodes

The quality control procedure for the electrodes used in the field welding of the containment steel liner plate is as follows:

a. Approved procedures for handling and storing of welding electrodes are used.

b. The subcontractor’s welding and quality control supervisors control the welding electrodes by:

   1. Performing receipt inspection and verifying the electrode identification with the certificate of compliance.

   2. Physically separating the various types of electrodes and ensuring that the approved storage requirements are met.

   3. Instructing the welders on type(s) of electrodes and welding procedures to be used.

4. Monitoring welders at their work stations to ensure proper welding electrodes and welding procedures are being used.

5. Surveying the work areas near the end of each work day to ensure electrodes are either discarded or returned to the proper storage facilities.

c. Bechtel quality control inspectors monitor the liner erection and welding on a continuous basis. Weekly inspection reports are prepared.

5.9.3.6 Interior Coatings (Original Construction)

All of the coating materials given have been tested by their manufactures under simulated operating and incident conditions and certified to fully comply with all the requirements of the ANSI Standard N.101.2 (1972) “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities.”

In addition, all shipments of these materials are accompanied by vendor certifications of compliance.

5.9.3.6.1 Containment Steel Liner Plate Coatings

Surface preparation of the interior (exposed) surfaces of the containment steel liner plate is accomplished in the shop by blast cleaning each plate from edge to edge in accordance with the Steel Structures Painting Council (SSPC) Specification SSPC-SP-6-63, “Commercial Blast Cleaning.” The plates are then primed with one coat of an inorganic zinc primer, Carbo-Zinc II, to within two inches of the edges of the plate. The minimum dry film thickness (dft) of the primer is 3 mils.

In applying the primer to the plates, the coating manufacturer’s written instructions and the SSPC-SP1-63, “Solvent Cleaning” are followed explicitly.

After the liner is erected, the field weld seams and the limited burnback of the Carbo-Weld 11 coating are power tool cleaned and recoated with Carbo-Zinc 11 of 3 mils dft.

Areas which are damaged by welding, such as arc strikes, or due to the removal of temporary attachments for erection, are required and recoated so that they are equivalent to the original conditions.

Finish coats for the steel liner plate are as follows:

a. Wainscot

Two coats of a modified, organic phenolics, Phenoline Number 305 finish, at 3 mils dft each coat.

b. Above wainscot, including:

One coat of Phenoline Number 305 at 3.0 mils minimum dft.

5.9.3.6.2 Containment Interior Coatings

5.9.3.6.2.1 Steel Surfaces Wainscot

Carbon steel surfaces, including structural and miscellaneous steels, uninsulated piping, and equipment which are located in areas subject to hard usage or radioactive contamination, are blast cleaned in accordance with Steel Structures Painting Council Specification SSPC-SP6-63, "Commercial Blast Cleaning."

Within eight hours after blast cleaning, the surfaces are primed with one coat of an inorganic zinc primer, Carbo-Zinc 11, at 3 mils dft. This is followed by two coats of a modified phenolic, Phenoline Number 305 Finish, at 3 mils dft.

Insulated piping is blast cleaned and primed the same as above, but receives no finish coating.

5.9.3.6.2.2 Steel Surfaces Above Wainscot

Carbon steel surfaces above wainscot height are blast cleaned and primed the same as in Section 5.9.3.6.2.1 except that they receive one organic finish coat of Phenoline Number 305 at 3.0 mils minimum dft.

5.9.3.6.2.3 Galvanized and Stainless Steel Surfaces are not Painted.

5.9.3.6.2.4 Concrete and Masonry Surfaces

All concrete and masonry surfaces, including floors, wainscot, walls, columns, pilasters, and ceilings are chemically cleaned by either caustic wash or acid etching, or by blasting. An organic surfacer, Keeler and Long Number 6548 Epoxy Block Filler is then applied over the surfaces at a thickness of 5.5 mils dft, and one coat of organic Keeler and Long Epoxy Enamel at 2.5 mils dft.

5.9.3.7 Interior Maintenance Coatings (first implemented during Mid cycle 13, 1997)

All maintenance coating materials applied to surfaces inside or to be installed in the reactor containment have been tested to withstand Millstone Unit 2 design basis loss of coolant accident (DBA-LOCA) conditions. The coating materials and their application comply with the intent of Regulatory Guide 1.54 within the following clarification and exception.

Clarification

Compliance with Regulatory Guide 1.54 will not be invoked for equipment of a miscellaneous nature and all insulated surfaces. It is impracticable to impose Regulatory Guide requirements on the standard shop process used in painting valve bodies, handwheels, electrical cabinetry, control panels, loud speakers, emergency light cases, and other miscellaneous equipment. Wrapped or rigid insulation captures and retains any coating which may come off equipment surfaces, thereby

preventing the coating material from reaching and blocking sump drains or interrupting water flow in the containment spray system.

Exception

Quality Assurance Program recommendations stated in Regulatory Guide 1.54 are followed except that inspection will be in accordance with Section 10 of ANSI N5.12-1974 in lieu of Section 7 of ANSI N5.9 as referenced in Section 6.2.4 of ANSI N101.4.

Each coating was tested in accordance with ASTM D3911, "Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions," to the DBA conditions represented by the pressure (70 psig) and temperature (340°F) curve of Figure 1, therein.

Prior to exposure to DBA conditions, each coating was irradiated to an accumulated dose of at least $1x10^9$ Rads in accordance with ASTM D4082, "Effect of Gamma Radiation on Coatings for Use in Light-Water Nuclear Power Plants."

The coating manufacturers provide certification with each shipment that the supplied coating materials are identical to the batches of coating materials satisfactorily tested to DBA conditions.

Maintenance coating materials applied to containment surfaces are used to repair and maintain the existing coatings, to coat existing surfaces that were intended to be, but were not previously coated, and to coat new surfaces to be installed into the containment. Application of the new coating materials requires complete prior removal of the existing coating from the surface within the repair area. Overcoating the existing coatings with the new coating materials requires prior testing of the material combinations to radiation and simulated DBA conditions in accordance with Regulatory Guide 1.54.

5.9.3.7.1 Stainless Steel Surfaces

Stainless steel surfaces are not painted.

5.9.3.7.2 Galvanized Surfaces

Galvanized surfaces are spot repaired with a qualified organic coating material, as required, to maintain corrosion protection. Galvanized surfaces are not otherwise coated.

5.9.3.7.3 Carbon Steel Surfaces

Existing carbon steel surfaces are repair coated as required. New carbon steel surfaces are coated prior to or upon installation into the containment. Carbon steel surfaces may remain uncoated when substantiated by appropriate engineering evaluation.

5.9.4 QUALITY CONTROL PROCEDURES FOR FIELD WELDING AND NONDESTRUCTIVE EXAMINATIONS

5.9.4.1 Scope

These procedures outline the general quality control requirements for welding to ensure that all field welding is performed in full compliance with the applicable job specifications.

5.9.4.2 Qualifications for Welding Inspectors

All welding inspectors who inspect welds covered by this specification are qualified by meeting the following minimum requirements:

a. Inspectors must have a thorough knowledge of the various welding processes and techniques employed in field construction and be able to demonstrate the proper methods for shielded metal-arc welding, gas tungsten-arc welding, gas metal-arc welding, and oxyacetylene welding.

b. A minimum of two years previous welding inspection experience or equivalent experience and training in welding fabrication and nondestructive testing is required for all inspectors.

c. Inspectors are required to demonstrate to the satisfaction of the responsible Bechtel Material, Fabrication and Quality Control Services Representative, their knowledge of the fundamentals, techniques, and applications of the inspection methods set forth in this standard, i.e., visual, vacuum box, magnetic particle, dye penetrant and radiographic inspections.

5.9.4.3 Welding Performed by Bechtel Construction Personnel

5.9.4.3.1 Welding Procedures

All welding performed by Bechtel construction personnel is in strict accordance with the approved Bechtel Welding Procedure Specifications. The appropriate Bechtel Welding Procedure Specifications for field welds are prepared and qualified by the Bechtel Material, Fabrication and Quality Control Services Department and issued to the field by the Bechtel Project Engineer.

5.9.4.3.2 Welder Qualification

All welders who are welding under Bechtel Welding Procedure Specifications are qualified by performing the test required in the applicable Bechtel Welder Performance Specification WQ-F-1 for ferrous materials and WQ-NF-1 for nonferrous materials. These Bechtel specifications encompass the requirements of Section IX of the ASME Code. Number welder is permitted to perform production welding until he has passed the necessary tests and has the appropriate Welder Performance Qualification Test Record.

5.9.4.4 Welding Performed by Bechtel Subcontractors

5.9.4.4.1 Welding Procedures

All welding performed by Bechtel subcontractors are in strict accordance with the applicable job specifications. All welding procedures used on the project are submitted to Bechtel Engineering for approval. Production welding is not permitted without prior approval of these procedures. In all cases, field welding inspectors are responsible for determining that the subcontractor's welding is being performed in accordance with properly qualified and engineering approved welding procedure specifications.

5.9.4.4.2 Welder Qualification

All welders are welding operators employed by subcontractors who are making welds under a code or standard which requires qualification of welders are tested and qualified accordingly before beginning production welding. Each subcontractor is responsible for testing and qualifying his own welders. The Bechtel field welding inspector is responsible in all cases for determining that the subcontractor's welders have successfully passed the necessary qualification tests and that the subcontractor has the proper qualification test records for each qualified welder on file at the jobsite.

5.9.4.5 Instructions for Field Welding Inspectors

The general instructions for field welding inspectors which follow cover welding performed by both Bechtel construction and Bechtel subcontractors.

5.9.4.5.1 Welding Procedures

It is the responsibility of the field welding inspectors to assure that all welding is performed in strict accordance with the appropriate qualified welding procedure specifications. Specific items to be checked follows:

a. Determine that the proper welding procedure specification has been selected to match the base materials being welded and the welding processes being employed.

b. Permit only welders who are properly qualified under the essential variables of each welding procedure specification to make welds under that procedure.

c. Check to see that the welding electrodes, bare filler rod, consumable insert strips, and backing strips all match those which have been specified.

d. Inspect weld joints as necessary prior to welding to ensure proper edge penetration, cleaning, and fit up.

e. Check to see that the welding machine settings are correct and fall within the range of current and voltage specified.

f. Check for proper preheat and interpass temperature.

g. Inspect the inprocess welding for proper techniques, cleaning between passes, and appearance of individual weld beads.

5.9.4.5.2 Postweld Heat Treatment

The field welding inspectors inspect each postweld heat treatment (thermal stress relieving) operation to ensure conformance with the applicable job specifications. Specific items to be checked include the following:

a. A sufficient number and proper location of thermocouples are selected to accurately record temperatures.

b. The thermocouples are connected to temperature indicator recorders which provide a permanent record of the heating rate, holding temperature and time, and the cooling rate.

c. Temperature charts are checked for proper heating rate, holding temperature, holding time, cooling rate, and to see that the proper weld identification is recorded on the chart.

5.9.4.5.3 Visual Inspection of Weldments

The field welding inspectors are responsible for carrying out the necessary welding surveillance to ensure that all welding meets the following requirements for visual qualify and general workmanship. Visual inspections are performed during and after welding.

a. All weld beads, passes, and completed welds are free of slag, cracks, porosity, incomplete penetration and lack of fusion.

b. Cover passes are free of coarse ripples, irregular surfaces, nonuniform bead patterns, high crowns, deep ridges or valleys between beads, and that all blend smoothly and gradually into the surface of base metal.

c. Butt welds are slightly convex, of uniform height, and have full penetration.

d. Fillet welds are of specified size, with full throat and, unless otherwise specified, the legs are of approximately equal length.

e. Repairing, chipping, or grinding of welds is done in such a manner as not to gouge, groove, or reduce the base metal thicknesses.

f. Where different base metal thicknesses are joined by welding, the finished joint is tapered no steeper than one to four (1:4) between the thick and the thin sections.

5.9.4.5.4 Magnetic Particle Inspection

The field welding inspector is responsible for determining that all magnetic particle inspection is properly performed. He ensures that the proper techniques are followed and that the results properly interpreted. The field welding inspector requires that the subcontractor’s responsible inspection personnel demonstrate their knowledge and understanding of the applicable specifications prior to performing any production testing.

Special attention is given to the following items for all magnetic-particle inspection:

a. Determine that surfaces to be inspected have been properly cleaned and are free of crevices which could produce false indications by trapping the iron powder.

b. Determine that the power source, current density, prod spacing and application of iron powder comply with the applicable requirements.

c. Permit no arcing between the prods and weld surfaces.

d. Interpret all linear or linearly disposed indications as defects.

e. Probe questionable indications by thermal cutting, chipping, grinding, or filing to confirm the presence or absence of actual defects.

5.9.4.5.5 Dye Penetrant Inspection

The field welding inspector is responsible for determining that all dye penetrant inspection is properly performed. He ensures that the proper technique is followed and that the results are properly interpreted. The field welding inspectors require the subcontractor's responsible inspection personnel to demonstrate their knowledge and understanding of the applicable specifications prior to performing any production testing.

Special attention is given to the following items for all dye penetrant inspection:

a. Determine that surfaces to be inspected have been properly cleaned and are free of crevices which can product false indications by trapping the dye penetrant.

b. Check to see that cleaner, dye penetrant, and developer are properly applied and the specified time intervals for dye penetration and developing are followed.

c. Determine that indications are properly interpreted. Defects will be identified as red stains against the white developer background. Red lines or linearly disposed red dots are indicative of cracks. Porosity and pinhole leaks appear as local red patches or dots.

d. Examine questionable indications by a 5X or stronger hand lens, and probe by grinding or filing to confirm the presence or absence of defects.

5.9.4.5.6 Radiographic Inspection

The field welding inspector is responsible for determining that all radiographic inspection is properly performed. He ensures that radiographic techniques are followed and that the completed films are properly interpreted. The field welding inspectors require the subcontractor's responsible inspection personnel to demonstrate their knowledge and understanding of the applicable specifications prior to beginning the radiographic inspection. The field welding inspector also reviews each completed radiograph.

Special attention is given to each of the following items for all radiographic inspection:

a. Check the type of film intensifying screens, penetrameters, and sources of radiation for conformance to the job specifications.

b. Check the relative location of film, penetrameters, identifying numbers, and radiation source for each typical exposure.

c. Review all completed film for quality and interpretation of defects. Check the exposed and developed film for proper density and visibility of penetrameters. Radiographic film of unacceptable quality or with questionable indications of defects are reradiographed.

5.9.4.5.7 Other Welding Inspections

The field welding inspectors are responsible for determining that all other types of welding inspection, where specified, are properly performed.

5.9.4.5.8 Repairs

It is the responsibility of the field welding inspectors to determine that all weld defects in excess of specified standards of acceptance are removed, repaired, and reinspected in accordance with the applicable job specifications.

5.9.4.5.9 Records

It is the responsibility of the field welding inspector to ensure that proper records of welding and nondestructive testing are kept on file at the jobsite.

TABLE 5.9-1 AGGREGATE TESTS

ASTM Number	Title	Results To Be Achieved	Initial Test	User's Test	Periodic Test
C-33	Specification for concrete aggregates	To conform with specification	X	X	X
C-40	Organic impurities in sands for concrete	To conform with specification	X	X	X
C-87	Effect or Organic Impurities in Fine Aggregate on Strength of Mortar	To conform with specification	X	X	
C-88	Soundness of Aggregates	To conform with specification	X	X	
C-117	Materials finer than Number 200 sieve	Design mix calculations	X	X	
C-127	Specific gravity and absorption (coarse aggregates)	Design mix calculations	X	X	
C-128	Specific gravity and absorption (fine aggregates)	To conform with specification	X	X	
C-131	Los Angeles Machine abrasion	To conform with specification	X	X	
C-136	Sieve analysis of fine and coarse aggregates	To conform with specification	X	X	
C-142	Clay lumps	To conform with specification	X	X	
C-227	Potential Alkali reactivity (mortar bar)	To conform with specification.	X	X	
C-289	Potential reactivity (chemical)	To conform with specification	X	X	
C-295	Petrographic examination of aggregates	To conform with specification	X		

TABLE 5.9-2 CEMENT TESTS

ASTM Number	TYPE OF TEST	INITIAL TEST	USER'S TEST	PERIODIC TESTS
C-109	Compressive Strength	X	X	X
C-114	Chemical Analysis	X	X	
C-115	Fineness-Turbidimeter	X	X	
C-151	Auto-clave expansion (soundness)	X	X	
C-183	Sampling	X		
C-185	Air content of mortar	X		
C-186	Heat of hydration	X		
C-191	Time of setting by Vicat needle	X	X	X
C-204	Fineness by air permeability	X		
C-266	Time of setting by Gillmore needles	X		
C-451	False set (paste)	X		

TABLE 5.9-3 FLY ASH TESTS

ASTM Number	TYPE OF TEST	INITIAL TEST	USER'S TEST	PERIODIC TESTS
C-109	Compressive strength	X	X	X
C-114	Chemical analysis	X	X	
C-151	Autoclave expansion (Soundness)	X	X	
C-188	Specific gravity	X	X	
C-204	Fineness	X	X	
C-311	Sampling and testing	X	X	

TABLE 5.9-4 TYPICAL CHEMICAL ANALYSIS OF FLY ASH USED

Chemical Analysis	NUSCO Devon Plant	RG&E Co. Russell Plant	ASTM C-618
Silicon dioxide (SiO_2) plus aluminum oxide ($A1_2O_3$) plus iron oxide (Fe_2O_3) (minimum %)	93.22	86.20	70.0
Magnesium oxide (MgO) (maximum %)	1.20	1.22	-
Sulfur trioxide (SO_3) (maximum %)	0.60	1.16	5.0
Moisture content (maximum %)	0.13	0.13	3.0
Loss on ignition (maximum %)	4.01	5.90	12.0
Available alkalies as Na_2O (maximum %)	0.77	0.73	1.5

TABLE 5.9-5 ALLOWABLE VOID LIMITS FOR CADWELDING

Bar Size Number	Splice Catalog Number	Maximum Allowable Void Limits	
		Void Area Standard Splices, (1) (2) (in^2)	Vertical - Full Circumference Low, (3) (inches)
6-6	RBT-6101-(-H)	1.05-1.05	5/8 - 5/8
6-7	RBT6-7101 (-H)	1.05-1.03	5/8 - 9/16
7-7	RBT-7101 (-H)	1.03-1.03	9/16 - 9/16
7-8	RBT7-8101 (-H)	1.03-1.02	9/16 - 1/2
8-8	RBT-8101 (-H)	1.02-1.02	1/2 - 1/2
8-9	RBT-8-9101 (-H)	1.02-1.02	1/2 - 1/2
9-9	RBT-9101 (-H)	1.02-1.02	1/2 - 1/2
9-10	RBT9-10101 (-H)	1.02-1.03	1/2 - 7/16
10-10	RBT-1091 (-H)	1.03-1.03	7/16 - 7/16
10-11	RBT10-11101 (-H)	1.03-1.53	7/16 - 9/16
11-11	RBT-11101 (-H)	1.53-1.53	9/16 - 9/16
11-14	RBT-11-14101 (-H)	1.53-1.52	9/16 - 5/8
11-18	RBT11-18101 (-H)	1.53-1.99	9/16 - 1/2
14-14	RBT-1476 (-H)	2.15-2.15	5/8 - 5/8
14-14	RBT-14101 (-H)	2.15-2.15	5/8 - 5/8
14-18	RBT14-18101 (-H)	2.15-1.99	5/8 - 1/2
18-18	RBT-1876 (-H)	2.64-2.64	9/16 - 9/16
18-18	RBT-1891 (-H)	3.00-3.00	5/8 - 5/8
18-18	RBT-18101 (-H)	3.00-3.00	5/8 - 5/8

Void Area = W (D-3/16) (Normal void due to asbestos packing)

NOTES:

(1) The maximum allowable void area computed separately for each end of the splice sleeve as shown in the sketch.

(2) This column is used for all standard splices including vertical, horizontal, horizontal side fill, angled splices and B-series structure splices.

(3) This column is used for vertical splices only with low filler metal around entire circumference. For spot voids in vertical splices, standard splices column is used.

5.A DESCRIPTION OF FINITE ELEMENT METHOD USED IN CONTAINMENT ANALYSIS

5.A.1 INTRODUCTION

The initial development of the finite element method was done by Turner, et al. (Reference 5.A-1) for future application in aerospace technology. Turner, et al. later used the two-dimensional plate elements in the analysis of aircraft structures. These first applications of the method were used to analyze the plane stress problem. (References 5.A-1 and 5.A-2) Continued development of the method has extended its applicability to the plane strain problem, flat plate bending, flat plate stability studies, three dimensional axisymmetric stress analysis and for general shell analysis. (References 5.A-3 through 5.A-6) Most recently, a textbook (Reference 5.A-7) by Zienkiewicz was published which contains many examples of solutions to practical problems using finite elements. This text also presents an excellent treatment of this powerful approach to the solution of problems in continuum mechanics.

5.A.2 ANALYTICAL METHOD

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analyses of frames and trusses, for example, can be considered as the application of the finite element method using one-dimensional elements. In utilizing the method to an axisymmetric solid (e.g., a concrete containment), the continuous structure is replaced by a system of rings of circumferential joints. Based on the energy principles, a set of force equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are the unknowns of the system. A solution of this set of equations is inherent in the solution of the finite element system.

There are many advantages to the finite element method, when compared to other numerical approaches. The method is completely general with respect to geometry and material properties. Complex bodies composed of many different materials are easily represented; therefore, in the analysis of the containment, concrete, liner plate and foundation material can be realistically considered. Also, axisymmetric thermal, mechanical and gravity loadings can be analyzed.

It can be shown mathematically that the method converges to the exact solution as the number of elements in increased; therefore, any desired degree of accuracy may be obtained within the limits of computational capacity.

5.A.3 COMPUTER PROGRAM

The initial development of the computer program used in the analysis of the containment was conducted at the University of California at Berkeley in 1962, under a National Science Foundation Grant (G18986). Since that time the program has been further modified and refined by Dr. Edward L. Wilson. The validity of the specific program used in the containment analysis has been established by the analysis of axisymmetric solids with known exact linear solutions. It

is noted that the results of a finite element analysis always satisfy statistics, since the equations solved within the computer program are based on the force equilibrium requirements.

5.A.4 COMPARISONS WITH KNOWN SOLUTIONS

An exact analysis of the containment structure under consideration is impossible by classic methods. A preliminary approximate analysis of the structure was conducted based on the classical shell theory. In addition to the difficulty in representing the steel liner, thickened portions of the shell and foundation materials, shell theory neglects the members thicknesses and shear deformations. Since the finite element approach includes the members thicknesses and shear deformation, an exact comparison with shell theory cannot be expected. However, forces obtained from the finite element method at sections not near discontinuities or the foundation do agree with the results, based on the shell theory.

Figure 5.A-1(c) illustrates the comparison of stresses from a classical problem for which an exact closed form solution exists and those obtained by the finite element method. The figure also shows the effect of the fineness of the finite element mesh on the degree of accuracy of the solution.

The problem is the determination of the radial and tangential stresses in an infinitely long, thick-walled cylinder of radius r and wall thickness of r/2, which is subjected to an internal pressure, p. (see Figure 5.A-1(b)). The values of the stresses through the wall thickness can be determined by the Lame solution. (Reference 5.A-8) The solution to this problem, using the finite element method, was carried out by E. L. Wilson. (Reference 5.A-9)

For the containment, comparisons have been made between the finite element method and an analysis done in accordance with the general shell theory for homogeneous surfaces of revolution. The matrix of influence coefficients (the unknown forces — deflections, moments and rotations for the dome, ring and cylinder) has been solved for the condition of equal deflection and rotations. Similar analytical methods have been used for the intersection of the base slab and the cylinder wall. The base slab has been analyzed as an elastic plate on an elastic foundation.

In general, the results thus obtained are within five percent of those obtained by the more rigorous finite element method for ring girder, dome, and cylinder wall of a similar containment.

For a typical finite element analysis, the foundation material and base slab interaction is studied by extending the mesh of the finite elements into the foundation material. Since the locations of the boundaries of the finite element mesh within the soil mass are a function of the soil properties, (Reference 5.A-10) studies are performed to determine where these boundaries should be located. After this has been determined, the final analyses of the containment by the finite element method will follow. Agreement between the results of this approach and the results of hand calculations, based on the assumption of an elastic plate on an elastic foundation, can be expected to be only approximate due to the difficulty of representing varying soil properties in the governing boundary conditions for the hand calculations.

Stresses in the regions of anchors have been determined using a plane strain analysis by the finite element method. Not only have the stresses resulting from the prestressing forces been determined, but also the effects due to the thermal and pressure loadings have been studied in the anchorage regions. The plane strain analysis is a better approximation than plane stress for the three-dimensional problem. However, since the program is prepared for plane stress analysis, modifications must be made in the elastic constants. The modulus of elasticity, E, is modified to $E/(1-v^2)$; Poisson's ration, v, is modified to v/(1-v) and the linear thermal coefficient of expansion α, is modified to $\alpha(1+v)$.

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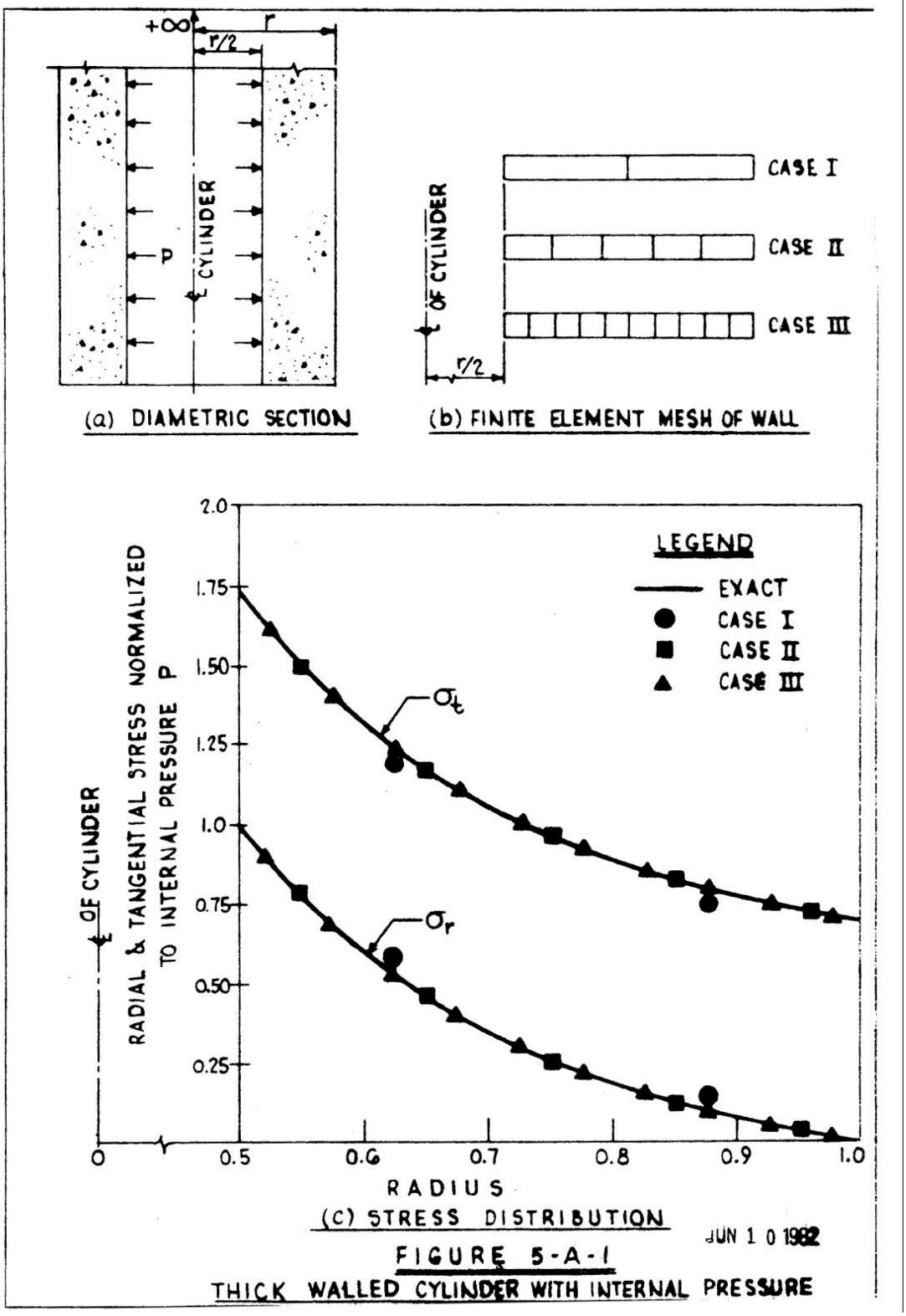
FIGURE 5.A–1 THICK WALLED CYLINDER WITH INTERNAL PRESSURE



5.B JUSTIFICATION FOR LOAD FACTORS AND LOAD COMBINATIONS USED IN DESIGN EQUATIONS OF CONTAINMENT

5.B.1 GENERAL

The load factors and load combinations in the design equations represent the consensus of the individual judgments of a group of Bechtel engineers and consultants who are experienced in both structural and nuclear power plant design. Their judgment has been influenced by current and past practice, by the degree of conservativeness inherent in the basic loads, and particularly by the probabilities of coincident occurrences in the case of incident, wind and tornadoes, as well as seismic loads.

The following discussions explain the justification for the individual factors, particularly as they apply to containment structures.

5.B.2 DEAD LOADS

Dead loads in a large structure such as this are easily identified and their effects can be accurately determined at each point in the structure. For dead loads in combination with the incident, seismic, or wind and tornado loads, a load factor representing a tolerance of five percent is chosen to account for dead load inaccuracies. The ACI Code allows a tolerance of +25 percent and -10 percent, but the code is written to cover a variety of conditions where weights and configurations of materials in and on the structure may not be clearly defined and are subject to change during the life of the structure.

5.B.3 LIVE LOADS

The live loads that would be present along with the incident, seismic, or wind and tornado loads would produce a very small portion of the stress at any point. Also, it is extremely unlikely that the full live loads would be present over a large area at the time of an unusual occurrence. For these reasons, a low load factor is felt to be justified and the live loads are considered together with dead loads at a load factor of 1.05.

5.B.4 SEISMIC LOADS

The operating basis earthquake that has been selected is considered to be the possible earthquake which could occur during the life of the plant. In addition, a design basis earthquake which defines the maximum credible earthquake which could occur at the site, is also considered in the design. Class I structures, systems, and equipment are designed so that no loss of function would result from the design basis earthquake. Consequently, the probability of an earthquake causing a loss-of-coolant incident is very small. For this reason, the two events, earthquake and the loss-of-coolant incident is very small. For this reason, the two events, earthquake and the loss-of-coolant incident, are considered together, but at much lower load factors than those specified for each separate event.

The seismic load factors of 1.25 and 1.00 are conservative for, respectively, the operating basis and the design basis earthquake in combination with the factored loss-of-coolant incident.

5.B.5 WIND AND TORNADO LOADS

Wind and tornado loads are determined from the velocities of the design wind and design tornado, respectively. With the containment designed for the extreme wind, it is inconceivable that the wind would cause a loss-of-coolant incident. Therefore, wind loads are not considered with the incident loads.

A load factor of 1.0 is applied to the tornado loads.

5.B.6 LOSS-OF-COOLANT INCIDENT

The design pressure and temperature are based on the operation of partial safeguards equipment using emergency diesel power.

European practice has been to use a load factor of 1.5 on the design pressure (Reference 5.B-1). This factor is reasonable and has been adopted for this design.

In all cases the design temperature is defined as that corresponding to the unfactored pressure. At 1.5 P, the temperature will be somewhat higher than the temperature at P. It would be unrealistic to apply a corresponding temperature factor of 1.5 since this could only occur with a pressure much greater than a pressure of 1.5 P.

5.B.7 REFERENCES

5.B-1 T. C. Waters and N. T. Barrett, “Prestressed Concrete Pressure Vessels for Nuclear Reactors,” Journal British Nuclear Society 2, 1963.

5.C JUSTIFICATION FOR CAPACITY REDUCTION FACTORS (ϕ) USED IN DETERMINING CAPACITY OF CONTAINMENT

The ϕ factors are provided to allow for variations in materials and workmanship. In the ACI Code 318-63, ϕ varies with the types of stresses or members considered; that is, with flexure, bond or shear stress, or compression.

The ϕ factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength assuming that materials are as strong as specified, sizes are as shown on the drawings, the workmanship is excellent, and that the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables by suing these ϕ factors:

ϕ = 0.90 for concrete in flexure

ϕ = 0.85 for diagonal tension, bond, and anchorage

ϕ = 0.75 for spirally reinforced, concrete compression members

ϕ = 0.70 for tied compression members

The ϕ value is larger for flexure because the variability of steel is less than that of concrete and the concrete in compression has a fail-safe mode of behavior; that is, material understrength may not cause failure. The values for columns are lower (favoring the toughness of spiral columns over tied columns) because columns fail in compression where concrete strength is critical. Also, it is possible that the analysis might not combine the worst combination of axial load and moment. Since the member is critical in the gross collapse of the structure, a lower value is used.

The additional ϕ values used represent the best judgment of Bechtel as to how much understrength should be assigned to each material and condition not covered directly by the ACI Code. The additional ϕ values have been selected, based on material quality in relation to the existing ϕ values.

Conventional concrete design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI uses ϕ = 0.90. The same reasoning has been applied in assigning a value of ϕ = 0.90 to reinforcing steel in tension, which now includes axial tension. However, the Code recognizes the possibility of reduced bond of the bars at the laps by specifying a ϕ of 0.85. Mechanical and welded splices will develop at least 125 percent of the yield strength of the reinforcing steel. Therefore, ϕ = 0.85 is recommended for this type of splice.

The only significantly new value introduced is ϕ = 0.95 for prestressed tendons in direct tension. A higher value than that specified for conventional reinforcing has been allowed because: during installation the tendons are each jacked to about 94 percent of their yield strength, so in effect each tendon has been proof tested; and, the method of manufacturing prestressing steel (cold drawing and stress relieving) ensures a higher quality product than conventional reinforcing steel.

5.D EXPANDED SPECTRUM OF TORNADO MISSILES

The spectrum of tornado missiles is expanded to include the following:

1. Utility pole 13.5 inch diameter by 35 feet long with density of 43 lbs/ft^3
2. 1inch solid steel rod 3 feet long with a density of 490 lbs/ft^3
3. 6 inch, schedule 40 pipe, 15 feet long with a density of 490 lbs/ft^3
4. 12 inch, schedule 40 pipe, 15 feet long with a density of 490 lbs/ft^3

For each of the above tornado-borne missiles, the following information is provided:

1. The maximum velocity and height attained. Assuming in the analyses that each of the missiles originates at ground level and at the highest structural elevation on the site capable of producing each missile.
2. The required thickness of a reinforced concrete missile barrier to stop the missiles without their penetrating the missile barrier. Discussing the adequacy of all tornado missile barriers protecting systems and components necessary for safe shutdown.
3. The required thickness of a reinforced concrete missile barrier to preclude the generation of secondary missiles within the structure.
4. The effects that secondary missiles could have on safety related equipment and systems in the event that they occur.

In developing the above information, the analytical approach presented in BC-TOP-9, Design of Structure for Missile Damage with the following exceptions is used, assuming the missiles do not tumble and are at all times oriented such as to have the maximum value of C_dA/W while in flight.

THE TORNADO MODEL: The tornado model will be patterned after the Dallas tornado of April 2, 1957, as studied by Hoecker (Reference 5.D-1). The model is basically that given in WCAP-7897 (Reference 5.D-2) but with a more rigorous extrapolation to the parameters desired for a design tornado than given by Bates and Swanson (Reference 5.D-3).

Hoecker summarized his findings by the use of a “pressure-time profile” for an average translational velocity of 27 mph and as a function of percentage of total pressure drop.

In Attachment A, it is shown that when this time-pressure profile is used to solve the cyclostropic wind equation, the tangential wind velocities correspond with the experimental ones when a total pressure drop of 60 mb, or 0.882 psi, and a translational velocity of 27 mph is substituted into the equation.

When a total pressure drop of 3 psi and a 60 mph translational velocity (88 fps) is substituted into the same equation, a 304 mph maximum tangential velocity at a 300 foot radius is obtained. This corresponds closely to the assumptions which have been made in the past when describing the design tornado.

The two exponential equations used by Hoecker to determine the time-pressure profile cross each other at a radius of 1,240 feet instead of the 300 feet at which they cross when a translational velocity is 27 mph. Therefore, it is only necessary to use one equation since the starting tangential velocity corresponding to this distance is 66 mph, which is less than the minimum 75 mph considered by Bates and Swanson.

By incorporating these two assumptions, namely, that the vertical component is equal to one third of the tangential and the radial component is a function of radial distances between minimum and maximum tangential components being considered, a complete windfield was defined by using the following equations:

$$V_t = 249\sqrt{\frac{\mathrm{Exp}(-48.3V_1^3/R^3)V_1^3D}{R^3}} \quad (1)$$

$$V_r = -\frac{(1240-R)R}{(1240-300)} \quad (2)$$

$$V_v = 1/3V_t \quad (3)$$

Where:

V_t = tangential velocity (fps)

V_r = radial velocity (fps)

V_v = vertical velocity (fps)

V_1 = translational velocity (fps)

D = total pressure drop (psf)

R = radius (feet)

Equation (1) has been left in a general form for use in future models to predict different total pressure drops or translational velocities. However, at this time D is taken as 432 psf and V_1 = 88 fps.

The relative conservatism with reference to the actual Dallas tornado is shown in Figure 5.D–1.

The assumption of constant velocities from the ground to a height of 500 feet is a degree of conservatism which is justified by the expanded view of these velocities. This information has been published by Hoecker and is reproduced in Figure 5.D–2.

TYPES OF MISSILES: Previous studies had considered a car as a missile for low elevations and a wooden plank for high elevations. Later a small cross-section missile in the form of a pipe was added. Five missiles are now being considered.

All the missiles are intended as prototypes of the many missiles generated by tornados. Considering the present (1973) state of the art, a detailed physical description of a missile is of little value when designing the missile proof target. Empirical formulations have to be used in areas where impactive energy and the impactive area are the points to be considered.

The more logical approach is to assume a generalized range of missiles with the required drag factors impacting at given elevations with the highest possible velocity. The impactive kinetic energy per square foot of impact area for each elevation would then be computed.

If a table is made with C_dA/W factors from 0.10 to 0.15, which is the smallest measurement for an airborne missile, it will be found that there is a drag factor that will give the highest velocity at each elevation. This is shown in Table 5.D-1.

It is interesting to show the small range and the gap left for the maximum drag factor proposed.

Wooden plank	0.06
Utility pole	0.026
Steel rod	0.031
6 inch pipe	0.029
12 inch pipe	0.021

Impactive energy per unit area measured in lb/ft as shown on Table II is readily found as follows:

$$K = \frac{W V^2}{2g A} = \frac{V^2}{2g F}$$

Where:

K = impactive factor (lb/ft)

V = velocity at impact (fps)

A = area of impact (ft^2)

F = C_dA/W for $C_d = 1$

W = weight of missile (pounds)

g = acceleration of gravity (ft/sec^2)

As expected, this impactive factor is much higher at lower elevations: it varies from 49,640 lb/ft for a 10 foot elevation to 22,070 lb/ft for a 110 foot elevation.

Penetrations can be computed by the required empirical formulation which is workable in terms of these impactive factors.

METHODS OF INJECTION AND PROPULSION: Bates and Swanson propose three methods of injection:

a. Explosive injection

b. Aerodynamic injection

c. Ramp injection

These are intended to limit the height at which a given object may be injected into a tornado. So many considerations and assumptions have to be made that they become of no practical value when it is to be assumed that the object will reach the highest point of a structure even if the missile has to be held at a convenient elevation for injection to occur.

If an explosive injection occurs some distance away from a structure, it is concluded that the object could clear the structure, if such an injection could occur. Aerodynamic injection will require aerodynamic objects or else the injection is overestimated. Likewise, a ramp injection will depend on the given ramp, a factor that is hard to generalize.

All three methods of injection required many assumptions which make it difficult for generalization. A fourth method which would be called the “Uplift Injection” offers the advantages of simplicity and applicability.

In the uplift injection it is assumed that the wind finds its way beneath a surface and the object will become airborne at the time when the vertical component of the wind produces an upward force equal to the weight of the object. While on the ground the object is assumed to be free to move on the horizontal plane in a frictionless manner as the tangential and radial components of the wind act on it.

When a missile flight is to be ascertained by applying the three components of the wind (tangential, radial and vertical) simultaneously, a random surface is assumed to be facing all three components. This random surface will produce what is called in WCAP-7897 an “effective drag factor” to be applied in all directions and which is computed as follows:

Cylinder:

$$C_e = \frac{0.389(h + 0.66D)}{P_{obj}\ hD}$$

Parallelpiped:

$$C_e = \frac{0.483\ w\,(h + d)}{P_{obj}\ w\ h\ d}$$

where:

C_e = effective drag factor

h = length, feet

D = diameter, feet

P_{obj} = density, lb/ft^3

w = width, feet

d = depth, feet

To date, this is the best method of computing an effective drag area for an object thrown into a tornado.

Using these effective drag areas in the computer program, DALLAS MISS GEN, the following results (velocities in ft/sec) were obtained:

Elevation	**Wooden Plank C_e = 0.03**	**Utility Pole C_e = 0.0082**	**Steel Rod C_e = 0.0097**	**6 inch Pipe C_e = 0.0015**	**12 inch Pipe C_e = 0.00078**
10	223	182	192	98	---
20	264	---	---	--	---
30	279	---	---	--	---
40	270	---	---	--	---
50	261	---	---	--	---
60	258	---	---	--	---

These results show that only the wooden plank type missile could be sustained in the air. It supports WCAP-7897, Chapter 5: “Investigation of Some Specific Missiles” which clearly states: “The results of Figure 3 indicate that objects with a C_dA/W less than 0.012 ft^2/lb will not be sustained by the vertical wind even if injected above immediate obstructions.” (Figure 3 is contained in WCAP-7897.)

It appears then that instead of assuming impossible C_dA/W factors for a given set of missiles, it is best to assume an infinite number of missiles, all with possible effective drag factors.

The required thickness of a concrete element that will just be perforated by a missile is given by:

$$T = \frac{427\ W}{\sqrt{f'_c}\ D^{1.8}} \left(\frac{V_s}{1000}\right)^{1.33}$$

where:

T = Thickness of concrete element to be just perforated (inches)

W = Weight of missiles (pounds)

D = Diameter of missiles (inches)

Note: For irregularly shaped missiles, an equivalent diameter is used. The equivalent diameter is taken as the diameter of a circle with an area equal to the circumscribed contact, or projected frontal area, of the non-cylindrical missile.

V_s = Striking velocity of missile (ft/sec)

f'_c = Compressive strength of concrete (psi)

This formula is known as the Ballistic Research Laboratory (BRL) formula as presented in Reference 5.D-5.

The thickness, t_p, of a concrete element required to prevent perforation must be greater than T. It is recommended to increase T by 25 percent, but not more than 10 inches, to obtain the t_p, required to prevent perforation.

$t_p = 1.25T \leq T + 10$ (in inches)

The results obtained by using the above formula are presented in Table 5.D-5. The concrete barriers furnished to protect systems and components necessary for safe shutdown exceed the required thickness to prevent perforation by the missiles.

Spalling of concrete from the side opposite the impact surface of the element is considered as a secondary missile. For an estimate of the thickness that will just start spalling, it is recommended that the following equation be used:

$$Ts = 2T$$

where:

T_s = Concrete element thickness that will just start spalling (inches)

T = Concrete thickness to be just perforated (inches).

The thickness, t_s, of a concrete element required to prevent spalling must be greater than T_s. It is recommended to increase T_s by 25 percent, but not more than 10 inches, to prevent spalling.

$$t_s = 1.25T_s \leq T_s + 10 \text{ (in inches)}$$

The results obtained by using the above formula are presented in Table 5.D-5.

The BRL formula was selected after a thorough study of all available formulae in the literature for concrete perforation and spalling due to missile impact. As with all other available formulae, the BRL formula represents an empirical expression based upon high velocity test data and was developed for use in the high velocity range (i.e., missile impact velocity in excess of 1,000 ft/sec). The range of missile velocities considered in a nuclear facility is generally below 500 ft/sec. In order to provide a confidence margin for the lower velocity range, and to assure that barrier thickness would exceed that at which perforation or spalling impends, the design thickness was increased.

Test data on the impact of a one-inch diameter steel rod having a velocity from 150 ft/sec to 320 ft/sec on concrete barriers of 3 inches, 6 inches and 9 inches in thickness indicate that these formulae provide conservative results for both concrete perforation and spalling in the velocity range as stated. A summary of the test results is presented in Table 5.D-6.

A procedure for determining thickness of spalling is presented in Reference 5.D-4. The spalling effects on concrete wall due to the impacts of wooden plank and utility pole were investigated and, in both cases, no spalling of concrete wall was indicated. Therefore, the secondary missiles are not considered credible.

The thickness, t_s, of a concrete element required to prevent spalling is more than the thickness, T_m, of a concrete element furnished in the cases of wooden plank and utility pole, as indicated in Table 5.D-5. The thickness, t_s, provides a simplified approach of determining a thickness required for a concrete barrier to stop a missile. A margin of safety, an increase of 25% of the calculated values with an upper limit of 10 inches, is a logical safety factor against spalling or perforation and is further reinforced by the test data presented in Figure 5.D–3. The formula used to determine the thickness of spalling does not consider reinforcing steel which tends to reduce the amount of spalling. If t_s is less than T_m, as in the case of 3 inch steel rod and 6 inch pipe, no additional analysis is required.

To determine the thickness of spalling, the following formula is presented in Reference 5.D-4.

$$X_{max} = \frac{282\ N\ W}{\sqrt{f'_c}\ d^{1.8}}\left(\frac{V}{1000}\right)^{1.8}$$

$$C_1 = \left(\frac{T}{X_{max}} + C_3 - 1\right)\left(\frac{C_s}{V}\right)^{1/3}$$

$$C_3 = \frac{0.877\ W^{1/3}}{X_{max}}$$

$$\Delta_{t(initial)} = C_2 X_{max}\left(\frac{V}{C_s}\right)^{1/3}$$

$$\Sigma\Delta_t = \frac{1}{8}[T + 0.877\ W^{1/3} + \Delta_{t(initial)} - X_{max}]$$

where:

V = velocity of missile, ft/sec

W = weight of missile, pounds

T = target thickness, inches

d = diameter of missile, inches

f'_c = concrete strength, psi

$\Delta_{t(initial)}$ = thickness of initial spall, inches

N = nose factor = 0.845 for hemi-spherical nose

C_2 = coefficient from Figure 4.10 of Reference 5.D-4

C_s = dilational velocity in concrete = 9,800 ft/sec

X_{max} = concrete thickness to just perforate, inches

By substituting the following values into the equations, the value of $\sum\Delta_t$, total thickness of all spalling, for the wooden plank and the utility pole was determined to be insignificant.

<parameter>	**Wooden Plank**	**Utility Pole**
Weight	105 pounds	1,500 pounds
Velocity	280 fps	182 fps
Target Thickness	12 inches	24 inches
Diameter of Missile	7.82 inches	13.5 inches
Concrete Strength	3,000 psi	3,000 psi

5.D.1 REFERENCES

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TABLE 5.D-1 IMPACTIVE VELOCITIES (fps) OF MISSILES OF DIFFERENT C_dA/W FACTORS AS PICKED FROM THE GROUND BY THE DESIGN TORNADO

Elevation	C_dA/W	0.1	0.09	0.08	0.07	0.06	0.05	0.04	0.03	0.02	0.015
10		142	148	153	161	171	183	200	223	244	219
20		167	175	184	194	206	226	244	264	246	---
30		194	203	214	226	241	259	278	279	238	---
40		223	233	246	259	274	291	300	270	238 [a]	---
50		253	263	276	290	305	313	301	261	---	
60		283	294	307	317	327	321	290	258 (a)	---	
70		312	322	333	339	336	313	279	---		
80		339	347	352	349	331	300	273	---		
90		361	364	360	346	318	290	271(a)			
100		375	369	355	331	304	283	---			
110		377	363	341	316	294	280	---			
120		367	346	324	303	287	280 (a)				
130		349	327	309	293	284	---				
140		330	312	298	287	283 (a)	---				
150		311	298	290	285	---					
160		297	290	284	285(a)	---					
170		287	283	283 (a)	---						
180		279	280	---							
190		276	280(a)	---							
200		275	---								

a. Missile reached the condensation funnel at some elevation above the previous one.

TABLE 5.D-2 KINETIC ENERGY PER FT^2 OF IMPACT AREA

C_d A/W	Max Impact Velocity (fps)	At Elevation (ft)	Kinetic Energy/Ft^2 (lb/ft)
0.015	219	10	49,640
0.02	246	20	44,747
0.03	279	30	40,290
0.04	301	40	35,171
0.04	301	50	35,171
0.05	321	60	32,000
0.06	336	70	29,217
0.07	349	80	27,019
0.08	360	90	25,155
0.09	369	100	23,492
0.10	377	110	22,070

Use the last figures for elevations above 110 feet.

TABLE 5.D-3 RADIUS VS. VELOCITY FPS/MPH (D=0.882 PSI/V_1 = 27 MPH)

RADIUS	VELOCITY FPS	VELOCITY MPH
1500	32.63	22.25
1400	33.75	23.01
1300	35.00	23.86
1200	36.39	24.81
1100	37.96	25.88
1000	39.76	27.11
900	41.85	28.53
800	44.29	30.20
700	47.22	32.20
600	50.83	34.65
500	55.40	37.77
400	61.48	41.92
300	127.30	86.80
290	133.15	90.78
280	139.39	95.04
270	146.05	99.58
260	153.15	104.42
250	160.71	109.58
240	168.74	115.05
230	177.22	120.83
220	186.14	126.91
210	195.43	133.25
200	204.97	139.75
190	214.57	146.29
180	223.90	152.66
170	232.49	158.52
160	239.58	163.35
150	244.08	166.42

TABLE 5.D-3 RADIUS VS. VELOCITY FPS/MPH (D=0.882 PSI/V_1 = 27 MPH)

RADIUS	VELOCITY FPS	VELOCITY MPH
140	244.40	166.63
130	238.39	162.53
120	223.34	152.27
110	196.44	133.93
100	156.08	106.41
90	104.68	71.37
80	52.23	35.61
70	15.07	10.27
60	1.45	.99
50	.01	.01
40	0	0
30	0	0
20	0	0
10	0	0

Where: D = 0.882 psi

V_1 = 27 mph

TABLE 5.D-4 RADIUS VS. VELOCITY FPS/MPH (D=3 PSI/V_1 = 60 MPH)

RADIUS	VELOCITY FPS	VELOCITY MPH
1500	88.63	60.43
1400	91.59	62.45
1300	94.88	64.69
1200	98.54	67.19
1100	102.67	70.00
1000	107.35	73.19
900	112.74	76.87
800	119.03	81.16
700	126.50	86.25
600	135.56	92.42
500	146.86	100.13
400	161.49	110.10
300	446.95	304.74
290	440.56	300.38
280	430.85	293.76
270	417.34	284.55
260	399.52	272.40
250	376.98	257.03
240	349.39	238.22
230	316.68	215.92
220	279.11	190.30
210	237.44	161.89
200	193.06	131.63
190	148.07	100.96
180	105.24	71.75
170	67.63	46.11
160	37.97	25.89
150	17.73	12.09

TABLE 5.D-4 RADIUS VS. VELOCITY FPS/MPH (D=3 PSI/V_1 = 60 MPH)

RADIUS	VELOCITY FPS	VELOCITY MPH
140	6.40	4.36
130	1.60	1.09
120	.23	.16
110	.02	.01
100	0	0
90	0	0
80	0	0
70	0	0
60	0	0
50	0	0
40	0	0
30	0	0
20	0	0
10	0	0

Where: D = 3 psi

V_1 = 60 mph

TABLE 5.D-5 VELOCITIES OF VARIOUS MISSILES

Missiles	Maximum Velocity (fps)	Maximum Height (ft.)	t_p (in.)	t_s (in.)	t_m (in.)
Wooden Plank	280	60	6.5	13.0	12.0
Utility Pole	182	10	14.0	28.1	24.0
1 inch Steel Rod	192	10	8.7	17.4	24.0
6 inch Pipe	98	10	5.1	10.1	24.0
12 inch Pipe	-	-	-	-	-

t_m = Minimum concrete thickness furnished.

TABLE 5.D-6 TEST DATA SUMMARY

Test Number	Test Date (July)	Target Thickness (inches)	Missile		Velocity		Penetration		Spalling
			Length (feet)	Weight (lbs)	Electronic (ft/sec)	Movies (ft/sec)	Depth (inches)	Volume (in^3)	
1	2	9	3	8.05		203	1.71	21	None
2	3	6	3	8.04	213	211	1.60	11.6	Small & cracks
3	3	3	3	8.05	218	213	Perforated	6.1/30.5 front/back	Maximum travel of 144 ft.
4	4	6	3	8.05	220	214	1.98	17.4	Large & cracks
5	4	6	1.4	3.78	322	312	1.68	21.6	Moderate & cracks
6	4	9	3 *	8.01	235 **	214	2.34	22	None
7	5	9 ***	3	8.04	217	216	1.84	19.9	None
8	5	6	3	8.04	150	151	1.23	7.6	Small & cracks

* Rounded end.

** Believed to be bad reading due to rounded missile end.

*** No reinforcing in slab.

FIGURE 5.D–1 TANGENTIAL VELOCITIES AS DERIVED FROM HOECKER'S PRESSURE - TIME PROFILE

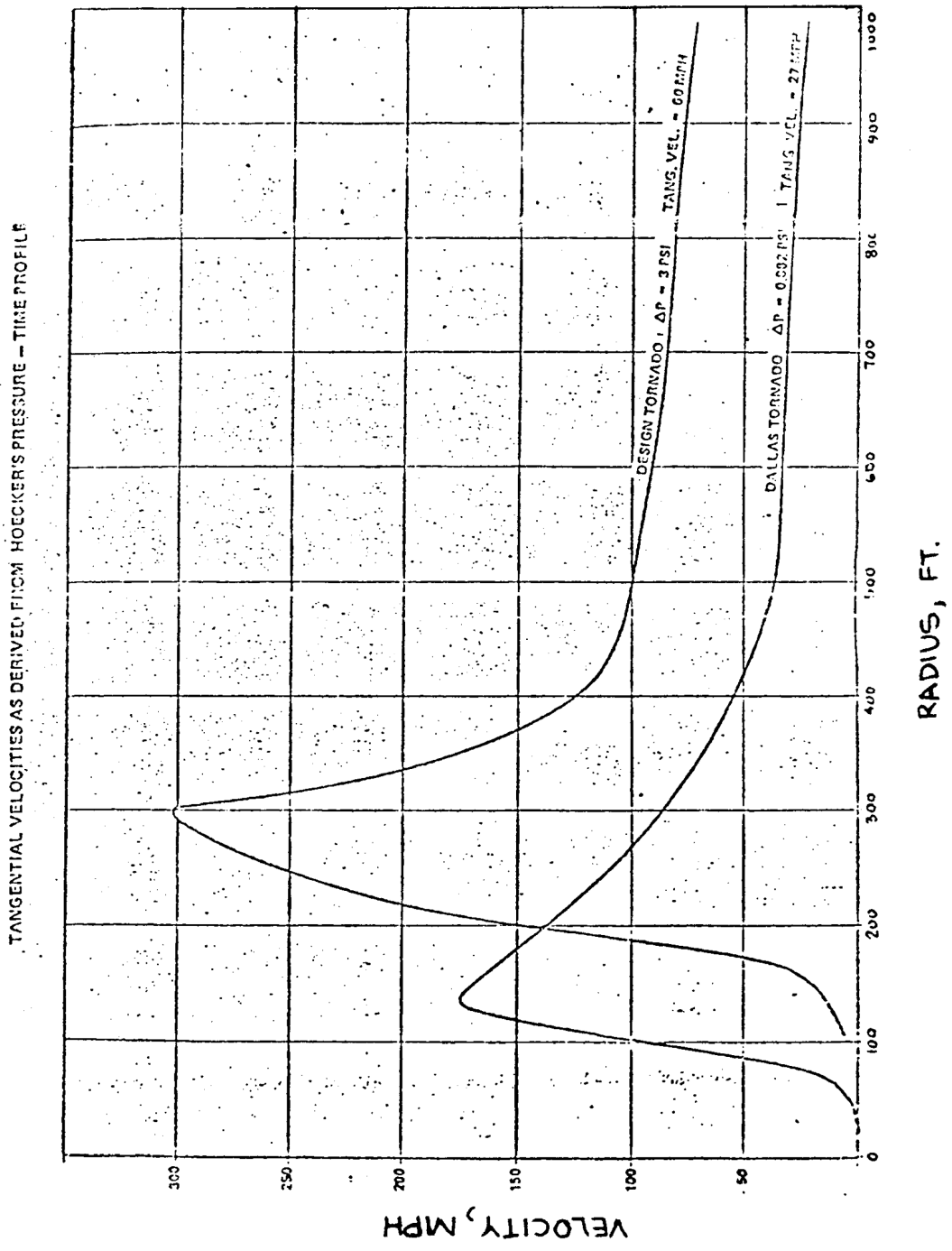

FIGURE 5.D-1

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FIGURE 5.D–2 MONTHLY WEATHER REVIEW - HOECKER

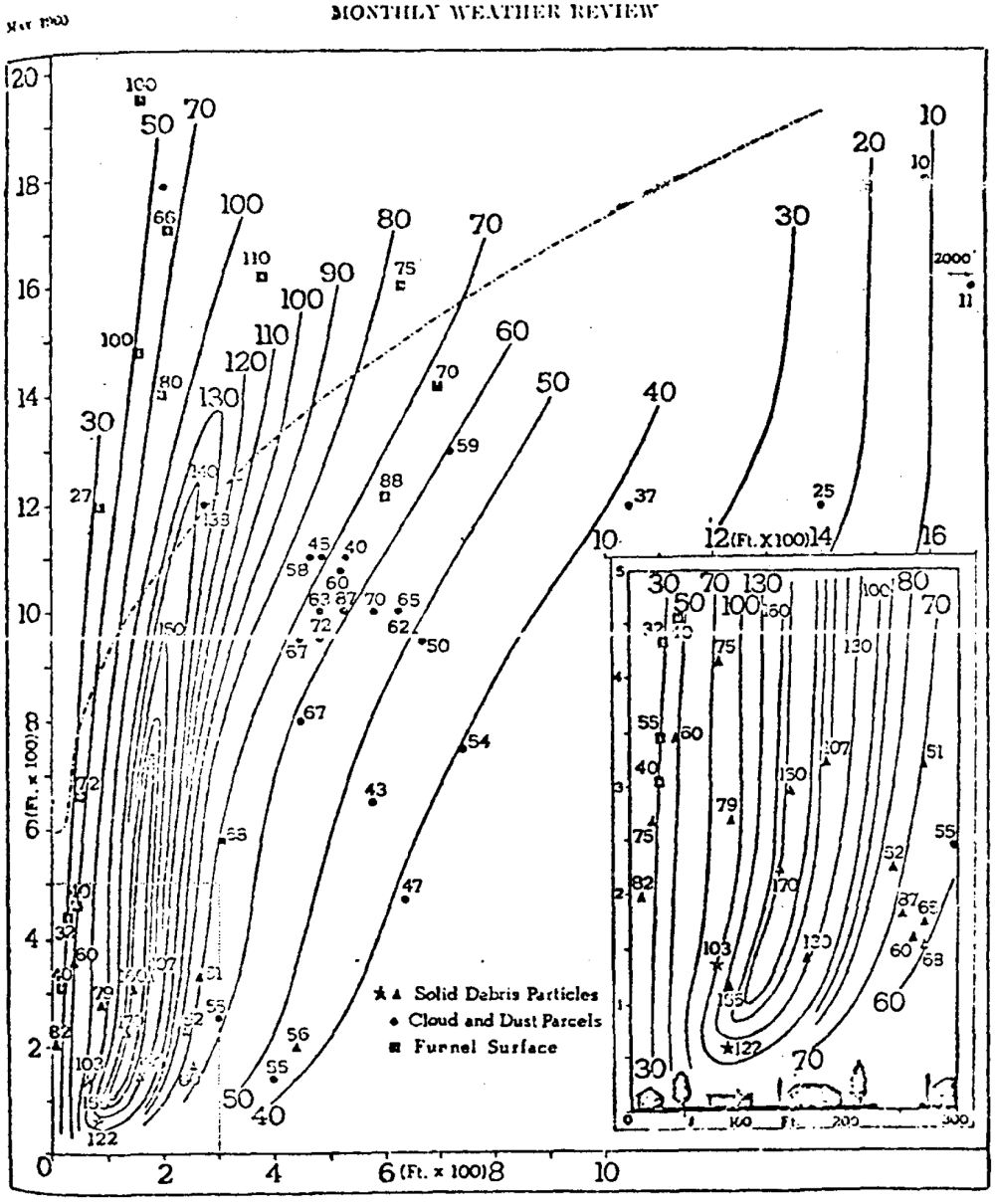

FIGURE 2.--Distribution of derived tangential speed from the center of the tornado to a radius of 2000 ft. and from near the ground to about 1500 ft. in elevation.

FIGURE 5.D–2

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FIGURE 5.D–3 TEST DATA SUMMARY

FIGURE 5.D-3

TEST DATA SUMMARY

Test No.	Test Date (July)	Target Thickness (ins.)	Missile		Velocity		Penetration		Spalling
			Length (ft)	Weight (lbs)	Electronic (ft/sec)	Movies (ft/sec)	Depth (in)	Volume (in^3)	
1	2	9	3	8.05		203	1.71	21	None
2	3	6	3	8.04	213	211	1.60	11.6	Samm & Cracks
3	3	3	3	8.05	218	213	Perforated	6.1/30.5 front/back	Maximum travel of 144 ft.
4	4	6	3	8.05	220	214	1.98	17.4	Large & cracks
5	4	6	1.4	3.78	322	312	1.68	21.6	Moderate & cracks
6	4	9	3*	8.01	235***	214	2.34	22	None
7	5	9**	3	8.04	217	216	1.84	19.9	None
8	4	6	3	8.04	150	151	1.23	7.6	Small & cracks

* Rounded end
** No reinforcing in slab
*** Believed to be bad reading due to rounded missile end.

[redacted] 10 1982

5.D.A THE DEVELOPMENT OF THE WIND-FIELD, TANGENTIAL VELOCITY

A wind-field may be generated by using the experimental findings of Hoecker (1) in the Dallas Tornado of April 1954, which was expressed in mathematical form as a "pressure-time profile" as follows:

$$p = [1 - \text{Exp}(-0.755/t)]\ D \text{ for 7.6 to 37.9 seconds} \quad (1)$$

$$p = [1 - \text{Exp}(-48.3/t^3)]\ D \text{ for 0 to 7.6 seconds} \quad (2)$$

(Intended only for all positive values of the time t)

Where:

t = time of arrival

D = total pressure drop in psf

In this publication it is clearly established that the distance from axis (R) varied from a radius of 1,500 ft at 37.9 seconds to 300 ft at 7.6 seconds, in other words for a translational velocity (V_1) of 27 mph. That is to say a relation between t and R is readily found as follows:

$$t = (R/V_1) \quad (3)$$

Therefore equations 1 and 2 may be written in terms of radius as follows:

$$p = [1\text{-Exp}(-0.775V_1/R)]D \quad (4)$$

$$p = [1\text{-Exp}(-48.3V^3{}_1/R^3)]D \quad (5)$$

The wind cyclostrophic equation was defined in the same publication as

$$dp/dR = \rho V^2/R \quad (6)$$

Where the partial differentiation of pressure to radius is equated to the mass density of the wind (ρ) times the square of the tangential velocity (V) divided by the radius (R).

By differentiating equations 4 and 5 it is found

$$dp/dR = -[\text{Exp}(-0.755\ V_1/R)](0.755V1/R^2)(D) \quad (7)$$

$$\frac{dp}{dR} = -[\text{Exp}(-48.3\ V_1^3/R^3)](144.9\ V_1^3/R^4)(D) \quad (8)$$

And substituting in equation 6 the following expressions are obtained which will relate tangential velocities to radius.

Introducing these differentiations in the cyclostrophic equation:

$$-[\mathrm{Exp}(-0.755\ V_1/R)](0.755\ V_1/R^2)(D) = \frac{V^2}{-R} \qquad (9)$$

$$-[\mathrm{Exp}(-48.3\ V_1^3/R^3)](144.9\ V_1^3/R^4)(D) = \frac{V^2}{-R} \qquad (10)$$

By assigning a constant value of 0.075/g for the mass density of air and solving for V, the following expressions are obtained:

$$V = 18\sqrt{\frac{[\mathrm{Exp}(-0.755\ V_1/R)](\ V_1)(D)}{R}} \qquad (11)$$

$$V = 249\sqrt{\frac{[\mathrm{Exp}(-48.3\ V_1^3/R^3)](\ V_1^3)(D)}{R^3}} \qquad (12)$$

This equation will give us the tangential velocities as a function of the radius. Equation 11 and 12 are applied, first to the Dallas Tornado where D = 60 mb = 0.882 psi and V_1 = 27 mph; then to the Design Tornado where D = 3 psi and V_1 = 60 mph. The results are shown in Tables 5.50-3 and 5.50-4.

Equations 11 and 12 coverage where R* = 1,240 feet when both equations give a tangential velocity of about 97 ft/sec about 66 mph which could be taken as the initial wind velocity to be considered at a radius of 1,240 feet. This will add some conservatism to our computations while obviating the use of two equations.

In other words: Starting at 1,240 foot radius with 66 mph wind is more conservative than starting at 600 foot radius with 75 mph winds, as done by WCAP 7897.

RADIAL WIND VELOCITY

The radial component of the wind will be computed using the same expression as that of WCAP except that the radius of 66 mph instead of 75 mph is chosen. So that:

$$V_2 = -\left(\frac{1240 - R}{1240 - 300}\right)R$$

Where 300 is the radius of the maximum tangential velocity.

VERTICAL COMPONENT

The vertical wind component will be taken as one third of the tangential as done previously.

$$V_3 = 1/3(V_1)$$

CONSERVATISM

A. CONSERVATISM IN THE WINDFIELD

a. Three (3) psi Total Pressure Drop

The equation of maximum tangential velocity was presented as follows:

$$V_t = 249\sqrt{\text{Exp}[-48.3(V_1/R)^3](V_1/R)^3 D} \qquad (1)$$

Where:

V_1 = translational velocity (fps)

D = total pressure drop (psf)

R = radius (ft)

Substituting $R/V_1 = t$, Equation (1) becomes

$$V_t = 249\sqrt{D}\sqrt{\frac{1}{e^{48.3/t^3}(t^3)}} \qquad (2)$$

The last term in Equation (2) maximizes at t = 3.64 seconds when its value is equal to 0.08727. Therefore:

$$V_{t(max)} = (249)(0.08727)\sqrt{D} \qquad (3)$$

From Equation (3), we relate the maximum velocities to the total pressure drops.

Pressure Drop (psi)	Pressure Drop (psf)	Maximum Tangential Velocity (fps)	Maximum Tangential Velocity (mph)
0.50	72	184.4	126
1.00	144	260.8	178
1.50	216	319.4	218

Pressure Drop (psi)	Pressure Drop (psf)	Maximum Tangential Velocity (fps)	Maximum Tangential Velocity (mph)
2.00	288	368.8	251
2.50	360	412.3	281
3.00	432	451.7	308

The tornado model, as described above, is applicable only to the Dallas Tornado as studied by Hoecker (Reference 5.D-1). Nevertheless, E. M. Brooks (Reference 5.D-9) submitted to the U.S. Weather Bureau a report in which he equated kinetic energy per unit volume to one-half the work done on the unit volume. He computed the following values:

Pressure Drop (inches Hg)	Pressure Drop (psi)	Tangential Velocity (mph)
5	2.5	285
6	3.0	316
7	3.5	348

There is exceptionally good agreement between these figures and those previously presented. One set is computed by extrapolating experimental evidence, and the other is computed theoretically.

The 308 mph and the 3 psi are the result of extrapolating Hoecker's pressure profile which shows a good agreement with experiments. Therefore, the most important parameter, the tangential velocity, is validated by both experiment and theory.

The degree of conservatism of this windfield should be recognized since the pressure drops recorded are less than the 3 psi proposed here. Dr. Reynolds, in his report (Reference 5.D-6), stated, "As far as I know, no pressure drop of as much as an inch of mercury (about 0.5 psi) has been officially recorded. Although unofficially made, spectacular atmospheric pressure measurements have been reported for tornadoes."

b. Translational Velocity of 60 mph

The conservatism of assuming a three (3) psi total pressure drop is presented above and shows that the maximum tangential velocity of 308 mph with respect to three (3) psi total pressure drop is obtained. The same tangential velocity can be obtained for tornadoes with various translational velocity by varying the radius of a tornado. If 60 mph translational velocity is used, as in the case for Millstone Unit 2, the maximum tangential velocity for a missile is more than that associated with 27 mph translational velocity, as in the case of the Dallas tornado. Therefore, the

translational velocity of 60 mph, as assumed in the analysis for Millstone is more conservative than the actual record as illustrated in the following table.

Assume a flight parameter = 0.10

Height	Tangential Velocity with 60 mph	Tangential Velocity with 27 mph
10	142	142
20	167	214
30	194	276
40	223	296
50	253	285
60	283	276
70	312	273
80	339	---
90	366	---
110	377	---

B. CONSERVATISM IN THE COMPUTATIONAL PROCEDURE

Maximum velocity and height associated with a flight parameter are obtained from the following procedures:

a. Relative Wind Velocity

It has been customary for WCAP (Reference 5.D-2) and others to consider the relative wind velocity with respect to the velocity of the flying object in the following manner:

$$\ddot{X} = \frac{C_d A}{W}\frac{\rho}{2} V_r(V_x - U_x) \qquad (4)$$

Where:

$$V_r = \sqrt{(V_x - U_x)^2 + (V_y + U_y)^2 + (V_z - U_z)^2}$$

$V_{x,y,z}$ = Wind velocities in the three coordinates

V_r = Relative wind velocity

$U_{x,y,z}$ = Object velocities in the three coordinates

$\overline{X}$ = Acceleration

$C_d = 1$

A= Projected area of missile in flight

W = Weight of missile

ρ = Density of missile

It should be apparent that, as the object reaches the wind velocities, not only will the terms in parentheses in Equation (4) decrease, but the value of V_r also will decrease.

b. Asolute or Actual Wind Velocities

It isn't quite clear that the relative velocity of the fluid should be used when the moving body is assumed to be immersed in the fluid propelling it. Gwaltney (Reference 5.D-7), when considering an object propelled by steam, uses the absolute or actual velocity of the steam in the following manner:

$$\ddot{X} = \frac{C_d A}{W}\frac{\rho}{2} V_a(V_x - U_x) \qquad (5)$$

Where:

V_a = Absolute wind velocity

$V_a = (V^2_x + V^2_y + V^2_x)^{1/2}$

Here, the absolute or actual velocity of the wind obviously will increase to its actual maximum value irrespective of the missile velocity. Consequently, the accelerations are greater.

c. Missile is Assumed to Stay in the Windfield

In the windfield presented by WCAP, at a time when the tangential velocity increases and the radial velocity decreases rapidly, the missile may leave the windfield. An analysis shows that the velocity at which the missile leaves the windfield is less than that calculated above by assuming the missile stays in the windfield. Therefore, it is more conservative to assume a missile strikes a target at its maximum velocity while it is within a windfield.

C. CONSERVATISM IN FLIGHT PARAMETER

The magnitude of the flight parameter depends essentially on the missile area projected to the wind. Accelerations and velocities are directly proportional to this area.

However, the governing parameter, as far as penetration is concerned, is a function of the weight, velocity, and impactive area of a missile.

$$P = f(W,V/A) \quad (6)$$

Where:

P = Penetration

W = Weight of missile

A = Impactive area

This function in most penetration formulae can be identified as the energy/area ratio.

When a flying object is following a spiral trajectory, it is difficult to calculate its velocity unless a common area is used. Dr. Bates (Reference 5.D-3) produced what he called the "effective area" to solve this problem. Justification for using the effective area in our computation is as follows:

The important thing to consider here is the significance of assuming one area for the flight parameter (even if it is the effective area) and another for impact. This situation is maximized in the following table:

<object>	Effective area in flight Minimum area on impact	Maximum area in flight Maximum area on impact
Plank	711,000 lb/ft	29,200 lb/ft
Pole	771,500 lb/ft	30,800 lb/ft
Rod	848,000 lb/ft	24,900 lb/ft
6 inch Pipe	732,900 lb/ft	25,700 lb/ft
12 inch Pipe	414,800 lb/ft	27,800 lb/ft

It is hypothetical to consider that a missile can maintain its maximum projected area flight and strike a target with its minimum area. Therefore, it is indicated from the above table that the assumption of using effective area in flight and minimum area on impact is a very conservative approach.

D. CONSERVATISM IN THE MAXIMUM HEIGHT

A distinction should be made between the height that a tornado missile could reach before its velocity is zero and the height at which it obtains maximum tangential velocity. The damaging effect of a missile depends on its impactive velocity and not the height it may reach.

The heights reported for Millstone are those at which the missiles are traveling at their maximum velocities.

From the analyses, we indicated that only the wooden plank would attain such a height conincident with the height of the missile proof siding. However, from a hypothetical viewpoint, should a missile be assumed to enter the spent fuel pool area through the missile proof siding, the following analysis is performed.

E. Energy Taken by Metal Siding

The formula used to determine the energy (ft-lb) taken by metal siding is given by Reference 5.D-7.

$$E = \frac{DS}{46500}\left(16,000t^2 + 1500\frac{W}{W_s}t\right) \quad (7)$$

Where:

D = Diameter of missile (inches)

S = Yield strength of target (lb/in^2)

t = Target thickness (inches)

W/W_s = Window factor = length of square side between rigid supports/length of a standard width (4 inches)

F. Striking Velocity of Missile at Water Surface

$$E_\gamma = E_i - E,\ V1 = (2E_r/m)^{1/2} \quad (9)$$

Where:

$E\gamma$ = Residual energy of a missile

E_i = Initial energy of a missile

m = Mass of a missile

G. Impactive Energy on Fuel Rack

The method used to determine the impactive energy on fuel rack is given in Reference 5.D-8.

$$\frac{E_f}{W} = \frac{1}{\varepsilon\rho_w}\left\{\left(\frac{\varepsilon\rho_w V_1^2}{2g} - 1\right)e^{-\varepsilon\rho_w H} + 1\right\} \quad (10)$$

where:

E_F = Kinetic energy at fluid depth H

W = Weight (lb)

$$E = \frac{C_d A}{W}(ft^2/lb)$$

A = Projected area (ft^2)

ρ_w = Fluid density (lb/ft^3)

C_d = Drag coefficient

g = Gravity (ft/sec^2)

H = Fluid depth (feet)

The assumed missile, namely the 1 inch rod with impactive energy of 4600 ft-lb, is analyzed with various possible trajectories entering the spent fuel pool through the missile-proof siding as shown in Figures 5.D.A–1 and 5.D.A–2.

By substituting the following values into Equation 7,

D = 1 inches

S = 50,000 lb/in^2

t = 0.2092 inches

$W/W_s = 2$

then

$$E = \frac{1.075}{\cos\Theta}\left[\frac{700.234}{\cos^2\Theta} + \frac{627.6}{\cos\Theta}\right] \quad (11)$$

where θ = angle of entry as demonstrated in Figure 5.D.A–3.

The values of energy taken by the metal siding, E, and the residual energy of the missile, E_r, with corresponding angles of entry, θ, are given in Table 5.D.A–2A, from which the following summary is drawn:

(a) At 0° entry.

The assumed missile would have no direct impact on the fuel pool. Free falling of the assumed missile is less critical than the trajectories considered below.

(b) At 45° entry.

From Table 5.D.A–2A, the residual energy for the missile is equal to 1,120 ft-lb. The missile velocity at impacting the water surface as obtained by Equation 9 is calculated to be 95 fps. Assuming the missile impacts with its minimum cross sectional area on the pool and travels vertically downward, the kinetic energy per pound impactive at the fuel rack is calculated from Equation 10 to be

$$E \div W = 66$$

Therefore E = 66(8) = 528 ft-lb for the impactive area of the rod. The kinetic energy per impactive unit area thus obtained (528 ft-lb/0.785 in^2 = 672.6 ft-lb/in^2) is the same as the design capability of fuel rack. The fuel racks are designed to withstand a fuel assembly weighing 1,300 pounds dropped from approximately 26 feet (assuming no drag resistance from the water). The resulting impactive energy thus obtained is approximately 33,800 ft-lb which when applied over an eight inch diameter area atop of the fuel rack gives the same impactive energy per unit area as the rod.

(c) At angles greater than 45°.

The residual energy of the missile impacting the siding at angles greater than 45° is less than that discussed in Item b above. As indicated in Table 5.D.A–2A, the missile would not penetrate the missile proof siding at an angle of about 55° or greater.

It is therefore concluded that even if the missile could attain such a height as to enter the spent fuel area through the missile proof siding, no possible damage to the fuel element could occur.

APPENDIX 5D

ATTACHMENT A

TABLE 5.D.A–1A RELATIVE CONSERVATISM IN STEPS A, B, AND C

F.P.	Maximum Velocities (fps)			Maximum Heights (feet)		
	(a)	(b)	(c)	(a)	(b)	(c)
0.10	250	323	377	40.7	79.9	194.8
0.09	243	314	369	36.8	74.4	181.2
0.08	236	303	360	30.0	69.0	166.8
0.07	229	291	349	28.2	61.8	150.1
0.06	220	278	336	23.7	54.6	131.4
0.05	209	262	321	18.8	45.7	110.5
0.04	196	243	301	13.3	35.6	86.7
0.03	180	220	279	7.8	24.4	59.1
0.02	159	189	246	2.1	11.4	27.8
0.015	144	170	219	0.005	4.9	6.16

(a) Relative wind velocity.

(b) Absolute wind velocity. Missile on own path.

(c) Absolute wind velocity. Missile following windfield.

APPENDIX 5D

ATTACHMENT A

TABLE 5.D.A–2A

ANGLE OF ENTRY θ	**RESISTANCE ENERGY BY SIDING, E (FT-LB)**	**RESIDUAL ENERGY, E_r (FT-LB)**
0	1430	3170
20	1670	2930
30	2060	2540
45	3480	1120
50	4460	140
55	6030	-
60	8720	-

APPENDIX 5D

ATTACHMENT A

TABLE 5.D.A–3A

ANGLE OF ENTRY θ	**RESISTANCE ENERGY BY SIDING, E (FT-LB)**	**RESIDUAL ENERGY, E_r (FT-LB)**
0	1430	3170
20	1670	2930
30	2060	2540
45	3480	1120
50	4460	140
55	6030	-
60	8720	-

FIGURE 5.D.A–1 PLAN OF SPENT FUEL AREA

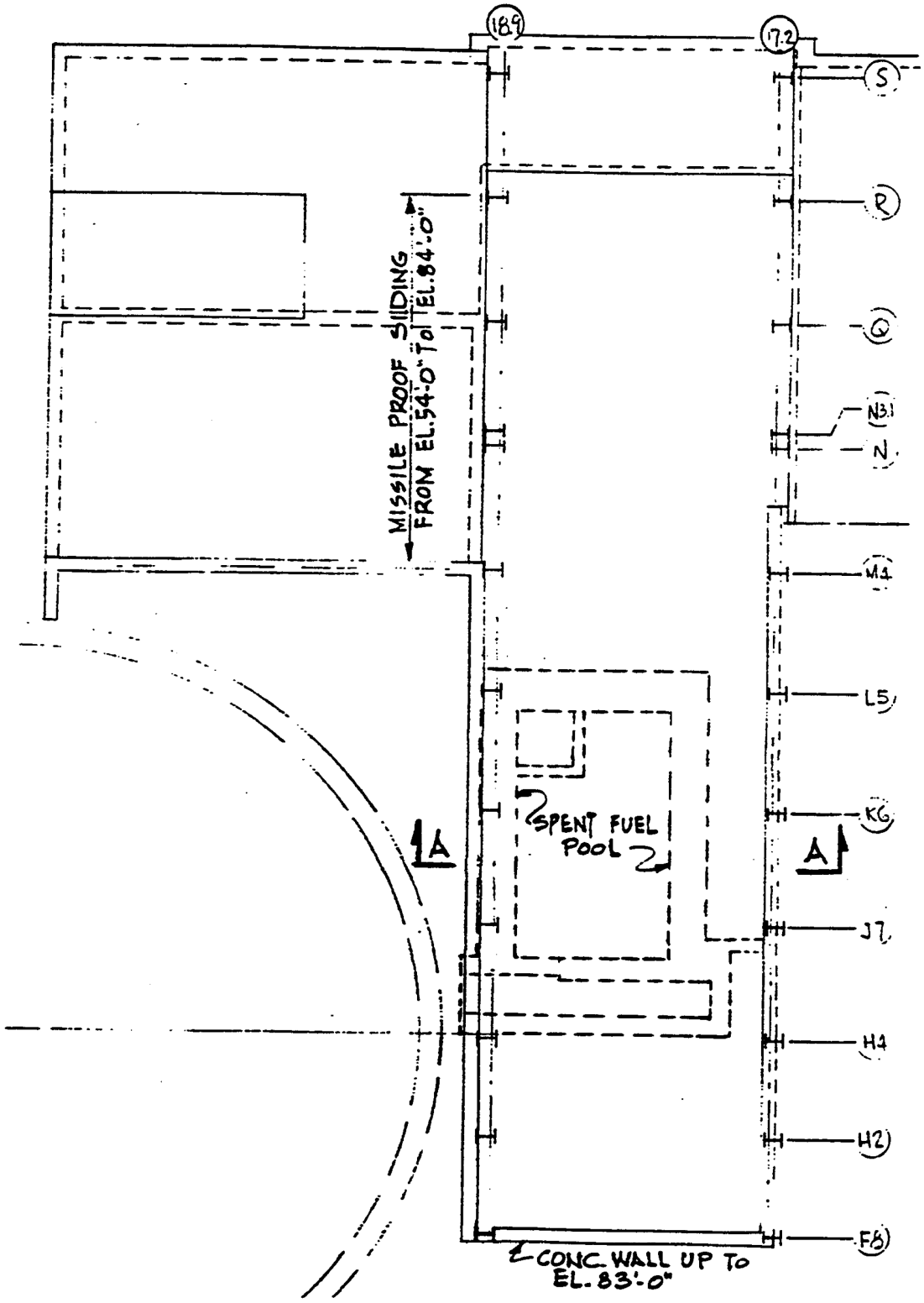


FIGURE 5.D.A–2 SPENT FUEL POOL SECTION VIEW

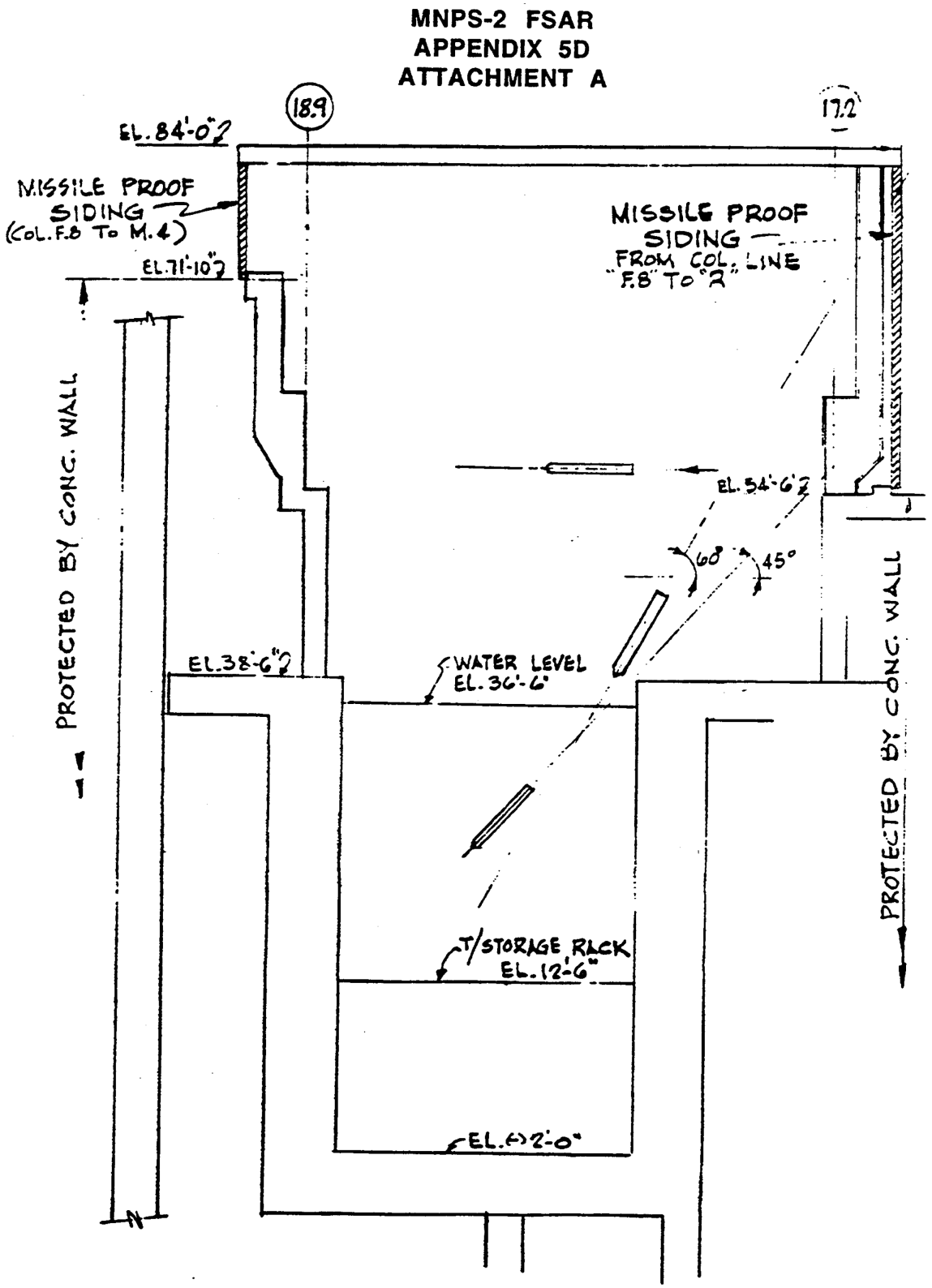
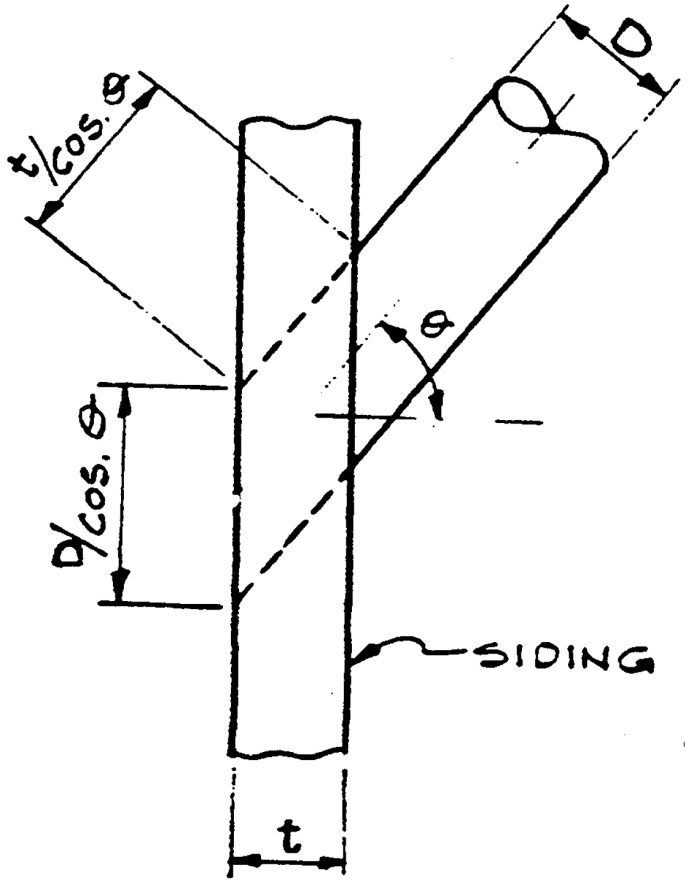


FIGURE 5.D.A–3 MISSILE STRIKE ANGLE TO MISSILE PROOF SIDING



5.E COMPUTER PROGRAM LIST AND DESCRIPTIONS

I. BLOWDOWN LOADS

The following codes provide the basis for the hydraulic forces acting on the structures during the subcooled and two phase periods of blowdown.

WATERHAMMER

a. Description

Fundamental to the determination of the mechanical loads during blowdown is the hydrodynamic solution of the fluid decompression. The major loads imposed on the reactor vessel internals occur during the subcooled period of decompression.

The digital computer program WATERHAMMER[(1)] is used to describe this subcooled response. The WATERHAMMER code is a digital computer program developed under contract to the AEC for use on the LOFT program. It is based on assumptions of one dimensionality and ignores fluid friction at the walls. These assumptions result in simplified forms of the equations of mass and momentum. These equations are transformed to their wave conjugate form and solved by the superposition method. Details of the method of solution are presented in Reference 1. The basic assumptions, which provide for convenient solution by the wave equations (Hookes' law treatment of fluid properties and neglect of conservation of energy) limit the extent of applicability of the solution to subcooled fluid conditions only.

b. Reference

1. 65-28-RA, "EARLY BLOWDOWN (WATERHAMMER) ANALYSIS FOR LOSS OF FLUID TEST FACILITY", by Stanislav Fabic, dated June 1965 and revised April 1967.

CEFLASH-4

a. Description

CEFLASH-4, is employed to provide information during the Two-phase transition period. CEFLASH-4 is a C-E modified version of the FLASH-4 [(1)] code. This code, as shown in References 2 and 3, has demonstrated applicability for the two-phase transient. It is a multinode-multiflowpath code which simultaneously solves the finite difference form of the equations of mass, momentum, and energy conservation in conjunction with tabularized values of the fluid properties. The numerical solution technique employs a backward difference integration scheme which leads to numerically stable results for the depressurization predictions.

The two codes described above provide the basis for determination of the hydraulic forces acting on the structures during the subcooled and two-phase periods of blowdown.

b. References

1. WAPD-TM-840, “FLASH-4: A FULLY IMPLICIT FORTRAN IV PROGRAM FOR THE DIGITAL SIMULATION OF TRANSIENTS IN A REACTOR PLANT”, by T. A. Porsching, J. H. Murphy, J. A. Redfield, and V. C. Davis, dated March 1969.

2. CENPD-26, Combustion Engineering Report, “Description of Loss-of-Coolant Calculational Procedures”, August 1971 (Proprietary).

3. CENPD-42, Combustion Engineering Report, “Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions With Application of Analysis to CE 800 Mwe Class Reactors”, August 1972 (Proprietary).

II. REACTOR VESSEL INTERNALS

The following codes were used in the analysis of the reactor vessel internals.

MRI/STARDYNE

a. Description and Assumptions

The program uses the finite-element method for the static and dynamic analyses of two and three dimensional solid structures subjected to any arbitrary static or dynamic loading or base acceleration. In addition, initial displacements and velocities may be considered. The physical structure to be analyzed is modeled with finite elements which are interconnected by nodes. Each element is constrained to deform in accordance with an assumed displacement field that is required to satisfy continuity across element interfaces. The displacement shapes are evaluated at nodal points. The equations relating the nodal point displacements and their associated forces are called the element stiffness relations and are a function of the element geometry and its mechanical properties. The stiffness relations for an element are developed on the basis of the theorem of minimum potential energy. Masses and external forces are assigned to the nodes. The general solution procedure of the program is to formulate the total assemblage stiffness matrix [K] and apply it to either of the following equations:

$$[k]\,(\delta) = (p) \qquad (1)$$

$$W^2[m](q) - [k](q) = 0 \qquad (2)$$

where:

(δ) = the nodal displacement vector

(p) = the applied nodal forces

[m] = the max matrix

w = the natural frequencies

(q) = the normal modes

Equation (1) applies during a static analysis which yields the nodal displacements and finite elements internal forces. Equation (2) applies during an eigenvalue/eigenvector analysis, which yields the natural frequencies and normal modes of the structural system. Using the natural frequencies and normal modes together with related mass and stiffness characteristics of the structure, appropriate equations of motion may be evaluated to determine structural response to a prescribed dynamic load.

The finite element used to date in CE analyses is the elastic beam member. The assumptions governing its use are as follows: small deformation, linear-elastic behavior, plane sections remain plane, no coupling of axial, torque and bending, geometric and elastic properties constant along length of element.

b. Extent of Program's Applications

The program is used to obtain the response of the reactor vessel internals due to prescribed seismic excitation. The structural components are modeled with beam elements. The geometric and elastic properties of these elements are calculated such that they are dynamically equivalent to the original structures. The response analysis is then conducted using both model response spectra and modal time history techniques. Both methods are compatible with the program.

c. Reference

"MRI/STARDYNE-Static and Dynamic Structural Analysis System: User Information Model", Control Data Corporation, June 1, 1970.

ASHSD

a. Description and Assumptions

The program uses a finite-element technique for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three-dimensional axisymmetric continuum is represented either as an axisymmetric thin shell or as a solid of revolution, or as a combination of both. The axisymmetric shell is discreted as a series of frustums of cones and the solid revolution as triangular or quadrilateral "toroids" connected at their nodal circles.

Hamilton's variational principle is used to derive the equations of motion for these discrete structures. This leads to a mass matrix, stiffness matrix and load vectors which are all consistent with the assumed displacement field. To minimize computer storage and execution time, the non-diagonal "consistent" mass matrix is diagonalized by adding off-diagonal terms to the appropriate diagonal terms. These equations of motion are solved numerically in time domain by a direct step-by-step integration procedure.

The assumptions governing the axisymmetric thin shell finite element representation of the structure are those consistent with linear orthotropic thin elastic shell theory.

b. Extent of Program's Applications

ASHSD is used to obtain the dynamic response of the CSB/TS system due to a LOCA. An axisymmetric thin shell model of the structure is developed. The spatial Fourier series components of the time varying LOCA loads are applied to the modeled structure. The program yields the dynamic shell and beam mode response of the structural system.

c. Reference

Ghosh, S., Wilson, E. L., "Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading", Report Number EERC 69-10, U. of California, Berkeley, September, 1969.

ICES/STRUDL - II

a. Description and Assumptions

The ICES/STRUDL-II computer program provides the ability to specify characteristics of problems - framed structures and three-dimensional solid structures, perform analyses - static and dynamic, and reduce and combine results.

Analytic procedures in the pertinent portions of ICES/STRUDL-II apply to framed structures. Framed structures are two or three dimensional structures composed of slender, linear members which can be represented by properties along a centroidal axis. Such a structure is modeled with joints, including support joints, and members connecting the joints. A variety of force conditions on members or joints can be specified. The member stiffness matrix is computed from beam theory. The total stiffness matrix of the modeled structure is obtained by appropriately combining the individual member stiffnesses.

The stiffness analysis method of solution treats the joint displacements as unknowns. The solution procedure provides results for joints and members. Joint results include displacements and reactions and joint loads as calculated from member end forces. Member results are member end forces and distortions. The assumptions governing the beam element representation of the structure are as follows: linear, elastic, homogeneous, and isotropic behavior, small deformations, plane sections remain plane, and no coupling of axial, torque and bending.

b. Extent of Program's Applications

The program is used to obtain stiffness properties of lower support structure and upper guide structure grid beams due to transverse loads. The results of the analyses are incorporated into overall reactor vessel internals' models, which calculate the dynamic response due to seismic and LOCA conditions. These latter results, at a given time, are fed back into the grid beam models to yield dynamic stresses.

c. Reference

"ICES/STRUDL-II, The Structural Design Language: Engineering User's Manual, Volume I", Structures Division and Civil Engineering's Systems Laboratory, Department of Civil Engineering, MIT, Second Edition, June, 1970.

SHOCK

a. Description and Assumptions

The computer program SHOCK solves for the response of structures which can be represented by lumped-mass and spring systems and are subjected to a variety of arbitrary type loadings. This is done by numerically solving the differential equations of motion for an nth degree of freedom system using the Runge-Kutta-Gill or a specialized Newmark integration technique. The equations of motion can represent an axially responding system or a laterally responding system; i.e., an axial motion, or a coupled lateral and rotational motion. The program is designed to handle a large number of options for describing load environments and includes such transient conditions as time-dependent forces and moments, initial displacements and rotations, and initial velocities. Options are also available for describing steady-state loads, preloads, accelerations, gaps, and nonlinear elements.

The output from the code consists of minimum and maximum values of translational and angular accelerations, forces, shears and moments for the problem time range. In addition, the above quantities are presented for all printout times requested. Plots can also be obtained for displacements, velocities and accelerations as desired.

b. Extent of Program's Applications

The program is used to obtain the transient response of the reactor vessel internals due to LOCA loads. Lateral and vertical lumped-mass and spring models of the internals are formulated.

Various types of springs - linear, compression only, tension only, or nonlinear springs - are used to represent the structural components. Thus, judicious use of load-deflection characteristics enables effects of components impacting to be predicted. Transient loading appropriate to the horizontal and vertical directions is applied at mass points and a dynamic response - displacements and internals forces - is obtained.

c. Reference

Gabrielson, V. K., “SHOCK - A computer Code For Solving Lumped-Mass Dynamic Systems”, SCL-DR-65-34, January, 1966.

SAMMSOR-DYNASOR

a. Description and Assumptions

SAMMSOR-DYNASOR provides the ability to perform nonlinear dynamic analyses of shell structures represented by axisymmetric finite-elements and subjected to arbitrarily varying load configurations.

The program employs the matrix displacement method of structural analysis utilizing a curved shell element. Geometrically nonlinear dynamic analyses can be conducted using this code.

Stiffness and mass matrices for shells of revolution are generated utilizing the SAMMSOR part of this code. This program accepts a description of the structure in terms of the coordinates and slopes of the nodes and the properties of the elements joining the nodes. Utilizing the element properties, the structural stiffness and mass matrices are generated for as many as twenty harmonics and stored on magnetic tape. The DYNASOR portion of the program utilizes the output tape generated by SAMMSOR as input data for the respective analyses.

The equations of motion of the shell are solved in DYNASOR using Houbolt’s numerical procedure with the nonlinear terms being moved to the right-hand side of the equilibrium equations and treated as generalized pseudo-loads. The displacements and stress resultants can be determined for both symmetrical and asymmetrical loading conditions. Asymmetrical dynamic buckling can be investigated using this program. Solutions can be obtained for highly nonlinear problems utilizing as many as five circumferential Fourier harmonics.

b. Extent of Program’s Application

This program is used to analyze the dynamic buckling characteristics of the core support barrel during a LOCA hot-leg break. The program’s nonlinear characteristics provide this capability.

A finite element model of the CSB is formulated which is consistent with the computer program. Taking into account the initial deviation of the structure and the shell mode which is most likely to give the minimum critical pressure, the time-dependent pressure load is applied to the barrel. The maximum displacement occurring in the barrel is obtained. This result is used as a basis for an overload analysis to satisfy ASME Code requirements.

c. References

Tillerson, J. R. and Haisler, W. E., “SAMMSOR II - A Finite Element Program to Determine Stiffness and Mass Matrices of Shells of Revolution”, Texas A&M University, TEES-RPT-70-18, October, l970.

Tillerson, J. R. and Haisler, W. E., “DYNASOR II - A Finite Element Program for the Dynamic Non-Linear Analysis of Shells or Revolution”, Texas A&M University, TEES-RPT-70-19, October, 1970.

SAAS

a. Description and Assumptions

The program performs finite element static analyses of axisymmetric solids. The continuous body to be analyzed is replaced by a system of ring elements with triangular or quadrilateral cross sections. The elements are interconnected at their apexes which are referred to as nodes. The displacement method of finite element analysis is used to derive the element stiffness matrix. This method proceeds by selecting a displacement expansion over the element, consistent with elemental boundary conditions, and assuming displacements in the interior of the element depend only on nodal quantities. The elemental stiffness matrices are computed and combined to yield the total stiffness matrix of the modeled structure. The principle of minimum potential energy is then applied to yield displacements and elemental forces (stresses). Since these elements are of relative arbitrary shape, the procedure can be applied to bodies of complex geometry. The program performs static analyses due to both boundary forces and thermal loads by converting these effects into equivalent nodal quantities.

Assumptions governing the use of the aforementioned finite elements are those consistent with linear elasticity theory of solid structures.

b. Extent of Program’s Applications and Assumptions

The program is employed to determine the stiffness properties of flanged regions as related to axisymmetric loads. Specifically, the CSB upper and lower flanges with connecting cylinders and the UGSP flange with connecting cylinder were analyzed. Displacements due to a known external load were determined. The resulting stiffnesses were incorporated into an overall model of the reactor vessel internals, which were employed in determining the dynamic response during vertical seismic and LOCA excitation. The results of the dynamic analyses were fed back into the flange analyses to determine their maximum stresses and deformations.

Instrumentation flanges, which are used as primary pressure boundaries were analyzed with the subject program to determine their acceptability with respect to ASME Section III criteria.

c. Reference

Wilson, E. L., Jones, R. M., “Finite Element Stress Analysis of Orthotropic, Temperature-Dependent Axisymmetric Solids of Revolution”, Aerospace Report Number TR-0158 (S3816-22)-1, September, 1967.

NAOS

a. Description and Assumptions

The program applies the finite element analysis to axisymmetric solids subjected to arbitrary nonaxisymmetric loadings by expanding the various kinematic and forcing functions into Fourier series.

The continuous body to be analyzed is replaced by a system of ring elements with triangular or quadrilateral cross sections and/or thin conical shell elements. The elements are interconnected at their apexes and ends (for the case shell elements) which are referred to as nodes. The displacement method of finite element analysis is used to derive the element stiffness matrices. This method proceeds by selecting a displacement expansion over the elements, consistent with elemental boundary conditions and assuming displacements in the interior of the element depend only on nodal quantities. The elemental stiffness matrices are computed and then combined to yield the total stiffness matrix of the modeled structure. The principle of minimum potential energy is then applied to obtain displacements and element forces (stresses). The program performs static analyses due to both boundary forces and thermal loads by converting these effects into equivalent nodal quantities. Since the elements employed are of relatively arbitrary shape, the procedure can be applied to bodies of complex geometry.

Assumptions governing the aforementioned analyses are those consistent with linear elasticity theory of solids and thin shell structures.

b. Extent of Program’s Applications

The program is employed to determine the stiffness properties of flanged regions due to lateral loads. Specifically, the CSB upper and lower flanges with connecting cylinders and the UGSP flange with connecting cylinders were analyzed for lateral shear and bending moment loads. Displacements due to known magnitudes of these loads were determined. The resulting stiffnesses were incorporated into overall models of the reactor vessel internals which were employed in determining the dynamic response during horizontal seismic and LOCA excitation.

The results of the dynamic analysis were fed back into the flange analyses to determine maximum stresses and deformations.

c. Reference

Dunham, R. S., Nickell, R. E., et. al., “NAOS - Finite Element Analysis of Axisymmetric Solids with Arbitrary Loadings”, Structural Engineering Laboratory, University of California, Berkeley, California, June, 1967.

EAC/EASE

a. Description and Assumptions

The EAC/EASE computer program provides static structural analyses of linear, three-dimensional systems subjected to sets of arbitrarily prescribed mechanical and thermal loads and displacement boundary conditions. The analysis used in the program is an application of the direct stiffness method.

As the first step, the actual system is approximated by an assemblange of discrete structural elements interconnected at a finite number of points called nodes. The behavior of the “discretized structure” is an approximation to the response of the real system.

Next, each element is constrained to deform in accordance with an assumed displacement field that is required to satisfy continuity across element interfaces. The displacement shapes are evaluated at convenient locations within the element (usually at points on element boundaries), and their amplitudes are sometimes called “generalized coordinates”.

The equations relating generalized coordinates and their associated forces are called the element stiffness relations and are a function of the element geometry and its mechanical properties. The stiffness relations for an element are developed on the basis of a governing variational principle, the theorem of minimum potential energy.

The “complete stiffness” of the discretized structure is obtained in the following manner: The element equations are combined into the system equations using the requirement that the summation of all internal forces contributed from those elements common to a particular node must be equal to the externally applied load at that joint. The resulting set of equations (appropriately modified for displacement boundary conditions) represent the equilibrium equations for the discretized structure connected only at the joints.

The solution of the equilibrium equations results in six displacement components at each node arising from the loads applied to the system. Having defined the displacements at all nodes in the system, the internal stresses are calculated from displacements for each element in the structure.

The elements used to model structures are the triangular plate membrane and bending elements. The stiffness relations for the elements are developed according to the following assumptions: small deformations, linear-elastic isotropic behavior, uniform element thickness, negligible through-the-thickness stress, normals to the original mid-surface do not distort and remain normal to the deflected mid-surface (applies to plate bending element).

b. Extent of Program's Application

The program is used to perform thermal stress analysis of the core shroud. A symmetrical section of the core shroud is modeled with triangular plate membrane and bending elements. The thermal load is applied by specifying temperatures at each of the nodal points. The results of the analysis are compared to stress criteria defined in Section III of the ASME Boiler and Pressure Vessel Code.

c. Reference

“EAC/EASE - Elastic Analysis for Structural Engineering: User's Information Manual”, Control Data Corporation, April, 1971.

III. REACTOR COOLANT SYSTEM

The computer programs that were used in the dynamic seismic analysis of the reactor coolant system components, as discussed in Section 4.A.2.3, Appendix 4.A, of the FSAR, include:

ICES/STRUDL-11

a. Description and Assumptions

The ICES/STRUDL-II computer program provides the ability to specify characteristics of problems - framed structures and three-dimensional solid structures, perform analyses - static and dynamic, and reduce and combine results.

Analytic procedures in the pertinent portions of ICES/STRUDL-II apply to framed structures. Framed structures are two or three dimensional structures composed of linear members which can be represented by properties along a centroidal axis. Such a structure is modeled with joints, including support joints, and members connecting the joints. A variety of force conditions on members or joints can be specified. The member stiffness matrix of the modeled structure is obtained by appropriately combining the individual member stiffnesses.

Masses may be specified for selected joint degrees-of-freedom represented in the total stiffness matrix. The total stiffness matrix is then “condensed” to yield a dynamic stiffness matrix in which only those joint degree-of-freedom for which mass is specified are retained. Using the (condensed) dynamic stiffness matrix and the associated diagonal mass matrix, an eigenvalue solution is performed by a diagonalization process (Jocobi's method) to yield the natural frequencies and mode shapes corresponding to the free vibrations of the structure.

Using the total stiffness matrix, the program will also solve the general statics problem to yield influence coefficients which relate member end forces and moments and support reactions to unit displacement imposed, in turn, at each of a designated group of joint degrees-of-freedom. In addition, the program will condense the total stiffness matrix to

yield a set of stiffness coefficients which relate the forces corresponding to selected degrees of freedom of one group of designated joints to imposed displacements corresponding to selected degrees-of-freedom of a second group of designated joints.

The stiffness analysis method of solution used in STRUDL treats the joint displacements as unknowns. The solution procedure provides results for joints and members. Joint results include displacements and reactions and joint loads as calculated from member end forces. Member results are member end loads and distortions. The assumptions governing the beam element representation of the structure are as follows: linear, elastic, homogeneous, and isotropic behavior, small deformations, plane sections remain plane, and no coupling of axial, torque and bending.

b. Extent of Program’s Application

The program is used to define the dynamic characteristics of the structural models used in the dynamic seismic analyses of the reactor coolant system components. The natural frequencies and mode shapes of the structural models and the influence coefficients which relate member end forces and moments and support reactions to unit displacements are calculated. The influence coefficients are calculated for each dynamic degree-of-freedom of each mass point and for each degree-of-freedom of each support point at which relative motion is imposed. In addition, stiffness coefficients are calculated which relate the forces corresponding to those joint degree-of-freedom for which mass is specified to the imposed displacements corresponding to those (support) joint degrees-of-freedom at which relative motion will be specified during subsequent seismic response calculations. As appropriate, these data are stored for later use in response spectra or time-history seismic response calculations (see Appendix 4.A of the FSAR).

c. Reference

“ICES/STRUDL-II, The Structural Design Language: Engineering User's Manual, Volume I”, Structures Division and Civil Engineering's Systems Laboratory, Department of Civil Engineering, MIT, Second Edition, June, 1970.

TMCALC

a. Description and Assumptions

Using normal mode theory and Newmark’s Beta-Method, with Beta equal to 1/6, the C-E computer program TMCALC solves the differential equations of motion which represents the dynamic characteristics of a singly or multiply supported, multi-degree-of-freedom linear structural system subjected to seismic excitations. The program provides for separate, time-dependent, inputs at each support point at which relative motion produced by the seismic event may be imposed. In the step-by-step numerical integration process (Newmark's Beta-Method) employed by TMCALC, the time step selected is less than one-tenth of the period of the highest frequency mode.

Inputs to TMCALC consist of:

1. Output from STRUDL:

   a. frequencies,

   b. mode shapes,

   c. stiffness coefficients which relate mass point degree-of-freedom forces to support point degree-of-freedom (relative) displacements.

2. Digitized time histories which describe the seismic event in terms of time-dependent motion imposed at the support points of the structural system. These consist of time histories of absolute accelerations at the reference support point of the system and corresponding time histories of relative displacements at all nonreference support points at which relative motion is imposed (see Appendix 4.A of the FSAR).

   The output from TMCALC consists of complete time histories of absolute accelerations, relative velocities and relative displacements corresponding to each dynamic degree-of-freedom of each of the mass points of the structural system. These data are calculated at each point in time during the integration process for the entire duration of the seismic event.

   The formulation of the program assumes linear elastic behavior of the structure and a linear variation in accelerations over an integration time step.

b. Extent of Program's Application

The program is used to calculate the dynamic response of structural models used in the dynamic seismic analysis of the reactor coolant system components. These data include time histories of absolute accelerations, relative velocities and relative displacements corresponding to each dynamic degree-of- freedom of the structural system. The data are stored for use in subsequent seismic response calculations.

c. References

1. Przemieniecki, J. S., "Theory of Matrix Structural Analysis", Chapter 13, McGraw-Hill Book Company, New York, New York, 1968.

2. Hurty, W. C., and Rubinstein, M. F., "Dynamics of Structures", Chapter 8, Prentice Hall, Inc., Englewood Cliffs, New Jersey, 1964.

3. Newmark, N. M., "A Method of Computation for Structural Dynamics", Volume 3, Journal of Engineering Mechanics Division, A.S.C.E., July, 1959.

FORCE

a. Description and Assumptions

The formulation of the computer program FORCE assumes a linear elastic structural system modeled as a three-dimensional assemblage of joints, or modes, which are interconnected by elastic structural elements or members. The program calculates a discrete time history of the loads, forces and moments, at each designated member end and the reactions at each designated support joint, induced by specified seismic conditions (see Section 4.A.2.3.4, Appendix 4.A of the FSAR). The program selects the maximum absolute values of each component of load at each member end and at each support joint, and the times at which the maximum values occur, over the entire duration of the specified seismic event.

As input, the program FORCE uses a matrix of influence coefficients calculated by ICES/ STRUDL-II, the dynamic response of the structure, i.e., the time history of relative displacements corresponding to each mass joint degree-of-freedom, as calculated by the program TMCALC, and the time histories of relative displacements prescribed for each support joint degree-of-freedom for the seismic event under consideration. With this input, the program FORCE forms appropriate linear combinations of the time-dependent relative displacements to yield a complete loads analysis of the deformed shape of the structure at each point in time over the entire duration of the seismic event.

b. Extent of Program's Application

The program is used to calculate the time-dependent response of the reactor coolant system components to specified seismic conditions.

c. References

Przemieniecki, J. S., "Theory of Matrix Structural Analysis", McGraw-Hill Book Company, New York, New York, l968.

SHAKE

a. Description and Assumptions

The computer program SHAKE performs a normal mode response spectrum analysis of a three-dimensional linear elastic structural system modeled as an assemblage of joints, or modes, which are interconnected by elastic structural elements, or members. In the formulation, mass is assumed "lumped" at selected joints, each of which may have up to three translational dynamic degrees-of-freedom.

Input to SHAKE consists of frequencies and mode shapes, corresponding to each normal mode of vibration of the structure, the relevant diagonal mass matrix, and the response spectrum value, acceleration, corresponding to the period of each normal mode. The

output from SHAKE consists of the modal inertial loads, forces, corresponding to each mass joint dynamic degree-of-freedom of the structure, for each normal mode.

b. Extent of Program's Application

The program SHAKE is used to calculate the dynamic response, modal inertial loads, for the reactor coolant system components in those cases where spectrum analysis is applied (see Appendix 4.A of the FSAR). In turn, the modal inertial loads are applied to the structure, mode-by-mode, using the ICES/STRUDL-II program, which calculates the member end loads and support joint reactions for each mode, and combines the modal values to give the total response to the specified seismic event.

c. References

Hurty, W. C., and Rubinstein, M. F., "Dynamics of Structures", Prentice Hall, Inc., Englewood Cliffs, New Jersey, 1964.

IV. REACTOR COOLANT SYSTEM - STATIC STRUCTURAL ANALYSIS

MEC-21

a. Description and Assumptions

The program is designed to compute the reactions and stresses in complex piping systems, including closed-loop configurations, due to thermal expansion and contraction; pressure effects; concentrated loads such as valves, fittings and fixtures; and uniform loads such as weight. The computational method is a tensor analysis treatment of Castigliano's theorem and assumes linear elastic behavior.

b. Extent of Program's Application

The program is used extensively to calculate the loads which act on the components and supports of the reactor coolant system for the various operating conditions. The calculated loads are included in the equipment specifications and, subsequently, used in the design calculations performed for the individual components.

c. References

1. J. A. Olson and R. V. Cramer, "Pipe Flexibility Analysis Program MEL21", Report Number 10-66, Rev. 1, "A Modification of Program MEC21S", San Francisco Bay Naval Shipyard, Mare Island Site, Vallejo, California, August 26, 1969.

2. J. A. Olson and R. V. Cramer, "Pipe Flexibility Analysis Program MEC21S", Report Number 35-65, San Francisco Bay Naval Shipyard, Vallejo, California, November 22, 1965.

3. James Griffin, MEC21 7094, "A Piping Flexibility Analysis Program for the IBM-7090 and 7094", Los Alamos Scientific Laboratory, Report LA-2929, Los Alamos, New Mexico, July 14, 1964.

V. ASME SECTION III, CLASS I COMPONENTS

The following programs were used to determine stresses in ASME Section III, Class I components.

SAAS

a. Description and Assumptions

The program performs finite element static analyses of axisymmetric solids. The continuous body to be analyzed is replaced by a system of ring elements with triangular or quadrilateral cross sections. The elements are interconnected at the apexes which are referred to as nodes. The displacement method of finite element analysis is used to derive the element stiffness matrix. This method proceeds by selecting a displacement expansion over the element, consistent with elemental boundary conditions, and assuming displacements in the interior of the element depend only on nodal quantities. The elemental stiffness matrices are computed and combined to yield the total stiffness matrix of the modeled structure. The principle of minimum potential energy is then applied to yield displacements and elemental forces (stresses). Since these elements are of relative arbitrary shape, the procedure can be applied to bodies of complex geometry. The program performs static analyses due to both boundary forces and thermal loads by converting these effects into equivalent nodal quantities.

Assumptions governing the use of the aforementioned finite elements are those consistent with linear elasticity theory of solid structures.

b. Extent of Program's Application

The program is used to determine stresses in the primary head-tube sheet-secondary shell regions of the steam generators.

c. References

Wilson, E. L., Jones, R. M., "Finite Element Stress Analysis of Orthotropic, Temperature-Dependent Axisymmetric Solids of Revolution", Aerospace Report Number TR-0158 (S3816-22)-1, September, 1967.

SEAL SHELL - 2

a. Description and Assumptions

The program uses the stiffness matrix method to solve the equations of thin elastic shells of revolution. The formulation assumes homogeneous, isotropic, elastic material properties. Thickness, radii of curvature, applied surface loads, temperature and material properties may vary along the generating curve of the shell, but not around its circumference. Circumferential uniformly distributed, line-loads and moments can be applied. In the formulation, the stiffness matrix is calculated from strain energy considerations by use of the principal of virtual work. Thick shell effects are included by using known results from the theory of beams to determine the appropriate contribution to the strain energy from the normal stress and shear deflections.

b. Extent of Program's Applications

The program is used to determine stresses and deformations in various axially-symmetric regions of Class I components.

c. References

"Seal-Shell-2-A Computer Program for the Stress Analysis of a Thick Shell of Revolution with Axisymmetric Pressures, Temperatures, and Distributed Loads", by C. M. Friedrich, WAPD-TM-398, UC-38: Engineering and Equipment, TID-4500, 24th Edition.

ANALYSIS OF AXISYMMETRIC SOLIDS

a. Description and Assumptions

The finite element method is applied to the determination of stresses and displacements in axisymmetric solids of arbitrary geometry subjected to thermal and mechanical loadings. The formulation is based upon energy principals and assumes linear elastic, isotropic materials.

b. Extent of Program's Application

The program is applied in the analysis of the regions of reactor vessel-to-vessel head bolted closure, and other axisymmetric regions of Class I components.

c. References

E. L. Wilson, "Analysis of Axisymmetric Solids", University of California, February, 1967.

WIN - 12100

a. Description and Assumptions

The CE computer program WIN–12100 is a general purpose thermal analysis program which determines transient and steady-state temperature distributions in physical systems.

The physical system, or structures analyzed may have irregular geometries, may be composed of several different materials and may be subjected to varying, time dependent, temperature transients at the boundaries. The formulation of the program is based upon the general finite difference equation of heat flow which is solved by relaxation techniques. Provisions are included to accommodate radiation, convection and conduction modes of heat transfer and to include internal heat generation.

b. Extent of Program's Application

The program is used extensively to determine the temperature distributions which are subsequently used in stress analysis of Class I components.

c. References

Computer Program Number WIN - 12100, “Heat Transfer by Relaxation”, T. R. McCormack, Combustion Engineering, Inc., Windsor, Connecticut, 1968.

Stanley Hellman, George Habetler and Harold Babrov, “Use of Numerical Analysis in Transient Solution of Two-Dimensional Heat Transfer Problems with Natural and Forced Convection”, ASME Transactions, 1956.

TAP

a. Description and Assumptions

The program uses finite element techniques in obtaining numerical solutions to the general, three-dimensional, time-dependent differential equation of heat conduction. The formulation is general with respect to geometry, material properties and boundary conditions, permitting the solution of complex transient and steady-state heat transfer problems.

b. Extent of Program’s Application

The program is used to obtain temperature distributions in the reactor coolant pump case-cover assembly for subsequent use in stress analysis.

c. References

Peterson, F. E., Hui, H., “Three Dimensional Transient Heat Transfer Using a Finite Element Procedure”, Theoretical Basis and Sample Solutions, Engineering/Analysis Corporation, 1611 South Pacific Coast Highway, Redondo Beach, California, August, 1971.

SOLIDS II

a. Description and Assumptions

The program is a finite element formulation for the analysis of axisymmetric and plane solids with variable orthotropic, temperature-dependent material properties. The program accommodates both axisymmetric and non-axisymmetric loadings. The formulation assumes linear elastic behavior of materials.

b. Extent of Program's Application

The program is used to determine both thermal and mechanical stresses in the axisymmetric regions of the reactor coolant pumps.

c. References

Crose, J. C. and Jones, R. M. "Finite Element Stress Analysis of Axisymmetric and Plane Solids with Different Orthotropic, Temperature-Dependent Material Properties in Tension and Compression", TR-0059(56816-53)-1, The Aerospace Corporation, San Bernardino, California, June 1971.

BJS

a. Description and Assumptions

The program is a general purpose, three-dimensional, finite element formulation for the analysis of elastic structures of arbitrary shape sustaining arbitrary loadings and temperature distributions. The program BJS was adapted from the program SAP (September 1970 version) by the Engineering/Analysis Corporation, 1611 South Pacific Coast Highway, Redondo Beach, California for the Byron Jackson Pump Division, Borg-Warner Corporation, P. O. Box 2017, Terminal Annex, Los Angeles, California. The program assumes linear elastic behavior of materials and runs on the CDC-6600 Computer System.

b. Extent of Program's Application

The program is used for stress analysis of the reactor coolant pump casings.

c. References

"BJS A First Generation Three Dimensional Finite Element Computer Program for the Analysis of Structures of Arbitrary Geometry", Byron Jackson Pump Division, Borg-Warner Corporation, P. O. Box 2017, Terminal Annex, Los Angeles, California.

Wilson, E. L., "SAP A General Structural Analysis Program", Structural Engineering Laboratory, University of California, Berkley, California (September 1970).

VI. CLASS I PIPING SYSTEM

The following codes are used in the analysis of the Class I piping systems.

MRI/STARDYNE

a. Description and Assumptions

Refer to Section II, Reactor Vessel Internals, of this question.

b. Extent of Program's Applications

The program is used to obtain the responses of Class I piping system using three-dimensional model. The model response spectrum technique is used to compute forces, moments and displacements at each point specified in the piping system.

ADLPIPE

a. Description and Assumption

The program provides an elastic analysis of redundant piping systems subjected to thermal, static and dynamic loads. The system may contain a number of sections, a section being defined as a sequence of straight and/or curved members lying between two points.

The basic approach to be used in computing the response of piping systems to ground spectra consists of generating the dynamic properties of the system and applying the modal super-position method or normal mode method to obtain the structural response.

The two eigenvalve routines used in ADLPIPE are the Jocobi rotation scheme and the Givens-Householder scheme; the later has been modified to incorporate a suggestion made by Wilkinson.

b. Extent of Program's Application

The program is used to obtain the modal responses of Class I piping by the response spectrum technique.

c. References

1. Greenstadt, J., "The Determination Characteristic Roots of a Matrix by the Jocobi Method", John Wiley, New York, 1959.

2. Wilkinson, "The Algebraic Eigenvalve Problem".

NUPPIE II

a. Description and Assumptions

The program provides an elastic analysis of redundant piping systems subjected to thermal, static and dynamic loads. The system may contain a number of sections, a section being defined as a sequence of straight and/or curved members lying between two points.

The basic approach to be used in computing the response of piping systems to ground spectra consists of generating the dynamic properties of the system and applying the modal super-position method or normal mode method to obtain the structural response.

b. Extent of Program's Application

The program is used to structurally analyze Class I piping systems. It performs static analysis as well as obtains modal responses by the modal analysis response spectra method and dynamic response to time history loads using the time history modal superposition method.

VII. CLASS I STRUCTURES

The following codes were used in the dynamic analysis of the Class I Structures.

CE 309 STRESS

a. Description and Assumption

STRESS (Structural Engineering Systems Solver) is a programming system for the solution of structural engineering problems. It was originally developed at M.I.T. in 1964 and intended for IBM-7094 application. In 1968 Bechtel implemented the program into GE-600 Series. The program is capable of executing a large variety of structural problems in two or three dimensional structures with different joint conditions. The program is primarily used to obtain the stiffness characteristics of a structure. Stiffness matrix of the structure obtained is used in the dynamic analysis.

b. Extent of Program's Applications

The program is used to obtain the stiffness matrix of the containment structure and containment internal structures.

c. Reference

Fenves, S. J., Logcher, R. D., and Mauch, S. P., "STRESS Reference Manual", The M.I.T. Press, Cambridge, Massachusetts, 1964.

CE 611 TIME-HISTORY ANALYSIS

a. Description and Assumptions

This program performs the time history analysis of a structure subjected to an earthquake motion. The analytical technique is based on modal synthesis. The general solution of the program is to formulate the equation of motion of the structure in terms of its mode shapes, frequencies and mass distribution as follows:

$$q_j + 2B_j w_j q_j + W_j^2 q_j = -F_j U_g \ (j=1,2....N)$$

where:

N = member of modes

q = generalized coordinates

B = modal damping

w = frequency

$$F_j = \frac{[\varnothing]T}{M_j}[M]$$

[Ø] = mode shape matrix

[M] = mass matrix

M = mass

U_g = input acceleration

The above equation is solved using Runge-Kutta method.

b. Extent of Program's Application

The program is used to generate acceleration time history at all Class I equipment locations in the containment, auxiliary building (including warehouse portion), turbine building and intake structure.

c. References

Hildebrand, F. B., "Introduction to Numerical Analysis," McGraw-Hill Book Company, 1956.

Kuo, S. S., "Numerical Methods and Computers," Addition Wesley Publishing Company, 1965.

Hurty, W. C., Rubinstein, M. F., "Dynamics of Structures," Prentice Hall, Inc., 1964.

Biggs, J. M., "Introduction to Structural Dynamics," McGraw-Hill Book Company, 1964.

CE 617 MODES AND FREQUENCIES EXTRACTION

a. Description and Assumptions

The program provides a means for obtaining the natural frequencies, w, and modes shapes, Ø, of structural models. The structural input consists of the models lumped masses and either the stiffness or flexibility matrix. If the flexibility matrix is entered, the program provides an option for automatic inversion to a stiffness matrix.

The program uses the method of diagonalization by successive rotations. A detailed description of the method can be found in Engineering Analyses, A Survey of Numerical Procedures, by S. Crandall, published by the McGraw Hill Book Company, New York, 1966.

The modes shapes can be printed out on an optional basis for normalization such that either $[Ø]^T$ [m\] [Ø] = [1] or with respect to a unit relative deflection of some arbitrary point. It is recommended to use the first scheme for complex structures where it is not known which mass point will show the largest activity. Otherwise the second scheme may result in the selection of a nodal point with virtually no activity. For purposes of plotting the modes and to provide a quick inspection of modal behavior, a third scheme is automatically provided to normalize each mode with respect to its largest relative value. These modes should not be further utilized.

As a last step an orthogonality check is provided by forming the product $[Ø]^T$ [m\] [Ø]. The resulting product should show the off diagonal terms virtually zero in comparison to the main diagonal. This automatic check should always be reviewed.

b. Extent of Program's Applications

The program is used to obtain the mode shapes and frequencies containment structure and containment internal structures.

c. Reference

S. Crandall, S., "Engineering Analyses, A Survey of Numerical Procedures," McGraw Hill, 1966.

CE 641 RESPONSE SPECTRUM TECHNIQUE

a. Description and Assumption

The program assumes that the structure has been previously analyzed and that the natural frequencies, w, and mode shapes Ø, for the structure has been obtained. The earthquake input is described in terms of response spectrum curves associated with different damping

values. Acceleration values are obtained from the curve for each mode corresponding to its natural frequency and damping value.

Fore each mode j, a modal inertial force at each mass point is calculated as follows:

$$Fij = \frac{Fj\phi ij\ Mi}{\sum^{\eta} \phi ij\ M_i}$$

$$Fj = \frac{\left(\sum^{\eta} \phi ij\ Mi\right)^2}{\sum^{\eta} \phi 2\ jM\ i\ i}\ Aj$$

where:

$$Fj = \frac{\left(\sum^{\eta} \phi ij\ Mi\right)^2}{\sum^{\eta} \phi 2\ jM\ i\ i}\ Aj$$

A_j = special acceleration for mode j

Then static analysis is used to obtain modal shear, moment and displacement by applying the inertial force at each mass center. Model responses are combined on an absolute sum basis.

b. Extent of Program's Application

The program is used to obtain the model responses of the containment structure and containment internal structures by the response spectrum technique.

CE 784 RESPONSE SPECTRUM TECHNIQUE

a. Description and Assumption

This program combines the previously described programs, CE 309, CE 617 and CE 641, and calculates the dynamic responses of a structure. The program utilizes the mathematical model of the structure which is represented by its cross sectional properties and lumped masses. The program then forms the structure stiffness matrix. By using the modified Jocobi method of symmetric matrix diagonalization, natural frequencies and normal modes of the structure are obtained. Using the natural frequencies and normal modes together with input acceleration, modal responses of the structure are calculated.

b. Extent of Program's Applications

The program is used to obtain the modal responses of the auxiliary buildings (including warehouse portion) turbine building and intake building.

c. References

1. Gere and Weaver, "Analysis of Framed Structures," Van Nostrand, 1968.

2. Weaver, W. J., "Computer Programs for Structural Analysis," Van Nostrand, 1967.

3. Biggs, J. M. "Introduction to Structural Dynamics," McGraw Hill, 1964.

4. Ralston, A., Wilf, H. S., "Mathematical Methods for Digital Computers," John Wiley & Sons, 1962.

CE 792 RESPONSE SPECTRUM CALCULATION

a. Description and Assumptions

This program computes the response spectra for specified acceleration time histories which are generated by CE 611 Time History Analysis. The input acceleration time histories is digitalized at equal time intervals. The numerical method used for integration is based on the exact solution to the governing differential equation, assuming that the input acceleration time history varies linearly between consecutive data points.

b. Extent of Program's Applications

This program is used to generate response spectra at all equipment locations in the containment, auxiliary building (including warehouse portion), turbine building and intake structure.

c. References

Nigam, N. C., Jennings, P.C., "Digital Calculation of Response Spectra from Strong-Motion Earthquake Records," CIT, 1968.

VIII. DELETED

IX. CLASS I STRUCTURES

CE 316-4 Finite Element Stress Analysis (FINEL)

a. Description and Assumptions

This program does static analysis of plane or axisymmetric structures using the finite element method. The finite element library contains orthotropic quadrilateral reinforcement elements, and isotropic triangles and quadrilaterals. Element stresses and joint displacements are solved due to applied loads or temperature distributions. Applied loads can be concentrated, distributed or inertial and must be axisymmetric for axisymmetric structures. The total load can be applied in small increments and the solution is iterated within each increment if necessary to establish equilibrium. The program will consider up to eight different materials and allows for material properties changing with temperature. Materials can have bilinear stress-strain curves to model elasto-plastic behavior. Prestress forces are simulated by using appropriate concentrated forces.

b. Extent of Program's Applications

The program is used to obtain stresses in the containment structure due to thermal and pressure loads.

CE 779 Structural Analysis Program (SAP)

a. Description and Assumptions

This program performs the static and dynamic analysis of linear elastic three-dimensional structures using the finite element method. The finite element library contains truss and beam elements, plane and solid elements, plate and shell elements, axisymmetric (torus) elements, and special boundary (spring) elements.

Element stresses and displacements are solved due to either applied loads or temperature distributions. Concentrated loads, pressures or gravity loads can be applied. Temperature distributions are assigned as an appropriate uniform temperature change in each element. Prestressing can be simulated by using artificial temperature change on rod elements.

The available dynamic response routines will solve for arbitrary dynamic loads or seismic excitations using either modal superposition or direct integration. The program also does response spectrum analysis.

b. Extent of Program's Applications

The program is used to obtain stresses in the concrete shell which is designed to protect the condensate storage tank from missiles.

5.E.1 COMPUTER PROGRAM APPLICABILITY AND VALIDATION

I. BLOWDOWN LOADS

The computer programs WATERHAMMER and CEFLASH-4, described in Appendix 5.E were derived from programs in the public domain. Changes have been made to each to increase its utility and improve its treatment of the blowdown problem.[1, 2]

The WATERHAMMER [3] code is recognized for its applicability to the analysis of the subcooled decompression. The code manual (Reference 3) demonstrates the program's validity through comparison of its predictions to LOFT Semiscale experimental results.

CEFLASH-4 is the C-E modified version of the FLASH-4 code [4]. The C-E modifications are discussed in Reference 1. CEFLASH-4 has been accepted by the AEC via the Interim Acceptance Criteria of December 1971 [5].

The FLASH-4 program was written in FORTRAN IV for use on the CDC-6600 computer. It has been converted, at C-E such that it may also be run on the CDC-7600 computer.

Verification of this conversion was obtained by running test cases when this change was made.

a. References

1. CENPD-26, “Description of Loss-of-Coolant Calculational Procedures,” August 1971 (Proprietary).

2. CENPD-42, “Topical Report on Dynamic Analysis of Reactor Vessel Accident Conditions With Application of Analysis to CE 800 MWe Class Reactors,” August 1971 (Proprietary).

3. 65-28-RA, “Early Blowdown (WATERHAMMER) Analysis For Loss-of-Fluid Test Facility,” by Stanislav Fabic, dated June 1965 and revised April 1967.

4. WAPD-TM-840, “FLASH-4: A Fully Implicit FORTRAN IV Program for the Digital Simulation of Transient In a Reactor Plant,” by T. A. Porsching, J. H. Murphy, J. A. Redfield, and V. C. Davis, March 1969.

5. “Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors,” Federal Register, pp 24083, Volume 36, Number 244, Saturday, December 18, 1971.

II. REACTOR VESSEL INTERNALS

The following programs are in the public domain and have had sufficient use to justify their applicability and validity.

MRI/STARDYNE June 1, 1970 Version

The program was developed by Mechanics Research, Inc., for use on the CDC-6600 Computer System. The program is run on the CDC-6600 Computer System located at the Boston, Massachusetts data center.

ICES/STRUDL Version 1.4

This version of the program was developed by the McDonnell Automation Company/Engineering Computer International. This version was purchased by Combustion Engineering and is run on the IBM-360 computer system.

EAC/EASE March 1970 Version

The program was developed by Engineering/Analysis Corporation for use on the CDC-6600 Computer System. The program is run on the CDC-6600 Computer System located at the Boston, Massachusetts data center.

SAAS Version I

The program was developed by E. L. Wilson and R. M. Jones and is documented in the following reference:

> Wilson, E. L., Jones, R. M., "Finite Element Stress Analysis of Orthotropic, Temperature-Dependent Axisymmetric Solids of Revolution," Aerospace Report Number TR-0158 (S3816022)-1, September 1967.

The theoretical basis of the program was developed in the following reference:

> Wilson, E. L., "Structural Analysis of Axisymmetric Solids," AIAA Journal, Volume 3, Number 12, December, 1965, pp 2269-2274.

The program is compatible with the CDC-6600 Computer System. It is run on the CDC-6600 Computer System located at Combustion Engineering's Windsor, Connecticut data center.

III. REACTOR COOLANT SYSTEM - DYNAMIC ANALYSIS

The following program is in the public domain and has had sufficient use to justify its applicability and validity.

ICES/STRUDL Version 1.4

This version of the program was developed by the McDonnell Automation Company/Engineering Computer International. This version was purchased by Combustion Engineering and is run on the IBM-360 computer system. (See, also, Appendix 5.E.3)

IV. REACTOR COOLANT SYSTEM - STATIC STRUCTURAL ANALYSIS

MEC-21 June 1970 Version

The program was developed by the Mare Island Naval Shipyard and is run on the CDC 6600/7600 computers.

V. ASME SECTION III, CLASS I COMPONENTS

The following programs are in the public domain and have had sufficient use to justify their applicability and validity.

SAAS Version I

The program was developed by E. L. Wilson and R. M. Jones and is documented in the following reference:

> Wilson, E. L., Jones, R. M., "Finite Element Stress Analysis of Orthotropic, Temperature-Dependent Axisymmetric Solids of Revolution," Aerospace Report Number TR-0158 (S3816-22)-1, September 1967.

The theoretical basis of the program was developed in the following reference:

> Wilson, E. L., "Structural Analysis of Axisymmetric Solids," AIAA Journal, Volume 3, Number 12, December 1965, pp 2269-2274.

The program is compatible with the CDC-6600 Computer System. It is run on the CDC-6600 Computer System located at Combustion Engineering's Windsor, Connecticut data center.

SEAL-SHELL-2

The program was developed by C. M. Friedrich and is documented in the following reference:

> "Seal-Shell-2-A Computer Program for the Stress Analysis of a Thick Shell of Revolution with Axisymmetric Pressures, Temperatures, and Distributed Loads," by C. M. Friedrich, WAPD-TM-398, UC-38: Engineering and Equipment, TID-4500, 24th Edition.

The program is run on the CDC-6600 Computer Systems, located at Combustion Engineering's Windsor, Connecticut data center.

Analysis of Axisymmetric Solids - 1967 Version

The program was developed by E. L. Wilson, University of California. The theoretical basis of the program is given in the following reference:

> Wilson, E. L., "Structural Analysis of Axisymmetric Solids," AIAA Journal, Volume 3, Number 12, December 1965, pp 2269-2274.

The program is run on the CDC-6600 Computer System located at Combustion Engineering's Windsor, Connecticut data center.

SOLIDS II - June 1971 Version

The program was developed by the Aerospace Corporation and runs on the CDC-6600 Computer System. The program is described in the following reference:

> Crose, J. C. and Jones, R. M., "Finite Element Stress Analysis of Axisymmetric and Plane Solids with Different Orthotropic, Temperature-Dependent Material Properties in Tension and Compression," TR-0059(56816-53)-1, The Aerospace Corporation San Bernardino, California (June 1971).

VI. CLASS I PIPING SYSTEMS

The following programs are in the public domain and have had sufficient use to justify the applicability and validity.

MRI/STARDYNE September 1, 1972 Version

The program was developed by Mechanics Research, Inc., for use on the CDC-6600 Computer System.

ADLPIPE

The program was developed by Arthur D. Little, Inc., Cambridge, Massachusetts for use on the CDC-6600 and UNIVAC 1108 Computer Systems.

VII. CLASS I STRUCTURES

None

5.E.2 COMPUTER PROGRAM TEST PROBLEM SOLUTIONS

I. BLOWDOWN LOADS

Additional demonstration of the WATERHAMMER's validity is given in CENPD-42 [(1)] which presents comparisons of the code's predictions to LOFT Semiscale and Battelle Northwest CSE experimental results.

CEFLASH-4 has also been tested against blowdown test data from the LOFT Semiscale and CSE experiments. The results of these comparisons are published in CENPD-26 and CENPD-42. These comparisons support the validity of the CEFLASH-4 method of analysis.

a. Reference

1. CENPD-42, "Topical Report on Dynamic Analysis of Reactor Vessel Accident Conditions With Application of Analysis to CE 800 MWe Class Reactors," August 1971 (Proprietary).

II. REACTOR VESSEL INTERNALS

The following programs' solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by computer program, SABOR-5-DRASTIC, developed at the Aeronautics at the Massachusetts Institute of Technology:

ASHSD

SAMMSOR/DYNASOR

The comparison appears in "Topical Report on Dynamic Analysis of Reactor Vessel Internals under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Connecticut, Appendix D.

III. REACTOR COOLANT SYSTEM - DYNAMIC ANALYSIS

None

IV. REACTOR COOLANT SYSTEM - STATIC STRUCTURAL ANALYSIS

None

V. ASME SECTION III, CLASS I COMPONENTS

The following program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by:

a. closed form solutions

b. EASE, a computer program developed by the Engineering/Analysis Corporation

c. SOLIDS II, TR-0059 (56816-53)-1, the Aerospace Corporation, San Bernardino, California

d. SAP, Structural Engineering Laboratory, University of California, Berkeley, California

BJS

The following program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by computer program MARC-HEAT, described in MARC-CDC, User Information Manual, Volume I, CDC Publication Number 17309500.

WIN-12100

VI. CLASS I PIPING SYSTEMS

None

VII. CLASS I STRUCTURES

CE 309 STRESS

This program has been demonstrated to be substantially identical to those obtained by the original program, STRESS, developed at the Massachusetts Institute of Technology. The traceability of this program can be obtained at the Pacific International Computer Corporation.

5.E.3 COMPUTER PROGRAM TEST PROBLEM SIMILARITIES

I. BLOWDOWN LOADS

Minor modifications were made to the WATERHAMMER code to increase its utility such as the addition of a plot routine, revision of the output edit format, and increase of the number of legs which may be employed.

WATERHAMMER was written in FORTRAN IV for use on the CDC-3600 computer. It has been converted, at C-E, for operation on both the CDC-6600 and CDC-7600 machines. A test case, given in the manual, was reproduced with the code and by hand techniques. Agreement was excellent.

II. REACTOR VESSEL INTERNALS

The following programs' solutions have been demonstrated to be substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results. The references below each program indicate where details of these comparisons can be found.

SHOCK

1. "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Connecticut, Appendix B.

2. Gabrielson, V. K., "SHOCK - A Computer Code for Solving Lumped-Mass Dynamic Systems," SCL-DR-65-34, January 1966, pp 58-79.

ASHSD

1. "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Dept., Combustion Division, Windsor, Connecticut, Appendix A.

2. Ghosh, S., Wilson, E. L., "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Report Number EERC 69-10, University of California, Berkeley, September 1969, pp 69-81.

SAMMSOR/DYNASOR

1. "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 MWe Class

Reactors," Combustion Engineering Report CENPD-42, Combustion Engineering, Inc., Nuclear Power Department, Combustion Division, Windsor, Connecticut, Appendix C.

NAOS

1. Dunham, R. S., Nickell, R. E., et al., "NAOS - Finite Element Analysis of Axisymmetric Solids with Arbitrary Loadings," Structural Engineering Laboratory, University of California, Berkeley, California, June 1967, pp 32-39.

III. REACTOR COOLANT SYSTEM - DYNAMIC ANALYSIS

ICES/STRUDL-II

The basic version of this code, Version 1.4, is described under 5.31.1, above. To facilitate dynamic analysis, additional Input/Output options and an eigenvalue analysis routine have been incorporated by Combustion Engineering. The validity of these additions have been carefully verified by appropriate test problems.

TMCALC

The program was developed by Combustion Engineering, Inc., and its validity has been carefully confirmed. The various matrix manipulations employed by the program were confirmed by hand calculations and the numerical integration procedure, Newmark's Beta-Method, used in the analysis of the reactor coolant system components for Millstone Nuclear Power Station, Unit Number 2, was confirmed by an alternate, independent, integration procedure which was, subsequently, incorporated into the program.

The alternate integration procedure employs a closed-form solution of the modal equations of motion over each time step of the integration process.

Input excitation is provided in digitized form and varies linearly between input points. The validity of the closed-form solution over a time increment was verified by hand calculations. After incorporation of the alternate integration procedure, TMCALC was again used to calculate the dynamic response of the reactor coolant system model shown in Figure 4.A–1, Appendix 4.A of the FSAR. The results obtained, using the two, independent, integration routines are substantially identical.

FORCE

The program was developed by Combustion Engineering, Inc., and its validity has been carefully verified by a series of test problems which were confirmed by hand calculations.

SHAKE

The program was developed by Combustion Engineering, Inc., and its validity has been carefully verified by a series of test problems which were confirmed by hand calculations. The test problems include those presented in the paper, "Response of Structural Systems to Ground Shock," by Dana Young, presented at the Annual Meeting of the ASME, November 30, 1960.

IV. REACTOR COOLANT SYSTEM - STATIC STRUCTURAL ANALYSIS

None

V. ASME SECTION III, CLASS I COMPONENTS

None

VI. CLASS I PIPING SYSTEMS

None

VII. CLASS I STRUCTURES

The following programs' solutions have been demonstrated to be substantially identical to those obtained by hand calculations from accepted experimental test on analytical results. The documents traceability of the following programs can be obtained at the Bechtel Power Corporation.

CE 611 TIME-HISTORY ANALYSIS

CE 617 MODES AND FREQUENCIES EXTRACTION

CE 641 RESPONSE SPECTRUM TECHNIQUE

CE 784 RESPONSE SPECTRUM TECHNIQUE

CE 792 RESPONSE SPECTRUM CALCULATION

5.F CONTAINMENT WATER INTRUSION INTO TENDON GALLERY DURING CONSTRUCTION [1]

Due to the recurrent appearances of water in some of the tendon sheaths and on the gallery floor, a formal inspection was made by Engineering personnel to determine a method to stop the water leakage. This inspection was performed on October 28, 1971.

The vertical tendon sheaths and trumpets were examined from the tendon access gallery, where water was observed running down the interior surfaces of the vertical tendon sheaths and dripping to the floor from the bearing plates. The number of the trumpet/bearing plate assemblies affected was 70 out of 124.

There were indications of light rusting on the interior surfaces of the trumpets and around the rims of the holes in the bearing plates, which had been painted; and in some trumpets, deposits of minerals were building up inside and around the edges of the holes. However, the galvanized surfaces of the inside of the tendon sheaths appeared to be free of rust and deposits.

Mineral deposits and damp areas were also noted in various locations along the gallery walls at construction joints.

A sample of the dripping water was taken by the Site personnel and submitted laboratory analysis.

From the report it was observed that:

1. The water is very alkaline, with a pH = 11.68.

2. The water is very hard.

3. The iron content is very low.

Two possible sources of water were investigated:

1. Curing water sprayed on the exterior concrete containment wall.

2. Ground water seeping into the concrete from below grade.

No tendons have been affected by water leakage since no tendons were installed in any sheaths while leaking water was present. Extreme care had been exercised to ensure that all water leakage was stopped in each tendon sheathing before installation of the tendon.

After consulting with American Drilling and Boring Company, they provided a proposed method of stopping the ground water via letter of April 4, 1972. After review and approval of procedures, work started on May 15, 1972.

NOTE: [1] From response to NRC Question Number 5.57.

The first scheme utilized was that of drillings upward through the access gallery ceiling, at elevation (-)32 feet 6 inches, to the construction joint between the Containment Mat and the Containment Wall at approximate elevation (-)26 feet 0 inches, and injecting chemical grout into these holes under a maximum pressure of 30 psi. It was theorized that the grout would travel along the same path as the water and seal the joint against leaking. This scheme worked around many of the sheathings, but in other instances the grout would travel to and leak from a sheathing near the drilled hole to the extent that no pressure could be built up during the grouting operation. Without a pressure build-up, the grout would not seal off the water. Approximately 40 of the 70 sheathings, which were leaking, were sealed by this method. On July 5, 1972, a second scheme for sealing off the water was initiated. This scheme involved putting a plug in the sheathing above and below the first sheathing joint, at elevation (-)22 feet 6 inches, and pumping the grout out through the joint. This scheme proved very successful and by July 29, 1972, the water leaking into the remaining 30 tendon sheathings was essentially stopped. However, the very humid weather during the weeks of July 30 and August 6, 1972, caused considerable condensation on the access gallery ceiling and the grouting operation was delayed until additional ventilation fans were installed to supplement the two fans then in use.

On August 14, 1972, with the use of one additional fan and with the arrival of less humid weather, the access gallery ceiling had dried up enough for grouting to proceed. By August 25, 1972, the leaks had been stopped in all tendon sheaths except 31V36, 31V31, 31V20, and 23V26. Several further attempts, utilizing scheme two, failed to completely stop the water seepage into these four tendon sheathings. However, the water seepage was in all cases under no pressure and in negligible quantities.

In early December 1972, a third scheme for sealing the tendon sheathing was initiated. This scheme involved the installation of a plug in the tendon sheath just above the bottom vertical trumpet. The sheath was then filled with chemical grout to approximately elevation 14 feet 6 inches. The top end of the sheath was then sealed with another plug and the air in the sheath was pressurized to approximately 40 psi. This pressure was held for 20 minutes to allow the grout to gel. The plugs were then removed and the grout in the sheath was expelled using air pressure. All remaining grout in the sheath was removed with a high pressure (6000 psi) water jet. Visual inspection of the sheathing showed it to be free of grout.

As of February 1973, there was only one location at which water was entering the access gallery. This location is near the construction opening and the water leakage was through holes drilled to the construction joint, through the gallery ceiling. These holes were not initially grouted in order to be able to observe this leakage during the post-tensioning of the containment. On July 11, 1973, the remaining holes were grouted.

As of July 16, 1973, vertical tendon installation, stressing, and greasing were completed. At all locations where water was indicated inside the sheathings, the grouting operation completely arrested all water seepage into the sheathings. Examination inside the sheathings prior to tendon installation and greasing showed a dry condition. Therefore, all tendons are installed in a non-corrosive environment.

A pressurized grease system connected to the tendon filler caps was installed after the fourth tendon surveillance inspection in 1986. As monitored in the sixth surveillance inspection report in 1996, the number of tendons in which water was found has been greatly reduced. No significant corrosion was identified beyond that noted at installation and reported during the fifth surveillance.

See FSAR Section 5.9.3.3.4, "Corrosion Protection", for additional data to prevent ground water intrusion.

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Chapter 6: Engineered Safety Features Systems

CHAPTER 6—ENGINEERED SAFETY FEATURES SYSTEMS

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CHAPTER 6 – ENGINEERED SAFETY FEATURES SYSTEMS

6.1 GENERAL

6.1.1 DESIGN BASES

6.1.1.1 Functional Requirements

Engineered safety features is the designation given to those systems which are provided for the protection of the public and station personnel against the postulated release to the environment of radioactive products from the reactor coolant system, particularly as the result of loss-of-coolant accidents (LOCA). These safety features function to localize, control, mitigate, and terminate such incidents and to hold exposure levels below limits of 10 CFR Part 50.67.

Following a LOCA, these systems function to cool the core to limit fuel damage, to limit the magnitude and duration of pressure transients within the containment, to provide post-incident cooling for extended periods and limit fission products release. The adequacy of the design is demonstrated and the protection of the public assured by the use of the conservative assumptions outlined in Section 14.6.5 and 14.8.4 in the analysis of the effects of the LOCAs.

6.1.1.2 Design Criteria

The following criteria have been used in the design of the engineered safety features systems:

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

b. Safety related structures, systems, and components shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

    1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with

sufficient margin for the limited accuracy, quality, and period of time in which the historical data have been accumulated.

2. Appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena.

3. The importance of the safety functions to be performed.

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

d. Safety related structures, systems, and components shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

e. Safety related structures, systems, and components shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

6.1.2 SYSTEM DESCRIPTION

6.1.2.1 System

The engineered safety features systems are shown schematically on Figures 6.1–1, 9.9–1, and 9.9–2, and are described in detail in the subsequent sections.

The systems defined as engineered safety features are as follows:

a. The safety injection system, including as subsystems the safety injection tanks, high pressure safety injection pumps and low pressure safety injection pumps, injects borated water into the reactor coolant system. This provides cooling to limit core damage and fission product release, and assures adequate shutdown margin. The injection system also provides continuous long term post incident cooling of the core by recirculation of borated water from the containment sump.

b. The containment spray system removes heat by spraying cool borated water through the containment atmosphere. The sprayed heated water is then collected in the containment sump and cooled by the reactor building closed cooling water system through the shutdown heat exchangers and recirculated into the containment atmosphere.

c. The containment air recirculation and cooling system removes heat by passing containment air over coils cooled by the reactor building closed cooling water system.

d. The enclosure building filtration system collects and processes potential containment leakage to minimize environmental activity levels resulting from containment leakage.

e. The containment hydrogen control systems including independent systems provided to mix and monitor the hydrogen in the containment atmosphere.

The supporting systems include the engineered safety features actuation system, the normal and emergency electrical power systems, the reactor building closed cooling water system, service water and chemical and volume control system (CVCS). These systems are described in detail in Chapters 7, 8 and 9. All pneumatically operated values and dampers are designed to assume the safe position following loss of instrument air.

6.1.2.2 Components

The engineered safety features systems components are procured to detailed engineering specifications and tested to applicable codes listed in following sections.

6.1.3 SYSTEM OPERATION

The engineered safety features systems generally operate only under certain emergency conditions. Some systems operate during normal operation as the containment air recirculation and cooling system and hydrogen monitoring system. Other components such as the low pressure safety injection system and shutdown heat exchangers operate during shutdown operations.

6.1.3.1 Emergency Conditions

In the unlikely event of a LOCA or Main Steam Line Break (MSLB) accident, the engineered safety features systems with the exception of the hydrogen control system are automatically initiated to perform their respective safety function. The hydrogen control system may be manually initiated within several hours after the incident.

6.1.4 AVAILABILITY AND RELIABILITY

6.1.4.1 Special Features

All components of the engineered safety features systems and associated critical instrumentation which must operate are designed to operate in the most severe environment to which they could be exposed in the event of the worst postulated LOCA or MSLB accident. These design requirements are of primary significance for those portions of the engineered safety features systems which are located inside the containment. All components of the engineered safety features located inside containment, the auxiliary building, or the enclosure building, are environmentally qualified per 10 CFR 50.49 to operate in the most severe post-accident environment to which they are exposed.

The forces generated by the maximum hypothetical earthquake, combined with the rupture of a reactor coolant pipe, are considered in the design of the engineered safety features. The design assures that the functional capability of the system will be retained. Vessels which are connected to the engineered safety features systems are supported and restrained to allow controlled movement during this load condition, and piping is designed to accept these imposed movements. Analysis of the flexibility of the systems have been performed to verify that the piping can accept these additional vessel movements and still remain within code allowable stress limits. Flexibility calculations have been performed in accordance with the applicable codes, ASME Section III Nuclear Power Plant Components – 1971 and Power Piping ANSI B31.1.0-67. The double-ended rupture of the largest reactor coolant pipe is used as the design basis since the forces are the most severe.

The maximum post-incident temperature, pressure and humidity in the containment are found to be developed by the MSLB accident. A discussion of the analyses performed for a spectrum of break sizes, to determine these limiting service conditions, is presented in Section 14.8.2.

The pipe rupture protection requirements are listed in Table 6.1-1. The following pipe whip criteria are used in the layout of piping and location of restraints protecting the redundant engineered safety features systems.

6.1.4.1.1 Assumptions

In event of a pipe rupture in the reactor coolant system, it is assumed the only loss of engineered safety features piping is limited to those connected to that specific segment of the coolant loop.

6.1.4.1.1.1 Damage Protection Criteria

The criteria for protection from results of pipe whip or rupture inside and outside of the containment are summarized below:

a. Whipping does not preclude safe shutdown.

b. A secondary system failure does not cause a failure in the primary system.

c. No secondary system pipe whip results in the loss of its counterpart in the other secondary loop.

d. The containment liner plate is protected against a rupture on the primary system or piping that is in operation during or after a LOCA.

e. The containment liner plate is protected against main steam line whip.

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, or (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. Subsequent to the commission review and approval, weld overlays were applied to dissimilar metal welds (DMWs) at the shutdown cooling, the safety injection and the pressurizer surge nozzles. A revised LBB analysis was performed for these nozzles and based on a 50.59 evaluation, commission review and approval of the DMW LLB analysis is not required to invoke the exemption from protection against pipe rupture dynamic effects for these nozzles.

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.

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. SA 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.

b. Circumferential (Guillotine) break in pipes exceeding 1 inch nominal size (in which complete severance of the pipe is assumed perpendicular to the axis of the pipe).

In either case, the total dynamic steady thrust force in the event of a pipe rupture is computed from the following equation:

$$\frac{F_T}{A_B} = (P_2 - P_\infty) + \frac{G^2 \upsilon}{gc}$$

where:

F_T = Total thrust force

A_B = Break flow area

P_2 = Exit plane pressure

P_∞ = Pressure outside pipe prior to rupture

G = Mass flow rate per unit area

ν = Fluid specific volume

g_c = Newton’s constant

A typical example of the existing pipe whip restraints designed according to the criteria in Section 5.2.5.7 of the FSAR and above is shown in Figure 6.1–2. This restraint prevents the main

steam line from impacting against the containment liner plate as a result of the postulated pipe break shown. The restraint must limit both lateral and/or outward movement of the pipe at this location. Of these two conditions, the most critical one is the outward movement away from the restraint supports.

The total equivalent static reaction force of the pipe on the restraint, including the thrust coefficients and dynamic load factors, amounted to 2·PA,

where:

P = maximum normal operating pressure
A = internal cross-sectional area of the pipe

This procedure resulted in an effective static design force of 1,600 kips on the main steam line pipe whip restraint.

The pipe whip restraints inside the containment were reevaluated by conducting a dynamic analysis, considering the restraint as a single degree-of-freedom system. Since a gap must be maintained between the restraint and the pipe, due to construction tolerances and thermal considerations, the effects of a plastic collision were also considered.

The procedures to determine the effective thrust force for the dynamic analysis were revised from the original assumption and are reported above. This procedure was used in all cases except for the main feedwater line inside the containment, where the thrust force was determined by using the computer code Relaps III and the conservation of momentum equations. In all cases, the thrust force was considered as a step function, having an instantaneous rise time and maintaining a constant (maximum) level of thrust until the pipe-restraint system has come to rest.

Since the thrust force was taken as independent of time, it is also independent of displacement. Being independent of displacement, it can be shown mathematically that the dynamic analysis discussed above can be simplified to an energy balance calculation (i.e., the energy input into the restraint by the pipe equals the strain energy of the restraint responding to the pipe impact).

Employing the Conservation of Momentum for a plastic collision and conservatively neglecting the energy absorbed by bending (or hinging) of the pipe, the total deflection of the restraint is computed to be:

$$y_m = \frac{1/2 R_m y_e + T e y_g}{R_m - T}$$

where:

y_m = maximum deflection of the restraint

y_e = elastic deflection of the restraint

y_g = initial gap between pipe and restraint

R_m = (maximum elastic structural resistance + plastic structural resistance) ÷ (2)

T = thrust force

e = m_p divided by (m_p + m_r)

m_p = effective mass of the pipe

m_r = effective mass of the restraint

Figure 6.1–3 shows the idealized dynamic pipe thrust force and the pipe whip restraint force as a function of pipe deflection. Elastic deflection at this load is conservatively computed to be 0.093 inches, resulting primarily from flexure. The ultimate allowable deflection that provides resistance to the 1,600 kip force is taken to be 0.93 inches (i.e., elasto-plastic behavior, and a ductility ratio of 10 inches flexure is considered).

All pipe restraints outside the containment are designed to satisfy the requirements of Section 6.1.4.1.1.3. These pipe restraints are designed for elasto-perfectly plastic behavior, considering the impact effects resulting from gaps between the pipe and restraint. An energy balance approach similar to the one discussed above is used.

In all cases these restraints are designed as flexible structures having a ductile failure mode. As a result the dynamic load factors (measured as ratio of total restraint strength to steady thrust force) ranged from 1.2 to 2.0. Strain limits associated with these values are within 50% ultimate strain of the pipe whip restraint material. The criteria for protection against critical pipe failures for Millstone Unit 2, including the criteria, as outlined in Section 5.2.5.7 of the FSAR, are in complete accordance with Bechtel Topical Report BN-TOP-2 (Rev. 1), “Design for Pipe Break Effects,” except as noted in the following items.

(1) Reference: Fluid Jet Geometry as shown in Figure 2-3 of BN-TOP-2 (Rev. 1). The jet expansion model used for Millstone Unit 2 and the referenced Figure are both shown in Figure 6.1–4 for comparison. The model presented in BN-TOP-2 consists of three regions. For subcooled water blowdown, this model assumes half-angle approach of 10 degrees uniformly in all three regions. For steam and steam/water blowdown, the jet expands to the asymptotic area at a half-angle which exceeds 10 degrees. The area remains constant in region 2 and expands again at a ten degree half-angle in region 3.

For steam and steam/water mixtures, the model used for Millstone Unit 2 assumes the jet expands to the asymptotic area (taken to be located at five inside pipe diameters from the break exit plane)

at a ten degree half-angle and remains constant thereafter for the remaining extent of its zone of influence. The applicable limit for steam and steam/water mixtures is based on the premise that the expansion of the fluid jet from a postulated pipe rupture attenuates the system pressure within ten nominal pipe diameters from the source of the jet to ambient conditions as demonstrated by NUREG/CR-2913, Reference 6.1-2. Unprotected components at a distance beyond ten nominal pipe diameters from the broken pipe are considered undamaged by the jet without further analysis.

The jet shape used for cold water and non-flashing/subcooled fluid is treated as a non-expanding jet with its area equal to that of the break exit area. In this case, conservation of load effects will be assumed and the zone of influence of the jet will be considered to end only by interaction with a component or structure with the capacity to block and resist the full jet load. Therefore, the Millstone Unit 2 model provides jet pressure intensities which are more conservative than BN-TOP-2.

(2) Reference: Moment capacity for steel beam, Page 3-9 of BN-TOP-2.

$$M_u = 0.9\, f_y\, S_p \qquad (1)$$

$$M_u = 1.0\, f_y\, S \qquad (2)$$

where

M_u = yield moment

f_y = yield stress

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).

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

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.)

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 Section 5.2.5.7.1.

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

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. Non seismic 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 Non seismic 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.

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.

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

The refueling pool skimmer filters are located in shielded compartments at elevation (+) 14 feet 6 inches 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.

6.1.5 REFERENCES

6.1-1 Plastic Design in Steel, 2nd Edition, ASCE Manual Number 41, 1971.

6.1-2 NUREG/CR-2913, “Two-Phase Jet Loads,” January 1983.

TABLE 6.1-1 PIPE RUPTURE PROTECTION REQUIREMENTS

LINE SERVICE	CATEGORIES					
	A	B	C	D	E	F
Safety Injection Lines	X	X	X	X		X
Charging Line	X	X				X
Reactor Coolant Letdown Lines	X	X		X	X	
Steam Generator Blowdown Lines	X	X	X	X		
Feedwater Lines	X	X	X	X	X	X
Main Steam Lines	X	X	X	X	X	X
Reactor Coolant System Piping	X		X	X	X	

The following categories have been listed in Table 6.1-1 to describe the criteria mentioned:

Category A Lines that must be restrained from damaging the reactor coolant system.

Category B Lines that must be restrained from damaging the containment liner plate.

Category C Lines that must be protected from damage by ruptured reactor coolant system piping

Category D Lines that must be restrained from damaging the secondary system.

Category E Lines that must be protected from damage by the secondary system.

Category F Lines that must be protected from damage by or restrained from damaging their parallel redundant lines.

TABLE 6.1-2 OMITTED

TABLE 6.1-3 DESIGN NON-SEISMIC CATEGORY I COMPONENTS LOCATED INSIDE CONTAINMENT AND ENCLOSURE BUILDING

I. Containment Building

A. Clean Liquid Waste Processing System Components (FSAR Figure 11.1–1)

1. Primary Drain Tank and Pumps
2. Primary Drain Tank and Quench Tank Heat Exchangers
3. Associated piping and valves

B. Reactor Coolant System Components (FSAR Figure 4.1–1)

1. Quench Tank and associated piping and valves
2. Vent, sampling and drain piping and valves beyond second isolation valve

C. Drain Systems (FSAR Figure 11.1–3)

1. Sump pumps
2. All drain piping and valves for non Category I equipment beyond second isolation valve of primary pressure boundary.

D. Reactor Vessel Head Decontamination System

1. Mixing tank
2. Pump
3. Filters (2)
4. Piping and valving

E. Miscellaneous

1. Containment elevator
2. Stud storage racks
3. CEDM Coolers
4. Revolving maintenance truss
5. Containment auxiliary circulation fans and ductwork
6. Instrument air receiver tank & associated equipment
7. Refueling pool skimmer filter & associated equipment
8. Containment Surveillance camera & associated equipment

TABLE 6.1-3 DESIGN NON-SEISMIC CATEGORY I COMPONENTS LOCATED INSIDE CONTAINMENT AND ENCLOSURE BUILDING (CONTINUED)

II. Enclosure Building

A. Main Steam System Components (FSAR Figure 10.3–1)

1. Blowdown Tank
2. S. G. Blowdown Quench Tank, Heat Exchanger, Recirculation Pump, valves and associated piping

B. Containment Penetration Cooling System

1. Fans
2. Ductwork

C. Miscellaneous

1. Miscellaneous ductwork
2. Unit heaters

TABLE 6.1-4 PIPE RUPTURE CRITERIA APPLICABILITY (OUTSIDE CONTAINMENT) BASED ON NORMAL OPERATING PLANT CONDITIONS

Operating Condition	Temperature (Degrees F)	Pressure (PSIG)
A	Equal to or Greater than 200	Equal to or Greater than 275
B	Equal to or Greater than 200 [1]	Less than 275
C	Less than 200	Equal to or Greater than 275
D	Less than 200	Less than 275

Pipe Rupture postulation at MP2 is separated into four distinct groupings, which are dependent on the normal service temperature and pressure of the line; e.g., the state of temperature and pressure of the system during Normal Plant Operation (NPO) from plant heatup through power operation and plant shutdown to the hot standby condition. Where a piping system operates within the parameters of Operating Conditions A, B, or C for only a short period of its operating time (i.e., less than 2%) it is considered to be a Moderate Energy System and is treated as an Operating Condition D System.

1. Lines whose normal operating condition is defined by condition A are defined as High Energy Lines with postulated pipe rupture effects of pipe whip and jet impingement and the related effects of pressurization, flooding and environment. Includes Charging System which operates substantially above 275 psig, but less than 200°F.

2. Lines whose normal operating condition is defined either by condition B or C are defined as High Energy Lines with postulated pipe rupture effects of jet impingement and the related effects of pressurization, flooding and environment.

3. Lines whose normal operating condition is defined by A, B, or C are defined as High Energy Lines with postulated effects of piping cracks (includes flooding and environmental conditions) at the most adverse equipment locations.

4. Lines whose normal operating condition is defined by operating condition D are considered Moderate Energy Lines (MELs) and require no postulation of rupture effects.

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1 Lines with temperature ≥ 200°F, and pressures approximately atmospheric are not considered as High Energy Lines since the calculated jet impingement force will be negligible for such low pressure.

FIGURE 6.1–1 SAFETY INJECTION P&ID (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 6.1–1 SAFETY INJECTION P&ID (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 6.1–1 SAFETY INJECTION P&ID (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

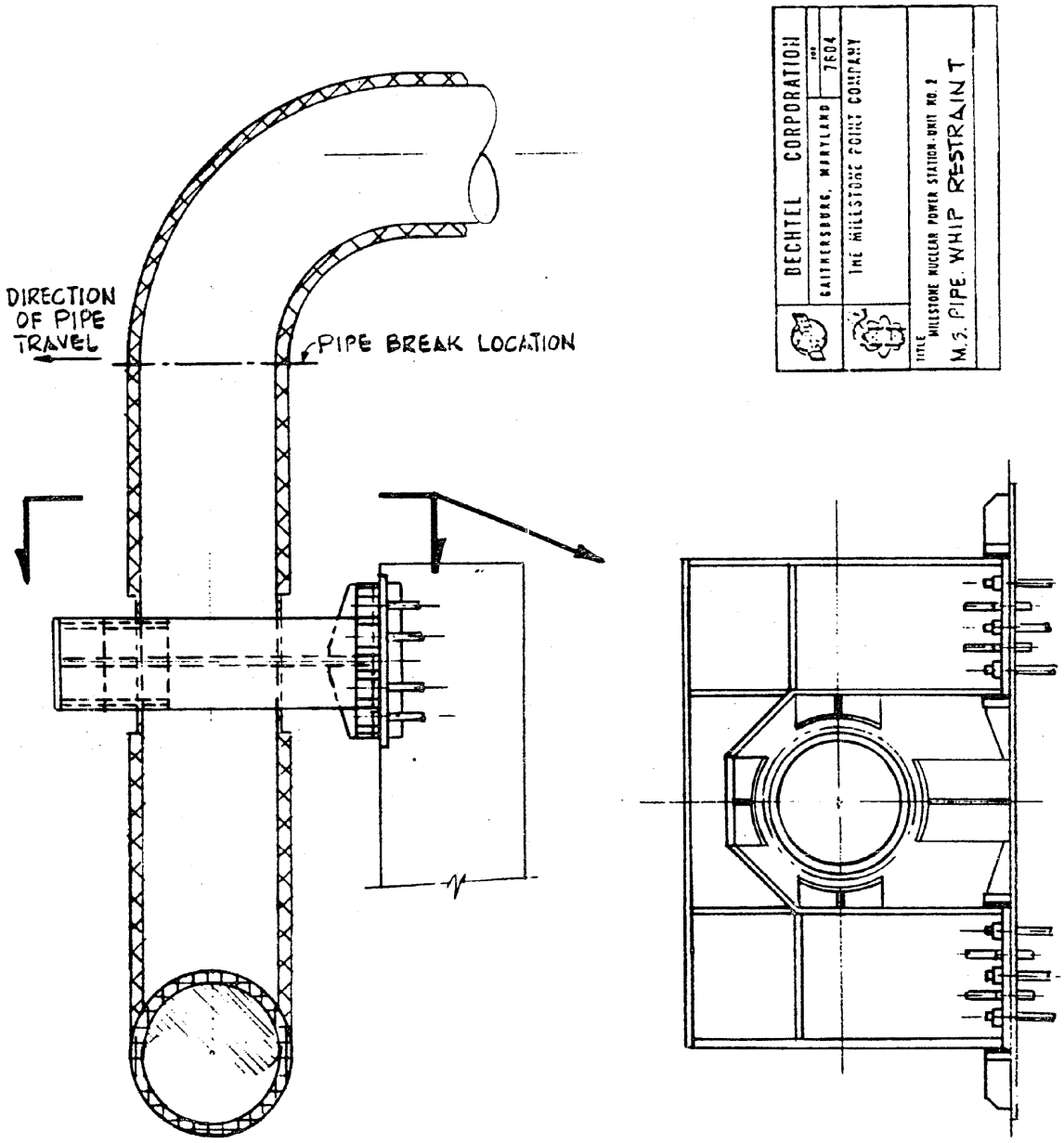
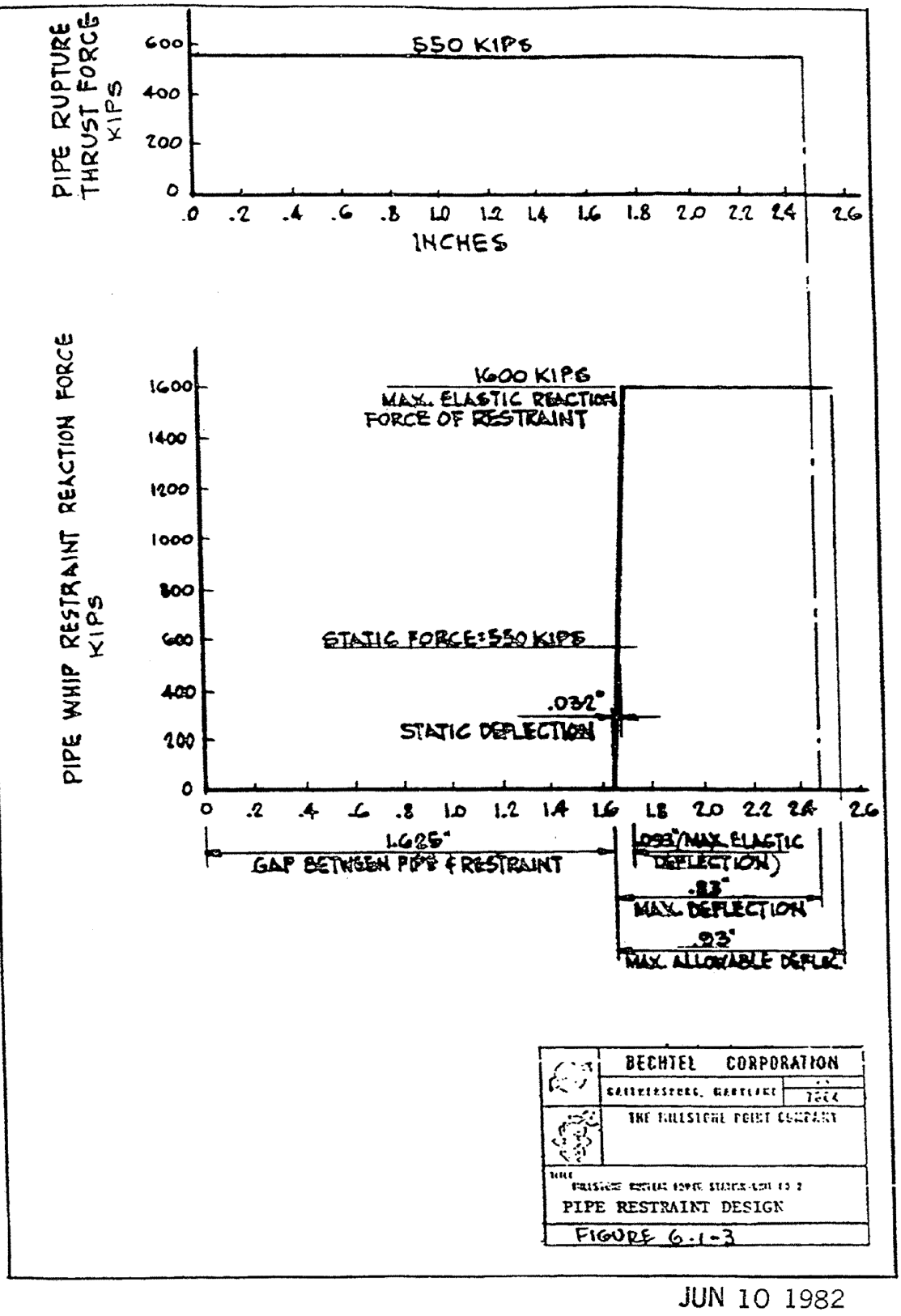


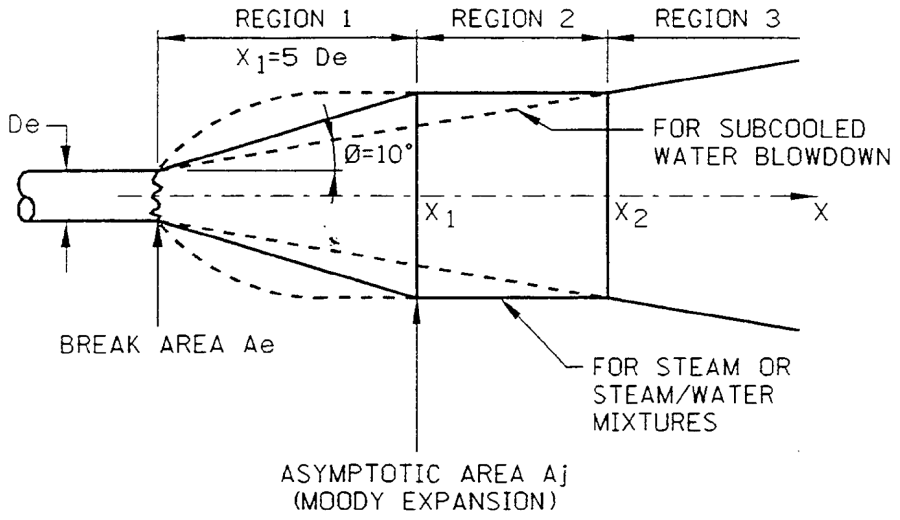
FIGURE 6.1–2 MAIN STEAM PIPE WHIP RESTRAINT

FIGURE 6.1–3 PIPE RESTRAINT DESIGN

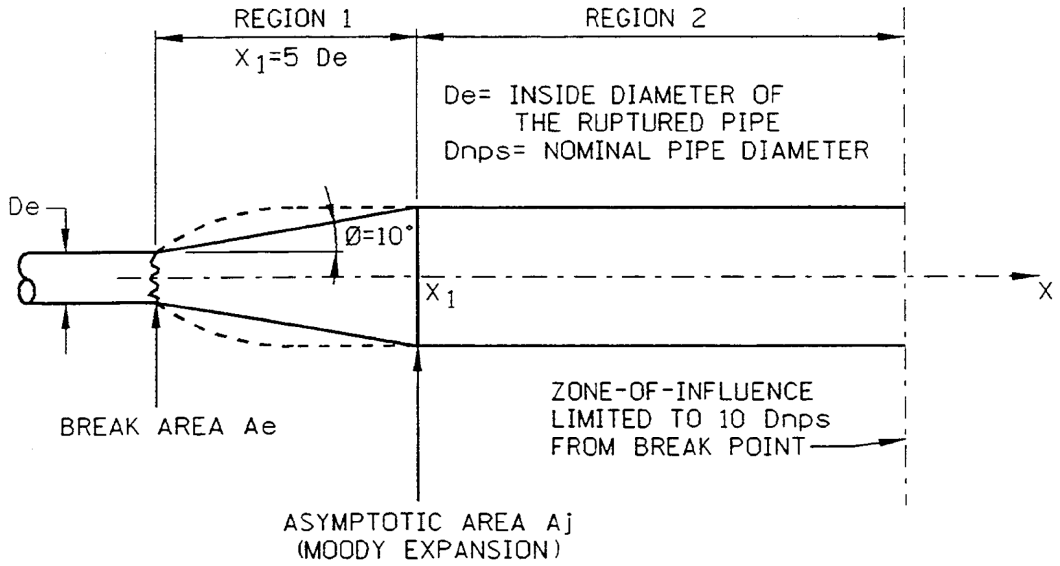


JUN 10 1982

FIGURE 6.1–4 FLUID JET EXPANSION MODEL



BN-TOP-2 MODEL
(FOR REFERENCE ONLY- NOT USED ON MP2)



MP2 MODEL FOR STEAM OR STEAM/WATER MIXTURES

NOTE:
NON-FLASHING/SUBCOOLED WATER BLOWDOWN IS TREATED AS
A NON-EXPANDING JET WITH CONSTANT AREA (Ae) AND FORCE.

6.2 REFUELING WATER STORAGE TANK AND CONTAINMENT SUMP

6.2.1 DESIGN BASES

6.2.1.1 Functional Requirements

The refueling water storage tank (RWST) functions to provide the initial source of borated water for the safety injection and containment spray pumps.

The containment sump functions to collect the water from the safety injection, containment spray and reactor coolant blowdown for recirculation after the water has been nearly exhausted from the RWST.

6.2.1.2 Design Criteria

The following criteria have been used in the design of the RWST and containment sump:

a. The system shall provide a reliable source of borated water for the safety injection system and the containment spray system.

b. The system shall be designed to permit inspection of important components, such as sump, tank, valves and piping to assure the integrity and capability of the system.

c. The components shall be designed to the general criteria as described in Section 6.1.

6.2.2 SYSTEM DESCRIPTION

6.2.2.1 System

The location and interconnection of the RWST and containment sump in the system are shown schematically in Figure 6.1–1.

The RWST is an atmospheric tank which stores borated water during normal operation. The RWST is the initial source of suction for the high pressure and low pressure safety injection pumps (Section 6.3) and the containment spray pumps (Section 6.4). The RWST is provided with two separate, independent outlet headers, each of which supplies borated water to each grouping of minimum engineered safety feature pumps as described in Section 6.1.4.1. The “A” RWST outlet header also supplies a suction connection that facilitates portable diesel driven RCS Injection Pump deployment. This connection supports the discharge connection described in Section 6.3.2.1 and is a defense-in depth design feature that is available for coping with an extended loss of AC power (ELAP) event. The location of this BDB RCS FLEX suction connection is shown on Figure 6.1–1, Sheet 2.

The RWST is designed to store sufficient borated water to fill the refueling cavity in the containment for refueling operations. This requires 421,000 gallons of borated water containing 1,720 ppm boron.

The containment sump is formed by the floor and the lowest elevation of containment. The borated water from the safety injection (Section 6.3), containment spray system (Section 6.4) and reactor coolant system is collected and subsequently recirculated to the pump's suction. Two 24 inch containment sump recirculation pipes are provided from the sump to the suction of the safety injection and containment spray pumps. The recirculating piping inside the containment extends approximately 11 inches above the floor Elevation (-) 22-6. A vortex breaker is provided on each containment sump recirculation pipe inlet to avoid any air ingestion.

The containment sumps are located at the bottom of the containment building at elevation (-) 22-6. The containment sump strainer has an enclosure that surrounds the pump suction pipes located in this sump. Two headers extend out from this enclosure in an eastern and northern direction. Perforated hollow fins are attached to these headers and flow is draw through these fins and into the headers and directed to the sump enclosure. The sump enclosure is solid plate and has a manway in the side to allow access to the sump for inspections.

The strainer is constructed of 304/304L SS or equivalent materials. The fins are made of thin corrugated stainless steel perforated with 1/16 inch holes. This perforation size prevents larger particles from passing and thus avoids any clogging of any downstream equipment including pump flow clearances, containment spray nozzles or HPSI throttle valves. The total filtration surface area of the strainer is approximately 6,000 square feet.

The strainer is designed to support the full flow rate from both trains of ECCS simultaneously. The strainer is a single unit that is shared by both trains. It is a passive component with no moving parts. Evaluations in accordance with Section 6.1.4 have determined that there are no credible failure modes that will cause loss of the strainer and redundant trains are not required.

The strainer is designed to accommodate the debris load from either a SBLOCA or a LBLOCA. This debris load consists of fibrous and particulate materials generated by the force of the pipe break in the steam generator cubicle. In addition, it considers latent particulate and fibrous debris from the containment walls, floors and equipment that is washed to the sumps by containment spray. The design also considers the formation of particulate due to chemical reactions between the substances suspended or dissolved in the containment sump water post accident. The head loss across the strainer, including this debris, is limited to a value that will not adversely affect the available net positive suction head for the safety injection and containment spray pumps.

The effects of debris ingested through the containment sump recirculation pumps via the containment sump strainer during recirculation mode were evaluated in response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors" (Reference 6.2-2). The Generic Letter Response (including RAI responses and Supplemental responses) includes evaluations of potential erosive wear, abrasion, and blockage of flow paths within recirculation spray system components. The

Generic Letter 2004-02 evaluations conclude that the system design is adequate to maintain the sump recirculation system functions during and post DBA LOCA event (Reference 6.2-1).

The post-LOCA sump recirculation phase downstream fuel and vessel effects (i.e., accumulation of fiber inside the reactor vessel) was performed in accordance with WCAP-17788-P, Volume 1, Revision 1 (Reference 6.2-3) to confirm long-term core cooling is not challenged. This evaluation used conservative debris quantities generated by and transported from the break location, which maximized the fibrous debris generated. The in-vessel debris effects evaluation utilized a conservative strainer bypass fraction, which was calculated assuming a sump strainer debris bed comprised exclusively of fine-sized fibrous debris to maximize debris transport to the reactor vessel. The calculated total reactor vessel fiber load for Millstone Power Station Unit 2 meets the in-vessel fiber limit defined by Section 6.4.4 of Reference 6.2-3. Thus, the accumulation of fibrous debris in the reactor vessel will not challenge the unit's ability to maintain adequate long-term core cooling or establish Boric Acid Precipitation Control (BAPC) actions at Millstone Power Station Unit 2.

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.

b. The recirculation lines from the containment out to and including the first isolation valve are enclosed in leakage controlling encapsulation barriers. The barrier will control leakage resulting from postulated pressure failure of the pipe or valve. This leakage control function is provided by gaskets, seals, and the concrete to steel interfaces at the encapsulation embedment plates. These features will physically limit potential fluid releases from the encapsulations to an extent that does not significantly deplete the water level in the containment sump during sump recirculation; that does not impact ECCS pump NPSH requirements or measurably alter ECCS flow capabilities; and that does not pose a threat of flooding, spraying, or otherwise shorting-out the ECCS pumps in the "A" safeguards room of the Auxiliary Building. The leakage control function of the encapsulations will also assure that post accident radioactive fluids released from these structures will be collected in the "A" safeguards room or in the liquid radwaste system in the Auxiliary Building, and processed or retained in a manner that protects public health and safety. The barrier is tightly attached to the exterior concrete of the containment. The encapsulation barrier is designed for 60 psig and is in accordance with the criteria for Seismic Class I structures.

c. Pipe material is Type 304 stainless steel.

The first isolation valve in each recirculation line is a motor-operated double-disk gate valve. Rupture disks are installed to prevent the possibility of thermal expansion of trapped fluid in the valve bodies causing the valves to become pressure locked prior to initiation of sump recirculation. The rupture disks are installed in piping connected to the body drain of each isolation valve. The discharge of the rupture disks is contained within the closed piping system.

To adjust the pH of the containment sump water following a postulated LOCA, wire-mesh baskets containing trisodium phosphate dodecahydrate (TSP) are provided on the lowest level of the containment as shown on Figure 1.2–11. The baskets contain a Technical Specification required volume of solid granular TSP, a quantity sufficient to raise the sump water pH to a value equal to or greater than 7.0.

6.2.2.2 Components

The major system components and associated fabrication data are listed in Table 6.2-1.

6.2.3 SYSTEM OPERATION

6.2.3.1 Emergency Conditions

The valves on the RWST suction headers are normally in the open position. The safety injection and containment spray systems are flooded from the RWST to their respective containment isolation valves, 2-SI-615, 625, 635, 645, 616, 626, 646, 617, 627, 637, 636, 647, 2-CS-4.1A and 4.1B (Figure 6.1–1).

In the unlikely event of a LOCA, borated water from the RWST is immediately available to the pumps suction without additional active operations.

When the RWST level reaches 46 inches, the safety injection and containment spray pump suction is automatically transferred to the recirculation mode with suction from the containment sump. This is initiated by the sump recirculation actuation signal (SRAS) and is described in Section 7.3.

The recirculation mode can also be initiated manually by the operator.

The SRAS opens the isolation valving (2-CS-16.1A and 16.1B on Figure 6.1–1) on the containment sump recirculation piping and therefore, permits the pumps to take suction from the containment sump.

An external heater is provided to maintain the contents of the RWST ≥ 50°F in Modes 1 and 2, and ≥ 35°F in Modes 3 through 6 (Section 6.2.4.1). RWST temperature is monitored by temperature indication in the control room. Low water temperature is alarmed to alert operators that remedial action is required. RWST water level is indicated in the control room. Low water level alarms are provided to alert operating personnel. Sample points located within the safety

features system indicate water chemistry. Provisions are incorporated for filling draining and altering the water chemistry of the tank contents. Purification of the RWST contents is provided by the spent fuel pool purification system.

Following a postulated LOCA, the TSP stored in dissolving baskets at the lowest elevation of the containment will dissolve in the sump water as the safety injection water fills the containment. Containment sump water recirculation will provide the required mixing to achieve a final pH greater than or equal to 7.0 for the sump water.

6.2.3.2 Cold Shutdown and Refueling

Following cold shutdown and after the reactor vessel head is removed, the refueling cavity is filled by the LPSI pumps with borated water from the RWST. The RWST is sized to provide a sufficient quantity of borated water to maintain a water level of 24 feet above the reactor vessel flange. Shutdown cooling operation and refueling operations are described in Section 9.3.

Following the refueling operation, the water from the refueling cavity is pumped back to the RWST.

6.2.4 AVAILABILITY AND RELIABILITY

6.2.4.1 Special Features

To assure availability of borated water for post-accident operation, the RWST is normally aligned for operation with the outlet valves opened. The tank contents are maintained ≥ 50°F in Modes 1 and 2, and ≥ 35°F in Modes 3 through 6 by recirculation through an external heat exchanger which is heated by unit steam. The supply and return connections for the RWST heating system penetrate the upper portions of the tank, above the minimum water level of 32.5 feet, to prevent excessive tank drainage following a postulated pipe rupture. The supply connection to the heat exchanger is provided with a siphon breaker to prevent siphoning of the tank following a postulated pipe rupture. The siphon breaker is located immediately above the minimum tank water level to assure suction for the RWST heating system and to minimize drainage following a postulated pipe rupture.

To maintain the RWST contents at 50°F with zero (0)°F outside air temperature, heating is required at a rate of 1.21×10^6 Btu/hr. The RWST borated water temperature is measured and alarmed should it decrease below 55°F. Remedial action can be taken by the operator by routing the tank contents through one shutdown cooling heat exchanger using one containment spray pump and aligning a recirculation path to the RWST by opening valves 2-SI-456 (or 2-SI-457), and 2-SI-460 (Figure 6.1–1). The RBCCW is used as the heating medium. Assuming a conservative 60°F RBCCW temperature, the tube side (RWST) water leaving temperature is 55°F.

The spent fuel pool cooling system may be used as back-up for the above to maintain the RWST contents at the required temperature.

The motor-operated isolation valves on the RWST headers remain open following the switch over to the sump recirculation mode of operation. These valves may be closed manually. However, this operation is not required for system operation. The elevated pressure within containment under post-accident conditions maintains the check valves (2-CS-14A and 14B) on the RWST headers in a closed position. Under no normal circumstance can the header from the RWST be emptied since the elevation head from the containment sump at Elevation (-) 22-6 maintains the water level in the RWST headers at this minimum elevation. The interconnection between the containment sump recirculation line the RWST header is located at Elevation (-) 30-1 5/8 and (-) 33-7.

The containment sump recirculation piping is embedded in the concrete containment floor to prevent leakage of the post-accident containment atmosphere. Although completely redundant sump recirculation piping is provided, one pipe is encased in a guard pipe for protection where both pipes are located in the same safety features pump room. Therefore, any failure of one line does not render the redundant line inoperable.

The RWST is located at grade Elevation 14-6. The containment sump inlet is located at the lowest elevation in the auxiliary building, (-) 45-6. Therefore, all suction lines are flooded to assure priming.

The Technical Specification required volume of TSP stored in the wire mesh baskets located on the containment floor is sufficient to raise the pH of the containment sump water to a value equal to or greater than 7.0 when considering the total quantity of borated water that could be available in the containment sump following a LOCA, the amount of hydrochloric acid from the degradation of electric cable insulation, and the amount of nitric acid from the irradiation of the containment air and sump water. Maintaining sump pH equal to or greater than 7.0 ensures iodine retention capability following a LOCA. Tests have demonstrated that the TSP will readily dissolve in the containment sump water. In addition TSP experiences no significant deterioration or decomposition upon storage.

6.2.4.2 Tests and Inspection

The RWST was acceptance tested in accordance with the ASME Code, Section III, Class 3. Since the RWST performs no active function, online testing is not provided. The RWST header valves underwent acceptance testing prior to initial start up. The test procedure is described in Section 13.

The RWST is accessible for periodic maintenance and inspection during normal operation.

The containment sump recirculation isolation valves (2CS-16.1A and 16.1B) may be tested for opening capabilities during normal operation. A check valve is located downstream of each of these valves to prevent flow from the RWST entering the containment.

The containment sump recirculation isolation valves undergo a preoperational test prior to start up. Test procedures are described in Section 13.

The containment sump and TSP baskets are accessible for inspection and maintenance during normal shutdown.

6.2.5 REFERENCES

6.2-1 Letter from Richard Guzman (NRC) to Eric Carr (Dominion), Millstone Power Station, Units 2 and 3 - Closeout of Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” Docket Nos. 50-336 and 50-423, July 26, 2023.

6.2-2 NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” dated September 13, 2004.

6.2-3 WCAP-17788-P, Volume 1, Revision 1, December 2019, Comprehensive Analysis and Test Program for GSI-191 Closure.

TABLE 6.2-1 REFUELING WATER STORAGE TANK AND CONTAINMENT SUMP

RWST

Manufacturer Company	Richmond Engineering Company
Model Number	3736-4x6-13DV
Quantity	1
Design Pressure	Atmospheric
Design temperature (°F)	120
Shell Material	SA-240, Type 304
Net Capacity (gal.)	475,000
Code	ASME Code, Section III, Class 3 (1971 edition)
Seismic	Class 1

Piping

Material	ASTM A-312, Type 304
2.5 inch and larger	Sch 10S
2 inch and smaller)	Sch 40S
Fittings	
2.5 inch and larger	Butt-welded except at flanged equipment
2 inch and smaller)	Socket-welded

Valves

2.5 inch and larger	Butt-welded, ANSI 600 lb rating stainless steel
2 inch and smaller)	Socket-welded, ANSI 600 lb rating stainless steel*
Standard - Piping	ANSI B 31.1.0
Code - Valves	Draft ASME Code for Pumps, and Valves for Nuclear Power, Class 2, 1968
Seismic	Class 1

TABLE 6.2-1 REFUELING WATER STORAGE TANK AND CONTAINMENT SUMP (CONTINUED)

Containment Sump	
Containment Sump Recirculating Piping	
Material	ASTM A-312, Type 304
24 inches	Sch 10S
Fittings	Butt-welded
Valves	Butt-welded, ANSI 150 lb rating, stainless steel
Standard	ANSI B 31.7 Class II
Seismic	Class 1
Containment Sump Screen	
Quantity	1
Approximate Overall dimensions (feet):	Two branches of headers 4 feet high, one header is 42 feet long, and one is 39 feet long. They are both connected to a 6 foot wide by 12 foot long by 4 foot high enclosure over the sump inlet piping.
Material:	304 / 304L or approved equal
Fin size:	37.75 inches high by 2 inches thick by various lengths
Fin perforation opening	1/16 inch
Free area (%) approximately:	40
Seismic:	Class 1
TSP Baskets	
Quantity	3
Overall dimensions (feet)	5 L x 5 x 1 H (approx.)
Material	TP 304 stainless steel

TABLE 6.2-1 REFUELING WATER STORAGE TANK AND CONTAINMENT SUMP (CONTINUED)

Wire Mesh	80
Quantity	2
Overall dimensions (feet)	5 L x 5 x 5 H (approximate)
Material	TP 304 stainless steel
Wire Mesh	80

* 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 600 lb, based on the design conditions of the specific application.

6.3 SAFETY INJECTION SYSTEM

6.3.1 DESIGN BASIS

6.3.1.1 Functional Requirements

The safety injection system must function to supply core cooling in the unlikely event of a loss-of-coolant accident (LOCA) for all breaks in the reactor coolant system piping up to and including the equivalent of a double-ended break in the largest coolant pipe; i.e., up to a flow area of 19.2 square feet (Section 14.6.5). This cooling must be sufficient to ensure that consequences are within the acceptance criteria of 10 CFR 50.46. The cooling function must be maintained for extended periods of time following a LOCA – until decay heat is sufficiently low that recirculation of borated water is no longer required.

The safety injection system also must inject borated water into the reactor coolant system in order to limit fuel damage and increase shutdown margin following a main steam line rupture (Section 14.1.5).

6.3.1.2 Design Criteria

The safety injection system is designed in accordance with AEC General Design Criteria 35, 36, and 37 in Appendix A to 10 CFR Part 50 and General Criteria as described in Section 6.1.

6.3.2 SYSTEM DESCRIPTION

6.3.2.1 System

The safety injection system is treated as an integrated system consisting of three complementary subsystems; i.e., high pressure and low pressure injection subsystems which utilize centrifugal pumps, and a passive injection subsystem which maintains a reservoir of borated water under pressure in the safety injection tanks. Figure 6.1–1 shows the piping and instrumentation diagrams of the safety injection system.

The high pressure system is capable of delivering emergency coolant when the discharge pressure decreases below approximately 1200 psia. Two high pressure safety injection pumps take suction from two independent suction headers. These headers are initially supplied with borated water from the refueling water storage tank and after that tank is nearly exhausted, borated water is recirculated from the sump of the containment. The third high pressure pump is available as an installed spare.

The low pressure safety injection system utilizes two, low pressure safety injection pumps. Each of the two pumps is connected to one of the two independent suction headers to assure an adequate supply of borated water.

The water from the safety injection tanks recovers the core following a reactor coolant system blowdown to minimize core damage until the safety injection pumps can provide adequate water

for reactor coolant. The tanks are designed to inject large quantities of borated water into the reactor coolant system immediately following a large pipe break. The water covers and cools the core, thereby limiting clad temperature and metal-water reaction (see Section 14.6.5. One tank is connected to each of the four reactor vessel inlet pipes. The driving head for water injection is provided by nitrogen gas pressure within the tanks at a minimum pressure of 200 psig. As the reactor coolant system pressure falls below tank pressure, check valves open in the line connecting each tank to the system. The tanks operate as a passive stored-energy safety feature; no outside power or signal is required for their operation. A remotely operated valve is provided to isolate the tanks during a normal depressurization of the reactor coolant system. Position of this valve is displayed in the control room. A vent isolation valve is provided on each tank to vent nitrogen into containment, when the tanks cannot be isolated during RCS depressurization following small break LOCA. A small drain valve controlled manually from the control room is used to drain any in-leakage from the reactor coolant system. The safety injection tanks are protected from overpressure by relief valves. Piping to each tank is arranged such that the operability of each tank can be demonstrated.

Safety injection actuation signal is initiated either when the pressurizer pressure drops below 1714 psia, or when the containment pressure rises above 4.42 psig. Diversity of the safety injection actuation signal is thus provided.

Upon initiation of safety injection, two high pressure and two low pressure safety injection pumps start and 12 safety injection line isolation valves receive signals to open, allowing water to be pumped from the refueling water storage tank into the reactor coolant system. After most of the water has been transferred from the tank, a continuous source of borated water is provided by recirculating containment sump water directly to the pump suctions. Recirculation is automatically initiated by low water level in the refueling water storage tank. Transfer to the recirculation mode may also be manually initiated.

The minimum time at which switch over to the recirculation mode (SRAS) occurs is based upon operation of two high pressure pumps, both low pressure pumps, and the two containment spray pumps. The system is designed to keep the core covered following initial safety injection. Following SRAS high pressure pump has sufficient capacity assuming spillage from a pipe break to maintain core water level.

In the recirculation mode, the high pressure safety injection pumps take suction directly from the containment sump. At the discretion of the operator, a portion of the cooled water from the containment spray system may be diverted to the suction of the high pressure injection pumps. This method of operation provides additional cooling margin, but is not necessary to meet core cooling requirements. However, there is no design basis whereby this flow path is utilized. The above description is maintained here to provide a basis for the physical plant design (i.e., valving).

The NPSH available to the high pressure safety injection (HPSI), low pressure safety injection (LPSI), and containment spray (CS) pumps has been calculated based on the following assumptions:

a. Pipe and fitting losses based on actual plant configuration;

b. Maximum and minimum combinations of HPSI, LPSI, and CS pumps operating during LOCA injection, recirculation, and boron precipitation control modes;

c. Assumed pressure drop for sump strainer and debris bed;

d. Containment pressure is equal to the saturation pressure of the containment sump water. For the recirculation modes, the containment sump temperature is 212°F and the containment pressure is 14.7 psia;

e. Minimum containment sump water level following SRAS for LBLOCA and SBLOCA.

The calculated available NPSH for the HPSI, LPSI, and CS pumps based on the above assumptions is greater than required NPSH for all operating modes evaluated.

The assumption that the containment pressure will be equal to the saturation pressure of the containment sump water was used in these NPSH calculation. In this method of calculating available NPSH, no credit is taken for increased containment pressure above the saturation pressure of the containment sump water.

The operator is not required to perform any immediate actions in the event of a LOCA except to reaffirm that automatic action of equipment and the notification and protection of plant personnel is under way. Operator actions following a LOCA are provided in plant Emergency Operating Procedures.

The safety injection water contains at least 1720 ppm of boron; consequently, the safety injection system also provides additional shutdown capability whenever the system is required to operate. This shutdown capability assists in maintaining the reactor subcritical following the rapid cooldown of the reactor coolant system caused by a rupture of a main steam line (see Section 14.1.5). The negative reactivity provided by the safety injection system exceeds the minimum required for cold shutdown.

The low pressure safety injection pumps are also used to supply coolant flow to remove heat from the reactor following reactor shutdown and to maintain a suitable temperature for refueling and maintenance operations (Refer to Section 9.3).

The system does not differ in any significant respect from those described in other C-E applications such as Calvert Cliffs (Docket Number 50-317 and 50-318) and Palisides (Docket Number 50-255).

The High Pressure Safety Injection system contains a discharge connection that facilitates portable diesel driven RCS Injection Pump deployment. This connection is a defense-in-depth design feature that is available for coping with an external loss of AC power (ELAP) event. The location of the discharge BDB RCS FLEX connection is shown on Figure 6.1–1, Sheet 2.

6.3.2.2 Components

high pressure Safety Injection Pump

The high pressure safety injection pumps are sized to ensure that one high pressure pump will keep the core covered at the start of recirculation, assuming spillage outbreak in one leg at atmospheric pressure.

The requirements for boron injection for the steam line break and the injection requirements for small break sizes are also analyzed to ensure that the pumps are adequately sized. The high pressure pumps are designed for the thermal transient conditions of 40°F to 300°F in ten seconds and 300°F to 40°F in ten seconds.

The high pressure pumps are seven-stage horizontal centrifugal units. Mechanical seals are used and are provided with leakoffs to the radwaste system which collects any leakage past the seals. The seals are designed for operation with a pumped fluid temperature of 350°F. To permit extended operation under these conditions, a portion of the pump fluid is externally cooled by the RBCCW system and recirculated to the seals. The pump motor is capable of starting and accelerating the pump to full speed with only 70 percent of rated voltage in eight seconds. The pumps are provided with drain and flushing connections to permit reduction of the radiation before maintenance. The pressure containing parts of the pump are stainless steel with internals selected for compatibility with boric acid solutions. The materials selected were analyzed to ensure that differential expansion during the design transient can be accommodated. The following inspections and tests were performed on the high pressure safety injection pumps:

a. materials used for pressure containing parts were inspected in accordance with techniques and acceptance standards of the Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, dated November, 1968;

b. pressure-containing parts were hydrostatically tested in accordance with API Standard 610.

The pumps are provided with minimum flow protection to prevent damage resulting from operation against a closed discharge. One Millstone Unit 2 pump was subjected to the following transient tests:

The pump was operated under conditions where suction temperature both increased 260°F in ten seconds and dropped 260°F in ten seconds. The pump was operated at rated speed for over ten minutes following the test in order to demonstrate satisfactory survival of the transient.

A full-scale hydraulic test was performed on each Millstone Unit 2 pump assembly. All pump test setups, test procedures and instrumentation were in accordance with the Standards of the Hydraulic Institute and the ASME Power Test Code, PTC-8.2. This included verification of satisfactory operation at the stated NPSH. Figure 6.3–2 shows the pump performance obtained during the hydraulic testing.

The high pressure pump data are shown in Table 6.3-2.

The design temperature for the high pressure safety injection pumps is based upon the saturation temperature of the reactor coolant at the containment design pressure, about 300°F, plus a design tolerance of 50°F, resulting in design temperature of 350°F. The design pressure for the high pressure pumps is based upon the sum of the containment spray pump shutoff head and the shutoff head of the high pressure pump.

low pressure Safety Injection Pumps

The low pressure safety injection pumps are vertical, single-stage centrifugal units equipped with mechanical face seals backed up by a bushing, with a leakoff to collect the leakage past the seal. To permit extended operation at the design temperature of 350°F, a portion of the pump discharge is cooled by reactor building closed cooling water and is used to cool the seals. The pump motor is capable of starting and accelerating the pump to full speed with only 70 percent of rate voltage with eight seconds. The pumps are provided with drain and flushing connections to permit reduction of radiation levels before maintenance. The pressure-containing parts are fabricated from stainless steel; the internals are selected for compatibility with boric acid solutions. The pumps are provided with minimum flow protection to prevent damage when starting against a closed system. The low pressure pump data are summarized in Table 6.3-2.

The following inspections and tests were performed on the low pressure safety injection pumps:

a. materials used for pressure-containing parts were inspected in accordance with techniques and acceptance standards of Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, dated November, 1968;

b. pressure-containing parts were hydrostatically tested in accordance with API Standard 610.

A full-scale hydraulic test of the pumps was performed. Figure 6.3–3 shows the pump performance obtained during the testing.

“One Millstone Unit 2 pump was subject to a transient test consisting of operating while its suction temperature was increased 260°F in less than ten seconds. The pump was operated at rated speed for over ten minutes following the transient to demonstrate satisfactory survival of the test.”

The design temperature for the low pressure safety injection pumps is based upon the temperature of the reactor coolant at the initiation of shutdown cooling, about 300°F nominal, plus a design tolerance of 50°F, resulting in 350°F. The design pressure for the low pressure pumps is based

upon the sum of the maximum pump suction pressure, which occurs at the initiation of shutdown cooling, and the pump shutoff head. This yields a nominal design pressure of 500 psig.

Safety Injection Tanks

The four safety injection tanks are used to flood the core with borated water following a depressurization as a result of a LOCA.

The volume of water in the safety injection tanks is selected so that the contents of three of the four tanks injecting into the reactor coolant system following the worst LOCA will cover the top of the core.

The tank gas/water fractions, gas pressure, and outlet pipe size are selected to ensure sufficient safety injection is available such that 10 CFR 50.46 criteria are met for all postulated LOCAs.

The tanks contain borated water at a minimum boron concentration of at least 1720 ppm. The tanks are pressurized with nitrogen at a minimum pressure of 200 psig.

Level and pressure instrumentation is provided to monitor the availability of the tanks during plant operation. The position of the motor operated isolation valves (SI 614, 624, 634 and 644) between the safety injection tanks and headers is displayed by the use of lights on the main control boards. An alarm is annunciated if any of these valves are closed. Signals are also provided to open any of these valves on SIAS, or when RCS pressure during plant heatup increases to approximately 280 psig.

The tanks are provided with vent isolation valves SI-613, 623, 633 and 643 which are assisted by nitrogen from the tanks for remote operation.

Provisions have been made for sampling, filling, draining, venting and correcting boron concentration. The tanks are carbon steel internally clad with Type 304 stainless steel. Design construction and overpressure protection are in accordance with the ASME Code Section III, Class C, 1968 Edition through Summer 1969 Addendum. The data summary for the safety injection tanks is given in Table 6.3-2.

Shutdown Cooling Heat Exchanger

The shutdown cooling heat exchangers are used to remove decay and sensible heat during plant cooldowns and cold shutdowns. The units are sized to hold refueling water temperature (130°F) with the design RBCCW temperature (95°F) 27.5 hours after shutdown following an infinite period of operation.

The units are further specified to accept a 40°F to 289°F transient in ten seconds when the containment spray pump suction is switched to the containment sump. During this period of operation, the tube-side flow is specified as the output of two containment spray pumps and the reactor building closed cooling water inlet temperature is specified at 115°F. The units are designed and constructed to the standards of ASME, Section III, Class C; and TEMA Class R

requirements. The units are of a U-tube design with two tube-side passes and one shell-side pass. The tubes are austenitic stainless steel and the shell is carbon steel.

The design parameters for the shutdown heat exchangers are found in Table 6.3-3, Section 6.3. Data on containment spray pump operation is available in Table 6.4-1, Section 6.4. Data on reactor building closed cooling water system operation is available in Table 9.4-1, Section 9.4. A description of integrated system performance during a Main Steam Line Break or LOCA is contained in Section 14.8.2.

Table 9.3-1 shows the design cases for the shutdown heat exchangers for accident transients.

The design temperature and pressure for the shutdown cooling heat exchangers are compatible with the design temperature and pressure for the low pressure safety injection pumps (see Section 6.3.2.2).

Safety Injection Piping

The safety injection system piping is fabricated of austenitic stainless steel and conforms to the standards set forth in ANSI B31.7. Flexibility and seismic loading analyses have confirmed the structural adequacy of the system piping.

Piping is shop fabricated in accordance with the ANSI B31.7, Nuclear Power Piping. The piping is field erected and tested in accordance with ASME Section III, 1971. Material and shop welds were inspected, tested and documented in accordance with the applicable class of B31.7. Valves are manufactured in accordance with the Draft ASME Code for Pumps and Valves for Nuclear Power, dated November 1968. The material and welds were inspected, tested and documented to the same code.

6.3.3 SYSTEM OPERATION

6.3.3.1 Emergency Conditions

A condition which causes a sufficiently low pressurizer pressure or high containment pressure will result in a safety injection actuation signal (SIAS). This signal will start the two high pressure safety injection pumps, both low pressure safety injection pumps, open 12 safety injection system isolation valves and close the four check valve leakoff lines at the safety injection tanks. (The SIAS also performs some functions in the Chemical and Volume Control System; see Figures 7.3–2A through 7.3–2D.) The 8 HPSI injection throttle valves are manually positioned to the desired open position.

When reactor coolant system pressure falls below approximately 1200 psia, the high pressure safety injection pumps start delivering flow through both high pressure headers.

If reactor coolant pressure falls below approximately 200 psig, the passive pressurized safety injection tanks will start delivering borated water into each cold leg along with the low pressure safety injection pumps. In the event of a Small Break LOCA, the Safety Injection Tanks (SITs)

will begin discharging borated water into the RCS when the RCS pressure falls below approximately 200 psig. During recovery from a Small Break LOCA, once the operators have control over the plant cooldown and depressurization, it is undesirable to have the SITS completely discharge to the point where nitrogen is introduced into the RCS. Therefore, the operators are directed to close the SIT outlet valves once RCS pressure is reduced.

If a single failure loss of power to the SIT outlet MOV(s) was to occur, the SIT(s) would be unisolable from the control room. Therefore, the operator would be directed to operate the SIT venting system and depressurize the SIT(s) by bleeding off nitrogen cover pressure.

The safety injection pumps initially draw borated water from the refueling water storage tank (RWST). This tank has sufficient water volume to supply safety injection flow assuming two high pressure and two low pressure safety injection pumps and two containment spray pumps are running. When the refueling water storage tank is nearly empty, a sump recirculation actuation signal (SRAS) from refueling water storage tank water level closes the minimum flow recirculation lines, opens the isolation valves in the two lines from the containment sump and shuts down the low pressure safety injection pumps. The refueling water storage tank outlet valves remain open initially during the transfer to the recirculation mode to preclude the loss of supply to a high pressure safety injection pump in the unlikely event the isolation valve in the containment sump line should experience delay in opening. Back flow through either refueling water storage tank suction line is prevented by check valves. In addition, the operator may manually close the refueling water storage tank outlet valves after verifying the opening of the containment sump line valves. The earliest automatic recirculation (SRAS) would occur is when 2 LPSI, 2 HPSI, and 2 CS pumps are in operation following a SIAS and the RWST contains the minimum Technical Specification volume. The recirculation mode can also be initiated manually by the operator.

The ESF pump minimum-flow recirculation line contains two series air-operated stop valves whose SRAS function is to close and prevent sump water from being returned to the RWST.

These valves are provided with key-lock switches in the Control Room to disable the valves open during normal operations to ensure a recirculation flow path for the low pressure safety injection pumps. The valves are enabled to be closed by placing the key-lock switches to “operate” prior to recirculation following a LOCA.

In the recirculation mode the high pressure safety injection pumps take suction from the containment sump. The safety injection flow spilling from the break in the reactor coolant system is cooled by mixing in the containment sump with the cooled containment spray water.

During normal operation, the containment sump recirculation lines between motor operated valves (2-CS-16.1A and 2-CS-16.1B) and the high pressure safety injection pumps (See Figure 6.1–1) will be filled with stagnant water. The portion of piping between the containment sump inlets and the motor-operated valves is designed to be full of water.

These portions of line will flood gradually as the sump level rises. In portions of the lines where air pockets would form, vent and fill connections are provided.

Figure 6.3–1 shows the routing of the recirculation lines and the provisions for venting air that may be trapped on these lines. The internal dimensions of the piping and the valves are specified in Table 6.3-5.

The analysis of the design basis accident for the safety injection system (Section 14.6.5) is performed assuming that only a minimum amount of the system components function during the LOCA. Minimum safety injection is defined as follows:

a. For a cold leg large break LOCA, the entire contents of one of the four passive safety injection tanks spill through the break and is not available for emergency core cooling. For a Small Break LOCA downstream of the ECCS Injection location, ECCS is injected into the broken loop and the portion of this injection which spills out the break is calculated as a part of the transient simulation. For a hot leg break, all four safety injection tanks discharge into the intact cold legs;

b. One of the two online, high pressure safety injection pumps functions. For a cold leg large break, the flow from one of the four injection lines is assumed to be spilled via the break. For a cold leg small break, flow is injected into the broken loop and the portion of this injection which spills out the break is calculated as part of the transient simulation. For a hot leg break, the entire pump discharge reaches the core;

c. One of the two low pressure safety injection pumps functions. For a cold leg break, one-half of this pump’s discharge reaches the core. For a hot leg break, the entire discharge from this pump reaches the core.

The assumed maximum delay time for operation of the safety injection pumps is 45 seconds after the SIAS, a 45 second delay for the LPSI system and a 25 second delay for the HPSI system. This is based on the assumption that outside power has been lost and includes an allowance for the start up and loadings of the emergency diesel generators. These and other conservative assumptions are given in the Loss of Coolant Analysis, Section 14.6.5.

The safety injection system can provide both long term cooling and boron precipitation control in the event of a LOCA. This is accomplished by simultaneous injection to both the hot and cold legs of the RCS. Hot leg injection is necessary for cold leg breaks to provide a flushing flowpath through the core region such that boric acid solubility limits are not reached during post-LOCA boiloff periods. The preferred method of boron precipitation control is to employ one LPSI pump to inject via the shutdown cooling system warmup and return line piping, past 2-SI-400, 2-SI-709, 2-SI-651 and 2-SI-652, into the RCS hot leg. Cold leg injection would be provided by a HPSI pump and also by LPSI flow diverted through at least one of the four LPSI injection lines. No more than two of the LPSI injection valves 2-SI-615, 625 and 645 are to be open in this configuration and valve 2-SI-635 cannot be open in combination with any other open LPSI injection valve.

An alternate method of providing boron precipitation flushing flow is to use HPSI pump P-41A to inject via the charging system header to the pressurizer auxiliary spray line and thus to the hot leg

through the pressurizer surge line. This alternate alignment is provided by flow past manually opened 2-CH-440 and 2-CH-340 through the regenerative heat exchanger, and past 2-CH-517 to the auxiliary spray line. Normal HPSI injection lines to the cold leg from pump P-41A are blocked by closing 2-SI-617, 627, 637 and 647. In this arrangement, cold leg injection is accomplished by the LPSI system.

Adequate flushing flow to preclude boron precipitation as well as adequate long term cooling are provided by either the preferred LPSI or alternate HPSI hot leg injection methods.

6.3.3.2 Cold Shutdown and Refueling

To obtain cold shutdown, operation of the system is discussed in Section 9.3. During the refueling, the shutdown cooling heat exchangers are aligned with the low pressure safety injection pumps and used to cool the refueling water.

6.3.4 AVAILABILITY AND RELIABILITY

6.3.4.1 Special Features

The design basis and system requirements during a DBI are met with the operation of the safety injection tanks and one high pressure and one low pressure safety injection pump, delivering rated flow and assuming spillage through the break as defined in Section 6.3.3.1. During recirculation, one high pressure safety injection pump has sufficient capacity to maintain the water level in the reactor vessel above the core.

Ability to meet the core protection criteria is assured by the following design features:

a. A high capacity passive system (safety injection tanks) which requires no power source and will supply large quantities of borated water to rapidly recover the core after a major LOCA up to a break of the largest reactor coolant system pipe.

b. low pressure and high pressure pumping and water storage systems with internal redundancy which will inject borated water to provide core protection for reactor coolant system break sizes equal to and smaller than the largest line connected to the reactor coolant system (the 12 inch pressurizer surge lines or the shutdown cooling and safety injection lines). The pumping systems also provide borated water to keep the core covered and to continue cooling the core after the passive water supply has been exhausted. In addition, the high pressure system will remove reactor core decay and sensible heat during long term operation after the reactor coolant system rupture. Instrumentation and sampling provisions allow monitoring of the recirculated coolant.

c. Separated pump rooms and redundant pumping systems which will permit minimum safety features equipment to operate should one pump room flood in the event of a passive failure during long term operation.

d. Redundant on site power supplies in the form of two emergency diesel generators, each of which has sufficient capacity for minimum safety features operation.

e. All active components which must function individually for the system to meet the performance criteria stated for core protection can be tested during reactor operation. Instrument sensors are tested for function at operating conditions. In addition, extensive shop and preoperational tests are performed to verify adequate component and system operation.

f. Most of the active components are located outside the containment where they are protected from incident-generated missiles and from post-incident environmental conditions. Those active components located inside the containment need only operate for a short time period after the accident.

g. The four injection lines are arranged such that movement of a ruptured reactor coolant pipe will not cause a subsequent failure of injection lines in non ruptured loops. The maximum movement of the reactor coolant pipe at the injection nozzle in the non ruptured loop will not damage the injection line.

h. The safety injection system have been designed to meet the single failure criterion. This includes the fluid systems and the electrical control and instrument systems. All pumps and critical power-operated valves can be actuated from their respective switch gear or control centers.

i. All components, piping, cabling structures, power supplies, etc., in the safety injection support systems are designed to Class 1 seismic criteria.

Effectiveness of the safety injection system to satisfy the criteria stated for core protection can be shown by the blowdown and refill transient curves following a LOCA. This analysis is presented in Section 14.6.5, Loss-of-Coolant Accident. The analysis shows that 10 CFR 50.46(b), “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants,” is satisfied by this design.

The high pressure safety injection system is designed to minimize the amount of equipment which must operate when a safety injection signal is received. All valves not required to operate on initiation of safety injection are either isolated from the safety injection flow path or locked in the safety injection position during operation. Administrative controls ensure that the locked valves are in the correct position.

Three parallel pumps are provided in the high pressure system. One pump is available as a spare. These pumps are located outside the containment. The pump rooms are in a controlled access area.

The pump room location outside of the containment is most favorable for extended operation and equipment life following a major LOCA. This location is accessible for service and inspection of

the engineered safety features systems components during plant operation and during the period of long term cooling following the postulated LOCA.

The pumps are appropriately segregated in separate watertight rooms. This arrangement permits access to and operability of those pumps required for minimum safety features operation.

Redundant flow paths are provided from the discharge of the high pressure injection headers. These headers, in turn, both supply the four individual safety injection lines, one leading to each cold leg of the reactor coolant system. Normally closed, power-operated valves isolate the two high pressure safety injection headers from each other and also provide the capability to switch the discharge from the pumps to either header.

Normal plant operating procedures include routine testing to ensure the operability of the pumps. The attention given to the selection of these pumps, the redundancy of power supplied, the design margins, and the fact that multiple pumps are installed assures a high degree of pump availability.

The high pressure safety injection valves are designed for 2735 psig. These valves are located outside of the containment and are thus not subjected to the environmental conditions existing in the containment following any LOCA. The attention given to the selection of these valves, design margins, and the fact that eight high pressure safety injection valves are in parallel assure a high degree of valve availability. Both header supply valves are normally open.

Four high pressure safety injection valves on one header are powered from one emergency power bus. The remaining four high pressure safety injection valves are powered from the second emergency power bus. The valves are equipped with remote position indicators in the control room. The 8 HPSI injection throttle valves are manually positioned to the desired open position and do not open further on an actuation signal.

During recirculation the high pressure safety injection pumps continue to operate, taking suction from the containment sump. The pump recirculation lines, the heat exchangers, the containment spray pumps, and the recirculation suction headers are arranged to provide two independent flow paths. The low pressure safety injection pumps can also be used for recirculation, if necessary.

The low pressure safety injection system is also designed to minimize the amount of equipment which must operate when a safety injection signal is received. All valves not required to operate on initiation of safety injection are either isolated from the safety injection flow path or locked in the safety injection position during operation. Administrative controls insure that the locked valves are in the correct position.

Normal plant operations, augmented by routine testing, insure the operability of the pumps. The attention given to the selection of these pumps, the redundancy of power supplied, the design margins, and the fact that two pumps are supplied assures a high degree of pump availability.

The power-operated, low pressure injection valves are located outside of the containment and are thus not subjected to the environmental conditions existing in the containment following any LOCA. The attention given to the selection of these valves, design margins, and the fact that four

low pressure safety injection valves are in parallel assure a high degree of valve availability. The safety injection valves are open automatically on initiation of safety injection.

Should the inadvertent closure of valve 2-SI-306 in the Low Pressure Injection System line occur, it would result in the inability of that system to perform its function following a LOCA. It is equipped with both a power (pneumatic) actuator and a manual overriding actuator connected to opposite sides of the valve plug shaft. Each half of the valve plug shaft is splined into the valve plug. The power actuator is a spring-to-open diaphragm type. To prevent inadvertent closure by this device, solenoid valve HV-306, which is located in the pneumatic signal line to the diaphragm of the actuator, will be electrically disabled during plant operation to prevent air pressure from reaching the diaphragm. As a result, the actuator spring will hold the valve plug in an open position. Spring force is sufficient to hold the valve in the open position. To provide additional assurance that the valve will not close, the manual operator on the opposite side of the shaft is pinned and locked to the handwheel to prevent movement of the valve plug due to the mechanical advantage of the handwheel drive nut. Also the handwheel is locked in position to prevent inadvertent operation. Thus, multiple means are provided to assure that valve 2-SI-306 will remain in an open position during all operations except shutdown cooling and that it will allow the system to perform its function following a LOCA.

The four safety injection tanks comprise a completely independent and redundant source of low pressure injection water which requires no outside signal or source of power operation. The analysis in Section 14.6.5 shows that core recovery will occur for all postulated LOCAs such that 10 CFR 50.46 acceptance criteria are met.

A failure modes and effects analysis has been performed of the safety injection system. The analysis for both the injection and recirculation modes of operation can be found in Table 6.3-6.

From the analysis, it has been concluded that the SIS can withstand any single failure as defined therein and still perform its intended function.

There is no undue risk to the health and safety of the public from the failure of a single active component during the injection mode of operation or from a single failure of any passive or active component during the recirculation mode of operation.

The failure mode analysis presented in Table 6.3-6 was performed using the following assumptions.

a. The SIS is composed of the electrical, instrumentation and fluid segments. In compliance with single failure criteria, only one failure is assumed to occur in the system; e.g., a failure of a diesel generator cannot occur simultaneously with a safety injection valve failure.

b. Only one active failure is considered for injection mode of the safety injection operation. For the recirculation mode, the single failure considered can be either active or passive.

c. Failure to respond to an external signal is considered an active failure.

d. Failure of valve internals, including check and stop valves, is a passive failure. Passive failures are considered in the recirculation phase only, no earlier than 24 hours after an accident.

e. The transition to the recirculation mode of operation occurs upon initiation of recirculation from the sump.

f. The analysis considers only failures or malfunctions which occur during the time period of SIS operation. Failures or malfunctions that might occur during normal reactor operation are not considered.

Abbreviations used in the Table 6.3-6 are:

CRI - Control Room Indication

HPSI - high pressure Safety Injection

LPSI - low pressure Safety Injection

SIAS - Safety Injection Actuation Signal

SRAS - Sump Recirculation Actuation Signal

RWST - Refueling Water Storage Tank

RCS - Reactor Coolant System

SIS - Safety Injection System

An administrative error analysis has been performed which evaluates the effect of improper positioning of administratively controlled safety injection system valves. The results of this analysis can be found in Table 6.3-7.

The HPSI and LPSI systems may be relied upon to provide long term cooling and boron precipitation flushing flow in the event of a LOCA. A break size of sufficient magnitude that the reactor coolant system is not filled at 8 to 10 hours after the start of the accident requires a simultaneous hot and cold leg injection alignment to provide both long term cooling and boron precipitation control because the break location is unknown.

Operator action both inside and outside the control room would be required to align for simultaneous injection.

The preferred method of boron precipitation control is to have LPSI pump injection to the RCS hot leg past opened valves 2-SI-400, 709, 651 and 652 in the shutdown cooling system warmup and suction piping. Some of the LPSI flow would be diverted to the cold leg by having at most two of valves 2-SI-615, 625 and 645 open. LPSI injection valve 2-SI-635 cannot be open in combination with any other open LPSI injection valve. The ability to align an alternate vital power source to 2-SI-651 in the shutdown cooling return line ensures single-failure criteria are

met when aligning for boron precipitation control. Cold leg injection is provided by a HPSI pump and the diverted LPSI flow.

An alternate method of boron precipitation and long term cooling can be accomplished by aligning HPSI pump P-41A to the charging header past valves 2-CH-440, 340, the regenerative heat exchanger and 2-CH-517 for injection to the RCS hot leg through the pressurizer auxiliary spray line and surge line. The ability to align an alternate vital power source to 2-CH-517 and 2-CH-519 in the charging lines ensures single-failure criteria are met.

Adequate margin to HPSI and LPSI pump NPSH exists for the various boron precipitation control alignments.

A failure modes and effects analysis has been completed for post-LOCA periods of combined hot and cold leg injection necessary for boron precipitation control and long term cooling. The results of this analysis are presented in Table 6.3-8.

See Section 14.6.5.3 for a description of boron precipitation control and long term cooling under post-LOCA conditions.

6.3.4.2 Tests and Inspections

Each safety injection pump is shop tested for hydraulic performance at sufficient head-capacity points to generate complete performance curves. Figures 6.3–2 and 6.3–3 show the resultant curves for one high pressure and one low pressure safety injection pump, respectively.

Nondestructive examinations are performed on all pressure-retaining components of each safety injection pump and tank in accordance with the Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968, and ASME Boiler and Pressure Vessel Code, 1968 Edition through Summer 1969 Addendum, Section III, Class C, respectively. The safety injection system undergoes a preoperation test prior to plant startup. The test procedure is described in Chapter 13.

The following preoperational tests and checks are planned:

1. Each of the high pressure safety injection (HPSI) pumps will be capacity tested while discharging into the reactor vessel through the HPSI injection lines. Tests will be run with various discharge heads to verify pump capacity over the range of break sizes where its function is of most importance. During these tests, flow distribution through the four HPSI line flow orifices will be checked.

2. Each of the low pressure safety injection (LPSI) pumps will be capacity tested while discharging into the reactor vessel through the (LPSI) injection lines. Tests will be run with discharge heads associated with maximum, design and minimum recirculation flow rates. During these tests, flow distribution through the four LPSI line flow orifices will be checked.

3. Each diesel generator will be tested to check its ability to pick up and operate ECCS equipment.

4. A simulated SIAS signal will be used to verify functional operation of all valves and pump breakers.

5. A simulated sump recirculation actuation signal (SRAS) will be impressed on each detector and functional operation of all valves and pump breakers.

6. Each Safety Injection tank will be tested by allowing its contents to be discharged into the reactor. Tank volume and pressure change versus time will be checked.

Periodic tests and inspections of the Safety Injection System components and subsystems are performed to insure proper operation in the event of an accident. The scheduled tests and inspections are necessary to verify system operation reliability since during normal plant, safety injection system components are aligned for emergency operation and serve no other function. The tests defined permit a complete checkout on the subsystem and component level during normal plant operation. Satisfactory operability of the complete system may be verified during normal scheduled refueling shutdowns. The test data recorded during the periodic testing provides positive assurance of the system to perform its intended function. Deteriorated performance will be detected by comparison of the periodic performance test data to the preoperational test data.

Routine operational testing of major portions of the logic circuits, pumps and power-actuated valves in the safety injection system is described in Section 7.3.

The pumps are located outside the containment for access and to permit maintenance during normal plant operations. A recirculation line is provided on the discharge of each pump. Periodic testing will be performed by recirculating water back to the refueling water storage tank.

Surveillance requirements to verify that the safety injection system will respond promptly and perform its intended function, if required, are given in Sections 4.5.1 and 4.5.2 of the Technical Specifications.

TABLE 6.3-1 OMITTED

TABLE 6.3-2 PRINCIPAL DESIGN PARAMETERS OF THE SAFETY INJECTION SYSTEM

Injection Water Boron Concentration (ppm)	1720
High-Pressure Safety Injection Pumps	
Manufacturer	Ingersoll-Rand
Quantity	3
Type	Seven-stage Horizontal Centrifugal
Motor Voltage, volts	4160
Design Pressure, psig	1600
Design Temperature, °F	350
* Design Flow (per pump), gpm (excluding minimum flow)	315
* Design head, ft.	2500
Pumped fluid	Water (Borated)
Temperature of Pumped Fluid, °F	35-300
Shutoff Head, ft	2850
* Maximum Flow, gpm	640
* Head at Maximum Flow (one pump), ft	1620
Material	ASTM-A-351 Gr CF8M
Horsepower (motor)	400
Shaft Seal	Mechanical
Acceleration Time, seconds (at rated voltage)	4
Minimum Flow, gpm	20
* NPSH Available (minimum), ft (at 640 gpm)	21.9
* NPSH Required at 640 gpm, ft	20
Design Maximum Suction Pressure, psig	250
Low-Pressure Safety Injection Pumps	
Quantity	2
Type	Single stage Vertical Centrifugal
Motor Voltage, volts	4160
Manufacturer	Ingersoll-Rand
Design Pressure, psig	500

TABLE 6.3-2 PRINCIPAL DESIGN PARAMETERS OF THE SAFETY INJECTION SYSTEM (CONTINUED)

Design Temperature, °F	350
Design Flow (per pump), gpm (excluding min. flow)	3000
* Design Head, ft	350
* Pumped Fluid	Water (Borated)
Temperature of Pumped Fluid, °F	35-300
Shutoff Head, Ft	420
* Maximum Flow, gpm	4500
* Head at Maximum Flow (one pump), ft	275
Basic Material	ASTM A351 GR CF8M
Horsepower	400
Seals	Mechanical
Acceleration Time, seconds (at rated voltage)	4
* NPSH Available (minimum), ft (at 3000 gpm)	25
* NPSH Required at 3000 gpm, ft	13
Design Maximum Suction Pressure, psig	300
Minimum Flow, gpm	100
Safety Injection Tanks	
Manufacturer	Air Preheater Co.
Quantity	4
Total Volume, each, ft^3	2019
Water Volume, ft^3, nominal	1137
Water Volume, ft^3, minimum	1080
Design Pressure, psig	250
Operating Pressure, psig, nominal	215
Operating Pressure, psig, minimum	200
Design Temperature, °F	200
Operating Temperature, °F	120
Relief Valve Setpoint, psig	250
Height, inches	399.75
Outside Diameter, inches	109 3/4

TABLE 6.3-2 PRINCIPAL DESIGN PARAMETERS OF THE SAFETY INJECTION SYSTEM (CONTINUED)

* The pump flows, heads, and NPSH values presented in this table are based on the original design specifications. The pumps will actually operate over a range of conditions during safety injection, sump recirculation, long-term cooling and boron precipitation control post LOCA. System hydraulic and NPSH analyses have been performed for these operational modes based on the pump design and degraded capacities.

Note: The original design basis for available NPSH was based on a total sump recirculation flow on one suction header of 2980 gpm (consisting of 1 CS pump flow of 1700 gpm and 2 HPSI pumps flow of 640 gpm each).

TABLE 6.3-3 SHUTDOWN COOLING HEAT EXCHANGER DATA

Manufacturer	Engineers & Fabricators, Inc.
Quantity	2
Type	Shell and Tube
Codes	
Tube Side, Shell Side	ASME Section III, Class C 1968 Edition through Summer 1969 Addendum
Tube Side	
Fluid	Reactor Coolant, 1.5 wt. % Boric Acid
Design Pressure, psig	500
Design Temperature, °F	400
Maximum pressure drop at rated flow, psi	10
Materials	Austenitic Stainless Steel
Shell Side	
Fluid	Reactor Building Closed Cooling Water
Design Pressure, psig	150
Design Temperature, °F	250
Maximum pressure drop at rated flow, psi	10
Material	Carbon Steel
Heat Exchanger Design Parameters (1) (Shutdown Cooling Mode)	
Tube Side	
Flow, Million lb/hr	1.5
Inlet Temperature, °F	130
Outlet Temperature, °F	111.9
Shell Side	
Flow, Million lb/hr	2.41
Inlet Temperature, °F	95
Outlet, Temperature, °F	106.3
Heat Load, Million Btu/hr	27.2
Service Transfer Rate, Btu/hr - ft^2 - °F	256

(1) Refer to Table 9.3-1 for shutdown cooling minimum design basis parameters.

TABLE 6.3-4 OMITTED

TABLE 6.3-5 INTERNAL DIMENSIONS OF CONTAINMENT SUMP RECIRCULATION PIPING AND VALVES

Pipe

Component	Internal Diameter (inches)
24"-HCB-1	23.5
8"-HCB-1	8.329
8"-GCB-3	8.125
6"-GCB-3	6.357

Valves

Component	Internal Diameter (inches)
2-CS-16.1A, B	23.25
2-CS-15A, B	23.5
2-SI-401, 410	8.0
2-SI-402, 470	6.0
2-SI-411, 412	8.0

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS HIGH PRESSURE SAFETY INJECTION

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
HPSI Loop Flow Indicator (4)	Malfunction	None	Comparison of Flow to Other Flow Indicators and Valve Position Indicators	CRI	
HPSI Valve (8)	Fails to open	None	Flow Indication, Pressure Indication, Valve position Indication	CRI	The valves are normally open and positioned in the throttled position.
Pump Discharge Isolation Valve (3)	None	None, “Locked Open” valve	N/A*	N/A *	
Pump Minimum Flow Recirculation Stop Valve (3)	None	None, “Locked Open” valve	N/A *	N/A *	
HPSI Pump (3)	a. Fails to Start	Loss of HPSI flow from 1 pump	Pump Motor Lights, Header Pressure	CRI	Flow from at least one pump available.
	b. Stops	Loss of HSPI flow from 1 pump	Pump Motor Lights, Header Pressure	CRI	Flow from at least one pump available.
Pump Suction Isolation Valve (2)	None	None, “Locked Open” valve	N/A *	N/A *	

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS
HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
HPSI Discharge Check Valve (4)	Fails Closed	1. Valve in ruptured loop-no effect.	Flow Indication, Pressure Indication	CRI	Flow is sufficient to cool core.
		2. Valve in nonrupture loop --at least 67% of HPSI flow from 2 pumps reaches core.			
HPSI Loop Flow Indicator (4)	Malfunction	None	Comparison of flow to other flow indicators and Valve Position Indicators	CRI	Balancing of flow is not essential during long-term cooling.
Check Valve Associated with HPSI Valves (8)	Fails Closed	1. Valve in ruptured loop--no effect.	Flow Indication, Pressure Indication	CRI	Flow is sufficient to cool core.
		2. Valve in nonruptured loop--at least 70% of HPSI flow from 2 pumps reaches core			

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS
HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
HPSI Valve (8)	Fails Closed	1. Valve in ruptured loop--no effect.	Flow Indication, Pressure Indication	CRI	
		2. Valve in nonruptured loop but flow to core still greater than minimum required	Valve Position Indication		
HPSI Pump Discharge Relief Valve (3)	Fails Open	Partial Loss of HPSI flow	None	CRI	Loss of flow will be insignificant.
Pump Discharge Isolation Valve (3)	Fails Closed	Loss of HPSI flow from 1 pump	Flow Indication, Pressure Indication	CRI	At least 1 HP pump available. Each pump is full capacity.
Pump Minimum Flow Recirculation Stop Valve (3)	Fails Closed	None, valve is not necessary after RAS	None		
Pump Discharge Check Valve (3)	Fails Closed	Loss of HPSI flow from 1 pump	Flow Indication, Pressure Indication	CRI	At least 1 HP pump is available.

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
HPSI Pump (3)	a. Stops	Loss of HPSI flow from 1 pump	Pump Motor Lights, Header Pressure	CRI	At least 1 HP pump is available
	b. Seal Failure	Slight reduction in output from affected pump	Sump Level Alarm	CRI	At least 1 HP pump is available
	c. Loss of Seal Coolant	Loss of HPSI flow from 1 pump	None	None	At least 1 HP pump is available
Pump Suction Isolation Valve (2)	Fails Closed	Loss of suction to 1 HP pump	Pressure Indication, Possible Flow Indication	Local CRI	At least 1 HP pump is available
Pump Suction Check Valve (2)	a. Fails Closed (no cooled suction)	Loss of suction to 1 pump	Pressure Indication, Possible Flow Indication	CRI	At least 1 HP pump is available
	b. Fails Open (cooled suction)	Loss of suction to 1 pump	Flow Indication, Pressure Indication	CRI	At least 1 HP pump is available
Cooled Suction Isolation Valve (2)	Fails Closed when cooled suction is desired	Loss of cooled suction to the suction header	Pressure Indication, Flow Indication	CRI	Pump design is based on suction from the containment sump without use of the shutdown heat exchangers.

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
Safety Injection Pump Minimum Flow Recirculation Stop Valve (2)	Fails Open	None	Valve Position Indicator	CRI	2 Valves are in series.
LPSI Valve (4)	Fails to Open	1. Valve in ruptured loop - no effect	Valve Position Indication	CRI	50% of flow from one L.P.S.I. pump is adequate
		2. Valve not in ruptured loop - at least 50% of LP flow reached core	Flow Indication, Pressure Indication		
L.P.S.I. Segment to Shutdown Cooling Heat Exchange Isolation Valve (1)	None	None	N/A *	N/A *	
Shutdown Cooling Flow Control Valve (1)	None	None	N/A *	N/A *	Failed open valve
Pump Discharge Isolation Valve (2)	None	None	N/A *	N/A *	Locked open valves
Pump Minimum Recirculation Stop Valve (2)	None	None	N/A *	N/A *	Locked open valves

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
LPSI Pump (2)	a. Fails to Start	Loss of L.P.S.I. flow from one pump	Pump Indication Lights	CRI	Other pump starts.
	b. Stops	Loss of L.P.S.I. flow from one pump	Flow Indication	CRI	Other pump starts simultaneously. Each pump is full capacity
	c. Fails to stop with the SRAS	Pumps may cavitate unless flow is reduced	Pump Indication Lights, Flow Indicator Fluctuations	CRI	Operator may stop pumps or close LPSI valves. Operator action ensures pressure drop in ECCS suction strainer remains within design limits.
Pump Suction Isolation Valve (2)	None	None	N/A *	N/A *	Locked open valves
S.I. Tank Isolation Valve (4)	a. None	None	N/A *	N/A *	Valves are maintained open with closing coil removed
	b. Fails to close following Small Break LOCA	Intrusion of nitrogen into the RCS at low RCS pressure	SI tank level and pressure; pressurizer pressure < 600 psia	CRI	Vent Isolation valves on affected tanks can be operated remotely to release nitrogen into containment
Drain and Fill Isolation Valve (4)	None	None, valve is “Fail Closed” valve	N/A *	N/A *	
Check Valve Leakage Control Valve (4)	None	None, valve is “Fail Closed” valve	N/A *	N/A *	

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS
HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
S.I. Tank Purge Valve (4)	None	None, valve is "Fail Closed" valve	N/A *	N/A *	
N_2 Supply Line Stop Valve	None	Valve is "Fail Closed" valve	N/A *	N/A *	
S.I. Tank Vent Isolation Valves	a. None	None, valves are "Fail Closed"	N/A *	N/A *	
	b. Fails to open to bleed nitrogen following SBLOCA	Nitrogen intrusion into RCS	Valve position indication	CRI	Single failure of SIT outlet valve already considered. Therefore, this failure is not applicable
Refueling Water Storage Tank, Tank Isolation Valve (2)	Inadvertently closed during use of tank	Loss of suction to affected pumps	Valve Position Indication, Flow Indication	CRI	Second full capacity suction line available. Potential damage to isolated pumps.
Sump Isolation Valve (2)	Inadvertently opened during use of refueling water tank	None. Check valve restricts flow	Valve Position Indication. Possible loss of flow	CRI	Operator Error. Operator should detect abnormal valve position by indicator lights.
RWS Tank Check Valve (2)	a. Fails Closed	None during long-term cooling	None	None	
	b. Fails Open	None-Isolation valve in series stops flow to RWS tank	None	None	

TABLE 6.3-6 SAFETY INJECTION SYSTEM FAILURE MODE ANALYSIS
HIGH PRESSURE SAFETY INJECTION (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	EFFECT ON SYSTEM	METHOD OF DETECTION	MONITOR	REMARKS
RWS Tank Isolation Valve (2)	Inadvertently Open during recirculation cooling	None - Check valve restricts flow to RWS tank	Valve Position Indication	CRI	Operator error. Operator should detect position indicator light and close valve.
Sump Isolation Valve (2)	Inadvertently closed during recirculation cooling	Loss of suction to one pump header	Valve Position Indication	CRI	Operator error. Full flow is available through parallel leg. Operator should detect position indicator light and open valve. Potential damage to isolated pumps.
Suction Line Between Containment Wall and Isolation Valve (2)	Leakage	None - Piping Encapsulated	None	None	
Sump Check Valve (2)	a. Fails Closed	Loss of suction to one pump header.	None	None	Full flow is available through parallel leg.
	b. Fails Open	None	None	None	Potential damage to isolated pumps.

* Not applicable where no failure mode is indicated.

TABLE 6.3-7 ADMINISTRATIVE ERROR ANALYSIS-SAFETY INJECTION SYSTEM VALVES

Valve Number	Normal Position	Administrative Error Position	Result
SI-306	Locked Open	Closed	Low pressure safety injection flow not available.
SI-402	Locked Open	Closed	Closure of one of these valves will result in no flow from its associated high-pressure safety injection (HPSI) pump. One pump remains.
SI-406	Locked Open	Closed	
SI-428	Locked Open	Closed	
SI-470	Locked Open	Closed	
SI-653	Closed	Open	Loss of isolation between two HPSI headers. No effect during injection.
SI-655	Closed	Open	
SI-411	Closed	Open	Separation of the two redundant HPSI paths is lost during recirculation.
SI-412	Closed	Open	
SI-432	Locked Open	Closed	No flow from associated low- pressure safety injection (LPSI) pump. One pump remains.
SI-435	Locked Open	Closed	
SI-444	Locked Open	Closed	
SI-447	Locked Open	Closed	
SI-421	Locked Open	Closed	Loss of safety injection pump minimum flow. Slight increase in injection flow.
SI-423	Locked Open	Closed	
SI-425	Locked Open	Closed	
SI-449	Locked Open	Closed	
SI-450	Locked Open	Closed	
SI-452	Locked Closed	Open	Some of flow from one containment spray (CS) pump will be injected into Reactor Coolant System through LPSI header.
SI-453	Locked Closed	Open	
SI-456	Closed	Open	No effect
SI-457	Closed	Open	
SI-662	Closed	Open	One HPSI pump will take suction from discharge of one CS pump. Partial loss of CS flow.
SI-663	Closed	Open	

TABLE 6.3-7 ADMINISTRATIVE ERROR ANALYSIS-SAFETY INJECTION SYSTEM VALVES (CONTINUED)

Valve Number	Normal Position	Administrative Error Position	Result
SI-654	Locked Open	Closed	HPSI flow not available through associated header.
SI-656	Locked Open	Closed	
SI-657	Closed	Open	No effect
SI-659	Open	Closed	Loss of minimum flow for all pumps. Slight increase in injection flows. Valves controlled by key lock switches to disable them open during normal operation.
SI-660	Open	Closed	
SI-460	Locked Closed	Open	No effect on safeguards system performance.
SI-461	Closed	Open	
SI-462	Closed	Open	
SI-463	Locked Closed	Open	
SI-661	Closed	Open	No effect
SI-611	Closed	Open	No effect on safeguards system performance.
SI-621	Closed	Open	
SI-631	Closed	Open	
SI-641	Closed	Open	
SI-612	Closed	Open	If any one of these valves is in the incorrect position, it will result in an inability to maintain its corresponding safety injection tank within the Technical Specifications requirements. Redundant alarms will alert the plant operator who will take corrective action.
SI-622	Closed	Open	
SI-632	Closed	Open	
SI-642	Closed	Open	
SI-613	Closed	Open	
SI-623	Closed	Open	
SI-633	Closed	Open	
SI-643	Closed	Open	

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
1. Loss of Facility Z1 (**B51**) coincident with SIAS	LPSI	LPSI	1. B HPSI train (using P41C pump)	P42B LPSI Pump via 2-SI-306 through 2-SI-651 and 2-SI-652
	1. 2-SI-651, 2-SI-615, 2-SI-625 remain closed	Open 2-SI-651 with Facility Z2 power.	2. P42B LPSI pump via 2-SI-645	
	2. P42A LPSI pump out of service	Close 2-SI-635		
	3. 125V DC (DV10) will not be available (battery has 8 hour coping factor)	Open 2-SI-400 and 2-SI-709		
	HPSI	HPSI		
	1. 2-CH-518 fails open due to loss of DC power (unused)	Verify B HPSI train in service to cold leg injection		
	2. A HPSI train out of service	Close 2-SI-636 and 2-SI-646		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
2. Loss of Facility Z2 (**B61**) coincident with SIAS	LPSI	LPSI	P42A LPSI pump via 2-SI-615 and 625	HPSI pump P41A via 2-CH-517 to pressurizer auxiliary spray line
	1. 2-SI-652, 2-SI-635, 2-SI-645 remain closed	Establish LPSI to cold leg injection 2-SI-615 & 625 are open		
	2. P42B LPSI pump train out of service			
	3. 125V DC (DV20) will not be available (battery has 8 hour coping factor)			
	HPSI	HPSI		
	1. 2-CH-517 fails closed due to Facility Z2 DC battery	Power 2-CH-517 and 2-CH-519 with 125V DC (DV10)		
	2. 2-CH-519 fails open	Align A HPSI pump to pressurizer auxiliary spray line via: Open 2-CH-340, 440 and 517		
	3. B HPSI train out of service	Close 2-CH-518 and 2-CH-519		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
3. Loss of Facility Z1 (**B51**)Post SIAS	LPSI	LPSI	1. B HPSI train (using P41C pump)	P42B LPSI Pump via 2-SI-306 through 2-SI-651 and 2-SI-652
	1. 2-SI-651 remains closed	Open 2-SI-651 with Facility Z2 power	2. P42B LPSI pump via 2-SI-615 and 625	
	2. 2-SI-615 and 2-SI-625 remain open	Close 2-SI-635 and 2-SI-645		
	3. LPSI P42A out of service	Open 2-SI-400 and 2-SI-709		
	4. 125V DC (DV10) will not be available (battery has 8 hour coping factor)			
	HPSI	HPSI		
	1. 2-CH-518 fails open due to loss of DC power (unused)	Verify B HPSI train in service to cold leg injection		
	2. HPSI P41A out of service	Close 2-SI-616 and 626		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
4. Loss of Facility Z2 (**B61**) Post SIAS	LPSI	LPSI	P42A LPSI pump via 2-SI-635 and 645	P41A HPSI pump via 2-CH-517 to pressurizer auxiliary spray line
	1. 2-SI-652 remains closed	Establish LPSI to Cold Leg Injection		
	2. 2-SI-635 and 2-SI-645 remain open	Close 2-SI-615 and 2-SI-625		
	3. 125V DC (DV20) will not be available (battery has 8 hour coping factor)			
	HPSI	HPSI		
	1. 2-CH-517 fails closed due to Facility Z2 DC battery	Power 2-CH-517 and 2-CH-519 with 125V DC (DV10)		
	2. 2-CH-519 fails open	Align P41A HPSI pump to pressurizer auxiliary spray line via: Open 2-CH-340, 440 and 517		
	3. HPSI P41C pump out of service	Close 2-CH-518 and 2-CH-519		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
5. Loss of 125V DC (**DV10**) coincident with SIAS	LPSI	LPSI	1. B HPSI train (using P41C pump)	P42B LPSI Pump via 2-SI-306 through 2-SI-651 and 2-SI-652
	1. Facility Z1 Diesel is shut down	Open 2-SI-651 with Facility Z2 power	2. P42B LPSI pump via 2-SI-645	
	2. P42A LPSI pump out of service	Close 2-SI-635		
	3. 2-SI-615, 2-SI-625 remain closed	Open 2-SI-400 and 2-SI-709		
	4. 2-SI-651 remains closed			
	HPSI	HPSI		
	1. 2-CH-518 fails open (unused)	Verify B HPSI train in service to cold leg injection		
	2. P41A HPSI train out of service	Close 2-SI-636 and 646		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
6. Loss of 125V DC (**DV20**) coincident with SIAS	LPSI	LPSI	P42A LPSI via 2-SI-615 and 625	P41A HPSI pump via 2-CH-517 to pressurizer auxiliary spray line
	1. Facility Z2 Diesel is shut down	Establish LPSI to Cold Leg Injection		
	2. P42B LPSI pump out of service	2-SI-615 & 625 are open		
	3. 2-SI-635 and 2-SI-645 remain closed			
	4. 2-SI-652 remains closed			
	HPSI	HPSI		
	1. 2-CH-517 fails close	Power 2-CH-517 and 2-CH-519 with 125V DC (DV10)		
	2. 2-CH-519 fails open	Align P41A HPSI pump to pressurizer auxiliary spray line via: Open 2-CH-340, 440 and 517		
	3. B HPSI train out of service	Close 2-CH-518 and 2-CH-519		

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
7. Mechanical Failure of a LPSI pump	P42A or P42B LPSI pump fails to operate	LPSI	P42A or P42B LPSI via 2-SI-645	P42A or P42B LPSI pump via 2-SI-306 through 2-SI-651 and 2-SI-652
		Align the operating LPSI pump hot leg injection	HPSI B train	
		2-SI-645 is open	Close 2-SI-636 & 646	
		Close 2-SI-615, 625 and 635		
8. Mechanical or Loss of Air Failure of 2-SI-306	2-SI-306 is pinned and locked at preset throttled open position and remains open. Therefore, this case is not considered a credible failure case.			
9. Mechanical Failure of 2-SI-651, 2-SI-652, 400 or 709	Failure of any one of these valves will disable LPSI hot leg injection.	Establish HPSI to Hot Leg Injection	One LPSI pump through any two of the four LPSI injection valves.	P41A HPSI pump via 2-CH-517 to pressurizer aux spray line

TABLE 6.3-8 FAILURE MODES AND EFFECTS ANALYSIS FOR BORON PRECIPITATION CONTROL (CONTINUED)

Single Failure	Effects of Failure	Actions Required	Cold Leg Injection Path	Boron Precip Control Path
10. Mechanical Failure of P41A HPSI pump or 2-CH-340, 440, 429, 517, 518 or 519	A HPSI train out of service	None	1. B HPSI train (using P41C pump) 2. P42A/B LPSI pump via2-SI-645	P42A/B LPSI pump via 2-SI-306 through 2-SI-651 and 2-SI-652
11. Mechanical Failure of 2-SI-635	2-SI-635 remains open	Establish HPSI to Hot Leg Injection	One LPSI pump through any two of the four LPSI injection valves	P41A HPSI pump via 2-CH-517 to pressurizer auxiliary spray line

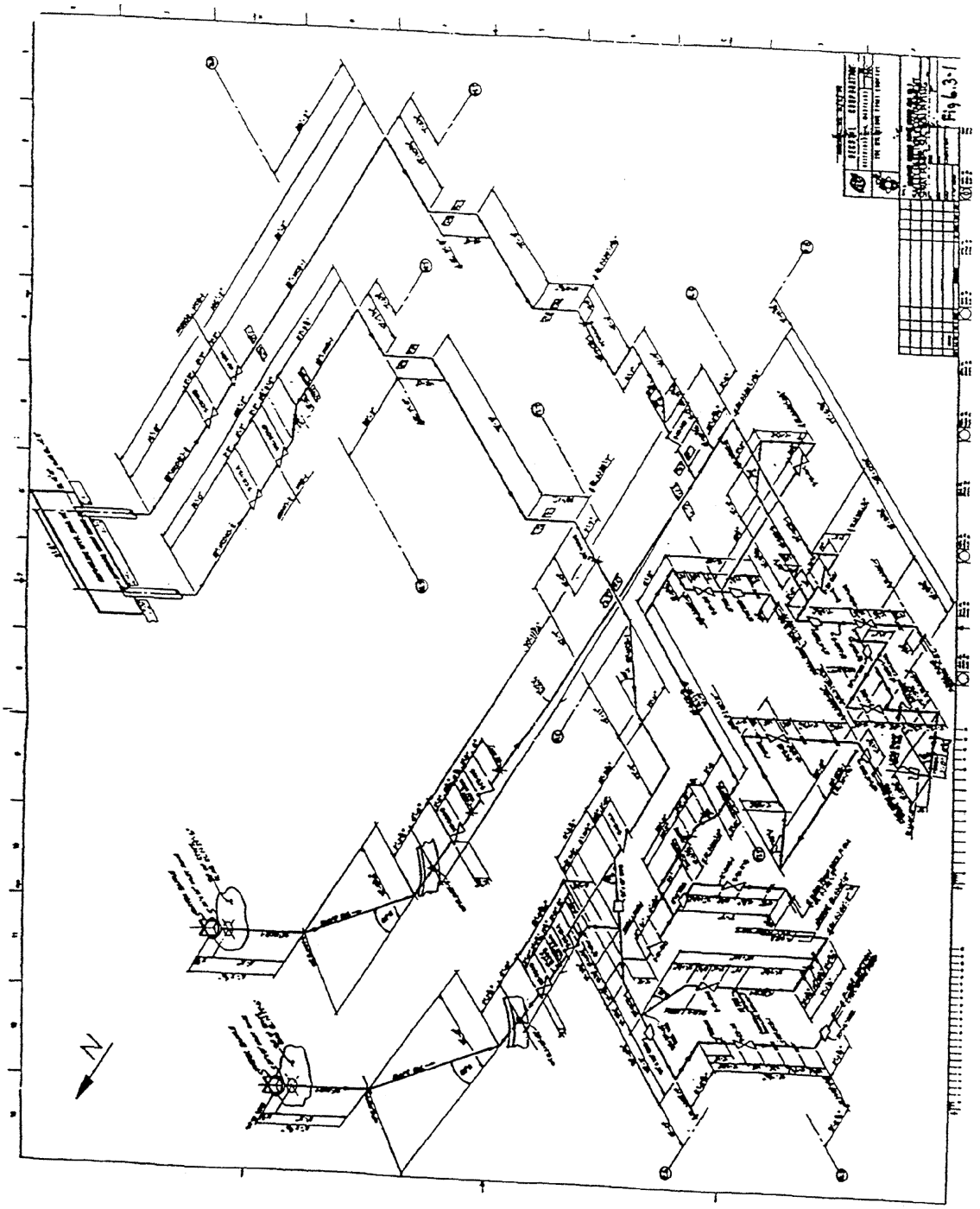


FIGURE 6.3–1 SAFETY INJECTION AND CONTAINMENT SPRAY PUMPS SUCTION PIPING

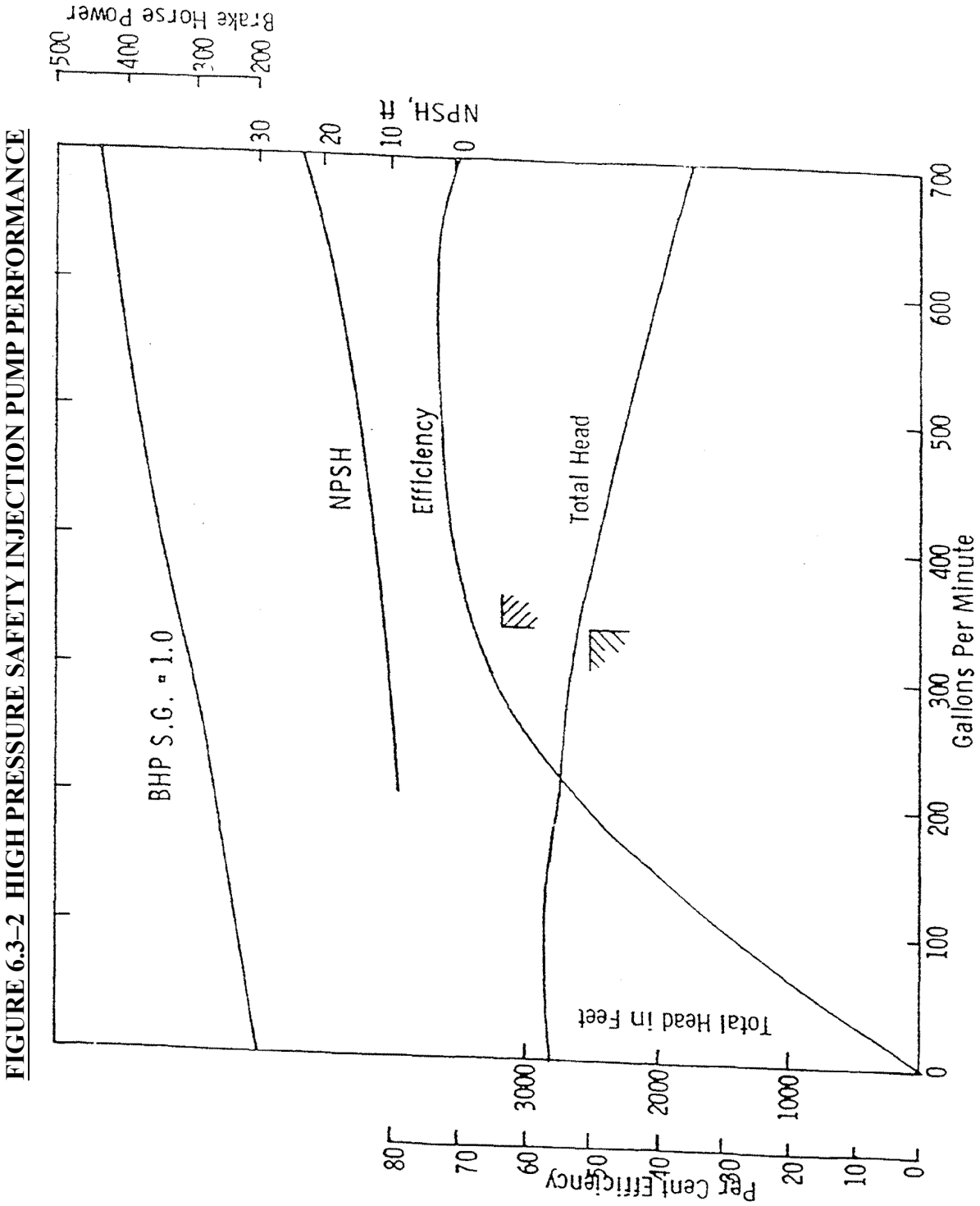
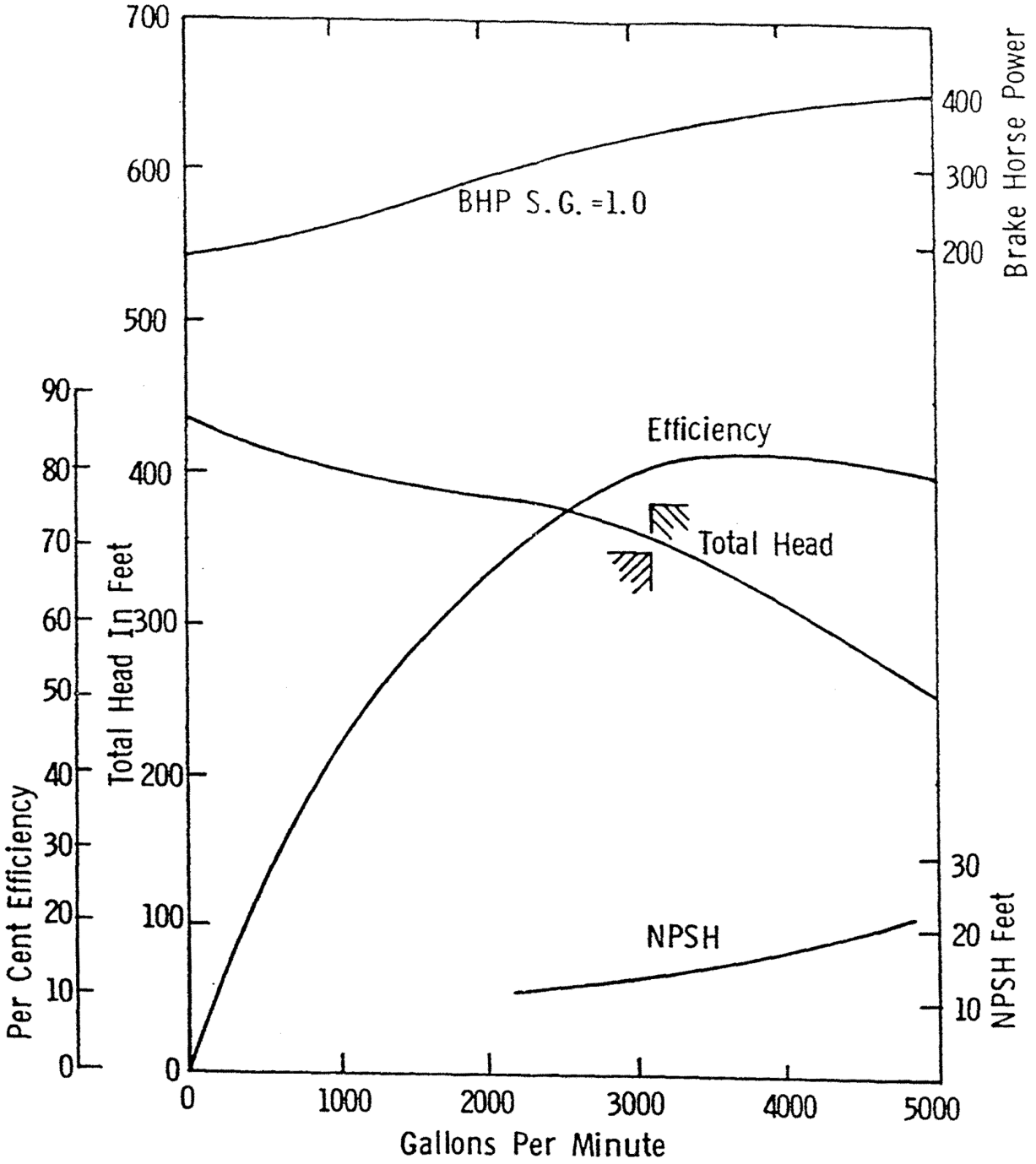


FIGURE 6.3–2 HIGH PRESSURE SAFETY INJECTION PUMP PERFORMANCE

FIGURE 6.3–3 LOW PRESSURE SAFETY INJECTION PUMP PERFORMANCE



6.4 CONTAINMENT SPRAY SYSTEM

6.4.1 DESIGN BASES

6.4.1.1 Functional Requirements

The containment spray system functions as an engineered safety feature to limit the containment pressure and temperature after a loss-of-coolant accident (LOCA) and Main Steam Line Break (MSLB) accident and thus reduces the possibility of leakage of airborne radioactivity to the outside environment. The containment spray system in conjunction with the containment air recirculation and cooling system (described in Section 6.5) provides sufficient heat removal capability to limit the post-accident containment pressure and structural temperature below the design values of 54 psig and 289°F, respectively (Section 14.8.2) by directing sprays of cooled borated water downward from the upper regions of the containment.

During shutdown or refueling, as a back-up to support maintenance activities of the shutdown cooling system, a containment spray system pump can be aligned to cool the spent fuel pool (see Section 9.5).

6.4.1.2 Design Criteria

The following criteria have been used in the design of the containment spray system:

a. The system has two redundant, independent subsystems, each having 50 percent of the required heat removal capability.

b. The system has suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities are provided to assure system operation using either on site power (assuming off site power is not available) or off site electrical power.

d. A single failure in either subsystem does not affect the functional capability of the other subsystem.

e. The system is designed to permit inspection of important components, such as containment sump, containment spray pumps, heat exchangers, spray nozzles, valves, and piping to assure the integrity and capability of the subsystem.

f. The containment spray system is designed to permit appropriate periodic pressure and functional testing to assure: (1) structural and leak-tight integrity of its components; (2) operability and performance of the active components of the system; (3) operability of the active components of the system as a whole. Under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation shall be demonstrated. This includes operation of applicable portions of the protection system, the transfer

between normal and emergency power sources, and the operation of the associated cooling water system.

g. The system is consistent with the design criteria as described in Section 6.1.

h. The components of the containment spray system are designed to operate in the most severe post-accident environment described in Section 6.1.

i. The containment spray pumps are designed in accordance and with the conditions given in Safety Guide 1.

6.4.2 SYSTEM DESCRIPTION

6.4.2.1 System

The containment spray system is shown schematically in Figure 6.1–1. The containment spray system consists of two identical, redundant, independent subsystems, each with a heat removal capability of 120 x 10^6 Btu/hr under design post-accident conditions. Each containment spray system consists of a containment spray pump, shutdown cooling heat exchanger, spray nozzles, piping, valves and instrumentation. The refueling water storage tank serves as the source of water and is described in Section 6.2. The shutdown cooling heat exchangers are described in Section 6.3.

Containment integrity is maintained following the limiting containment transient, the design basis MSLB, by utilizing one containment spray subsystem in combination with two containment air recirculation and cooling units (Section 6.5). This combination has a heat removal capability of 280 million Btu/hr under design post-accident conditions (Section 14.8.2). The heat sink for the containment spray system is shown in Figure 6.4–9.

The system is designed assuming the containment spray water is heated to the average temperature of the containment atmosphere while falling through the steam-air mixture within the containment. In order that the spray droplets approach thermal equilibrium during the fall, a minimum distance of 65 feet is provided between the spray nozzles and the highest obstruction in the containment.

Each spray subsystem is provided with spray headers oriented to provide near equal distribution within the containment cross section.

The containment spray header locations and spray nozzle orientations are shown in Figure 6.4–1. There are two different types of nozzles employed in the system as listed in Table 6.4-1. Of the 193 total nozzle locations, 64 are Lechler model 372.975.17.BL which deliver 8 gpm of flow at a 40 psid pressure differential. The remaining nozzles are Spraco model 1713A with a flow capacity of 15 gpm at the same 40 psid differential pressure. The mass-mean droplet diameter for the Lechler nozzles is less than that for the Spraco nozzles. Typical spray patterns for the Spraco nozzles are shown in Figures 6.4–2, 6.4–3 and 6.4–4 while spray patterns for the Lechler nozzles are given in Figure 6.4–4A.

The containment volume which is effectively sprayed, given a single failure of one containment spray train, is approximately 75% of the total containment net free volume.

6.4.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 6.4-1.

6.4.3 SYSTEM OPERATION

The containment spray system operates only under certain emergency conditions. With the exception of the cold shutdown and refueling conditions, valve alignment for all other operating modes is as described under emergency conditions.

During shutdown or refueling, as a back-up to support maintenance activities of the shutdown cooling system, a containment spray system pump can be aligned to cool the spent fuel pool (see Section 9.5).

6.4.3.1 Emergency Conditions

In the event of a LOCA or MSLB accident, the containment spray system is automatically initiated by the containment spray actuation signal (CSAS) as described in Section 7.3. The containment spray pumps initially take suction from the refueling water storage tank. When low level is reached in this tank, the pump suction, in the absence of operation action, is automatically transferred to the containment sump by a sump recirculation actuation signal (SRAS) as described in Section 7.3. The recirculated spray water is cooled by the shutdown heat exchangers prior to discharge into the containment through the spray headers. The RBCCW serves as the cooling medium and is discussed in Section 9.4. Emergency power is provided by the emergency buses as described in Section 8.3.

To prevent the refuel pool from capturing water, the pool drain line isolation valves are locked open during the operating cycle and a screened enclosure is installed over the two drain openings in the floor of the refuel pool.

System operation and performance is monitored in the control room. The containment spray pumps are monitored by the pump pressure gauges, motor trip alarms, and flow metering to indicate abnormal operation. The tube side of the shutdown heat exchangers is monitored by system water temperature to indicate abnormal heat exchanger operation. The shell side of the shutdown heat exchanger is described in Section 9.4.

The motor-operated valve in each containment spray header opens automatically upon the containment spray actuations signal (CSAS). The position of these valves is monitored in the control room.

The manual valves in the system are aligned for containment spray operation and are administratively locked in their respective operating positions.

The operation of the refueling water storage tank (RWST) and containment sump are described in Section 6.2.

6.4.3.2 Cold Shutdown and Refueling

During refueling operations the shutdown heat exchanger is isolated from the containment spray system and aligned with the low pressure safety injection system for shutdown cooling operations (see Section 6.3). This alignment is made by closing manual valves 2-CS-3A and 2-CS-3B on the containment spray pump discharge and 2-CS-4A and 2-CS-4B downstream of the shutdown heat exchanger, and opening the manual isolation valves upstream and downstream of the shutdown heat exchanger to the low pressure safety injection header 2-SI-452, 453, 456 and 457. This alignment is made only after the safety injection actuation signal is blocked (see Section 7.3).

During shutdown or refueling, as a backup to support maintenance activities of the shutdown cooling system, a containment spray system pump can be aligned to cool the spent fuel pool (see Section 9.5.

6.4.4 AVAILABILITY AND RELIABILITY

6.4.4.1 Special Features

The components of the containment spray system are designed to general requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physically separating duplicate equipment, as described in Section 6.1.

To assure the availability of water to the pumps, separate suction headers from the refueling water storage tank are provided for the spray pump located in the two separate and shielded pump rooms, which house the pumps of the engineered safety features systems. Each of the two pump rooms contains one spray pump, one low pressure safety injection pump and one high pressure safety injection pump. Two separate headers, one to each of these pump rooms, are also provided from the containment sump.

The containment spray pumps are located in the lowest elevation of the auxiliary building at Elevation (-) 45-6 to assure a flooded suction. This assures pump priming and protects the mechanical seals in the spray pumps. In this location, the available NPSH is always greater than the required NPSH (see Table 6.4-1).

To assure adequate design margins, the minimum available NPSH for the containment spray pumps is conservatively calculated during the recirculation mode in accordance with Safety Guide 1. Refer to Section 6.3.2.1 for assumptions used in calculating.

To increase system reliability, the containment spray pump motors have the capacity to start with the motor-operated valves on the discharge header fully opened.

The refueling water storage tank (RWST) and containment sump assure sources of water for the containment spray system. These components are described in Section 6.2.

A failure mode analysis is given in Table 6.4-2.

Inadvertent initiation of the spray system does not affect the safety of the unit, since within the containment all the instruments are drip-proof or weatherproof, all the motors are drip-proof or totally enclosed and signal cable runs are enclosed in waterproof jackets. All piping or equipment insulation which may come in contact with sprays are of the metal reflective type or jacketed to prevent large quantities of water from penetrating the insulation. Small amounts of seepage do not present any significant thermal shock to hot equipment.

Inadvertent operation of the system is monitored by pressure, flow and valve position indicators for the operator, so the situation would be quickly observed and remedial action taken.

The design basis case for containment pressure analysis is a Main Steam Line Break accident at zero power with the single failure of vital bus cabinet VA-10 and VA-20. The analysis assumes that one train of containment spray and one train (two units) of containment air recirculation (CAR) are also operable. The containment pressure response based on all other combinations of equipment which would be available assuming any other single failure is bounded by this assumed single failure.

Following a LOCA, the containment spray efficiency decreases with the containment post accident steam/air mass ratio (hence with containment temperature and pressure). The efficiency of the containment air recirculation and cooling units remains relatively constant since these units are designed to remove sensible heat (FSAR Section 6.5). The spray system, even operating at a lower efficiency over the long term, in combination with the containment air recirculation and cooling units, provides ample heat removal capabilities.

6.4.4.2 Test and Inspections

Each containment spray pump was shop tested for hydraulic performance at sufficient head-capacity points to generate complete performance curves. These performance tests were run for design NPSH at runout conditions and calculated back to rated conditions. This containment spray pump performance test curve is shown in Figure 6.4–8.

Nondestructive examinations were performed on all pressure-retaining components of each containment spray pump in accordance with the Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968.

Both types of spray nozzles listed in Table 6.4-1 were performance tested to assure the desired flow at a specified pressure drop across the nozzle is achieved. The 129 Spraco nozzles are capable of delivering 15 gpm of flow at a 40 psid pressure difference while the remaining 64 Lechler nozzles deliver 8 gpm at the same pressure drop. Type, location and orientation of the spray nozzles is shown in Figure 6.4–1. The containment spray nozzle performance characteristics for both types of nozzles are provided in Figure 6.4–7.

The containment spray system underwent a preoperation test prior to startup. The test is described in Section 13.

Provisions are incorporated for on-line testing of the active components in the containment spray system during normal unit operation. The automatically actuated containment spray pumps and motor-operated valves are tested for operability by manually initiating (Section 7.3.4.2.1) the CSAS for each component. Valve operation can be verified by position indication in the control room and locally by visual inspection. Pump operation is indicated by local pressure indication and by status lights in the control room.

Operability testing, response time testing and inservice testing of the containment spray system and its components are performed as required by plant Technical Specifications in accordance with plant procedures.

The design and location of the containment spray pumps and shutdown cooling heat exchangers permit access for periodic testing and maintenance during normal operation.

TABLE 6.4-1 CONTAINMENT SPRAY SYSTEM COMPONENT DESIGN PARAMETERS

Containment spray pumps	
Manufacturer	Goulds Pump
Model Number	3736-4x6-13DV
Quantity	2
Type	Vertically split, horizontally centrifugal with mechanical seals backed up with an auxiliary gland
Material	Casing ASTM A351 Gr CF8M; Bolting A-193 B7; Impeller A-296 Gr CG8M
Design temperature (°F)	300
Fluid pumped	Borated water (≥1720 ppm)
Codes	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968; Standards of Hydraulic Institute
Seismic	Class 1
Containment spray pump motor	
Manufacturer	General Electric
Horsepower rating (hp)	250
Horsepower rating (hp)	B
Type	Induction
Frame designation	8188S
Codes	NEMA, MG-1
Seismic	Class 1

TABLE 6.4-1 CONTAINMENT SPRAY SYSTEM COMPONENT DESIGN PARAMETERS (CONTINUED)

	Mode of Operation	
	Injection	**Recirculation**
* Capacity (each) (gpm)	1350 **	1650 **
* Head (ft)	450	360
* NPSH available (ft)	64.0	27.0
* NPSH required (ft)	15.5	21.0
Brake horsepower (bhp)	202	215

Temperature transient (°F)	50 to 300 inch, 10 seconds
Pump speed (rpm)	3560
Fluid pH	5.5 to 10.5

Piping, Fittings and Valves

Suction

Pipe Sizes	**Wall Thickness**
2 inch and smaller	Schedule 40S
2.5 inch and larger	Schedule 10S

Material	ASTM A-312, Type 304
Design pressure (psig)	60
Design temperature (°F)	300
Standard	ANSI B-31.7 Class II
Seismic	Class 1

TABLE 6.4-1 CONTAINMENT SPRAY SYSTEM COMPONENT DESIGN PARAMETERS (CONTINUED)

Construction

Piping	Valves
2 inch and smaller; Socket welded	2 inch and smaller: Socket-welded 600 lb ANSI rating, stainless steel ***
2.5 inch and larger: Butt-welded except at flanged equipment	2.5 inch and larger: Butt-welded, 150 lb ANSI rating, stainless steel

Discharge

Pipe Sizes	Wall Thickness
3 inch and smaller	Schedule 40S
4 inch through 6 inch	Schedule 10S
8 inch through 14 inch	Schedule 20

Material	ASTM A-312, Type 304
Design pressure (psig)	500
Design temperature (°F)	300
Standard	ANSI B-31.7, Class II; ANSI B-31.1.0 Modified (inside containment)
Code (valves)	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968
Seismic	Class 1

Construction

Piping	Valves
2 inch and smaller: Socket welded	2 inch and smaller: Socket welded, 600 lb ANSI rating, stainless steel ***
2.5 inch and larger: Butt welded except flanged equipment	2.5 inch and larger: Butt welded, 300 lb ANSI rating, stainless steel

TABLE 6.4-1 CONTAINMENT SPRAY SYSTEM COMPONENT DESIGN PARAMETERS (CONTINUED)

	Spray nozzles	
	Manufacturer	
	Spraco	Lechler
Model number	1713A	372.975.17.BL
Quantity	129	64
Type	Ramp Bottom	Ramp Bottom
Pattern	Hollow Cone	Hollow Cone
Material	Type 304 or 316 stainless	Type 316 Stainless
Nozzle size (inches)	1, 3/8 inch orifice	3/4, 0.28 inch orifice
Rated flow, each (gpm)	15	8
Rated pressure drop (psid)	40	40

* The pump flows, heads, and NPSH values presented in this table are based on the original design specifications. The pumps will actually operate over a range of conditions during containment spray from the RWST following a SIAS and from the containment sump following SRAS. System hydraulic and NPSH analyses have been performed for these operational modes based on the pump design and degraded capacities.

** Includes 50 gpm for minimum flow recirculation.

*** 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 600 lb, based on the design conditions of the specific application.

TABLE 6.4-2 CONTAINMENT SPRAY SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Motor-operated Valves (2) HV3010, HV3011	Fail as is	Position indication. CRI	None	Repair operator	Normal operation, valve normally open	Valve can be manually operated.
Motor-operated Valves (2) HV3008, HV3009	Fail as is	Position indication CRI	Loss of one (1) containment sump recirculation line	Repair operator	One header out of service. Alternate header is operable	One header sufficient for containment cooling with combination of two containment air recirculation & cooling units.
Containment Spray Pump (2)	Pump stops	Pressure indication & flow indication	Loss of water flow in CTMT spray header	Isolate pump and repair	One containment spray header out of service, alternate containment spray header in normal operation	One containment spray header sufficient for emergency cooling.
Motor-operated Valves (2) VHV3021, HV3022	Fail to open on CSAS	Pressure indication & flow indication. All CRI	Loss of flow from one containment spray header	Repair operator	One containment Spray header out of service, alternate containment Spray header in operation	One containment Spray header sufficient for emergency cooling

FIGURE 6.4–1 AREA 5 PIPING CONTAINMENT SPRAY AND HYDROGEN PURGE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 6.4–2 CONTAINMENT SPRAY NOZZLES SPRAY PATTERNS (1713A NOZZLE SPRAYING HORIZONTALLY)

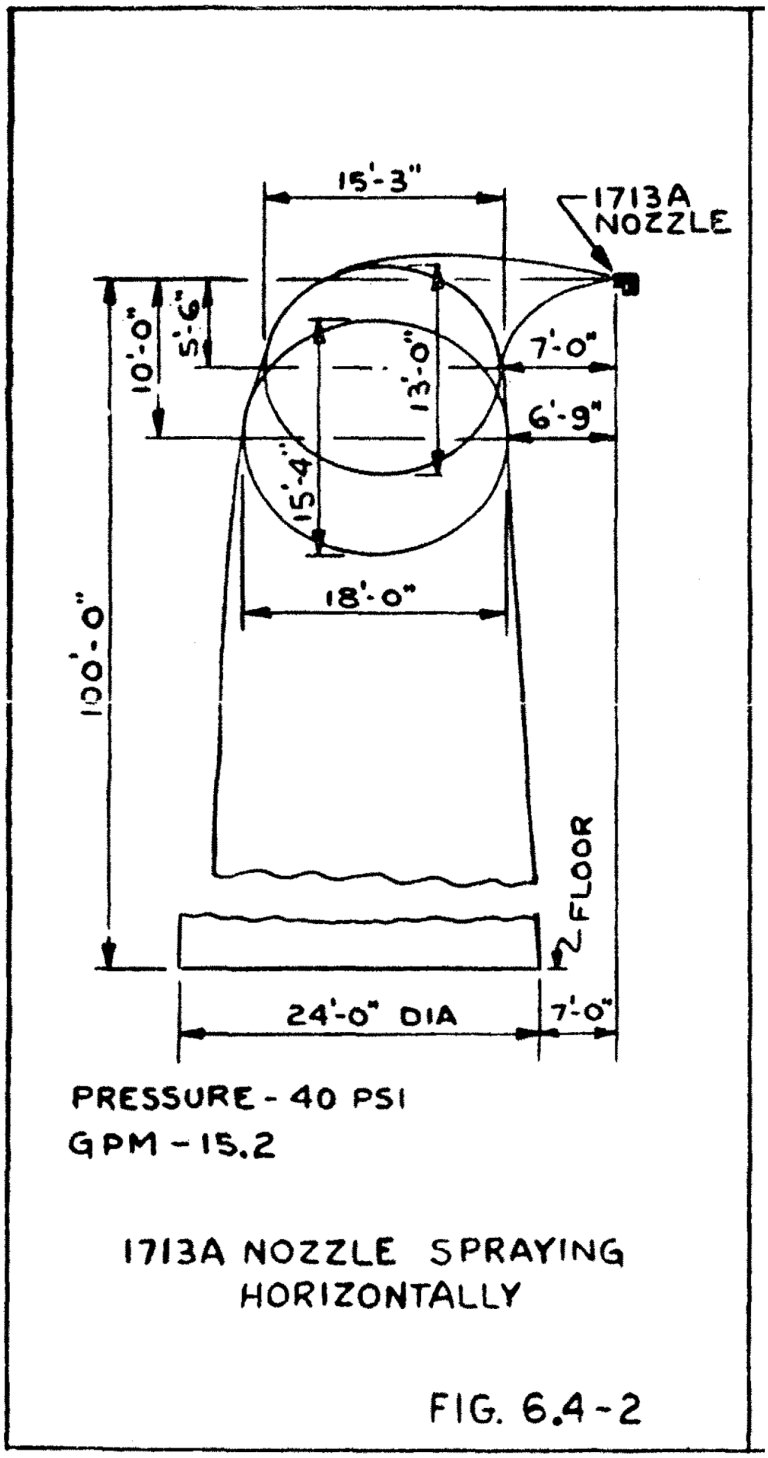


FIGURE 6.4–3 CONTAINMENT SPRAY NOZZLES SPRAY PATTERNS (1713A NOZZLE SPRAYING DOWNWARD ON 45°)

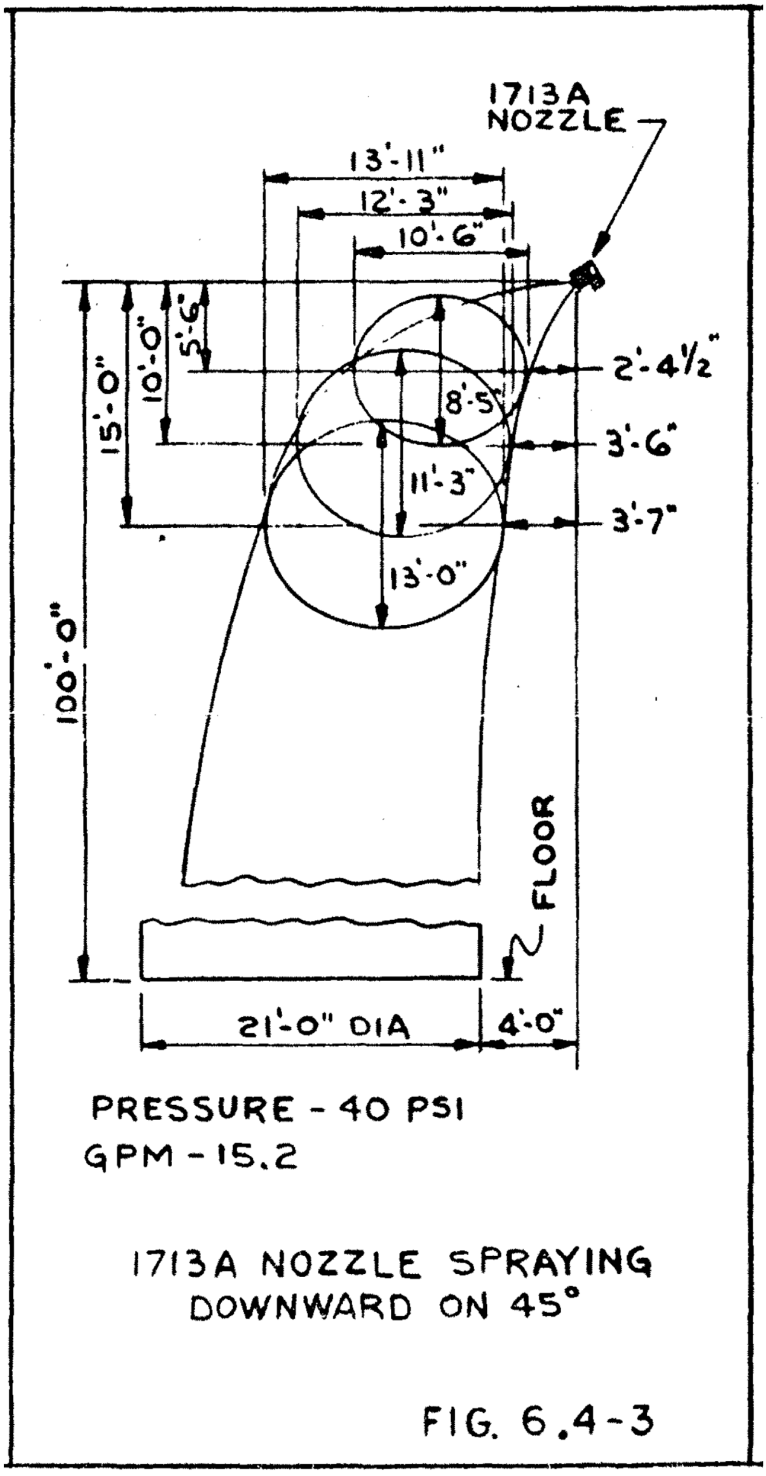


FIGURE 6.4–4 CONTAINMENT SPRAY NOZZLES SPRAY PATTERNS (1713A NOZZLE SPRAYING VERTICALLY DOWNWARD)

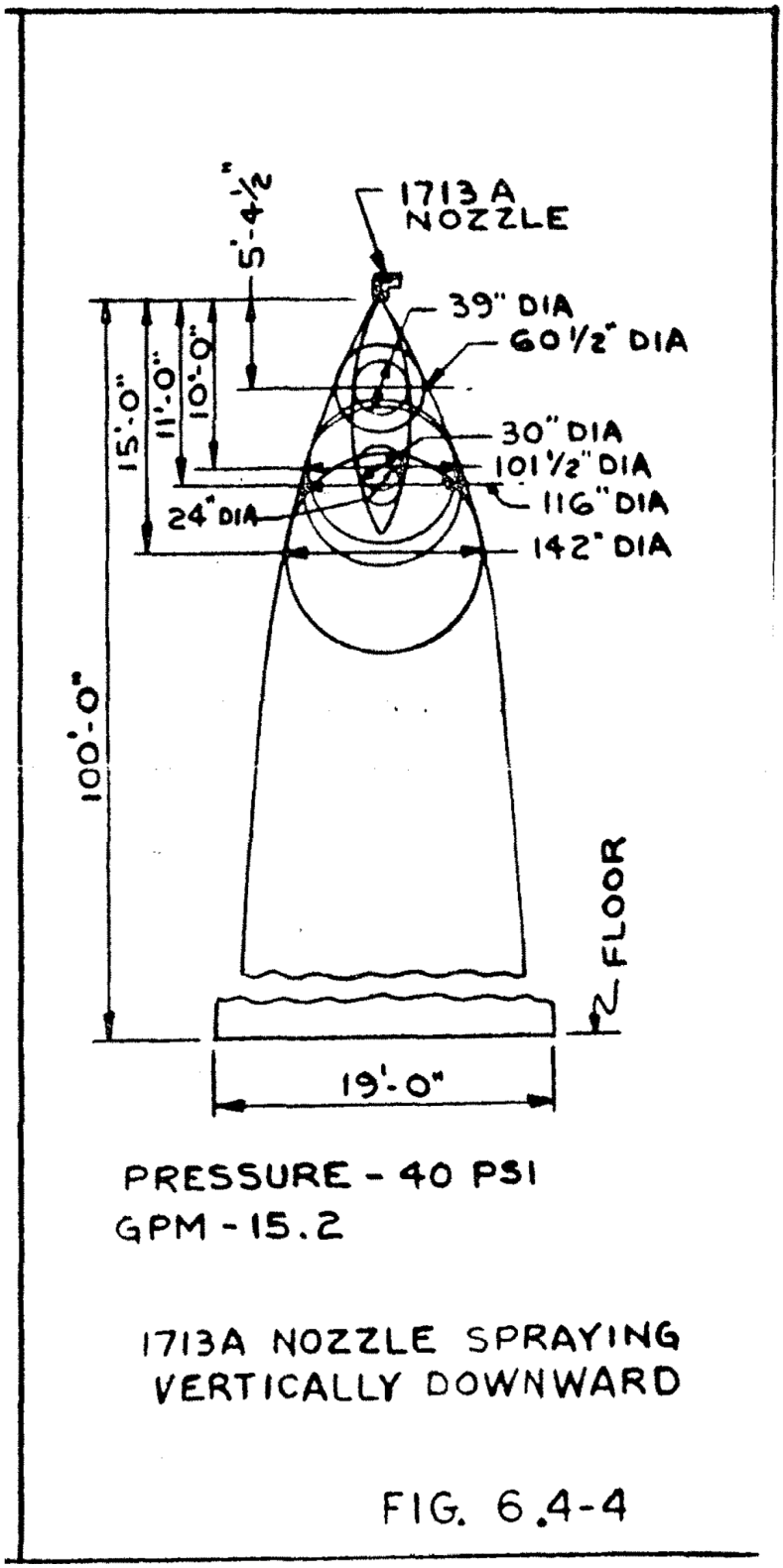


FIGURE 6.4-4A TYPICAL SPRAY COVERAGE PATTERNS

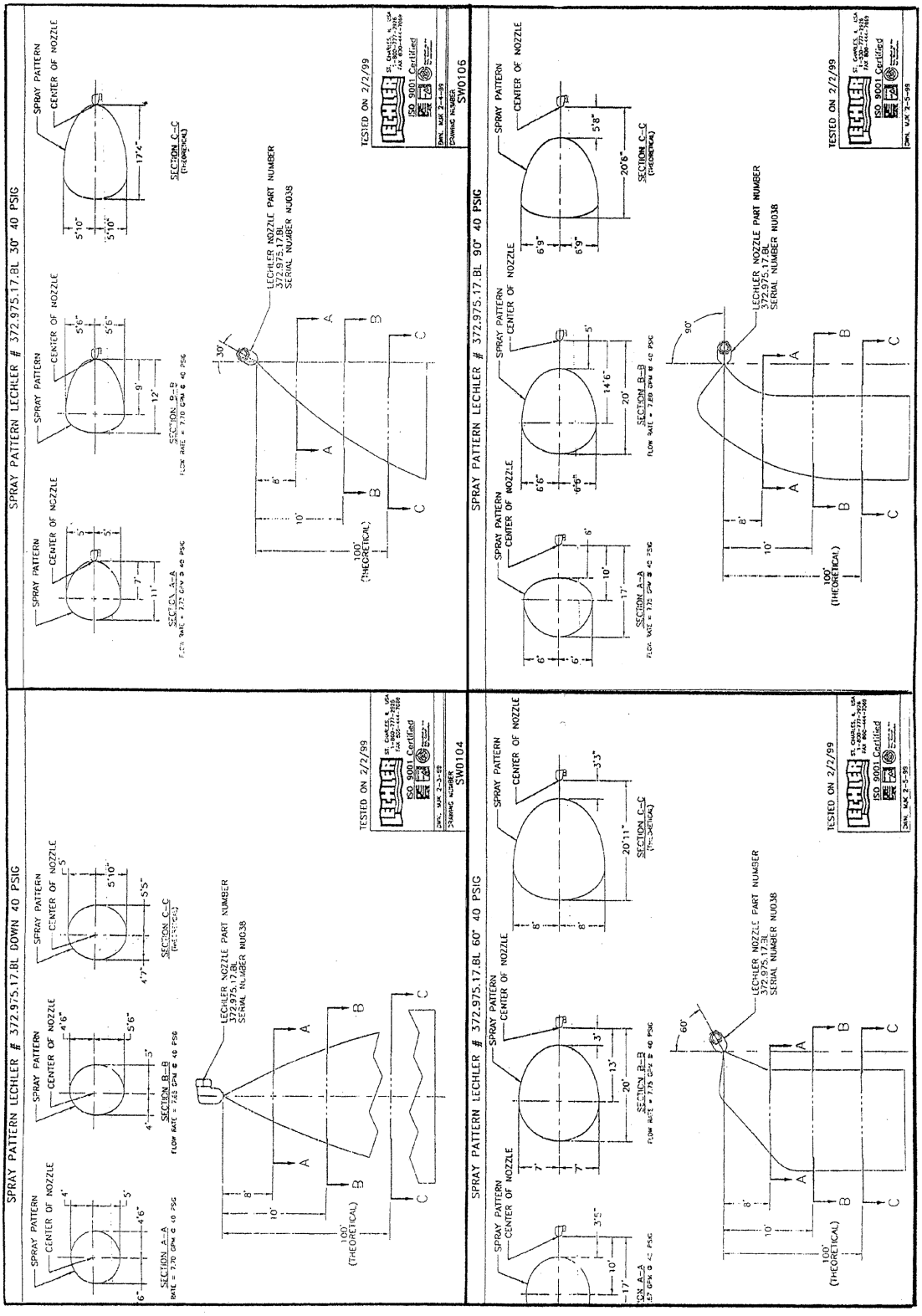


FIGURE 6.4–5 NOT USED

FIGURE 6.4–6 NOT USED

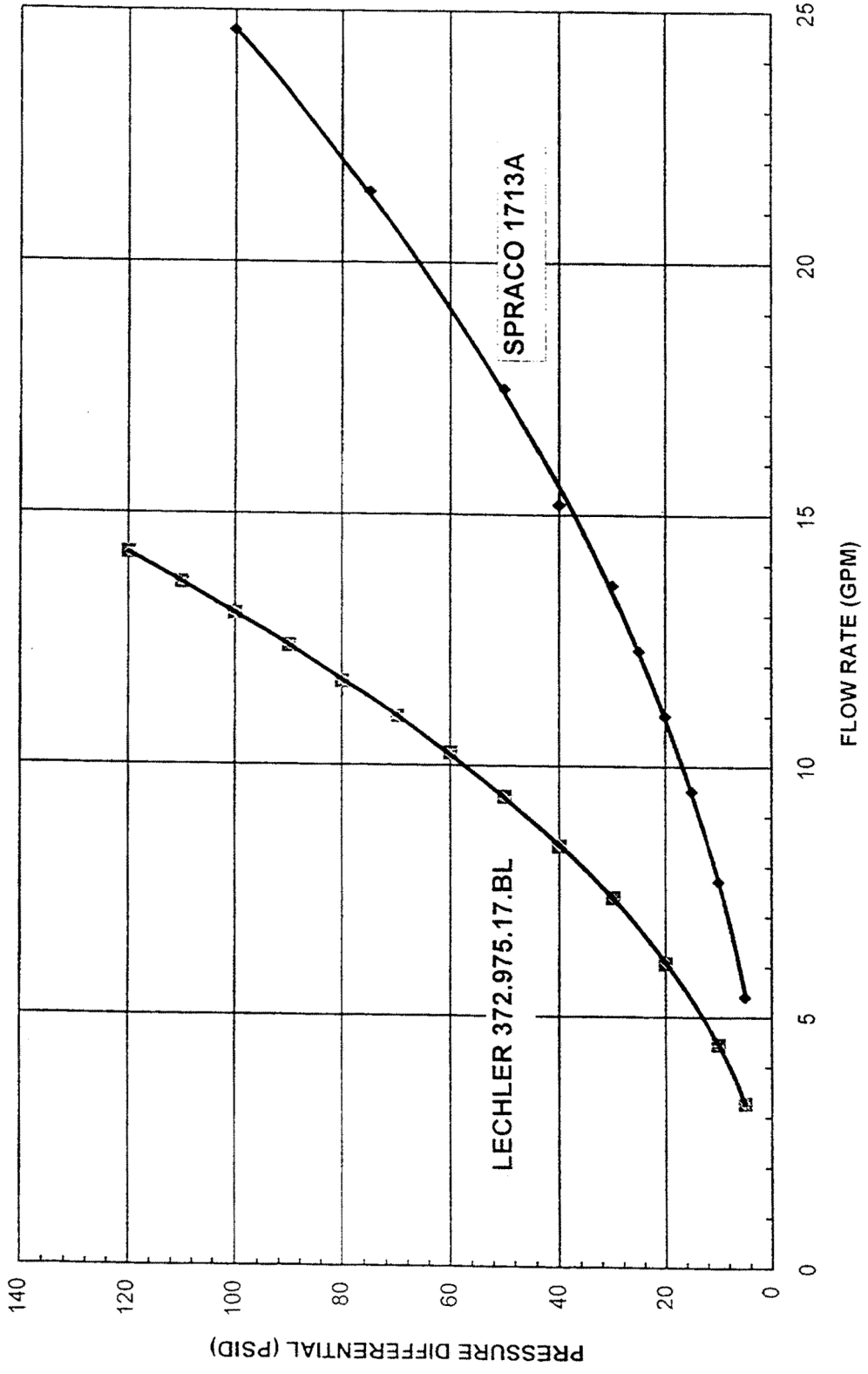
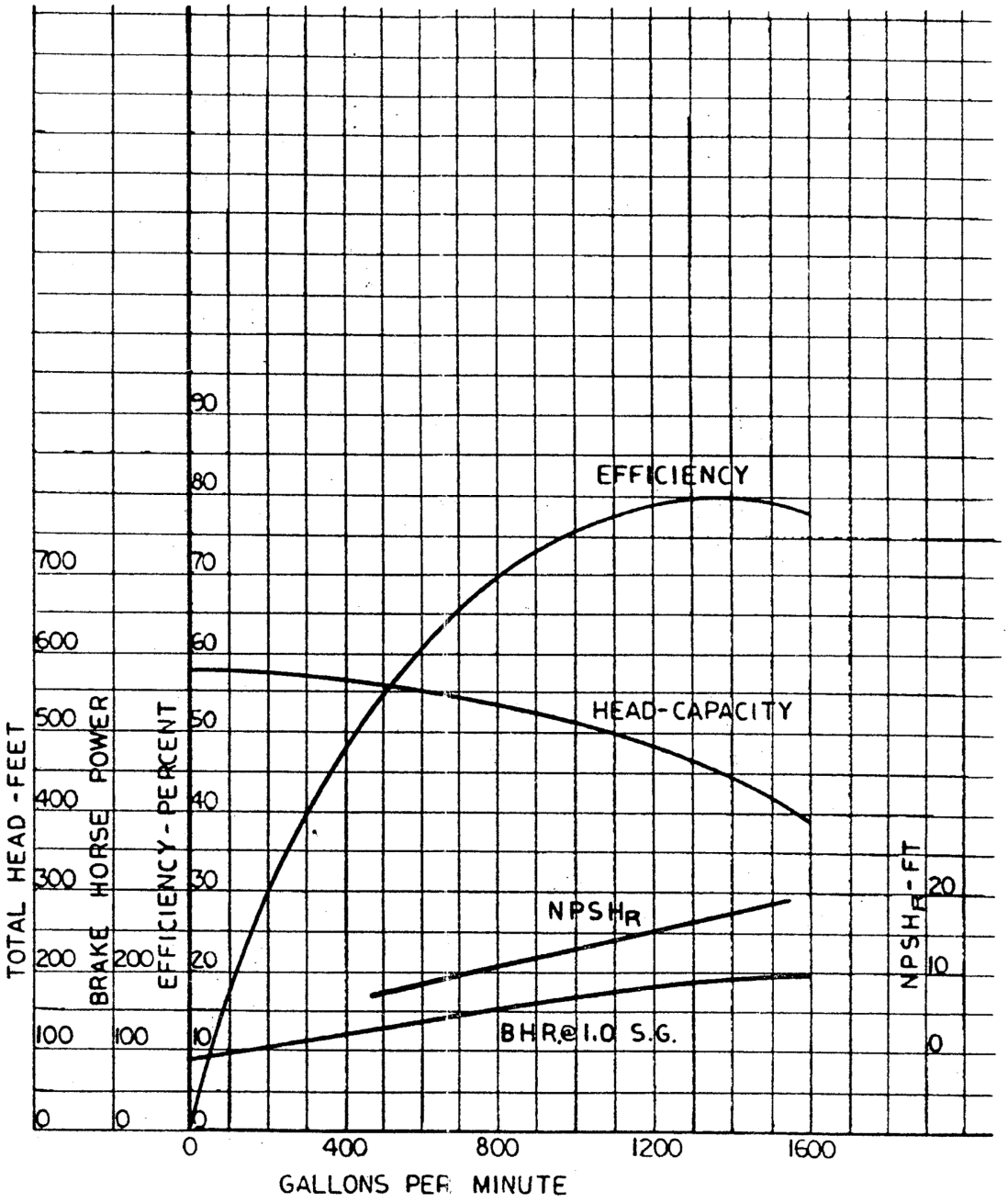


FIGURE 6.4–7 CONTAINMENT SPRAY NOZZLE PERFORMANCE CHARACTERISTICS

FIGURE 6.4–8 CONTAINMENT SPRAY PUMP PERFORMANCE CHARACTERISTICS.



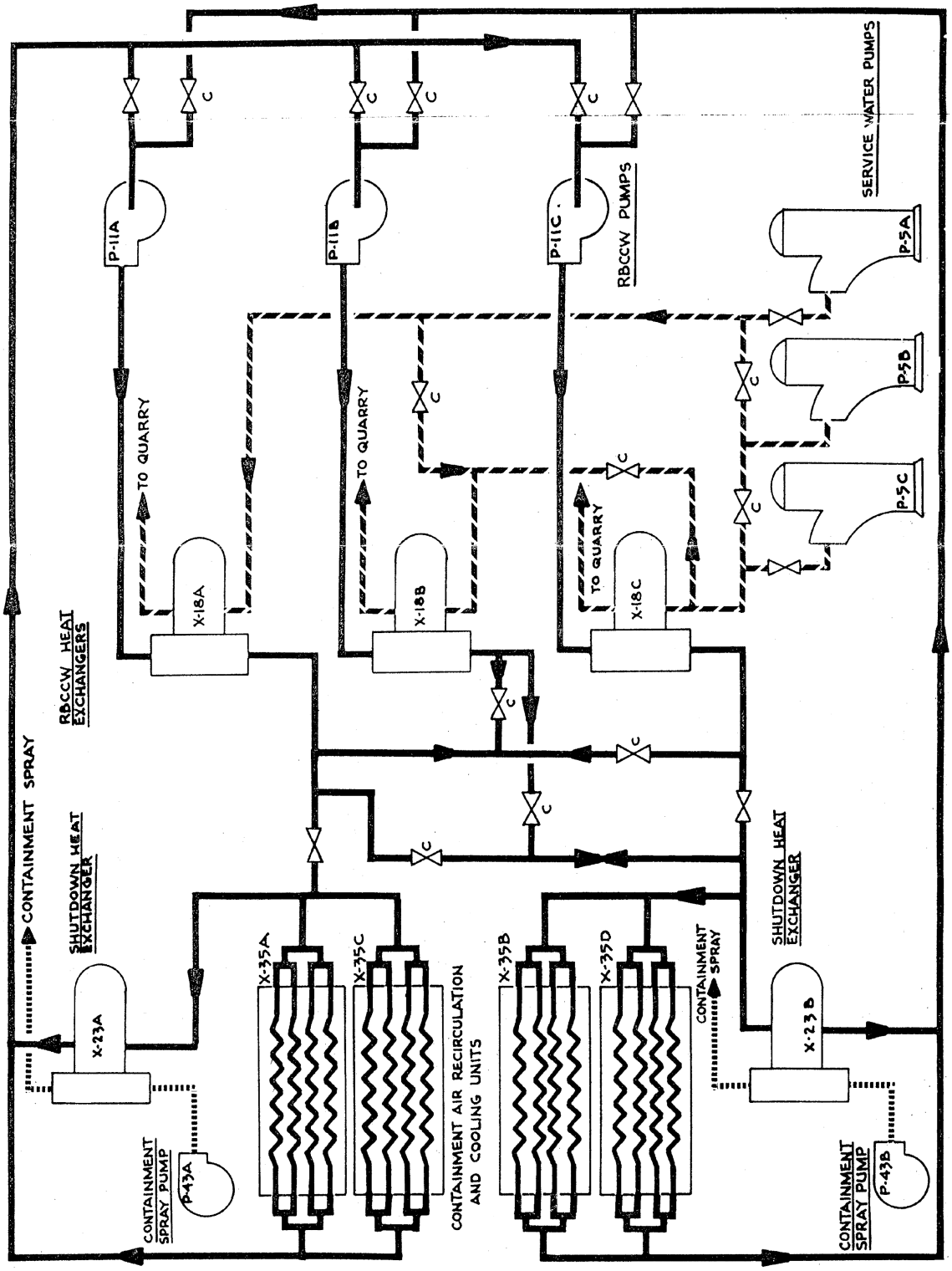


FIGURE 6.4–9 CONTAINMENT HEAT REMOVAL SYSTEMS SINK SCHEMATIC

6.5 CONTAINMENT AIR RECIRCULATION AND COOLING SYSTEM

6.5.1 DESIGN BASES

6.5.1.1 Functional Requirements

The function of the containment air recirculation and cooling system is to remove heat from the containment atmosphere during normal operation. In the event of a Loss-of-Coolant-Accident (LOCA) or Main Steam Line Break (MSLB) accident, the system, in conjunction with the containment spray system described in Section 6.4, provides a means of cooling the containment atmosphere to reduce the containment building pressure and thus reduce the leakage of airborne and gaseous radioactivity.

The containment air recirculation and cooling system is independent of the safety injection and containment spray systems. It is sized such that, assuming the most adverse containment heat-removal single failure, following a LOCA or MSLB accident, two of the four containment air recirculation units in conjunction with one train of the containment spray system limits the containment pressure and structural temperature to less than the containment design values (54 psig, 289°F).

6.5.1.2 Design Criteria

The following criteria have been used in the design of the containment air recirculation and cooling system:

a. The system has two redundant, independent and separate subsystems, each consisting of two containment air recirculation and cooling units.

b. The system has suitable subsystem and component alignment to assure operation of the complete subsystem with its associated components.

c. Capabilities are provided to assure that the system can operate using either on site power (assuming off site power is not available) or with off site electrical power.

d. The containment air recirculation and cooling system is designed to permit inspection of important components, such as cooling coils, fans, motors and reactor building closed cooling water (RBCCW) piping to assure the integrity and capability of the system.

e. The containment air recirculation and cooling system is designed to permit appropriate periodic pressure and functional testing to assure: (1) structural and leak-tight integrity of its components; (2) operability and performance of the active components of the system; and, (3) operability of the active components of the system as a whole. Under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer

between normal and emergency power sources, and the operation of the associated cooling water system shall be demonstrated.

The system is designed to the criteria described in Section 6.1.

The components of the containment air recirculation and cooling system are designed to operate in the most severe postaccident environment as described in Section 6.1.

6.5.2 SYSTEM DESCRIPTION

6.5.2.1 System

The containment air recirculation and cooling system is shown schematically in Figure 9.9–1.

The containment air recirculation and cooling system is designed to maintain the containment normal ambient temperature at 120°F with three of the four units operating. Each unit is designed for 2.2 x 10^6 Btu/hr, one-third of the normal containment heat load.

Each containment air recirculation and cooling unit is designed for removing 80 x 10^6 Btu/hr under Main Steam Line Break accident or LOCA conditions prior to the containment Sump Recirculation Actuation Signal (SRAS). Following the SRAS, a portion of the Reactor Building Closed Cooling Water (RBCCW) flow is directed through the shutdown cooling heat exchanger to cool the sump water prior to being sprayed into containment. RBCCW is isolated to nonvital inside containment loads by manual closure of 2-RB-30.1 A/B and 2-RB-37.2A/B. The resulting RBCCW flow is reduced from 2000 gpm (prior to SRAS) to 1600 gpm per CAR cooling unit. This flow reduction, coupled with the reduction in containment air/steam temperature at the time of SRAS, results in a reduction in the containment heat removal capability of a CAR cooling unit from 80 million to 60 million Btu/hr. Two containment air recirculation and cooling units in combination with one containment spray pump (see Section 6.4.2.1) provide complete containment cooling capabilities under post-accident conditions. Each component in this combination is powered from the same emergency power source (Section 8.3) and is provided with cooling water from the same RBCCW header (Figure 9.4–2). This combination is designed for removing 280 x 10^6 Btu/hr. The heat sink for the containment air recirculation and cooling system is shown in Figure 6.4–9.

While the failure of both trains of the containment spray system requires multiple failures and is beyond the licensing design basis of the plant, following a LOCA, containment cooling may be provided by three of the four containment air recirculation and cooling units without the containment spray system. This combination is designed for removing 240 x 10^6 Btu/hr under LOCA conditions. However, following a MSLB accident, the combination of three containment air recirculation and cooling units does not provide adequate containment heat removal. A MSLB accident can result in a dry steam blowdown. With a dry steam blowdown, the potential exists for a significant increase in and superheating of the containment steam/air temperature above 289°F. In the design basis event/Section 14.8.2), the initiation of the containment sprays quickly reduces

the containment air/steam temperature to saturation conditions, below 289°F. Without containment sprays following a MSLB accident, the potential exists that the containment air/ steam temperature remain above 289°F for extended periods of time, possibly resulting in an excessive heatup of structures and equipment inside containment.

The containment air recirculation and cooling system is designed to maintain an ambient temperature within the containment of 120°F during normal operation. The duct work distribution system is routed to provide cooling air near all major components.

The reactor cavity cooling air is supplied by two independent, physically separated air supply headers. These are located on opposite sides of the reactor cavity. The two air supply headers are fed by a common plenum, which in turn, is fed by the four (4) CAR Fans.

The cavity cooling system is designed to supply air at a maximum of 91°F to the bottom of the cavity and to the reactor vessel supports and nuclear instruments in the reactor cavity. The maximum estimated air temperature at the permanent reactor cavity seal plate is 132°F.

Air is supplied directly in the vicinity of the steam generators, reactor coolant pumps and motors and pressurizer. Additional air is supplied in the lower regions of the steam generator compartments. As the air is heated by sensible heat, a stack effect is created within the steam generator compartments.

Air is supplied directly into the pressurizer skirt assembly to provide a suitable environment for the pressurizer heaters. The ambient temperature in this region is limited to 120°F.

Air is supplied directly to cool the concrete interface where the concrete is in contact with environment operating above 150°F. In all cases, the concrete does not exceed 150°F the containment during normal operation.

Each of the four containment air recirculation and cooling units consists of a direct drive two-speed vaneaxial fan, eight cooling coil sections and steel housing which forms an integral air handling units.

Each coil house is equipped with eight individual coils piped to supply and return manifolds which connect to the RBCCW system.

The four containment air recirculation and cooling units (CARs) are arranged in two groups. Each group includes two CAR units that share an independent RBCCW supply header. Each header branches into two supply lines, one supply line to each of the two CAR units in the group (Refer to Figure 9.4–5).

Each of the four RBCCW supply lines has an air-operated stop valve for its associated CAR unit (for example: 2-RB-28.1A for one of the four units). Each of these four RBCCW supply stop valves is normally open and deenergized. Each of these valves is designed to fail in the open position.

Each of the four CAR units has an RBCCW return line that has two air-operated stop valves in a parallel arrangement (for example, valve 2-RB-28.2A and 2-RB-28.3A for one of the four CAR units). Each of the air-operated control valves in the four sets of return line valves can be operated from the control room. One of the two control valves in each return line set (e.g., 2-RB-28.3A in the example set) opens automatically on receipt of a safety injection signal (SIAS).

There are twenty air temperature measuring instruments located in the containment. Eight of these measure the containment atmosphere air temperature in different locations in the containment, including one location within the reactor cavity. By use of an eight position selector switch and a temperature indicator located on the main control board, the containment air temperature at the eight locations can be monitored in the main control room. In addition, the temperature readings are inputted into the computer. The other 12 temperature measuring instruments are associated with the various ventilation systems. These measurements are also inputted into the computer.

A malfunction in the cavity cooling system is indicated by an increase in cavity air temperature and also by an increase in the containment air temperature since these areas are served by the same air distribution system. The limiting conditions for operation is the primary containment average temperature of 120°F as described in the Technical Specification 3.6.1.5. During normal plant operation the CAR coolers are designed to maintain the cavity concrete temperature below 150°F.

6.5.2.2 Components

Ductwork for the containment air recirculation and cooling system outside the secondary shield wall is galvanized steel containing 1.25 oz/square foot of zinc. Inside the steam generator compartment duct work is 18-8, ANSI Type 304 stainless steel with a 2B mill finish. Transverse duct work joints are welded flange with angle reinforcing. Longitudinal seams are continuously welded.

Materials of construction and performance data for system components are listed in Tables 6.5-1 and 6.5-2, respectively. The fan performance curve is shown in Figure 6.5–1. The coil performance is shown in Figure 6.5–2.

Codes and Standards:

Cooling coils	ASME Section VIII
Fan	AMCA 211A
Motors	IEEE STD-334-1971, NEMA, MG-1
Duct work	SMACNA, High Velocity Standards
Seismic	Class 1 (Ductwork Class 2)

6.5.3 SYSTEM OPERATION

6.5.3.1 Normal Operations

During normal operation, the 120°F containment air is recirculated through the containment air recirculation and cooling system by the vaneaxial fans. The fans operate normally at high speed. Air is supplied through the duct work system into the steam generator compartments and reactor cavity at approximately 91°F. The cooling air is heated, by the sensible heat, to 120°F as natural circulation occurs.

The RBCCW system supplies cooling water at 85°F maximum to the cooling coils of each CAR unit (Section 9.4). During normal operation, the full sized valve in each of the four sets of two parallel RBCCW return valves (e.g., for one CAR unit, 2-RB-28.3A) is positioned according to the containment heat load requirements. (Each of these same full sized RBCCW return valves also provides service during a LOCA). Additionally, during normal operation, the corresponding smaller (6 inch diameter) return line valve in each parallel set is open.

During normal operation, three of the four containment air recirculation and cooling units are required to provide sufficient sensible cooling.

The loss of the normal containment cooling system is not considered credible.

However, postulating the above event, the containment temperature and pressure response would distinguish this occurrence from the LOCA. The containment conditions following the design basis accident (DBA) are described in Section 14.8.2). Assuming the loss of normal cooling event, the containment temperature would increase approximately 40°F for a one (1) psi increase in containment pressure. Therefore, for this event there is an elevated containment temperature to a relatively small increase in containment pressure. Other minor accidents, less severe than the DBA, would result in the elevated temperature and relatively large increase in pressure since additional mass is being introduced into the containment free volume. Thus, the postulated loss of normal cooling capability could be distinguished from an accident condition.

Section 3.6.1.5 in the Technical Specification states the containment temperature limit. Safety related instrumentation exposed to the containment atmosphere is capable of withstanding 150°F on a continuous basis.

In the unlikely event that the fusible link plates and blowout plates open during normal operation, the containment air distribution system would be out of service. As a result, the cool air from the containment air recirculation and cooling units would be discharged outside the secondary shield wall. However, the stack effect created by the temperature gradient within the steam generator compartments would induce portions of this cool air inside the compartments. Although partial cooling does exist, potential hot spots could be present around major components. Temperature elements located throughout the containment will alarm any elevated temperature condition. Upon receipt of a high-temperature alarm on the plant computer, the operator will take action in accordance with plant procedures.

6.5.3.2 Leak Rate Testing

The operation of the Containment Air recirculation and Cooling systems are not required during the Integrated Leak rate test. During the initial containment pressurization, the units may be operated to remove the heat from compression through RBCCW as the cooling medium. The fans and motors operate at low speed to reduce horsepower requirements at these elevated pressure conditions.

The units may be operated during the test, if required.

The units may operate without cooling water during the post-test containment depressurization to provide heating.

6.5.3.3 Emergency Conditions

Upon receipt of a SIAS, the idle containment air recirculation and cooling unit is automatically started on the low speed setting. Simultaneously, the running units are switched from their normal operating high speed setting to low-speed operation. The full flow (10 inch diameter) RBCCW valves at the outlet of each cooler are also opened upon receipt of the SIAS. (Figure 9.4–2)

With off site power available under this mode of operation, three units are switched to low speed and the fourth is started on low speed as described above. Each of two emergency buses carries the load of two containment air recirculation and cooling units.

Each CAR fan coil unit is provided with blowout doors equipped with fusible links downstream of the fan discharge. At high temperature in the containment environment after either a LOCA or a MSLB, these fusible links melt disengaging the doors allowing them to swing open. This allows air to circulate through the path of least resistance as described in Section 6.5.4.1.

The containment air recirculation and cooling units assist in uniform mixing of combustible gas within the containment environment as described in Section 6.6.2.1.

6.5.3.4 Refueling

The containment air recirculation and cooling units may be operated at reduced speed during refueling operations. The units provide mixing with the containment purge air (see Section 9.9.2) for uniform temperatures throughout the containment.

6.5.4 AVAILABILITY AND RELIABILITY

6.5.4.1 Special Features

The components of the containment air recirculation and cooling system are designed to the general requirements including seismic response as described in Section 6.1. Components are protected from missile damage and pipe whip by physical separation of duplicate equipment as described in Section 6.1.

The cooling units are designed to operate for the life of the station. There are no belts or flexible couplings; the motor is directly connected to the fan wheel.

Failure of the normal electrical power supply automatically places all four fans on emergency electric power source. A failure mode analysis is given in Table 6.5-3.

Associated system components, such as piping, valves, and instrumentation are located outside the secondary shield to minimize the possibility of missile damage.

The ductwork system which is located on the containment air recirculation fan discharge, downstream of the blowout doors, is not required during post-accident operation. To assure an unrestricted flow path, 4 doors downstream of the discharge of each fan open via the melting of the high-temperature fusible links. These doors consist of galvanized steel sheets attached rigidly to the duct work system by means of UL rated fusible links.

The fusible links are nominally rated at 165°F ± 7% based on the UL allowable tolerance for fusible links. Each containment air recirculation unit is provided with blowout doors equivalent to 30 square feet of flow area. This flow area is greater than the fan outlet area. This assures fan operability and prevents flow restrictions following any possible duct work failure during post-accident conditions. All blowout doors plates are hinged to the ductwork to prevent their becoming credible missiles after separation.

The post-accident distribution system is designed to discharge the air in four (4) horizontal directions from each unit through the openings left by the fusible link plates. Under design conditions, it is assumed that the existing duct work is restricted such that all the air is discharged through these openings. The throw (distance traveled by the air stream before reaching terminal velocity) is approximately 100 feet. Under these conditions, the discharge from the upper unit is well beyond the intake region of the lower unit, thus preventing any short circuiting. The air streams drop off toward the end of the throw and tend to settle toward the bottom of containment due to the slightly lower temperatures. This creates secondary stack effects throughout these regions of containment.

It is anticipated, but not necessary, that portions of the ductwork distribution system remains intact. This provides greater air distribution throughout containment. However, the velocity of air, and hence the throw, through the fusible link plate openings would be much less. This could create a slight short circuiting effect. Since the units are located one above the other, the bottom unit may suction portions of the discharge from the upper unit. However, this has negligible effect on the performance of the units since, during the short term, the units will be removing latent and not sensible heat. That is, the discharge temperature from the units will be only a few degrees lower than the inlet temperature. The effect of short circuiting is slightly more pronounced as the containment post-accident temperature begins to decrease, as the units are removing more sensible heat. However, at this time, the containment heat removal requirements are much less than the heat removal capabilities of the containment air recirculation and cooling units. Short circuiting is not expected to affect the performance of the system.

The containment air recirculation system in the post-accident mode of operation provides local containment cooling. The containment spray system is utilized in conjunction with the containment air recirculation and cooling system to provide a uniform containment atmosphere.

Relief dampers are provided for the ductwork headers on the side of the main CAR fan discharge plenum outside of the secondary shield wall. These dampers are provided to prevent a pressure front from reaching the containment air recirculation and cooling units through the ductwork system. The relief dampers are held closed by tension springs of Type 301 stainless steel. The springs are designed to permit the damper to be fully open (out from the duct) at a three-psi pressure differential. Each main ductwork header is provided with 30 square feet of relief area.

The condensate leaving the cooling coils is collected in individual drip pans. These pans cascade the liquid into the housing drain. The air side coil face velocity is approximately 180 fpm under LOCA conditions. This low velocity reduces moisture carry-over from the coils.

Each housing is equipped with two four-inch drains with loop seals. Clogging of these drains does not render the unit inoperable. Under this postulated case, the housing water level rises and overflows through the coil sections before reaching the fan inlet. The curb supporting the cooling coils extends approximately six inches above the housing floor and the fan inlet extend approximately 15 inches above the housing floor.

The fin pitch is 8.5 fins per inch of cooling coil length. With this pitch, water logging of the coil fins under post-accident conditions is minimized to avoid any problem.

A fouling factor of 0.0005 for the water side is included in the coil ratings. Demineralized water (RBCCW) fouling factor of 0.0005 is recommended by the Tubular Exchangers Manufacturing Association (TEMA). The water side of the cooling coil tubes is equipped with removable plugs on the return bends of the coils to permit cleaning.

The cooler housing is designed to ensure no loss of function when subjected to a pressure differential of 2 psi. The housing is open on four sides through the cooling coil sections and through the base by the fan barrel. Each of the four sides of the housing is approximately 40 percent free area through the coil face. This provides ample free paths to minimize differential pressure during the post-accident transient-pressure condition.

A simplified analysis was performed on the containment air recirculation and cooling units to demonstrate that the units will not experience a differential pressure greater than 2.0 psi during the post-incident conditions. The following is a summary of that analysis:

1. It was assumed that a given cooler was pressurized with air to a pressure of 2.0 psig.

2. Assuming air to be a perfect gas, the change in mass from atmospheric pressure to 2.0 psig was calculated.

3. Based on the 2.0 psi per second post-accident pressure transient, the mass flow rate, and hence, the volumetric flow rate, can be calculated.

4. Assuming the pressure drop across the coil banks is proportional to the square of the flow rate, for any given flow rate the corresponding pressure drop across the coil banks, and hence the cooler housing, can be calculated.

Based on the above-simplified analysis, a flow rate less than 7,000 CFM (normal flow rate is 70,000 CFM) was calculated based on the 2.0 psi per second pressure transient. The drop across the coil banks is much less than 0.10 inches w.g., which is negligible. This allows a factor of about 100 inches converting the model from the perfect gas to the post-accident steam-air mixture. The difference in densities between the model assumed (0.075 lb/ft^3) and the design post-accident atmosphere (0.18 lb/ft^3) is about a factor of 2. Therefore, there is adequate margin to demonstrate that the containment coolers will not experience a pressure gradient across the housing of 2 psi.

Each Containment Air Recirculation and Cooling unit is provided with a vibration-sensing switch. In the event of excessive, sustained vibration at the fan assembly, the switch will provide an alarm in the Control room.

The valves in the normal cooling water outlet lines (6 inches) will be open during normal operation and valves in the parallel emergency outlet lines (10 inches) can be opened from the control room and the flow rate can be monitored at any time.

The maximum RBCCW temperature on the discharge side of the containment air recirculation and cooling unit coils is 234°F during design post-accident conditions. The RBCCWS operates under pressure, approximately 50 psia in the downstream piping. The saturation temperature corresponding to this operating pressure is approximately 280°F. Therefore, the RBCCW temperature at the discharge side of the CAR units is approximately 46°F below saturation. In addition, the RBCCW surge tank (Section 9.4.2.1) maintains a minimum static head on the CAR coolers, corresponding to about 30 psia. The saturation temperature corresponding to this pressure is 250°F.

6.5.4.2 Tests and Inspections

The major components of the containment air recirculation and cooling system are tested for performance and integrity to assure its ability to operate within the containment post-accident environment.

The cooling coil sections are similar to a coil section which has been previously tested and has demonstrated the capability of condensing steam at conditions equal to or greater than the LOCA design conditions. The coil section performance test results are compared to computed data and the comparison is then used to predict the performance of the full-size coil assembly. A topical report, W-CAP-7336-L, verifying the performance of similar coils for this application was filed by Westinghouse with the NRC.

The fans and motors are similar to those which were tested and demonstrated as having the capability of starting and operating in the post-accident environment. Testing is conducted in accordance with the Draft IEEE Standard 334-1971. Westinghouse is preparing to file topical report, W-CAP-7829, on the test results. Topical reports, W-CAP-7343-L and W-CAP-9003, for the motor insulation and cable splice were filed with the NRC by Westinghouse. A complete commercial motor test is performed before and after the simulated LOCA tests on the production model motor.

Each fan and motor was tested as a unit to assure characteristic performance curves. Fan ratings are in accordance with Air Moving and Conditioning Association (AMCA) Standard Test Code 211-A.

All stators used in qualified Class 1 motors (such as these) are given a voltage test in water before assembly in a motor.

Hydrostatic testing was performed on the cooling coil sections in accordance with ASME Code Section VIII requirements. The cooling coil tubes were nondestructively tested in accordance with ASTM Standard E-243-68.

The containment air recirculation and cooling system ductwork is leak tested and balanced in accordance with Sheet Metal and Air Conditioning Contractors National Association (SMACNA) Standards.

Provisions are incorporated into system design for online testing during normal operation. Each group of two containment air recirculation and cooling units is tested separately. In accordance with applicable Technical Specifications and corresponding procedures, during Modes 1, 2, and 3, CAR unit emergency RBCCW outlet valves and slow CAR fan speeds are tested. The testing of the remotely operated RBCCW valves is performed by operating the corresponding manual control switch. The testing of slow fan speeds is also performed by use of manual control switches in the control room. The positions of the valves and fan speeds are verified by status lights in the control room.

The testing of these components during Modes 1, 2, and 3 does not use any manual initiation of SIAS. Such a manual SIAS actuation would result in unnecessary starting of diesel generators and unnecessary RBCCW flow transients.

Additionally, in accordance with Technical Specifications required surveillance frequencies, during outages, manual SIAS actuations are used for testing automatic actuation circuitry for emergency operation of CAR units, as well as other integrated safeguard features.

The fusible link blowout doors (plates) were tested as part of the initial plant pre-operational test of the containment air recirculation and cooling system. The blowout doors were tested for disengaging to open to allow an air flow path from the ductwork as the local ambient temperature exceeded 170°F. The blowout doors are mounted with compressed springs between the blowout doors and the ductwork to provide a spring action for an initial disengaging push to open the blowout doors, when the fusible links melt.

Assuming a failure of the fusible link plate to disengage, there would be a partial loss of that containment air recirculation and cooling unit. Each unit is provided with four (4) plates to insure at least partial flow. Should one of the four (4) plates on a given unit fail, there would be only a partial loss of flow due to the increased system resistance.

A preoperational test of the containment air recirculation and cooling system was performed prior to start up. The test procedure is described in Section 13.

Equipment and associated components located in the containment are accessible for inspection and maintenance during shutdown.

In lieu of test results under 10 CFR 50, Appendix J. Type C testing, the “T-Ring” seats on the containment isolation valves, for the RBCCW supply and return lines for the Containment Air Recirculation and Cooling heat exchangers, will be replaced based on observed degradation. These valves are not in the Type C testing program because the valves are open during accident conditions. This eliminates a commitment made under letter A06107 dated 1/16/87 under Docket Number 50-336.

TABLE 6.5-1 CONTAINMENT AIR RECIRCULATION AND COOLING UNITS MATERIALS OF CONSTRUCTION

Fans (F-14A/B/C/D)

Type	Vaneaxial
Standard	AMCA 211-A
Seismic Class	1

Motor

Type	Induction
Standard	NEMA
Seismic Class	1

Cooling Coils (X-35A/B/C/D)

Type	Water (RBCCW)
Standard	ASME Section VIII
Seismic Class	1

Ductwork

Type	In Accordance With Specification M-506
Standard	SMACNA
Seismic Class	1

TABLE 6.5-2 CONTAINMENT AIR RECIRCULATION AND COOLING UNITS PERFORMANCE DATA

Per Unit	Normal	Post-incident (prior to SRAS)
Heat removal capability (Btu/hr)	2.2×10^6 (nominal rating)	80×10^6 (nominal rating)
Air flow (cfm)	70,000 (nominal rating)	34,800 (nominal rating)
Static pressure (inches wg)	2.98	0.53
Brake horsepower (bhp)	70.0	18
Static efficiency (%)	56	50.7
RPM	1760	875
Motor horsepower, (hp)	75	37.5
Air temperature, inlet/outlet (°F)	120/90.9	289/281.8
Water temperature, inlet/outlet (°F)	85/94	130/210
Water side pressure drop (feet H_20)	3	25
Water flow rate (gpm)	* 500	2000

Note:

* The cooling water flow through the CAR coolers is maintained at approximately 2000 gpm during normal operation to maintain RBCCW pump flow closer to the normal pump design conditions.

TABLE 6.5-3 CONTAINMENT AIR RECIRCULATION AND COOLING SYSTEM

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Housing CTMT cooling unit (4)	Housing failure (Air bypasses cooling coils)	Temperature sensing device if available	Loss of cooling	Unit taken out of service	One unit out of service, three alternate units operable	Sufficient containment cooling provided by two units in combination with one containment spray subsystem or by two (2) containment spray subsystems.
Cooling coils (Eight (8) per unit)	Tube header of manifold rupture	Temperature sensing device, if available	Loss of cooling	Same as above	Two units on RBCCW Header taken out of service	Sufficient containment cooling by a combination of two (2) units with one (1) containment spray subsystem.
Fans (4)	Fails to operate	Status lights CRI	Loss of air flow	Same as above	One unit out of service, three alternate units operable	Same as Item 1
Fusible ink plates (4)	Fails to open	None	Partial loss to complete loss of air flow depending upon degree of restriction in ductwork system	None	Partial to complete loss of one unit, three alternate units operable	Same as Item 1

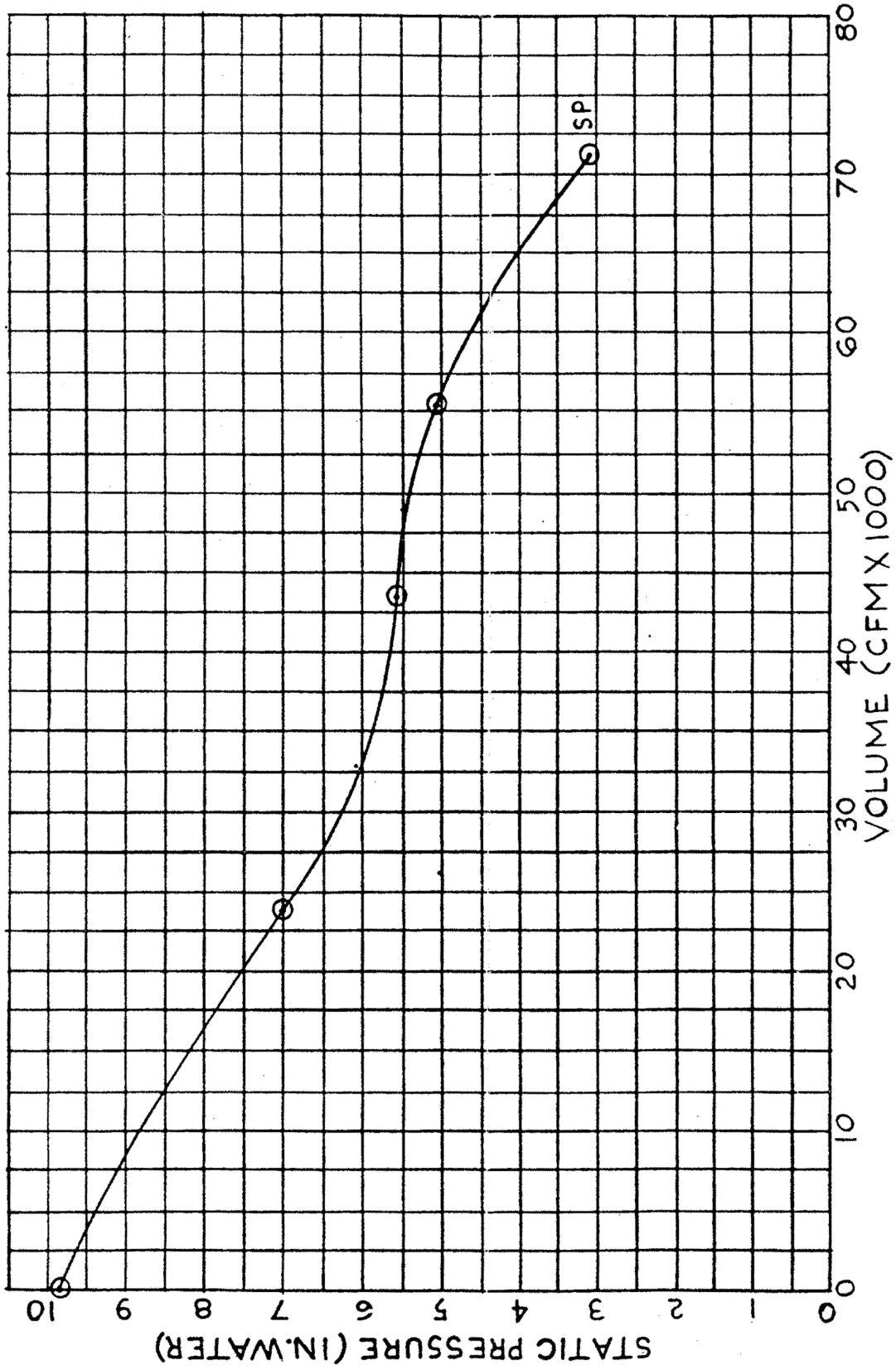
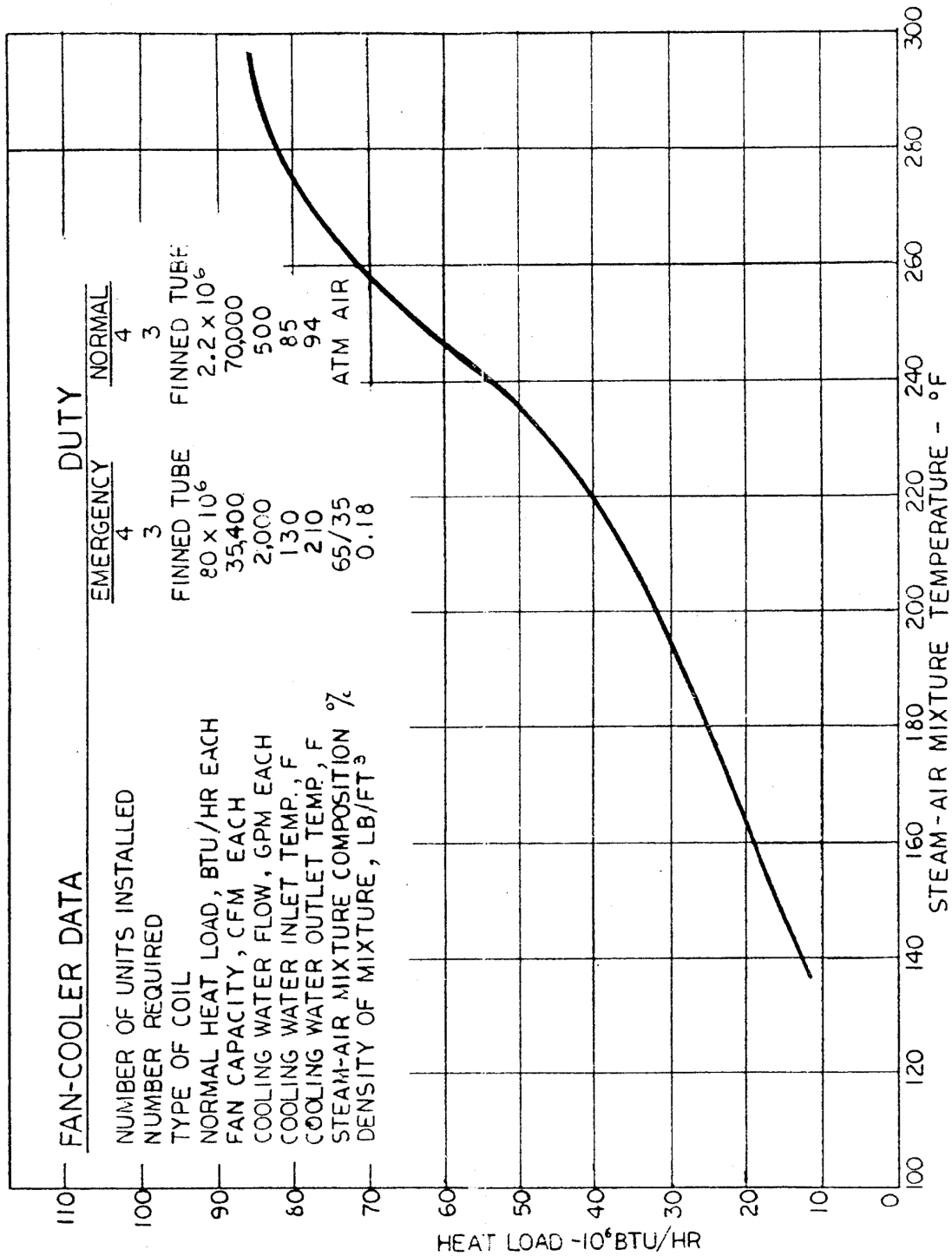


FIGURE 6.5–1 CONTAINMENT AIR RECIRCULATION AND COOLING UNIT FAN PERFORMANCE CURVE

FIGURE 6.5-2 CONTAINMENT AIR RECIRCULATION AND COOLING UNIT COIL PERFORMANCE CHARACTERISTICS



6.6 CONTAINMENT POST-ACCIDENT HYDROGEN CONTROL SYSTEM

The hydrogen recombiner portion of the Post-Accident Hydrogen Control System is installed, but not used for any mitigating function. The hydrogen recombiners, associated controls and alarms have been isolated awaiting abandonment. The following section describes the recombiners as originally installed and operated.

6.6.1 DESIGN BASES

6.6.1.1 Functional Requirements

The post-accident hydrogen control system functions to control the concentration of hydrogen which may be released within the reactor containment atmosphere following postulated accidents.

6.6.1.2 Design Criteria

The following criteria have been used in the design of the containment post-accident hydrogen control system:

a. Each system has two redundant, independent subsystems, each capable of performing the functional requirements.

b. The system has suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities are provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure of an active component in either subsystem will not affect the functional capability of the other subsystem.

e. The system is designed to permit periodic inspection of important components such as hydrogen recombination units, fans, filters, valves, piping, ductwork and analyzers to assure the integrity and capability of the system.

f. The post-accident hydrogen control system is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components; (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole. Under conditions as close to the design as practical, the performance will be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

g. The post-accident hydrogen control system is designed to the general criteria as described in Section 6.1.

h. The components of the containment post-accident hydrogen control system are designed to operate in the most severe post-accident environment as described in Section 6.1.

i. The post-accident hydrogen control system is designed in accordance with the conditions given in Regulatory Guide 1.7.

6.6.2 SYSTEM DESCRIPTION

6.6.2.1 System

The post-accident hydrogen control system is shown schematically in Figure 9.9–2.

The post-accident hydrogen control system includes independent, fully redundant subsystems to mix and monitor the hydrogen concentration in the containment following a loss-of-coolant accident.

The uniform mixing of the containment post-accident atmosphere is provided by the post-accident recirculation (PIR) system. This (PIR) system is provided inside the containment to mix any hydrogen accumulated in the upper portion of the containment with the rest of the containment atmosphere.

The post-accident recirculation system takes suction from the highest points in the containment and discharges air to the minus 3 foot elevation area of containment. Two CAR fans and coolers remove air from the minus 3 foot elevation area, and dilute this mixture with additional containment air and discharges it into the lower elevations of containment. Each PIR system is designed for one air change per hour from the upper 20 percent of the containment volume.

Two full capacity, completely redundant, hydrogen concentration monitoring systems are provided outside the containment for periodic or continuous analysis of hydrogen concentration in the containment atmosphere.

Each train of the hydrogen monitoring system consists of a sensor module located in the east electrical penetration room elevation 14 feet 6 inches, a remote calibration module located in the railroad and truck access bay elevation 14 feet 6 inches, and a control room module located in control room panel C101. Recorders in control room panel C101 provide indication and a permanent record of the hydrogen concentration.

The main control room and the remote calibration module are accessible at all times. Access to the sensor modules under post-accident conditions is not necessary. Control of all system functions including performing calibrations is available from the remote calibration modules. In addition, the control room module provides system power status indication, system alarms and system start control to the control room personnel. Since the two trains of the hydrogen monitoring system are completely independent, no single failure can prevent operating personnel from initiating hydrogen sampling or reading hydrogen concentration from the control room.

Hydrogen concentration measurements are made by two identical subsystems each of which monitors a sample from the containment atmosphere. High hydrogen concentration is alarmed at two volume percent.

Each hydrogen monitoring system draws containment gases through a small diameter sample line which limits the flow rate and reduces the sample transit time. From the analyzer, the sample is returned to the system from which it is drawn.

The active means for reducing the hydrogen concentration in the containment following a postulated accident are provided by hydrogen recombination subsystems. The recombiners have no mitigating function.

Each of the recombiners is capable of 100 SCFM.

The hydrogen recombiners are located within the containment and consist of two completely independent, full capacity thermal type recombiner units.

One recombiner unit is located on the operating floor at Elevation 38 feet 6 inches. The second unit is at Elevation 14 feet 6 inches.

The containment hydrogen-air atmosphere is circulated across electric heating elements. The influent is heated to approximately 1100°F at which temperature the recombination reaction takes place. Containment air is induced into the recombiner unit by the natural draft created by the temperature differential between influent and effluent. Additional driving forces are not required.

A method for hydrogen reduction in the containment following a LOCA is the hydrogen purge system. The hydrogen purge operation, is not credited in accident analysis. During purging operations, the hydrogen concentration is reduced by providing makeup air from the instrument air or station air compressors using containment atmosphere sample return line penetrations into the containment, and by remotely opening the containment isolation valves (2-EB-91 & 2-EB-92, or 2-EB-99 & 2-EB-100) to their respective 6 inch line connecting with the Enclosure Building Filtration System (EBFS) common plenum.

Each purge train includes a passive flow control station, consisting of a manual valve (in the open position) in the 6-inch line (valve 2-EB-193 Train “A” or valve 2-EB-194 Train “B”), and a small 1.5 inch bypass line equipped with a flow element and a remote flow indicator.

The containment purge flow is mixed with the enclosure building filtration system and is passed through the EBFS charcoal filters discharging through the Millstone Stack.

The hydrogen purge system does not need to be redundant or be designed Seismic Category I, consistent with the recommendations of Regulatory Guide 1.7, Rev. 2, except insofar as portions of the system constituting part of the primary containment boundary or containment filters.

The instrument air, used for purging, ties into line 1"-HCD-118 upstream of valve 2-AC-20 and enters the containment through penetration 87, (Figure 9.11–1). The station air used for purging

ties into line 1"-HCD-117 upstream of valve 2-AC-15 and enters the containment through penetration 62, (Figure 9.11–1). Two capped branch connections are also provided downstream of valves 2-AC-20 and 2-AC-15 for a temporary emergency air supply.

If the air compressors are not available after a LOCA, there is adequate time to restore the system to service. Portable air compressors can be connected to the capped connections provided for emergencies.

Since the purging of containment atmosphere could increase the radiological dose to the public, this system is initiated only when hydrogen concentration must be reduced due to a threat to containment integrity as the result of a beyond design basis accident.

6.6.2.2 Components

The components of the hydrogen purge system are described with the EBFS in Section 6.7.2.2. The major components with associated fabrication and performance data are listed in Table 6.6-1.

The Hydrogen Recombiner units supplied by Westinghouse, consist of two 100 percent capacity recombiners, thermal (electric) type, of completely passive design with no moving parts and with operating controls mounted on control boards located in the Main Control Room. Environment tests under simulated conditions have been performed by Westinghouse with results verifying the integrity and reliability of their units.

The hydrogen monitoring system is supplied by Whittaker Safety Systems. Two redundant trains are supplied each consisting of a sensor module, a remote calibration module, and a control room module.

Each sensor module contains valves, sensors, heat trace and a sample pump. Connections to the sample stream supply and return, calibration gases and Containment Air Post Accident Sampling System are provided. The hydrogen sensors use electrochemical gas measurement technology. The sensor modules are located in the east electrical penetration room elevation 14 feet 6 inches. The maximum air temperature in this room will be 140°F and maximum relative humidity will be 100 percent after a LOCA. Access to the sensor modules is not required for system operation or to perform system calibrations.

The sensor module will take a sample from the containment atmosphere and return it to containment after measuring the hydrogen concentration. Provisions to divert the sample stream to the Containment Air Post Accident Sampling System prior to returning it to containment are provided. Operation of the Containment Air Post Accident Sampling System with the hydrogen monitoring system discharge will not interrupt the ability to monitor hydrogen concentration.

Each remote calibration module contains a microprocessor, controls and relays necessary for operation, self-monitoring, performing calibrations and interfacing with the control room. Control room annunciation is provided for high hydrogen, system trouble and low ambient temperature local to the remote calibration modules. The system range is 0 to 10 percent hydrogen. Calibration of the system is performed using calibration gases with 1 percent and 4 percent hydrogen. The

remote calibration modules and the calibration gas bottles are located in the railroad and truck access bay elevation 14 feet 6 inches, which is accessible after a LOCA.

Each train has a control room module located in control room panel C101 which interfaces with the remote calibration module. Each control room module has system 120 VAC and 480 VAC status lights, alarms for loss of power, system trouble and high hydrogen, and a control switch for initiating hydrogen monitoring from the control room. Control room display of hydrogen concentration is provided on a recorder also located on control room panel C101.

The hydrogen monitoring system is QA Category 1. Two redundant trains of equipment meet the design and separation requirements for Class 1E and are powered from separate Class 1E power sources. Piping, tubing and other equipment is Seismic Class 1. The monitor system must be operable following, but not necessarily during a SSE. Pressure boundaries which are an extension of the containment building atmosphere are designed to maintain their integrity during a SSE.

Qualification tests for the hydrogen monitoring equipment include seismic testing, pressure testing and performance testing.

6.6.3 SYSTEM OPERATION

6.6.3.1 Emergency Conditions

In the unlikely event of a LOCA, the post-accident hydrogen control system is manually initiated by operation action. All subsystems are initiated and monitored in the control room.

The PIR system is initiated within the first few hours following the LOCA. The system takes suction from the dome region where the initial local concentration of hydrogen is anticipated. This potential hydrogen-air mixture is conveyed through carbon steel piping headers and exhausted near the inlet of the containment air recirculation and cooling units located on the minus three foot level of containment (Section 6.5) where it is diluted with the containment environment from lower elevations.

The operation of the PIR system can be monitored in the control room by indicating lights and motor trip alarms.

Two redundant, independent hydrogen monitors are provided for sampling during post-accident operation. Continuous sampling is manually initiated from the Control Room. Recorders provide the operators with visual indications of the measured hydrogen concentration between zero to ten volume percent. Grab sampling of containment atmosphere can be taken from the post-accident sample system (PASS) containment air remote operator modules C102A & B located at elevation 14 feet 6 inches of the auxiliary building.

In accordance with Regulatory Guide 1.7 Rev. 3, within 90 minutes following the initiation of safety injection, the hydrogen monitors are required to be fully functional.

The hydrogen recombiner system may be manually initiated from the control room. The operation of the recombiners is monitored in the control room by indicating lights and measurement of the effluent temperature. Performance of the recombiners is monitored by measurements of the local concentration of hydrogen in the containment by the monitoring system.

Operating at rated capacity of 100 scfm of hydrogen-air mixture, each recombiner removes two scfm of free hydrogen from the containment atmosphere when the hydrogen concentration is at least two percent by volume.

The hydrogen purge system is provided as an active means for reducing the local buildup of hydrogen. This operation is initiated manually.

The overall performance of the containment post-accident hydrogen control system is monitored by the recorders located in the control room. Each hydrogen monitoring system is set to alarm at two volume percent containment hydrogen concentration.

6.6.4 AVAILABILITY AND RELIABILITY

6.6.4.1 Special Features

The components of the containment post-accident hydrogen control system are designed to the general requirements, including seismic response, as described in Section 6.1. All components are protected from missile damage and pipe whip by physical separation of duplicate equipment, as described in Section 6.1.

To assure availability, all components located within containment are designed to withstand the pressure and temperature transients conditions resulting from a LOCA. Each PIR subsystem is provided with carbon steel suction headers to assure system integrity following the LOCA. Ductwork is not used in the PIR system. The motors associated with the PIR fans are designed to operate in a containment post-accident environment.

The PIR fans are located outside the secondary shield wall, below the operating floor, at an elevation well above the post-accident water level. In this location the fans are protected against credible missiles and flooding.

The PIR fans are completely redundant, are powered from independent emergency sources and are physically separated in the containment. A failure mode analysis of the PIR system is given in Table 6.6-4.

The hydrogen monitoring system includes two independent, fully redundant monitoring subsystems. Both are physically separated and located outside the containment. Independent emergency power sources are provided for each subsystem.

If the normally aligned Instrument Air System is unavailable, a safety related backup bottled air supply subsystem provides the source of air required to open the Containment Isolation Valves required to establish a flow path for the Hydrogen Monitoring System post-accident. The backup

bottled air supply system to the Hydrogen Monitoring System is not independent as both trains of inside and outside CIVs are supplied operating air by the same supply header (one supply header to the inside valves and one supply header to the outside valves), however, it is designed to meet Single Failure Criteria.

The hydrogen recombiner systems are completely redundant, are supplied from independently emergency power supplies and are physically separated. Each recombiner unit is designed to withstand the temperature and pressure transient following a LOCA and to operate under post-accident conditions. Both units are located well above the flood level in the containment and physically separated for missile protection.

The containment has redundant penetrations and pipe headers for the hydrogen purge systems and are physically separated for protection from credible missiles. Each hydrogen purge system is provided with independent valving and instrumentation and powered from independent emergency sources. Prior to initiation of the hydrogen purge system, the EBFS electric heaters, X-61A/B, are manually taken out of service. The EBFS is described in Section 6.7. Ductwork is utilized outside the containment only, under atmospheric pressure conditions.

Each hydrogen purge valve is provided with an auxiliary air accumulator to assure an air supply for post-accident operation. The cylinder air-operated valves if open, are closed by the accumulator air assuming a failure of the normal instrument air system (Section 9.11). Air is available in the accumulator for opening the purge valves, if required, for hydrogen control. Each accumulator is sized for four open or close operations.

The hydrogen purge containment isolation valves inside containment are designed to fail in their last position. These valves are normally closed and are therefore, expected to be closed during the LOCA condition.

Intermittent system operation or system isolation can be controlled by the outside containment isolation valve. These valves are provided with handwheels for manual operation should instrument air be unavailable.

Precautions have been taken in the design of the equipment, their support structures and building structures in the containment to avoid probable pockets or enclosures where hydrogen build-up could occur locally. Provisions have been incorporated for ventilation holes, sloping concrete floor slabs, openings and other means of allowing locally generated hydrogen to flow upward in the containment.

The major source of hydrogen generation following the accident is the containment sump. Since the sump region will be at a higher temperature than the dome region due to the cool containment spray water (Section 6.4) entering the containment, a stack effect will be created. Therefore, it is assumed that the higher energy hydrogen-air mixture from the containment sump region will tend to rise toward the dome. Vent areas, such as the floor grating outside of the secondary shield wall, around the steam generators and over the reactor coolant pump motors, offer negligible resistance to this induced flow. Taking a cross-section of the containment at the operating floor (elevation 38 feet 6 inches), the ratio of the free area created by the grating and other openings to the cross-

section area of the containment is 70 percent. Conservation analyses estimate an induced flow in the order of 15,000 CFM from one steam generator compartment at the containment conditions one (1) day after the accident. Although this flow rate decreases with containment post-incident temperature and pressure, the mechanism of transport is established.

Although the hydrogen concentration will tend to equalize itself, the PIR system accelerates this process by maintaining a slight negative pressure in the containment dome region, thereby supplementing the natural buoyancy forces, and by mixing the dome hydrogen-air mixture through the containment air recirculation and cooling units with lower level mixtures, thereby assuring a uniform mixture throughout the containment.

Each hydrogen recombiner unit is capable of 100 scfm to assure the reduction of hydrogen concentration and to shorten the duration of operation. The backup hydrogen purge system processes the containment effluent through the EBFS charcoal filters prior to discharge through the 375 foot Millstone stack.

Sufficient emergency electrical power is available to operate the post-accident hydrogen control components.

6.6.4.2 Tests and Inspection

The post-accident recirculation fans and motors are similar to those which are tested for starting and operating within the containment post-accident environment. Engineering data are provided to substantiate the extrapolation of environmental tests which are conducted at Joy Manufacturing Company on nuclear containment motors (Section 6.5.4.2) which are similar to the motors provided for the PIR fans. Components, materials, standards, method of manufacture and design criteria of the PIR fan motors are of identical origin and application to those motors tested by Joy. The only variation is in the motor size with relative horsepower and RPM. The tests performed by Joy are in accordance with the Draft IEEE Standard 334-1971. Identical QA records and reports are maintained on both the motors.

The motors used in the postincident recirculation units are made by Reliance Electric Company. Prototype motors have been successfully tested in a steam-air environment, hence the manufacturer does not consider a production test as described above to be necessary.

Prototype testing included a four hour run followed by four two-hour runs in a 300°F ambient. Record bearing temperatures were of the order of 300°F, and winding temperatures were of the order of 360-400°F. Each of the above cycles were separated by two, two hour runs in an ambient of 200-220°F.

Since this is a totally enclosed motor, no boric acid crystals can enter the internal air passages. All external air is directed along the ribbed motor enclosure.

No qualification test can insure that a motor has a service life greater than that demonstrated by service experience to date. It demonstrates that the equipment, in an aged condition, is capable of operating in a post design basis event environment for a period at least as long as the test cycles.

One of the two PIR fans has been rated in accordance with Air Moving and Conditioning Association (AMAC) Standard 211A.

The PIR system is incorporated with provisions for on-line testing. Each subsystem is tested individually. The PIR fan is manually initiated from the control room and is monitored for operation by motor trip alarms.

Full system tests are conducted during shutdown.

The PIR system is located within containment. The PIR fans are located near the operating floor to permit access for inspection and maintenance during shutdown.

The hydrogen monitoring system design has provisions for performing calibrations without accessing the sensor module in the east electrical penetration room elevation 14 feet 6 inches. The calibration process is initiated and controlled from the remote calibration module in the railroad and truck access bay elevation 14 feet 6 inches. The calibration process utilizes calibration gases of 1 percent hydrogen in nitrogen and 4 percent hydrogen in nitrogen. The gas cylinders with the calibration gases are located in the railroad and truck access bay near the remote calibration modules.

The hydrogen purge system is incorporated with provisions for on-line testing. The hydrogen purge valves are tested with the EBFS as described in Section 6.7.4.2. Purge valves 2-EB-91, 92, 99, and 100 opening is monitored by position indication in the control room.

Each purge valve is tested periodically to assure the operability of the accumulator tank. The normal air supply is isolated and operation is provided by the accumulator.

Purge valve elastomers and shaft packing are adjusted and/or replaced based on 10 CFR 50, Appendix J, Type C test results.

[1]Note: ORNZ-4749, VC4-Chemistry, Analytical Chem. Div. Annual Progress Report, September 30, 1971

TABLE 6.6-1 CONTAINMENT POST ACCIDENT HYDROGEN CONTROL SYSTEMS COMPONENT DESCRIPTION

PIR Fan (F-18A/B)

Type	Vaneaxial
Flow (CFM)	7,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	1

PIR Fan Motor

Type	Squirrel Cage Induction
Horsepower rating, (hp)	25
Code	NEMA
Seismic Class	1

Hydrogen Monitoring System

Manufacturer	Whitaker Safety Systems
Model	Containment Atmosphere Monitoring System
Accuracy	+/- 0.2% Hydrogen
Ambient Temperature	40 to 140°F (Analyzer and Remote Cabinets)
Sample Temperature	300°F
Sample Pressure	-5 to 65 psig

Hydrogen Recombiner

Manufacturer	Westinghouse
Quantity	2
Type	Thermal (electric)
Capacity	100 SCFRM
Power Input	75 kW
Housing Material	Stainless Steel 300 series
Inner Structure	Inconel 600
Heater Element Sheath	Incoloy 800
Maximum Sheath Temperature	1550°F
Temperature at Heater Outlet	1150°F to 1400°F
Exhaust Temperature	100°F above ambient

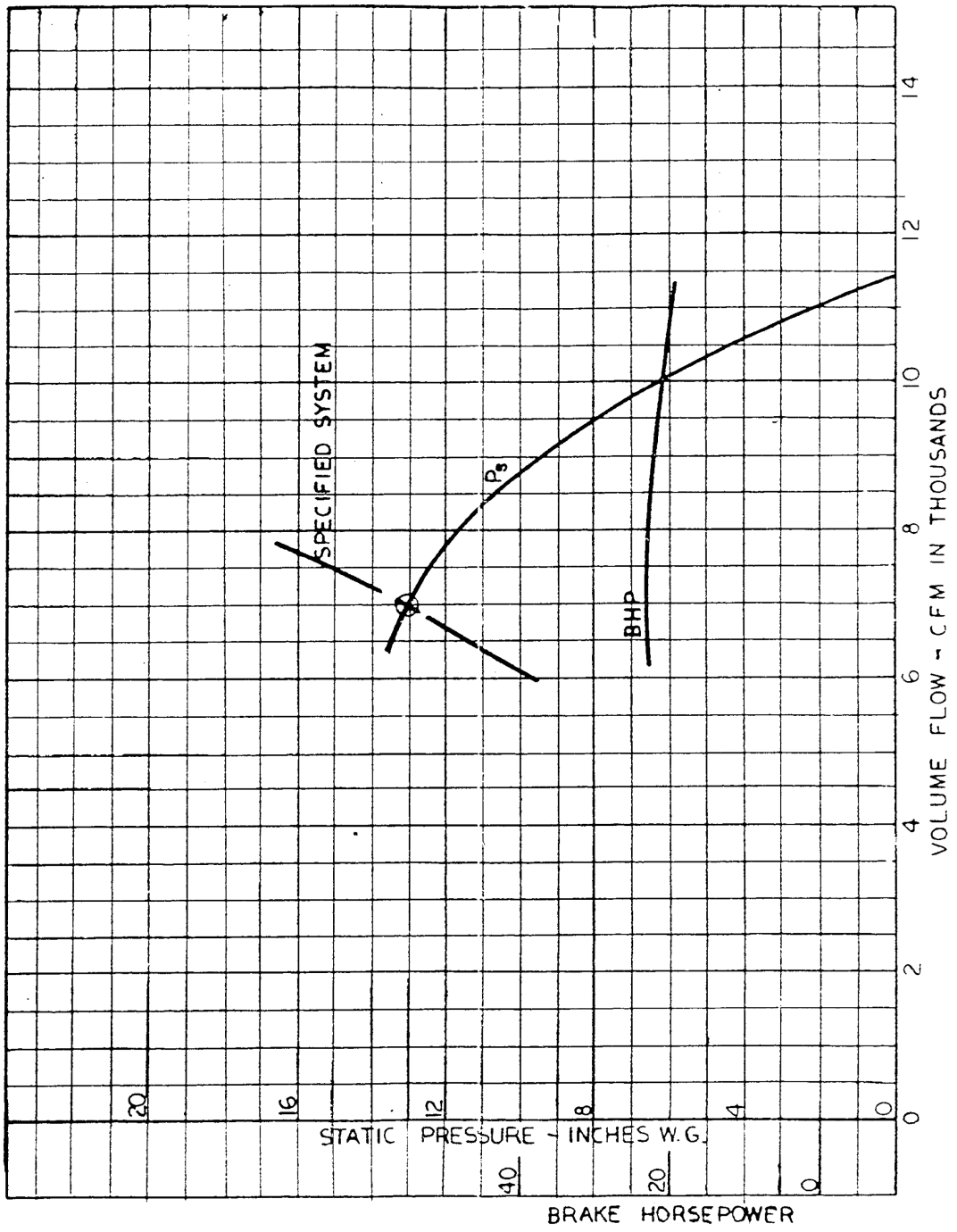
TABLE 6.6-2 DELETED BY FSARCR 04-MP2-018

TABLE 6.6-3 OMITTED

TABLE 6.6-4 POST-INCIDENT RECIRCULATION SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
PIR Fan (2)	Fails to start	Status light	Loss of one subsystem	None	One subsystem taken out of service	Redundant PIR Subsystem is available.
Pipe header	Rupture	None	Partial loss of suction from containment dome region	None	One subsystem does not perform the intended function	Same as above.

FIGURE 6.6–1 POST-INCIDENT RECIRCULATION FAN PERFORMANCE CHARACTERISTICS



6.7 ENCLOSURE BUILDING FILTRATION SYSTEM

6.7.1 DESIGN BASES

6.7.1.1 Functional Requirements

The functions of the enclosure building filtration system (EBFS) are to collect and process potential containment leakage, to minimize environmental activity levels resulting from sources of containment leakage following a loss-of-coolant accident (LOCA). The enclosure building filtration region (EBFR) contains potential containment leakage. Throughline leakage that can bypass the EBFR is discussed in Section 5.3.4. The EBFS is also designed to reduce the concentration of combustible gas built up in the containment following a LOCA in conjunction with the Hydrogen Purge System (Section 6.6). Although not credited in the fuel handling accident and cask drop analyses, EBFS is capable of being automatically or manually aligned to minimize the consequences of those accidents.

6.7.1.2 Design Criteria

The following criteria have been used in the design of the EBFS:

a. The system has two redundant, independent subsystems, each fully capable of the functional requirement.

b. The system has suitable subsystem and component alignments to assure operation of the complete subsystem and with its associated components.

c. Capabilities are provided to assure the system operation with either on-site power (assuming off site power is not available) or with off site electrical power.

d. A single failure of an active component in either subsystem will not affect the functional capability of the other subsystem.

e. The EBFS is designed to permit periodic inspection of important components such as fans, motors, filters, filter frames, ductwork, dampers, piping and valves to assure the integrity and capability of the system.

f. The EBFS is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole. Under conditions as close to the design as practical, the performance is demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

g. The EBFS is designed to the general requirements as described in Section 6.1.

h. The components of the EBFS are designed to operate in the most severe post-accident environment as described in Section 6.1, with the exception of the electric heaters, X-6/A/B.

i. The EBFS is designed to support the Hydrogen Purge System (Section 6.6) in accordance with the conditions given in Safety Guide 7. However, consistent with Regulatory Guide 1.7, Rev. 2, a backup hydrogen purge system is not required to satisfy the safety-related function to control post-accident containment hydrogen. Therefore, the backup hydrogen purge system is no longer credited for hydrogen control.

6.7.2 SYSTEM DESCRIPTION

6.7.2.1 System

The EBFS is shown schematically in Figures 9.9–1 and 9.9–2. The EBFR includes the region between the containment and the enclosure building, the penetrations rooms and the engineered safety feature equipment rooms.

The EBFS is designed to establish and maintain a negative pressure of 0.25 inches w.g. within the EBFR immediately following a LOCA and to reduce airborne radioactive products to the environment by filtration prior to release of air through the Millstone Stack. Neglecting wind and stack effects, each EBFS train has the capability to maintain the EBFR under the minimum negative pressure of 0.25 inches w.g. The required minimum negative pressure can be achieved using both trains or either train of the redundant filtration systems.

The EBFS is located external to the containment in an enclosed area adjacent to the enclosure building. The EBFS exhausts air from all areas of the EBFR. Makeup air is induced into the EBFR by infiltration through building cracks, doors, and penetrations between the EBFR and outside or surrounding structures, as well as from potential containment leakage. The entire system is designed to operate under negative pressure up to the fan discharge. In all cases, the flow rate from this region exceeds the total maximum containment leakage rate.

The in-leakage into the EBFR was originally estimated using analytical and experimental leakage data (Reference 6.7-1). This leakage rate included a conservative containment leakage rate of 0.03 containment volumes per day. The design in-leakage rate was established at a factor of 3.0 greater than the estimated in-leakage rate. The leak tightness of the containment is independent of the operation of the EBFS and is established through Integrated Leak Rate Testing (ILRT) (Section 5.2.8.1).

The original Bechtel building specification for the enclosure building (Reference 6.7-2), required that all metal siding and metal roof decking be designed to withstand the pressure created by a wind load of 140 mph (Section 5.3) at the girt spacing shown on the design drawings, and with a maximum deflection of L/180 of span under load. In addition to the wind pressure, the metal siding and roof decking shall also withstand a differential pressure equivalent to 2 inches water gauge and maintain air tightness for the completed installation. On May 25, 1972, a test

(Reference 6.7-3) was performed based on the Specification (Reference 6.7-2) for a "negative pressure of 2 inches of water plus a wind pressure of 140 miles per hour." The total test pressure of 9.75 inches w.g. was used and the leak-tightness capability was established at this pressure. The calculated maximum negative pressure (Reference 6.7-1) demonstrates that during two parallel fans operation, the EBFR negative pressure is less than the tested limit of 9.75 inches w.g.

The EBFS consists of independent, full redundant fans, filter banks, heating elements, isolation dampers, and ductwork with the exception of common plenums. Charcoal filters installed are demonstrated by tests to have a bypass leakage efficiency above 99 percent, and by a laboratory test to have an iodine removal efficiency above or equal to 95 percent in accordance with Technical Specification requirements.

The prefilters are provided to remove coarse airborne particles to prolong high efficiency particulate air (HEPA) filter life. The HEPA filters are provided to remove fine airborne particles that penetrate the prefilters. The activated coconut shell charcoal filters are impregnated to remove methyl iodine as well as elemental iodine contaminates resulting from a LOCA or a spent fuel handling accident in the spent fuel pool.

Electric heaters (X-61A/B), rated at 25 kW each, are provided to maintain the entering air stream relative humidity (RH) to the charcoal filters below 90 percent. These heaters may not be available throughout a LOCA due to high radiation. However, analyses considering the maximum possible relative humidity within the Enclosure Building Filtration Region during a LOCA, or a REA, or within the Fuel Handling area during a Fuel Handling Accident, determined that the entering EBFS air stream remains below 90 percent RH without the heaters. The electric heaters are energized when the fan is on and deactivated manually prior to initiation of Hydrogen purge.

The EBFS fans are belt driven centrifugal fans capable of operating singly or in parallel with the redundant system. EBFS fan operating parameters are shown in Table 6.7-1. A failure mode analysis for the EBFS is given in Table 6.7-2.

Ductwork for that portion of the EBFS located outside the enclosure building is round and/or rectangular. Longitudinal seams are continuously welded air tight.

The EBFS may be used for containment cleanup during cold shutdown and refueling (Modes 5 and 6). The EBFS can be operated in combination with the containment purge system (Section 9.9.2) for containment cleanup of minor releases of radioactivity. These interconnections between the two systems are shown in Figures 9.9–1 and 9.9–2. This process may be initiated to reduce activity levels prior to purging the containment.

When using the EBFS to purge containment, either train or both trains may be used to perform this activity. The incoming flow path is provided by opening the following dampers: 2-AC-1, 3, 4, & 5 and the exhaust flow path is provided by opening 2-AC-6 and 2-AC-7. Since the purge supply fan, F-23, is not activated, the purge rate is a function of the EBFS train(s) capacity.

The EBFS may also be used to vent (depressurize) the containment under all modes of operation, through the hydrogen purge system, by opening the six inch line valves 2-EB-99/100, or 2-EB-91/92.

The EBFS discharges to the Millstone stack. Containment cleanup is initiated manually by operator action. The connections to the EBFS are automatically closed by an EBFAS to assure system integrity for post-accident operations.

6.7.2.2 Components

The major components with their associated fabrication and performance data are listed in Table 6.7-1.

6.7.3 SYSTEM OPERATION

6.7.3.1 Emergency Conditions

In the unlikely event of a LOCA, the EBFS is automatically initiated by the EBFAS as described in Section 7.3. Air is exhausted from the EBFR, processed through the filter banks and discharged through underground piping to the Millstone 375 foot stack. Redundant on site emergency power is provided by the diesel generators as described in Section 8.3.

System performance is monitored in the control room. Differential pressure across the filter units indicates filter dust loading and replacement requirements. Low flow conditions are alarmed in the control room and local flow indication is provided upstream of each filter unit. A temperature sensor is located within each enclosure filter unit in the vicinity of the charcoal elements to alarm excessive temperature.

The pneumatically operated dampers are normally closed to isolate the filter unit. These dampers open automatically upon the EBFAS and are designed to fail in the open position. The positions of all power operated dampers are indicated in the control room.

The EBFS is designed to reduce the concentration of combustible gas buildup in the containment following a LOCA by controlled purging operations. This operation is described in Section 6.6.3.

Prior to initiation of the hydrogen purge system, the electric heaters in the EBFS filter units are manually taken out of service by tripping electric heater supply breaker. The backup purge operation is described in Section 6.6.2.1.

6.7.4 AVAILABILITY AND RELIABILITY

6.7.4.1 Special Features

The enclosure building structure is designed to retain structural integrity subsequent to a seismic event. However, the EBFS is not designed to be functional subsequent to a safe shutdown

earthquake (SSE). All components are protected from missile damage and pipe whip by physically separating duplicate equipment as described in Section 6.1.

The reliability of the EBFS is assured by providing two independent, with the exception of common ductwork plenums, full capacity subsystems. Each subsystem is capable of maintaining the design negative pressure within the EBFR and of discharging into the 375 foot stack.

Differential pressure is measured from various locations within the EBFR including four locations within the enclosure building proper. The acceptance criteria is that each EBFS train individually establish a minimum negative pressure of greater than or equal to 0.25 inches w.g. in the EBFR within one minute after an enclosure building filtration actuation signal (EBFAS). Neglecting wind and stack effects, each EBFS train has the capability to maintain the EBFR under the minimum negative pressure of 0.25 inches w.g. The acceptance criteria of 0.25 inches w.g. assures that the EBFR will be maintained at a negative or neutral pressure precluding exfiltration under most meteorological conditions. With one EBFS train exhausting the EBFR, the combined effects of certain wind and stack pressures may cause positive pressures in the upper regions of the enclosure building. The potential exfiltration and radiological consequences due to positive pressure in the upper regions of the enclosure building, were analyzed and determined to be fully bounded by the radiological consequences of the low wind speed LOCA case presented in Section 14.8.4.

The EBFR is constructed to limit leakage as described in Section 5.3. Piping, cable tray and ductwork penetrations through the EBFR boundary are sealed with foam or insulation to decrease leakage. All doors into the EBFR are designed to minimize leakage. Containment penetrations into the EBFR are described in Section 5.2.6 and Section 5.3. The metal siding is constructed to limit leakage as described in Section 6.7.2.1.

The EBFS fans F25A/B are fully redundant and are powered from separate emergency sources. The EBFS fans are connected on the suction side by cross-tie ductwork which is provided with a parallel arrangement of electric motor-operated dampers which are likewise powered from separate emergency sources. These dampers are normally closed but can be opened by operator action following a fan failure. The design of the EBFS is such that it renders a loss of cooling air to the filters due to fan failure as incredible. The EBFS requires no additional decay heat removal system. Cooling air is always available, but not necessary (Section 6.7.4.1.1), to prevent iodine desorption and, therefore, ignition of the charcoal elements.

Each subsystem has a nominal design flow of 9,000 CFM. The fan performance curve is shown in Figure 6.7–1.

A failure mode analysis for the EBFS is given in Table 6.7-2. Although there are common plenums, all ductwork is considered a passive component whose failure is not credible.

6.7.4.1.1 Minimum Air Flow Required to Prevent Desorption of Radionuclides

The charcoal filter elements within the EBFS are analyzed to ensure adequate iodine removal capacity, and residual heat removal capabilities following any single failure. The analysis assumed iodine loading is limited to one EBFS unit, and concluded:

- Iodine loading of the enclosure building charcoal filters at 30 days post-accident is 0.267 milligrams of iodine per gram of charcoal, or approximately 11% of Regulatory Guide 1.52 design loading.

- No minimum flow is necessary to maintain the charcoal temperature below 200°F.

However, a temperature sensor is mounted within each filter unit, in the vicinity of the charcoal beds. This sensor provides alarm capability in the control room at 200°F or less.

6.7.4.1.2 Single Failure Evaluation

The containment and enclosure building purge system functions to maintain a suitable environment in the containment building (during Modes 5 and 6) or the enclosure building during any mode of operation (see Section 9.9.2).

As noted in Table 9.9-2 of the FSAR, the containment and enclosure building purge system is a nonsafety-related system. Purge system isolation dampers 2-AC-1 and 2-AC-11, including control circuits, are safety related since they receive a containment isolation actuation signal (CIAS) to close to isolate the enclosure building in the event of a LOCA. For the same reason, the controls for purge fan F23 are also safety related.

Simultaneous occurrence of a LOCA and a seismic event is not a design basis for Millstone Unit 2. Although the enclosure building structure is designed to retain structural integrity subsequent to a seismic event, the EBFS is not designed to be functional subsequent to an SSE. Maintaining a negative pressure in the enclosure building after an SSE is not assured mainly because the sheet metal siding may not remain intact. The original design for the purge supply and exhaust ductwork is nonsafety related and nonseismic.

FSAR Question 6.15.4 addressed single failure (failure to close on CIAS) for dampers 2-AC-1 (purge supply) and 2-AC-11 (purge exhaust). The response to Question 6.15.4 documented the basis for concluding that with an assumed single failure of 2-AC-1 to close, the minimum negative pressure of 0.25 inches w.g. can be maintained within the EBFR. The response noted that the evaluation was conservative for a postulated single failure of the exhaust damper 2-AC-11.

Since the installation of damper 2-AC-130 eliminates the 2-AC-1 damper single failure, there is assurance that the EBFS can maintain an adequate negative pressure within the EBFR as described in Section 6.7.2.

The containment and enclosure building purge inlet was modified with the installation of a counterbalanced gravity damper (2-AC-130) to provide redundancy for isolation damper 2-AC-1.

This design change was a system upgrade to mitigate a postulated single failure of the Facility 1 CIAS signal to 2-AC-1.

The Main Exhaust System (MES) fans will trip on a CIAS actuation signal as described in Section 9.9.9.4.1, to mitigate the consequences of the 2-AC-11 single failure vulnerability. If the postulated single failure of 2-AC-11 occurred the MES fans will automatically trip. The enclosure building integrity is maintained by the design of the enclosure building as described in Section 6.7.2.1.

FSAR Question 6.17 addressed single failure (failure to open on CIAS) of enclosure building purge isolation dampers 2-AC-3 and 2-AC-8. The response stated that the assumed single failure of either damper was acceptable.

A design basis review of enclosure building isolation dampers 2-AC-1 and 2-AC-11 is documented in Millstone Unit 2 Nuclear Engineering Design and Program Services Department records.

6.7.4.2 Tests and Inspections

Individual components of the EBFS are tested to assure performance.

Prefilters are of the throwaway type to be replaced as necessary.

The HEPA filters characteristics are in accordance with MIL-STD-282 standard.

Each HEPA filter bank is tested, in place, periodically. The testing media will be Dioctyl-phthalate (DOP).

Charcoal filters are initially shop performance-tested and methyl iodide tracers for efficiency and Freon for leakage. After installation the charcoal filter banks were tested in place with Freon to ensure that there is no leakage across the filter bank and that the charcoal elements are not damaged. Each charcoal filter housing is provided with test canisters which contain a sample specimen of the charcoal used in the filter elements. These test canisters are analyzed periodically by an independent laboratory to determine remaining charcoal filter life and replacement requirements.

Each fan and motor is tested as a unit to assure characteristic performance curves. Fan ratings are in accordance with AMCA Standard Test Code 211-A.

The EBFS ductwork is leak tested and balanced in accordance with SMACNA Standards.

Provisions are incorporated to test the entire system for performance during normal operation. Each EBFS fan is tested simultaneously with the associated power operated valves and instrumentation, but independently from the redundant subsystem.

The EBFAS is initiated to start the fan and open the filter unit isolation dampers. The fan is tested at some point on the fan performance curve other than shutoff. Fan flow is verified by measuring the pressure differential across the filter elements. Opening of the power-operated dampers is monitored by the damper position indication in the control room. Since the containment purge ducts are vented back into the enclosure building by fail open dampers, additional testing is not necessary.

The EBFS is tested periodically to assure the negative pressure is maintained within the EBFR. EBFR negative pressure is monitored by differential pressure indicators located throughout the EBFR.

The EBFS undergoes a preoperational test prior to startup. The test procedure is described in Section 13.

The system equipment is fully accessible during all normal operation for maintenance and performance testing, including replacement of filter elements. The equipment is accessible for inspection and maintenance on components outside of the air stream during accident conditions.

6.7.5 REFERENCES

6.7-1 “Conventional Buildings for Reactor Containment,” NAA-SR-10100, dated May 1965, issued by Atomics International, a Division of North American Aviation Incorporated.

6.7-2 Bechtel Specification Number 7604-A-16A, Section 12.0, “Design Criteria.”

6.7-3 Pittsburgh Testing Laboratory witnessed an “Air Infiltration Test,” on Siding and Roof Deck Mock-up, of the enclosure building, for Elwin G. Smith Division, Cyclops Corporation.

TABLE 6.7-1 ENCLOSURE BUILDING FILTRATION SYSTEM COMPONENT DESCRIPTION

Fans F-25A & F-25B	
Type	Belt driven
Capacity (nominal each)	9,000 cfm
Standard	AMCA-211-A
Seismic Class	1
Motors	
Horsepower rating, hp	25
Code	Nema, MG-1
Seismic Class	1
Electric Heaters X-61A & X-61B	
Power rating, kW	25
Code	UL approved
Seismic Class	1
Particulate Filters (Non-QA)	
Quantity per unit	9
Type (prefilter)	Throwaway
HEPA Filters	
Quantity per unit	9
Type	High capacity
Rated air flow per filter	
at 1.30 inches wg	1,500 cfm
at 0.87 inches wg	1,000 cfm
Standard	MIL-STD-282

TABLE 6.7-1 ENCLOSURE BUILDING FILTRATION SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Charcoal Filters	
Quantity per unit (Trays)	27
Charcoal type	CNN-816 Activated coconut shell
Rated nominal air flow per unit	9,000 cfm
Standard	ANSI N-509
Ductwork	
Material/Type	IAW Specification M-506
Standard	SMACNA
Seismic Class	1
Piping, valves and fittings	
A. Suction	
Pipe sizes	Wall thickness
2.5 to 10 inches	SCH40
12 to 48 inches	0.375 inches wall
Material	Seamless ASTM A-53B (containment)
	Seamless ASTM A-333 (penetration)
Design pressure (psi)	60
Design temperature (°F)	289
Code	ANSI B-31.1.0 (containment)
	ANSI B-31.7 Class II (penetration)
Seismic	Class I
Construction	
Piping	2.5 inches and larger: butt-welded except at flanged equipment
Valves	2.5 inches and larger: butt-welded (except Butterfly Valves) 150 lb ANSI rating carbon steel
B. Discharge	
Pipe sizes	Wall thickness

TABLE 6.7-1 ENCLOSURE BUILDING FILTRATION SYSTEM COMPONENT DESCRIPTION (CONTINUED)

2 inches and smaller	Schedule 80
2.5 inches to 10 inches	Schedule 40
12 inches and larger	0.274 wall
Material	Seamless ASTM A-53A or B
Design Pressure (psi)	50
Design Temperature (°F)	120
Standard	ANSI B-31.1.0
Seismic	Class I
Construction	
Piping	2.5 inches and larger: butt-welded except at flanged equipment
	2 inches and smaller: 300 lb M.I. Screwed
Valves	2 inches and smaller: 125 lb WSP, screwed bronze
Ductwork	
Seismic	Class 1

TABLE 6.7-2 ENCLOSURE BUILDING FILTRATION SYSTEM FAILURE MODE ANALYSIS

Component Identification & Quantity	Failure Mode	Method of Detection & Monitor	Detrimental Effect On System	Corrective Action	Resultant System Status	Remarks
EBFS/AES Filtration Fan F-25A/25B	Fails to operate	Status lights on C01 Low flow alarm on RC 22	Loss of one subsystem	Monitor charcoal heatup and utilize opposite train	Redundant system operable	Both trains are automatically started upon EBFAS/AEAS signal. Each unit has 100% capacity. Utilize cross connect line for charcoal temperature control if needed.
Electric Heater X-61A/61B	Fails to operate	None See remarks*	None See remarks**	None	See remarks **	Both trains are automatically started upon EBFAS/AEAS signal. Each unit has 100% capacity. * There is a local status light at the unit, which may be off limits to personnel during LOCA or a Spent Fuel Pool Accident. ** Analyses determined that the entering air stream remains below 90% R.H. without heaters.

TABLE 6.7-2 ENCLOSURE BUILDING FILTRATION SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

Component Identification & Quantity	Failure Mode	Method of Detection & Monitor	Detrimental Effect On System	Corrective Action	Resultant System Status	Remarks
Filter Unit L-29A/ 29B	N/A See remarks	Charcoal beds temperature computer alarm points: AD8772 & AD8776	Possible charcoal bed heating	Opposite train available. Utilize cross-country line to maintain charcoal temperature if required	Redundant system operable	The filter unit itself and its components, HEPA filters, charcoal filters are passive components. However, if the air flow over the charcoal beds is cut off (i.e., fan fails), charcoal beds could heat up due to decaying iodine.
EBFS Plenum FD 2-HV-37A/37B	N/A See remarks	N/A	N/A	N/A	N/A	Fire damper is deemed a passive component.
EBFS Isolation Damper to Filter Unit 2-EB-40/50	Fails as is See remark	Status lights on C01	None	None	Redundant system operable	Open on EBFAS. Close on AEAS if EBFSAS not present.
EBFS/AES Common Isolation Damper to Filter Unit 2-EB-41/ 51	FO Fails open See remarks	Status lights on C01	None	None	Redundant system operable	Open on EBFAS or on AEAS.
Filter Fan F-25A/ 25B Isolation Discharge Damper 2-EB-42/52	FO Fails open See remarks	Status lights on C01	None	None	Redundant system operable	Open on EBFAS or on AEAS.

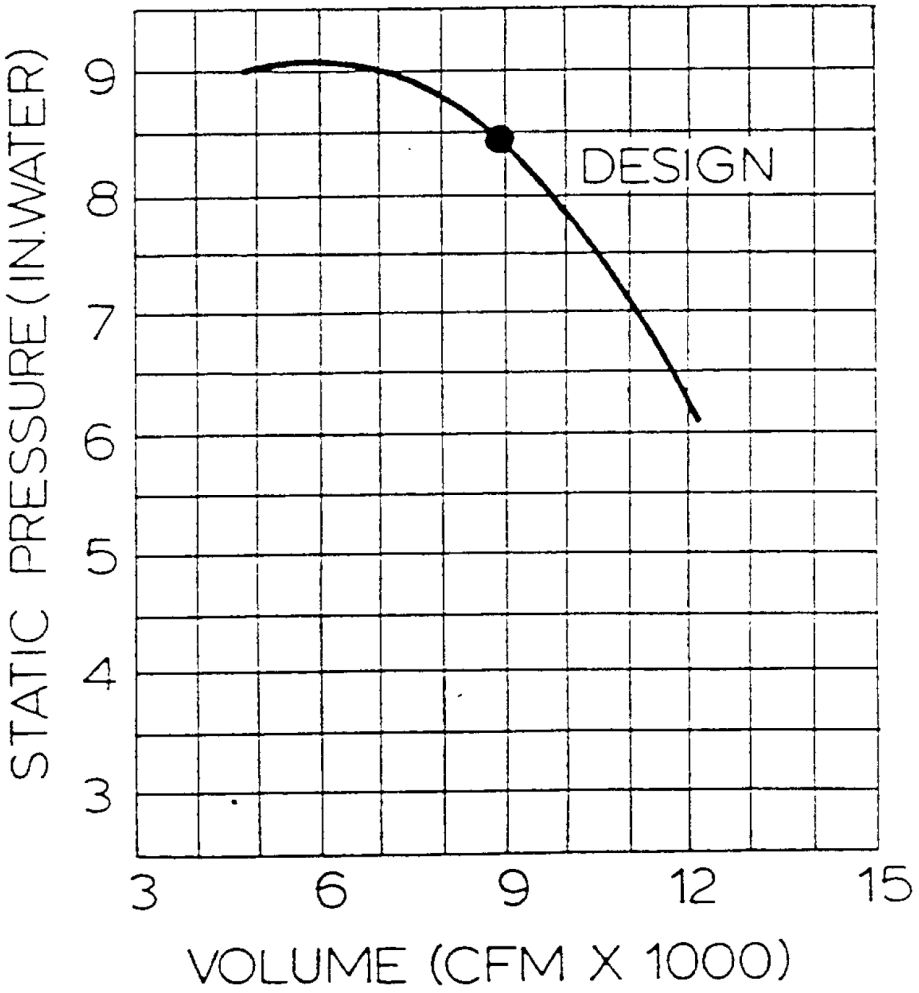
TABLE 6.7-2 ENCLOSURE BUILDING FILTRATION SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

Component Identification & Quantity	Failure Mode	Method of Detection & Monitor	Detrimental Effect On System	Corrective Action	Resultant System Status	Remarks
Filter Fan F-25A/ 25B Discharge Damper (Backdraft/ weighted) 2-EB-43/ 54	Fails as is See remarks	None	None	None	Redundant system operable	This damper function is to prevent gross recirculation in the idle train if damper 2-EB-42/52 fails open.
AES Isolation Damper to Filter Unit 2-EB-60/61	Fails closed NC Normally closed See remarks	Status lights on C01	None	None	Redundant system operable	Open on AEAS if EBFAS is not present.
Charcoal Beds Bypass Cooling Line Damper 2-EB-76/77	Fails as is NC Normally Closed See remarks	Status lights on C01	None	None	Redundant Cooling Path Available	If the air flow over the charcoal beds is cut off (i.e. fan fails), charcoal beds could heat up due to decaying iodine.
Purge Plenum Fire Damper 2-EB-131	N/A See remarks	N/A	N/A	N/A	N/A	Fire damper is deemed a passive component.
Hydrogen 6 inch Purge Line from Containment Penetration number 82 to EBFS Plenum Valve 2-EB-193	N/A See remarks	None	See remarks	See remarks	See remarks	Hydrogen Purge mode is manually controlled. These valves are normally open. These valves are passive components.

TABLE 6.7-2 ENCLOSURE BUILDING FILTRATION SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

Component Identification & Quantity	Failure Mode	Method of Detection & Monitor	Detrimental Effect On System	Corrective Action	Resultant System Status	Remarks
Hydrogen 6 inch Purge Line from Containment Penetration number 83 to EBFS Plenum Valve 2-EB-194						
All EBFS/AES vital ductwork	N/A See remarks	N/A	N/A	N/A	N/A	Ductwork is deemed a passive component.

FIGURE 6.7–1 ENCLOSURE BUILDING FILTRATION SYSTEM FAN PERFORMANCE CURVE



Millstone Power Station Unit 2
Safety Analysis Report

Chapter 7: Instrumentation and Control

CHAPTER 7—INSTRUMENTATION AND CONTROL

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CHAPTER 7 - INSTRUMENTATION AND CONTROL

7.1 INTRODUCTION

The plant systems are instrumented to provide information on plant conditions at selected locations, to protect equipment and personnel from undesirable conditions and to control the plant during startup, operation, and shutdown. The principal control station for the plant is in the control room located in the reactor auxiliary building.

The plant is started up and shut down under remote manual control. Annunciators, indicators and recording devices will alert the operator and provide data on plant conditions.

Instrumentation and controls essential to plant safety are located in the control room. The instrumentation is arranged in groups on the control boards so that when corrective action is required, all pertinent indicators, recorders and controllers are within easy reach and view of the operator. The control board is a duplex benchboard. Visible and audible alarms located in the top section of the main control board annunciate and identify abnormal operating conditions. Telephone systems provide both in-plant and external communication. The control room and computer room are kept at a controlled temperature which is well within the design temperature requirements of the instruments.

To ensure reliability, components of established quality are selected and used in the instrumentation and control equipment. The protection instrumentation consists of four independent channels to permit system testing without reducing the degree of protection provided. Reliable sources of electrical power are provided to ensure safe and reliable plant operation (see Chapter 8).

The operation of the reactor within established limits is achieved by its inherent characteristics, instrumentation and control systems, by operational procedures and administrative controls. Departures from these limits are audibly and visibly annunciated in the control room. A reactor protective system is designed to initiate reactor trips to protect the core and the reactor coolant system pressure boundary.

The engineered safety features actuation system instrumentation provides the equipment necessary to initiate the required safety features functions. This system also monitors the power sources acting to assure the availability of emergency power for operation of at least the minimum engineered safety features (see Chapters 6 and 8). This system is provided with the necessary redundant circuitry and physical isolation so that a single failure within the system would not prevent the proper system action when required. This system is provided with test facilities and alarms to alert the operator when certain components trip or malfunction, or are inoperable. The controls are designed to automatically provide the sequence of operations required to initiate engineered safety features system operation with or without off-site power available.

Those instruments which have been classified Seismic Class I have been type tested or validated by computation to ensure operability.

The instrumentation, protection and control systems designed and built by C. E. for the Millstone Unit Number 2 are functionally identical to corresponding C. E. systems provided for Calvert Cliffs (NRC Docket Numbers 50-317 and 50-318 through Amendment 19) except for the modifications mentioned below.

A reactor coolant pump underspeed trip has been added (Section 7.2.3.3.1). This trip is not credited to provide protection during any accident analysis event.

Automatic axial flux tilt protection has been added through the addition of a Local Power Density Trip (Section 7.2.3.3.10) and modification of the Thermal Margin/Low Pressure Trip (Section 7.2.3.3.7).

The High Power Trip has been modified to use a variable setpoint (Section 7.2.3.3.2).

The trip logic matrix relay power supplies have been split so that each matrix power supply supplies only two relays, thus preventing a single power supply failure from causing a spurious reactor trip (Figure 7.2–2).

The range of the T_{cold} channel inputs to the RPS has been changed in support of the modification to the Thermal Margin/Low Pressure Trip (Section 7.2.3.3.7) from 515°F to 615°F to 465°F to 615°F. This was to assure the existence of a valid ΔT power signal during transients involving significant decreases in core inlet temperature.

The function of the Zero Power Mode Bypass (Section 7.2.3.3.12) has been expanded to include removing the ΔT power component of the power signal Q in order to allow for special physics startups when system temperatures are below the operation range. In addition, the Zero Power Mode deletes the RCP underspeed trip.

The Low Reactor Coolant Flow Trip (Section 7.2.3.3.3) acts on a ΔP signal.

The Power Trip Test Interlock (Section 7.2.4) has been expanded to provide protection against the Nuclear Instrumentation Test Select switch being left in an off-normal position after testing.

The Control Element Drive System (Section 7.4.2) has been modified to include a CEA Motion Inhibit.

The Reactor Regulating System (Section 7.4.1) has been modified to delete the automatic mode of CEA movement.

7.1.1 IDENTIFICATION OF SAFETY RELATED EQUIPMENT

7.1.1.1 Protective Systems

Protective Systems encompass electrical and mechanical devices and circuitry from sensors through actuation devices.

Reactor Protective System (Section 7.2)

Engineered Safety Features Actuation System (Section 7.3)

Auxiliary Feedwater Automatic Initiation System (Section 7.3)

Protective System Power Supply (Section 8)

7.1.1.2 Safe Shutdown Systems

The Reactor Protective System will safely bring the Reactor to a hot standby condition if any of the input parameters deviates from its preselected operating range or upon manual initiation by the operator. The equipment normally used to maintain the plant in hot shutdown is listed in Table 7.6-1. Additional equipment normally used to bring the plant to cold shutdown conditions is listed in Table 7.6-2.

7.1.1.3 Safety-Related Display Instrumentation

Reactor Protective System input parameter indication. (Section 7.2, 7.5)

Engineered Safety Features Actuation System input parameter indication. (Section 7.3, 7.5.1)

Engineered Safety Features monitoring instrumentation. (Section 6)

7.1.1.4 Other Safety-Related Systems

Shutdown Cooling Interlocks (Section 9.3)

Refueling Interlocks (Section 9.8)

Auxiliary Steam Detection and Isolation System (Sections 7.10 and 9.13)

7.1.1.5 Control Systems

The systems listed below are not required for safety. Protective system action ensures that design limits are not exceeded.

Reactor Coolant Pressure Regulating Systems (Section 7.4.3)

Pressurizer Level Regulating Systems (Section 7.4.4)

Reactor Regulating Systems (Section 7.4.1)

7.1.2 IDENTIFICATION OF SAFETY CRITERIA

The design bases, criteria, safety guides, information guides, standards, and other documents implemented in the design of systems listed in Section 7.1.1 are stated in the corresponding descriptive sections of the FSAR.

The following guides referenced in Section 7.0 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." February 1972 were not available at the time of issue of the Millstone Unit Number 2 construction permit (December 11, 1970) and were therefore not included in the design or installation criteria used during construction. However, these guides are referenced in applicable description sections of the FSAR where the equipment provided meets the general guide requirements.

IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Systems," dated June 3, 1971.

IEEE Standard 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," dated September 16, 1971.

IEEE Standard 323-1971, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," dated April 1971.

IEEE Standard 336-1971, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations," dated September 16, 1971.

IEEE 338-1971, "Periodic Testing of Nuclear Power Generating Station Protection Systems," dated September 16, 1971.

IEEE 344-1971, "Seismic Qualifications of Class I Electric Equipment for Nuclear Power Generating Stations," dated September 16, 1971.

Safety Guide 22, "Periodic Testing of Protection System Actuation Functions," dated February 17, 1972.

A description of the quality assurance procedures to be used during equipment fabrication, shipment, field storage, field installation, system and component checkout, and the records pertaining to each of these is contained in Sections 1.7, Appendix 1.B (located in the original FSAR dated August 1972), and 13.

It should be noted that the Local Power Density Trip is not a new trip but rather a modified version of the Axial Flux Offset trip described in the Calvert Cliffs FSAR. The name has been changed relative to the Calvert Cliffs FSAR in order to more accurately describe its function. The

modifications made to the Thermal Margin, Axial Flux Offset and High Power trips described in the Calvert Cliffs FSAR can be categorized as follows:

1. Providing additional input to the trips, and

2. Providing additional processing equipment for the trips.

The Modifications were made to increase the operating flexibility over that provided by the equivalent trips described in the Calvert Cliffs FSAR.

Table 7.2.1.1–I shows the NSSS parameters that affect the acceptable fuel design limits mentioned above. Table 7.2.1.1–II shows what variables are monitored on both Calvert Cliffs and Millstone Unit 2, to determine the state of the NSSS parameters that affect the fuel design limits. It should be noted that the radial peaking factor is inferred from the core power measurement. This indirect measurement is accomplished through the Power Dependent CEA Insertion Limits (PDILS) given in the Technical Specifications. Also, the azimuthal tilt magnitude is not directly measured. The protective system set points are developed to: 1) take into account the worst core radial peaking factor that can result at any power level from CEAs full out up to CEAs inserted to their insertion limit, and 2) take into account an azimuthal tilt at its Technical Specification limit.

Table 7.2.1.1–III gives a comparison of the specific function of the subject trips, on Calvert Cliffs and Millstone Unit 2, with respect to the fuel design limits. The low flow trip is included in Table 7.2.1.1–III for completeness purposes only. Table 7.2.1.1–IV gives a comparison of the variables monitored, on Calvert Cliffs and Millstone Unit 2, for the trips listed in Table 7.2.1.1–III.

Thermal Margin Trip Modification:

The function of the Thermal Margin trip is the same on Millstone Unit 2 and Calvert Cliffs (see Table 7.2.1.1–III). The modification to the thermal margin trip consists of adding the axial flux offset as a measured input and the processing equipment needed to relate axial offset to thermal margin limits (see Table 7.2.1.1–IV). On Calvert Cliffs the combination of thermal margin and axial flux offset trip provides DNB and void fraction protection. This is accomplished by using the axial offset trip to limit the axial power distribution that can occur at any power level. This limiting distribution is then related to the thermal margin limit at that power level when generating the set points, since the thermal margin trip on Calvert Cliffs does not monitor the axial offset. The approach used on Calvert Cliffs does not allow credit to be taken for operation with axial offsets that are more favorable in terms of thermal margin than the offset that exists at the trip limit. Direct measurement of the axial offset in the thermal margin trip in Millstone Unit 2 allows this credit to be taken when it exists; thereby, 1) increasing margin to trip and improving operating flexibility, where 2) providing the same degree of protection as on Calvert Cliffs.

Axial Flux Offset Trip Modification (i.e., Local Power Density Trip):

The function of the axial flux offset trip on Calvert Cliffs is twofold:

1. assure that the fuel temperature limit (i.e, kw/ft limit) is not violated (See Table 7.2.1.1–III, and

2. limit the axial power distribution that can exist at any power level to that which was assumed in generating the thermal margin trip set points, as explained above.

The modifications made to the axial offset trip in Calvert Cliffs are:

1. a power dependent radial peaking penalty function has been included, and

2. the auctioneered higher of the ΔT and neutron flux power is used as the power input to generate the trip limits.

As shown in Table 7.2.1.1–I the radial peaking factor is one parameter that offsets the kw/ft of which the plant is operating. Table 7.2.1.1–II shows that this effect is inferred from the sensed power level. In Calvert Cliffs the axial offset trip setpoints are determined in a manner that is consistent with the worst core CEA insertion (radial peaking) that can occur at any power level. The trip functions to limit the axial peaking factor consistent with the assumed worst radial and measured power level such that the specified kw/ft limit is not exceeded. By generating trip setpoints in this manner, any change in CEA insertion limits during operation requires setpoints to be regenerated. The modification provided on Millstone Unit 2 is done to facilitate the regeneration of setpoints in the event that CEA insertion limits are modified.

In generating trip system setpoints, one must consider all measurement errors and uncertainties that can occur during anticipated operational occurrences. On Calvert Cliffs, the neutron flux power is used as the power input to the axial offset trip. One factor that is considered in generating trip setpoints is how much can this signal be distorted (i.e., uncertain) during transients that require this trip. Two major phenomena that result in neutron flux power measurement error are:

1. CEA shadowing: This effect results from the distortion of the radial power distribution. The out-of-core detectors “see” the fast neutron flux escaping from the peripheral fuel bundles. As CEAs are inserted or removed the power produced in the peripheral fuel bundles will vary relative to the core average power. The result is that the out-of-core detectors may indicate a power level different from the core average power.

2. Inlet Temperature Shadowing: This effect results from a change in density of the cold leg coolant which passes between the peripheral fuel bundles and the out-of-core detectors. The result is that as cold leg temperature decreases, from that value at which the out-of-core detectors had last been calibrated, the detectors will indicate a power level below the core average power. This is due to the fact that the incident neutron flux at the detectors will be decreased due to the increased coolant density.

The ΔT power, obtained from a calculation of hot and cold RTD measurements, is not influenced by these two phenomena. The use of the auctioneered higher of the ΔT power and neutron flux

power provides a means of decreasing the uncertainty imposed on the core power measurement when generating the local power density trip setpoints.

The modifications made to the Calvert Cliffs axial flux offset trip on Millstone Unit 2 (local power density trip) provide 1) increased operating margin and increased system flexibility, while 2) still satisfying the same criteria (i.e., kw/ft protection) and providing the same degree of protection.

High Power Level Trip Modification:

The function of the high power trip on Calvert Cliffs is to provide 1) a maximum upper load limit on the NSSS, and 2) a reactor trip to assist the engineered safety system in the event of an ejected CEA accident. The function of the high power trip on Millstone Unit 2 is to provide:

1. an upper load limit on the NSS that is always a given percent of rated power above the steady-state operating power, and

2. assistance during an ejected CEA accident.

In the Calvert Cliffs design, the combination of the thermal margin and axial flux offset trips assure the integrity of the fuel design limits. If during a power excursion an axial flux offset or thermal margin trip does not occur, in which case neither would be required, a high power trip eventually results if the indicated power reaches 106.5% of rated.

As mentioned previously, the thermal margin and axial offset trip setpoints are generated by assuming that worst case radial peaks can occur at various power levels. This analysis must take into consideration the possibility of “carrying-up” high radial peaks to high power levels. This effect may occur when CEAs are in manual control and a power excursion ensues and must be accommodated up to the point of the high power trip at 106.5% or rated power. The high power trip then essentially provides a mechanism by which to limit the radial peaking that must be assumed in generating the thermal margin and axial offset trip setpoints. The modification to the high power trip on Millstone Unit 2 provides a means of limiting any power excursion to approximately 10% increase above the initial power level. The trip then provides a means of limiting the radial peaking that can be “carried-up” during a power excursion starting below 100% power. When the initial power level is 100% of rated, the 110% high power trip setpoint provides this assurance. Since the maximum power excursion is always limited by a set amount, credit can be taken for the reduced radial peak that may be “carried-up.” This credit is reflected in the generation of the thermal margin and local power density trip setpoints and results in increased operating margins. The modification of using the auctioneered high of the ΔT and neutron flux power as the trip variable for the high power trip is provided to increase system accuracy as explained previously.

The design criteria for the thermal margin, local power density and high power level trip are discussed in Section 7.2.1.2.

7.2 REACTOR PROTECTIVE SYSTEM

7.2.1 DESIGN BASIS

7.2.1.1 Functional Requirements

The reactor protective system (RPS) consists of the sensors, amplifiers, logic, and other equipment necessary to monitor selected nuclear steam supply system (NSSS) conditions and to effect reliable and rapid reactor shutdown if any one or a combination of conditions deviates from a preselected operating range. The system functions to protect the core and reactor coolant pressure boundary (RCPB). The Millstone Unit 2 Protective System is functionally identical to that provided for Calvert Cliffs (NRC Docket Numbers 50-317, 50-318 through Amendment 19), except for the following modifications:

A reactor coolant pump (RCP) underspeed trip has been added (Section 7.2.3.3.1). This trip is not credited to provide protection during any accident analysis event.

Automatic axial flux tilt protection has been added through the addition of a Local Power Density (LPD) Trip (Section 7.2.3.3.10) and modification of the Thermal Margin/Low Pressure (TM/LP) Trip (Section 7.2.3.3.7).

The High Power Trip has been modified to use a variable setpoint (Section 7.2.3.3.2).

The trip logic matrix relay power supplies have been split so that each matrix power supply supplies only two relays, thus preventing a single power supply failure from causing a spurious reactor trip (Figure 7.2–2).

The range of the T_{cold} channel inputs to the RPS has been changed in support of the modification to the TM/LP Trip (Section 7.2.3.3.7) from 515°F to 615°F to 465°F to 615°F. This was to assure the existence of a valid ΔT power signal during transients involving significant decreases in core inlet temperature.

The function of the Zero Power Mode Bypass (Section 7.2.3.3.12) has been expanded to include removing the ΔT power component of the power signal Q in order to allow for special physics startups when system temperatures are below the operating range.

The Low Reactor Coolant Flow Trip (Section 7.2.3.3.3 acts on a ΔP signal.

Signal Commons between Nuclear Instrumentation Channels have been isolated to prevent channel interaction. (Section 7.2.3.2.1)

The Power Trip Test Interlock (Section 7.2.4) has been expanded to provide protection against the Nuclear Instrumentation Test Select switch being left in an off-normal position after testing.

The function of the RPS on all Combustion Engineering (CE) supplied plants has always been threefold:

1. To prevent overpressurization of the primary coolant systems,

2. To assist the engineered safety system in mitigating the consequences of accidents (e.g., loss-of-coolant accident (LOCA)), and

3. To assure that acceptable fuel design limits are not exceeded during anticipated operational occurrences.

The RPS supplied with Millstone Unit 2 is, in this sense, functionally identical with that supplied on Calvert Cliffs.

The assurance that the protective system fulfills its threefold function is provided by choosing a complement of trips that monitor pertinent parameters that: 1) are related to specified limits, and 2) that are affected during accidents which may lead to a violation of these specified limits.

The particular trips under discussion (i.e., Thermal Margin, LPD and High Power) have as their prime function the assurance that acceptable fuel design limits are not exceeded during anticipated operational occurrences. Anticipated operational occurrences are defined in General Design Criteria (GDC), Appendix A of 10 CFR 50 as "...those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit...".

Acceptable fuel design limits are described in Section 3.2.3.

7.2.1.2 Design Criteria

The RPS is designed to the following bases to assure adequate protection for the core:

a. Instrumentation conforms to the provisions of the Institute of Electrical and Electronic Engineers (IEEE), Criteria for Nuclear Power Plant Protection Systems (IEEE 279-1971).

b. No single component failure can prevent safety action.

c. Four independent measurement channels are provided for each parameter that can initiate safety action.

d. Channel independence is assured by separate connection of the sensors to the process systems and of the channels to vital instrument buses.

e. The four measurement channels provide trip signals to six independent logic matrices, arranged to effect a two-out-of-four coincidence logic having outputs to four independent trip paths.

f. A trip signal from any two-out-of-four protective channels on the same parameter causes a reactor trip.

g. When one of the four channels is taken out of service, the protective system logic may be changed to two-out-of-three coincidence for a reactor trip by bypassing the out of service channel.

h. The protective system AC power is supplied from four separate vital instrument buses.

i. Open circuiting, or loss of power supply for the channel logic, initiates an alarm and a channel trip.

j. The trip logic matrices assume the nonconducting state to provide a tripping function.

k. The RPS can be tested with the reactor in operation or shut down.

l. The manual trip system is independent of the automatic trip system.

m. Trip signals are preceded by pretrip alarms to alert the operator of undesirable operating conditions in cases where operator action can correct the abnormal condition and avoid a reactor trip.

n. The RPS components are independent of control systems.

o. All equipment, including panels, and cables associated with the RPS, are marked with colored markers or nameplates in order to facilitate identification.

p. Electrical circuit isolation is provided between the RPS, annunciators, and the plant computer.

q. There are no RPS instrumentation transmitters for which the trip set points are within 5 percent of the high or low end of the calibrated range, or within 5 percent of the overall instrument design range.

7.2.2 DISCUSSION

The RPS meets the general requirements of the applicable sections of the below listed guides although they were not available at this time the construction permit for Millstone Unit Number 2 was issued. (December 11, 1970).

IEEE Standard 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations,” dated June 3, 1971.

IEEE Standard 336-1971, “Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment during the construction of Nuclear Power Generating Stations,” dated September 1971.

IEEE Standard 338-1971, “Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems,” dated September 16, 1971.

Regulatory Guide 1.22, “Periodic Testing of Protection System Actuation Functions,” (Safety Guide 22) dated February 17, 1972.

Combustion Engineering Topical Report CENPD-11 “Reactor Protection System Diversity,” W. C. Coppersmith, C. I. Kling, A. T. Shesler, and B. M. Tashjian CENPD, February 1971) demonstrates that functional diversity has been incorporated in the protective system design.

7.2.3 SYSTEM DESCRIPTION

7.2.3.1 General

As shown in Figures 7.2–1 and 7.2–2, the RPS consists of four trip paths operating through the coincidence logic matrices to maintain power to, or remove it from, the control element drive mechanisms (CEDM). Four independent measurement channels normally monitor each plant parameter which can initiate a reactor trip. Individual channel trips occur when the measurement reaches a preselected value. The channel trips are combined in six two-out-of-two logic matrices. Each two-out-of-two logic matrix provides trip signals to four one-out-of-six logic units, each of which causes a trip of the reactor trip switchgear in the AC supply to the CEDM power supplies. Each CEDM power supply source is separated into two branches.

Reactor trip is accomplished by deenergizing the CEDM coils allowing the shutdown and regulating CEAs to drop into the core by gravity. Reactor trip is initiated by the conditions described in Section 7.2.3.3; the reactor trip and pretrip alarm set points are listed in Table 7.2-1.

All protective systems that actuate reactor trip and engineered safety features (ESF) components conform to the regulation in effect at the time of procurement including relevant sections of the GDC, Appendix A to 10 CFR 50.

The cabinets of the RPS are appropriately tagged A, B, C, D to distinguish between channels. The RPS is distinguished from nonsafety related equipment by the use of colored nameplates. At termination points the incoming and outgoing cables of the RPS are appropriately tagged to identify the channel and to distinguish between channels.

Physical separation between channels is accomplished by feeding each of the four independent signal inputs into a separate cabinet. The four cabinets are separated from each other by fireproof barriers. Logic matrix and other interconnections between the four cabinets are made by running interconnecting wiring through rigid metallic conduits penetrating the barrier between each of the cabinets. All barrier penetrations are sealed with fireproof material.

The ability of the RPS to protect the core and the RCPB has been established by the analysis discussed in Chapter 14. The analyses show that the plant and fuel design limits are not exceeded during anticipated operational occurrences such as loss of load or inadvertent withdrawal of regulating control element assembly (CEA) groups. The provision of more restrictive set points to

permit operation at less than full flow is described in Section 7.2.3.3.3 and where appropriate, such operation is considered in the analyses. No part of the RPS is dependent on plant instrument air systems.

Water is not used for cooling any part of the RPS.

7.2.3.2 System Components

7.2.3.2.1 Signal Generation

Four independent instrument channels are used to generate the signals necessary to initiate the automatic reactor trip action. The signal cable routing and readout drawer locations are separated and isolated to provide channel independence.

Isolation Assemblies provide independence and separation of signal common references through high impedance isolation of signals and reduction of common mode signals within FET repeating amplifiers. Isolation of signal common between channels prevents channel interaction. Isolation of signal common for signals originating outside the RPS cabinet prevents noise pick up.

7.2.3.2.1.1 Wide-Range Logarithmic Channels

The four wide range logarithmic channels each obtain signals from two high sensitivity fission chambers. The fission chambers are grouped axially and located on the reactor cavity wall around the reactor. The output from the fission chambers are conditioned and amplified in the cable vault amplifier assemblies and transmitted to the signal processing drawers in the control room. The signal processing drawers further process the detector signal into signals that represent the source range logarithm of count rate and the rate of change of count rate, and the wide range logarithm of reactor power and the rate of change of reactor power (see Figure 7.2–7).

7.2.3.2.1.2 Power-Range Safety Channels

The signals for each of the four power range safety channels are obtained from one of the four detector assemblies located on the reactor cavity wall around the reactor. Each assembly consists of two uncompensated ion chambers stacked vertically to monitor the full length of the core. The DC current signals from each set of ion chambers are fed separately and directly to the power range safety channel drawer assemblies located in the control room. The ion chambers cover the range from 0.1 percent to 200 percent power (see Figure 7.2–8).

7.2.3.2.1.3 Flow, Water Level, Pressure Temperature and Reactor Coolant Pump Speed

The flow, water level, pressure temperature and RCP speed signals are each generated by separate sets of transmitters. Flow is measured by monitoring the pressure difference between the hot leg piping and the steam generator outlet plenum. Steam generator water level and pressure are monitored in each steam generator. The reactor coolant system (RCS) pressure is measured in the pressurizer. Temperature measurements are taken from the reactor inlet and outlet piping in each loop.

Piping and connections for these transmitters are separated and isolated to provide independence. The output of each transmitter is an ungrounded current loop supplying signal receivers and bistable trip modules.

7.2.3.2.2 Logic

Refer to Figure 7.2–2 for the following discussion.

Each measurement channel which can initiate protective action operates a channel trip unit; each trip unit includes three sealed, conventional electromagnetic actuated relays and associated contacts. Four trip units are provided for each trip condition, e.g., high pressurizer pressure.

The relays in each trip unit are numbered one, two, and three. The normally open contacts from the Number 1 relay group of Channel A are connected into a two-out-of-two logic matrix with Channel B relay contacts. (The normally open contacts are used for the logic ladders so that the relays are energized and the contacts closed under operating conditions).

The Number 2 and Number 3 relay contacts are similarly connected into two other two-out-of-two logic matrices with Channel C and Channel D relay contacts. With the number 2 and number 3 relay contacts of Channels, B, C, and D similarly arranged in BC, BD and CD combinations of two-out-of-two logic matrices, there is a total of six two-out-of-two logic matrices, forming a two-out- of-four coincidence logic with respect to the input channels.

At the output of each of the six trip logic matrices is a set of four sealed, electromagnetically actuated relays. These sets are designated the AB, AC, AD, BC, BD, and CD logic trips. The contacts from one relay of the logic trip set from each logic matrix output are placed in series with corresponding contacts from the remaining sets in each of the four trip paths. Each of these paths is the power supply line to one of the trip breaker control relays, K1 through K4, whose contacts provide actuation of under voltage and shunt trips on the trip circuit breakers, thus interrupting the AC power to the CEDMs. Deenergizing of any one trip breaker control relay interrupts (opens) one trip path and trips the two breakers controlled by that trip path. Deenergizing any set of four logic trip relays causes an interruption of all trip paths which results in a reactor trip. This logic is shown on the RPS Block Diagram, Figure 7.2–1.

The CEDMs are separated into two groups. The CEDM power supplies in each group are supplied in parallel, with three-phase AC power from the motor-generator sets. Two full-capacity motor-generator sets, each with a one-second ride through capability, are provided. The loss of either set does not cause a release of the CEAs. Each power supply source is separated into two branches. Each side of each branch line passes through two trip circuit breakers (each actuated by a separate trip path) in series so that, although both sides of the branch lines must be deenergized to release the CEAs, there are two separate means of interrupting each side of line. This arrangement provides means for the testing of the protective system.

If one of the trip units is to be removed for maintenance, the logic matrices may be changed from a two-out-of-four trip to a two-out-of-three trip by the operation of the logic bypass switch (shown on the output of the trip module, Figure 7.2–2). One key-operated switch is provided for

each trip unit. Only one key is provided for the trips for any one variable to ensure that only one of the group of four units monitoring that variable could be bypassed at one time.

The operation of the key-operated switch to bypass the trip function of a single bistable trip unit is indicated by a light on the face of the bistable trip unit and an annunciator on the main control board. This meets the requirements of paragraph 4.13 in IEEE-279 in that it provides continuous indication of the bypass in the control room.

Where the trip is to be allowed only in selected power ranges, a neutron flux signal is utilized to inhibit the action of the trip units. A manually actuated inhibit action may, under administrative control, be applied to the low reactor coolant flow, thermal margin and low steam generator pressure trips for zero power testing. The inhibits on reactor coolant flow and thermal margin are automatically removed above a preset power. The inhibit on steam generator pressure is automatically removed above a preset pressure. An additional feature of this zero power inhibit is to remove the ΔT power component of the power signal Q which is described in Section 7.2.3.3.7. This prevents RCS temperature which would cause false trips on high power and LPD during low power testing. Protective system criteria are met by this use of neutron flux signals to provide multiple independent inhibit or reset signals.

7.2.3.3 System Operation

7.2.3.3.1 Reactor Coolant Pump Under-Speed

The trip is not credited in any accident analysis event. It is provided to improve response to a low-coolant-flow condition resulting from loss of supplied power to the RCPs.

RCP shaft speed is sensed by a magnetic sensor which transmits a signal to a frequency-voltage proportional transmitter. The voltage signal, which is proportional to pump shaft speed is then applied to a bistable trip unit. (See Figure 7.2–10). The trip is initiated by two-out-of-four coincidence logic from the four channels.

Each of the four RCPs is equipped with an independent speed sensing system, with each channel actuating independently.

The zero power mode bypass switch allows the trip to be bypassed below 10^{-4} percent power. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

7.2.3.3.2 High Power Level

A reactor trip in power level Q (see Section 7.2.3.3.7) is provided to trip the reactor in the event of a reactivity excursion too rapid to result in a high pressure trip and to help prevent violation of the CEA position vs. power level assumed in the Thermal Margin and LPD trips. The high power trip setpoint can be set no more than a predetermined amount above the indicated plant power. Operator action is required to increase the setpoint as plant power is increased. The setpoint is automatically decreased as power decreases.

The setpoint and Q are compared in a bistable trip unit in each of the four safety channels. The high power trip is initiated by two-out-of-four coincidence logic from the four safety channels.

Figure 7.2–11 shows the operation of the system. If Q decreases, the setpoint Q_{TR} follows it, remaining above Q by a fixed, adjustable bias Q_b. If Q now increases, the setpoint remains at the minimum value of $Q + Q_b$ last achieved, until reset by the operator.

The system must be capable of holding the setpoint Q_{TR} at the previous minimum of $Q + Q_b$ indefinitely. This requirement precludes storing Q_{TR} by purely analog means. For this reason, the signal is stored as a digital word.

The reset circuit is designed to apply a momentary signal to the appropriate terminal of the digital storage device when a pushbutton is pressed. This causes Q_{TR} to achieve the current value of $Q + Q_b$. The reset circuit is buffered to permit locating the pushbutton outside the RPS.

The signal Q_{TR} is limited so that, regardless of the logic described above, it cannot go above or below limits set by potentiometers.

Other circuits generate a pretrip limit for the bistable trip unit, as well as a contact closure to alert the operator when power increases after reaching a minimum. The pretrip alarm provides audible and visual annunciation in addition to CEA withdrawal prohibit signals.

Q and Q_{TR} are processed and buffered for remote display on the main control board. Q is also taken to the Control Element Drive System (CEDS) and the Plant Computer for use in the power dependent insertion limit calculation.

7.2.3.3.3 Low Reactor Coolant Flow

This reactor trip is provided to protect the core against DNB in the event of a coolant flow decrease.

The flow measurement signals are provided by measuring the differential pressure across each of the two steam generators. Each steam generator differential pressure signal is proportional to the square of the steam generator mass flow rate. These signals are summed to provide a signal that is proportional to the square of the reactor vessel mass flow rate. The measured signal is compared to a pre-determined trip setpoint in the reactor protection system trip bistable. This configuration is shown in Figure 7.2–4, and is repeated in each of four redundant channels. A reactor trip is initiated when the measured value falls below the bistable trip setting in two-out-of-four coincident channels.

Pre-trip alarms are similarly provided to warn of decreased coolant flow conditions.

The Zero Power Mode Bypass switch allows this trip to be bypassed for subcritical testing of CEDMs. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

Reactor coolant pump operating requirements and surveillance requirements are defined by Technical Specifications.

7.2.3.3.4 Low Steam Generator Water Level

An abnormally low steam generator water level indicates a loss of steam generator secondary water inventory. If not corrected, this would result in a loss of capability for removal of heat from the RCS.

The low steam generator water level reactor trip protects against the loss of feedwater flow accident (see Section 14.2.7) and assures that the design pressure of the RCS will not be exceeded. The trip set point specified in Table 7.2-1 assures that sufficient water inventory will be in the steam generator at the time of trip such that steam generator dryout does not occur before the auxiliary feedwater (AF) delivers sufficient flow to remove decay heat and recover steam generator water level.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two downcomer level differential pressure transmitters, one on each steam generator. Audible and visual pretrip alarms are actuated to provide for annunciation of the approach to reactor trip conditions.

7.2.3.3.5 Low Steam Generator Pressure

An abnormally high steam flow from one of the steam generators (e.g., that which would occur as the result of a steam line break (SLB)) would be accompanied by a marked decrease in steam pressure. To protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant following an SLB, a reactor trip is initiated by low steam generator pressure.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two pressure transmitters: one on each steam generator. Audible and visual pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions.

Signals from these pressure transmitters initiate closure of the main steam isolation valves (MSIV) on a two-out-of-four coincidence of low pressure in either steam generator.

A bypass is provided for the low steam generator pressure trip to allow performance of zero power physics testing. Bypass is accomplished manually by means of a key-operated switch in each channel. The manual bypass is enabled only below a preset steam pressure and is automatically removed above this set point. Figure 7.2–5 is a functional diagram of this circuit.

The trip bypass is initiated manually by turning a switch to the BYPASS position. The bypass is removed, regardless of the manual switch position if the auctioneered higher of the steam generator pressures exceeds a predetermined set point. When the manual switch is in the “Off” position, there is no bypass of the low steam generator pressure trip function.

For setting and testing the bistable device by use of the trip tester, a TEST SELECT switch is provided to disconnect the signal not being tested. This can only be done if the manual bypass switch is in the OFF position.

The contact testing system consists of two pushbuttons, one for auto bypass removal test and one for manual bypass removal test. The purpose of these tests is to check the status of the bypass circuit contacts. These tests do not alter or change the contacts from either an open or closed position.

Pressing the AUTO TEST pushbutton completes a path through a contact to the light; thus the light being on indicates that bypass is allowed by the automatic removal circuit. Pressing the MAN TEST pushbutton similarly tests the manual contact. Pressing both pushbuttons energizes the light regardless of bypass status; this tests the light. The light is also on for both the manual and automatic contacts closed, i.e., when the trip bypass is in effect.

The reactor trip set point of 691 psia (Table 7.2-1) is sufficiently below the full load operating pressure so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow.

7.2.3.3.6 High Pressurizer Pressure

A reactor trip for high pressurizer pressure is provided to prevent excessive blowdown of the RCS by relief action through the pressurizer safety valves. A reactor trip is initiated by two-out-of- four coincidence logic from the four independent measuring channels if the pressurizer exceeds 2397 psia. This signal simultaneously opens the power-operated relief valves (PORV).

The trip signals are provided by four independent narrow range pressure transducers measuring the pressurizer pressure. Pretrip alarms are initiated if the pressurizer pressure exceeds 2350 psia as indication of the approach to reactor trip conditions.

7.2.3.3.7 Thermal Margin/Low-Pressure Trip

The TM/LP trip is provided for two purposes. The thermal margin portion of the trip, in conjunction with the low reactor coolant flow trip, is designed to prevent the reactor core safety limit on DNB from being violated during anticipated operational occurrences. The low pressurizer pressure portion of the trip functions to trip the reactor in the event of a LOCA.

A reactor trip is initiated whenever the RCS pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of RCPs operating and the axial offset. Consistent with the Technical Specifications, the minimum value of reactor

coolant flow rate, the maximum azimuthal tilt, the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with the Technical Specifications is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a High Power Level trip is assumed.

Figure 7.2–6 and 7.2–11 describe the operation of this trip system. The higher of the two inlet temperatures is used in the TM/LP calculation.

Figure 7.2–13 shows a block diagram of the thermal power calculation.

The calculation begins with the generation, by temperature transmitters, of currents representing the cold and hot leg temperatures in each loop. By forcing these currents through precision resistors and utilizing the resulting voltage drops, voltages representing cold leg temperatures (T_{c1} and T_{c2}) and hot leg temperature (T_h) are sent to the calculator. The latter signal is the average T_h for the two loops.

In the calculator, the higher cold leg temperature signal is selected and subtracted from the hot leg temperature signal to determine the temperature rise. The calculator generates terms proportional to the first and second powers of the temperature rise and to the product of temperature rise and cold leg temperature. These three terms represent thermal power for four pump operation and steady state conditions, accounting for coolant density, specific heat, and flow rate variations with temperature and power.

The sum of these terms represents the core power for four-pump operation under steady state or mild transient conditions.

The coefficient of the term proportional to the temperature rise ($K\sigma$) is set by the potentiometer labeled “ΔT Power Calibrate” on the Reactor Protective System Calibration and Indication Panel (RPSCIP) front panel. This factor is adjusted to make the thermal power calculation agree with the plant calorimetric calculation.

The thermal power (B) is subtracted from the nuclear power (ϕ), generated by the NI Channel, and the difference is displayed on a meter with a range of -10 percent to +10 percent of full power. The meter has adjustable upper and lower setpoints. The contacts energize a local light when the deviation goes outside the range defined by the setpoints.

To make the nuclear power signal agree with the thermal power and/or the plant calorimetric calculation, a potentiometer labeled “Nuclear Power Calibrate” is provided. This potentiometer adjusts the gain of the NI Channel from 0.8 to 1.33. An auctioneering circuit selects the higher of nuclear power or thermal power for use in the remainder of the system.

The signal Q, the maximum of nuclear or thermal power, is modified by a CEA position function. The resulting signal is then augmented by an axial factor which is generated in the LPD Trip

section as shown on Figure 7.2–12 and described in Section 7.2.3.3.10. The resulting signal is called Q_{DNB}.

A pressure setpoint Pvar is calculated as a linear function Q_{DNB} and of the modified inlet temperature described above.

An auctioneering circuit selects the maximum of this calculated pressure setpoint and a constant pressure P_{min}, and sends the resulting signal to the trip unit as a downscale trip setpoint. Trip will occur if the primary pressure drops below the calculated setpoint or below 1865 psia, whichever is larger. A pretrip setpoint, 75 psi above the trip point, is also generated.

The trip signal is initiated by a two-out-of-four coincidence logic from four independent safety channels, and audible and visual pretrip alarms are actuated to provide for annunciation on approach to reactor trip conditions. The pretrip action also initiates a CEA withdrawal prohibit.

The zero power mode bypass switch allows this trip to be bypassed for low power testing. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

The Thermal Margin trip setpoint is processed and buffered for remote display on the main control board in four dual indicators which compare the trip setpoint with indicated pressurizer pressure.

7.2.3.3.8 Loss of Turbine

The trip for loss of turbine is an equipment protective trip and is not required for reactor protection. (Refer to Chapter 14).

This trip is initiated above a preset power level, by actuation of 2 of 4 low hydraulic fluid pressure switches associated with the turbine-generator control systems. Its purpose is to help avoid the lifting of the steam generator safety valves during the system transient after a turbine trip, thus extending the service life of these valves. Since credit has not been taken for the equipment protective trips in the Safety Analysis of the plant, they do not fall within the scope of IEEE 279. In the case of the Millstone Unit Number 2 RPS, the design criteria listed in Section 7.2.1.2 (which includes IEEE 279) apply to all trip functions including the equipment protective trips.

7.2.3.3.9 High Containment Pressure

A trip is provided on high containment pressure in order to assure that the reactor is tripped concurrent with safety injection actuation.

Four pressure measurement channels provide analog signals to bistable trip units which are connected in a two-out-of-four coincidence logic to initiate the protective action if the containment pressure exceeds a preselected value.

7.2.3.3.10 High Local Power Density Trip

The minimum power level required to produce centerline melt in Zircaloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in KW/ft. This FCMLHR is determined using the methodology of XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5 (“Qualification of Exxon Nuclear Fuel for Extended Burnup,” Exxon Nuclear Company, October 1986.) The high LPD trip is provided to prevent the peak LPD in the fuel from exceeding the FCMLHR limit during anticipated operational occurrences thereby assuring that the melting point of the UO_2 fuel will not be reached.

A reactor trip is initiated whenever the axial offset exceeds either a high or low calculated setpoint as described below. The axial offset is calculated from upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of the core power level with the CEA group position being inferred from the core power. The trip is automatically bypassed below 15 percent power.

Consistent with the Technical Specifications, the maximum azimuthal tilt and the maximum CEA deviation permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with Technical Specification is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a High Power Level Trip is assumed.

Figure 7.2–13 shows a block diagram of a typical channel. Circuits in the Power Range Safety Channel generate signals proportional to the sum of and the difference between the upper and lower detector outputs. An axial offset signal is formed as a linear function of the ratio of the difference to the sum and compared with upper and lower limits generated from a modified power signal described in Section 7.2.3.3.8. The offset signal is also used in the Thermal Margin Trip as previously described.

If the axial offset exceeds either calculated limit, a contact in the calculator opens and deenergizes the trip relays in an auxiliary trip unit. The pretrip relay is similarly released if a narrower envelope is exceeded.

7.2.3.3.11 Manual Trip

A manual reactor trip is provided to permit the operator to trip the reactor. Depressing two pushbutton switches on the control panel causes interruption of the AC power to the CEDM power supplies. The manual trip function is testable during reactor operation.

7.2.3.3.12 Bypass Operation

All trips are normally cleared before startup. (The loss-of-turbine trip is automatically bypassed below 15 percent power.) For some operations, it may be desirable to perform a reactor startup with some reactor parameters at values which would normally cause a trip. For these special operations, zero power mode bypass switches may be used to bypass the low flow, RCP underspeed, and the low TM/LP trip functions. Four bypass key switches are provided. Each

bypass switch removes all three trip functions from one of the four protective system channels. These bypasses are automatically removed above 10^{-4} percent power. A manual bypass is provided to allow startup with a low steam generator pressure.

7.2.4 TESTING

IEEE 338-1971, “Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems,” September 1971, provides guidance for development of procedures, equipment and documentation of periodic testing. The bases for and the scope and means of testing are described in this section. Test intervals and their bases are included in the Technical Specifications. The organization for testing and for documentation is described in Chapter 13.

Since operation of the protective system will be infrequent, the system is periodically and routinely tested to verify its operability. A complete channel can be individually tested without initiating a reactor trip or violating the single failure criterion, and without inhibiting the operation of the RPS.

The RPS is capable of being checked from the trip unit input through the power supply circuit breakers of the CEDMs. The majority of the components in the protective system can be tested during reactor operation. The remainder of the components can be checked by comparison with similar channels or channels that involve related information. These components, which are not tested during reactor operation, will be tested during scheduled reactor shutdown to assure that they are capable of performing the necessary functions. Minimum frequencies for checks, calibration and testing of the RPS instrumentation are given in Section 4 of the Technical Specifications. Overlap in the checking and testing is provided to assure that the entire channel is functional. The use of individual trip and ground detection lights, in conjunction with those provided at the supply bus, assure that possible grounds or shorts to another source of voltage may be detected.

During reactor operation the measuring channels are checked by comparing the outputs of similar channels and cross-checking with related measurements. The trip units are tested by inserting a voltmeter in the circuit, noting the signal level, initiating a test input and noting signal level required to effect trip action. This provides the necessary overlap in the testing process and also enables the test to establish that the trip can be effected within the required tolerances. The test signal is provided by a test signal generator which is connected to the trip module at the signal input terminals. With the test signal generator connected, the desired signal is selected and then inserted into the trip unit by depressing the manual test switch. The test circuit permits various rates of change of signal input to be used. Trip action (opening) of each of the trip unit relays is indicated by individual lights on the front of the trip unit. The pretrip alarm action is indicated by a separate light.

The sets of logic trip relays at the output of each logic matrix are tested one at a time. The test circuits in the logic permit only one logic ladder to be opened and one set of relays to be held at a time; the application of hold power to one set denies the power source to the other sets. In testing a logic trip set (e.g., AB) a holding current is initiated in the test coils of the logic trip relays by turning the matrix relay trip test switch to “off” and depressing the matrix logic AB test

pushbutton switch. Operation of the matrix trip test switch initiates a deenergizing current in the test coils of a parallel pair of trip unit relays. With the ladder logic relay contacts open, the logic trip relays may be deenergized one at a time (by rotating the matrix relay trip test switch) to open the associated trip breakers. Indicator lights on the trip status panel provide verification that coil operation and trip breaker actuation conditions have occurred.

The response time from an input signal to the protection system trip units through the opening of the trip circuit breakers is verified by measurement in accordance with the Technical Specifications and the Technical Requirements Manual.

Periodic testing can be carried out from the control room to ensure the continuity of the measurement loop. A supplementary signal is introduced into the measurement loop that is bypassed and the response to this signal is indicated on a meter in the protection system. This verifies the continuity of the loop and insures its operability.

The RPS is manufactured under strict engineering and quality control specifications. These specifications require that the equipment be inspected for workmanship, proper materials and channel separation as required by IEEE-279-1968. Furthermore, all intra- and inter-connection wiring is tested for continuity and an insulation test is performed between each conductor and chassis ground and between each individual pair of connectors. An operational test is performed on the system during which time input signals are simulated to ensure that the protective system is capable of producing the proper trip signals.

A Power Trip Test Interlock feature is provided to assure protection against the consequences of certain signal selection switches being left in off-normal positions at the completion of testing. The position of these switches, the Test Select switch in conjunction with the Test Enable switch in the Power Range Nuclear Instrumentation and the ΔT Power Calculator Test Switch in the RPSCIP, may be changed during testing without causing a trip by bypassing the High Power, TM/LP, and LPD Trips. If these switch positions should be changed during normal operation or if any of these channels that have been bypassed for testing should be returned to normal prior to returning the switches to normal, trips will occur on the affected channels.

The reactor trip circuit breakers are provided with the capability of disabling the shunt trip coils for time response testing of the undervoltage trip coils. This feature is provided by means of a handswitch located on the trip breaker switchgear. These handswitch contacts are shown in Figure 7.2–1 and 7.2–2. During this testing process, no more than one handswitch should be placed in the test position at the same time and annunciation is provided on C04 that warns the operator that the shunt trip coil is disabled.

7.2.5 SYSTEM EVALUATION

7.2.5.1 General

The RPS is designed to limit reactor power and coolant conditions to levels within the design capability of the reactor core. Instrument performance characteristics, response time, and accuracy are selected for compatibility with and adequacy for the particular function. Trip

setpoints are established by analysis of the system parameters. Factors such as instrument inaccuracies, bistable trip times, CEA travel times, valve travel time, circuit breaker trip times, and pump starting times are considered in establishing the margin between the trip setpoints and the safety limits. The time response of the sensors and protective systems are evaluated for abnormal conditions.

Since all uncertainty factors are considered as cumulative for the derivation of these times, the actual response time may be more rapid. However, even at the maximum times, which are added to the CEA drop time, the system provides conservative protection.

The wiring in the protective system is grouped so that no single fault or failure, including either an open or shorted circuit, will negate protective system operation. Signal conductors are protected and routed independently.

Loss of or damage to any one path will not prevent the protective action. Sensors are piped so that blockage or failure of any one connection does not prevent protective system action. The process transducers located in the containment building are specified and rated for the intended service.

These components which must operate in the LOCA environment are rated for the LOCA temperature, pressure and humidity conditions. Results of type test are used to verify these ratings. In the control room the nuclear instrumentation and protective system trip paths are located in four compartments. Mechanical and thermal barriers between these compartments reduce the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltage does not cause channel malfunction. Where signals originating in the RPS feed the computer, buffering is used to ensure circuit isolation; where the RPS is feeding annunciators, isolation is ensured through the use of relay contacts.

The protective system is designed such that the deenergized state initiates a channel trip. This feature ensures that if channel continuity is lost, that channel will fail in a safe condition. Module withdrawal is indicated by lights on the RPS panel and annunciators and alarms above the main console. The modules are not interlocked to prevent withdrawal but are set up such that withdrawal of one module causes a channel trip and withdrawal of a second module causes full trip. If a channel is in the bypassed condition, withdrawal of any other two modules of that parameter will cause a full trip since the system is in the two-out-of-three trip mode. Only one set of keys is available to the plant operator allowing only one of the four channels of any one parameter to be bypassed at any one time. Strict administrative control ensures that this requirement is not violated. Indication of test and bypass conditions or removal of any channel from service is given by lights on the protection system front panel and an audible/visible alarm annunciator. If a protection system channel is removed from service either by a failure in that channel or by deenergization for maintenance purposes, then that channel will go to the tripped condition which is indicated and alarmed. Bypasses are alarmed and indicated on the main control panel. Automatic removal of a bypass is also indicated by the main control panel annunciator.

The protective system is designed and arranged to be able to perform its function with a single failure of any component. Some of the faults and their effects are described below.

7.2.5.2 Analog Portion of the System

a. A loss of signal in a measurement channel initiates protective channel trip action for those parameters which normally trip on a decreasing input. These include Low RCP Speed, Low Reactor Coolant Flow, Low Steam Generator Water Level, Low Steam Generator Pressure, and Thermal Margin/Low Pressure pressure input. Parameters which trip on an increasing input, including high power level, high pressurizer pressure and high containment pressure, will not trip on a loss of input signal.

b. Shorting of the signal leads to each other has the same effect as a loss of signal. Shorting a lead to a voltage has no effect since the signal circuit is ungrounded. Periodic testing will determine if grounds or applications of potential to the signal circuit exist.

c. Single grounds of the signal circuit have no effect. Periodic checking of the system will assure that the circuit remains ungrounded.

d. Open circuit of the signal leads causes a channel trip signal.

7.2.5.3 Logic Portion of the Circuit

a. Inadvertent operation of the relay contacts in the matrices will be identified by the indicating lights.

b. Shorting of the pairs of contact in the matrices will prevent the trip relay sets from being released. Such shorts are detectable in the testing process by observing that the trip relay sets cannot be dropped out. Testing is accomplished by successive opening of the logic matrix contact pairs.

c. Shorting of the matrices to an external voltage has no effect since the matrix is ungrounded. The testing process will indicate accidental application of potential to the matrix. Equipment is provided to detect grounds on the matrices.

d. The logic matrices will each be supplied by two power sources. Loss of a single power source has no effect on operation. Loss of power to a logic matrix initiates a trip condition.

e. Failure of a logic trip relay set to actuate has no effect since there are six sets in series in the trip action and any one set initiating trip action will cause the action to be completed.

f. The failure of one trip breaker control relay in a trip breaker circuit has no effect since there are two trip breakers in series, either of which will provide the necessary action.

g. Single grounds in the trip breaker control relay circuits have no effect since the circuit is ungrounded. Ground detectors on each 125 VDC bus also indicate an accidental ground.

h. The AC circuit supplying power to the trip breaker control relay coil is fed from an isolation transformer. The circuit has a local ground detection system. Each of the four trip paths are fed from a separate 120 VAC vital instrument bus.

i. The CEDM power supply circuits operate ungrounded so that single grounds have no effect. A ground detection system is provided. The CEDMs are supplied in two groups by separate pairs of power supplies to further reduce the possibility of a CEA being improperly held. The CEDM load requirements are such that the application of any other local available voltage would not prevent CEA release.

7.2.6 SYSTEM RELIABILITY AND AVAILABILITY

7.2.6.1 Power Supply

The power for the protective system is supplied from four separate and independent vital 120 volt AC buses. Each vital bus is supplied from one of the two station battery systems through separate inverters (as shown in Figure 8.5–1). During normal operation, the battery chargers maintain a floating charge on each battery while at the same time, supply power to the vital inverters. Upon loss of auxiliary AC power, the batteries provide the power for inverter operation. In the event of loss of vital bus, the protective channel associated with the bus goes into a trip condition. Each vital bus also has automatic rapid transfer to a regulated instrument ac supply used as a backup.

The distribution circuits from the vital buses are provided with fuses and circuit breaker protection to assure that individual circuit faults are isolated close to the fault.

7.2.6.2 Environment Capability

RPS components were specified for environmental conditions existing in the area of the plant in which the components are installed.

Radiation design criteria for RPS components located within normally radioactive areas are specified at a gamma level of 1R/hour for 10 years, except for the main coolant resistance temperature detectors (RTDs), which are specified at 10R/hour for 40 years. Protective system equipment not located in normally radioactive areas or located in areas of very low activity has been specified accordingly. Periodic tests and calibration will assure detection of gradual equipment deterioration and will assure capability of the system to operate as required by the original design bases.

The design criteria for all electrical cable are that the cable shall not fail when subjected to any accident conditions after the long-term normal operating conditions. Cable details are specified in Chapter 8.

7.2.6.3 Seismic Capability

The specification for each RPS component or assembly includes the seismic requirements for that equipment.

These components are qualified by either of the two following methods.

In most cases, the supplier is required to qualify his equipment by calculation or testing, or a combination of both. This qualification is formally documented and submitted for approval prior to official acceptance of the equipment by Quality Control.

In other cases, tests or calculations are performed by independent consultants or laboratories who submit a formal report. Acceptance of the equipment from the supplier is contingent upon the proof of suitability as established by the results of those tests or calculations.

The choice of the analytical or experimental qualification procedure is determined by the size, shape and structural or functional complexity of the equipment in accordance with the criteria outlined in IEEE 344 “Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations.” Racks, panels or other supporting structures are generally qualified by analysis, while bistable trip units and other modules are generally qualified through testing. Tests and calculations are performed following the guidelines of IEEE 344, 1971.

Type testing is the preferred qualification method in accordance with IEEE-323, 1971, “General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Station.” A report on the qualification of the RPS panels, racks, and equipment including the results of type tests, was submitted by CE in December of 1972 in the form of a topical report, CENPD-61, “Seismic Qualification of Category I Electric Equipment for Nuclear Steam Supply Systems.”

7.2.6.4 Physical Separation

The locations of the sensors and the points at which the sensing lines are connected to the process loop have been selected to provide physical separation of the channels, thereby precluding a situation in which a single event could remove or negate a protective function. Process transmitters located inside the containment and required for short term operation following a LOCA are rated for the intended service in the LOCA environment. The routing of cables from these transmitters is arranged so that the cables are separated from each other and from power cabling to minimize the likelihood of common event failures. This includes separation at the containment penetration areas. In the control room, the four nuclear instrumentation and protective system trip channels are located in individual compartments. Mechanical and thermal barriers between these compartments minimize the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltages do not cause channel malfunction.

7.2.6.5 Bistable Trip Unit Drift

The bistable trip units have been specified and designed for a maximum drift of 32 mv.

TABLE 7.2-1 REACTOR TRIP AND PRETRIP SET POINTS

No.	Reactor Trip	Pretrip Alarm Set Point	Trip Set Point
1.	Reactor Coolant Pump Underspeed * (not credited)	N.A.	≥ 830 rpm
2.	High Power Level	2% below trip setpoint	≤ 10% above measured power Q
3.	Low Reactor Coolant Flow * 4-Pump Operation, %	N.A.	≥ 91.7
4.	Low Steam Generator Water Level, % (Auctioneered low of SG #1, SG #2)	54	≥ 48.5
5.	Low Steam Generator Pressure **, psia (Auctioneered low of SG #1, SG #2)	780	≥ 691
6.	High Pressurizer Pressure, psia	2350	≤ 2397
7.	Thermal Margin/Low-Pressure *	75 psia above trip set point	Variable trip set point with minimum of 1865 psia
8.	Loss of Turbine *** (Low Hydraulic Fluid Pressure) psig	N.A.	≥ 500
9.	High Containment Pressure, psig	N.A.	≤ 4.42
10.	Manual Trip (Push Buttons)	N.A.	N.A.
11.	Local Power Density	< FCMLHR	FCMLHR

* Manual inhibit permitted below 10^{-4} percent power: automatically removed prior to increasing reactor power to 10^{-4} percent power.

** Manual inhibit permitted below 800 psia: automatically removed above 800 psia.

*** Inhibited below 15% power.

TABLE 7.2.1.1–I PARAMETERS AFFECTING FUEL DESIGN LIMIT

I. DNBR and Void Fraction

1) Core Power
2) Core Inlet Temperature
3) Axial Power Distribution
4) Radial Peaking Factor
5) Primary System Pressure
6) Core Mass Flow Rate
7) Azimuthal Tilt Magnitude

II. Fuel Temperature (or Equivalent kw/ft)

1) Core Power
2) Axial Power Distribution
3) Radial Peaking Factor
4) Azimuthal Tilt Magnitude

TABLE 7.2.1.1–II PERTINENT NSSS PARAMETERS AND MONITORED VARIABLES

NSSS Parameter	Monitored Variable
Core Power	Neutron Flux Power/ΔT Power
Core Inlet Temperature	Cold Leg Temperature
Primary System Pressure	Pressurizer Pressure
Axial Power Distribution	Axial Flux Offset
Radial Peaking Factor	(Inferred from Neutron Flux Power/DT Power and Technical Specifications PDILS)
Core Mass Flow Rate	Steam Generator Differential Pressures
Azimuthal Tilt Magnitude	Value assumed consistent with Technical Specifications Limit

TABLE 7.2.1.1–III TRIP FUNCTIONS

Acceptable Fuel Design Limit	Calvert Cliffs	Millstone Unit 2
DNBR & Void Fraction	Thermal Margin Trip Axial Flux Offset Trip Low Flow Trip	Thermal Margin Trip High Power Trip Low Flow Trip
Kw/Ft (Fuel Temperature)	Axial Flux Offset Trip	Local Power Density Trip High Power Trip

TABLE 7.2.1.1–IV TRIP VARIABLE MONITORED

Trip	Calvert Cliffs	Millstone Unit 2
Thermal Margin	Pressurizer Pressure Neutron Flux Power/ΔT Power	Pressurizer Pressure Neutron Flux Power/ΔT Power
Core Inlet Temperature	Core Inlet Temperature	Core Inlet Temperature
Axial Flux Offset	Neutron Flux Power Axial Flux Offset	
Local Power Density		Neutron Flux Power/ΔT Power Axial Flux Offset
High Power	Neutron Flux Power	Neutron Flux Power/ΔT Power
Low Flow	Steam Generator Differential Pressure	Steam Generator Differential Pressure

FIGURE 7.2–1 REACTOR PROTECTIVE SYSTEM BLOCK DIAGRAM

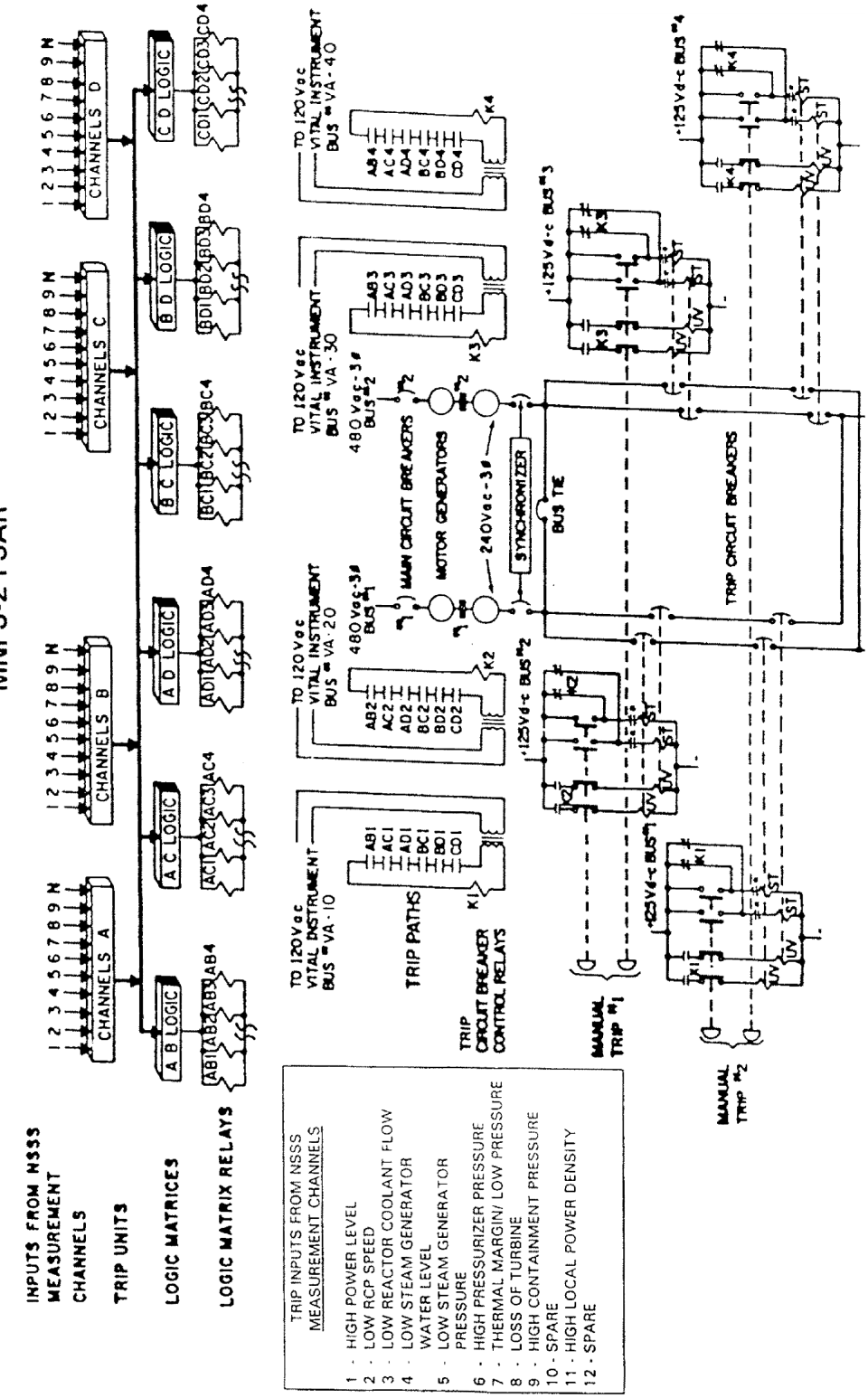


FIGURE 7.2–2 REACTOR PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.2–3 NUCLEAR INSTRUMENTATION SYSTEM FUNCTIONAL DIAGRAM

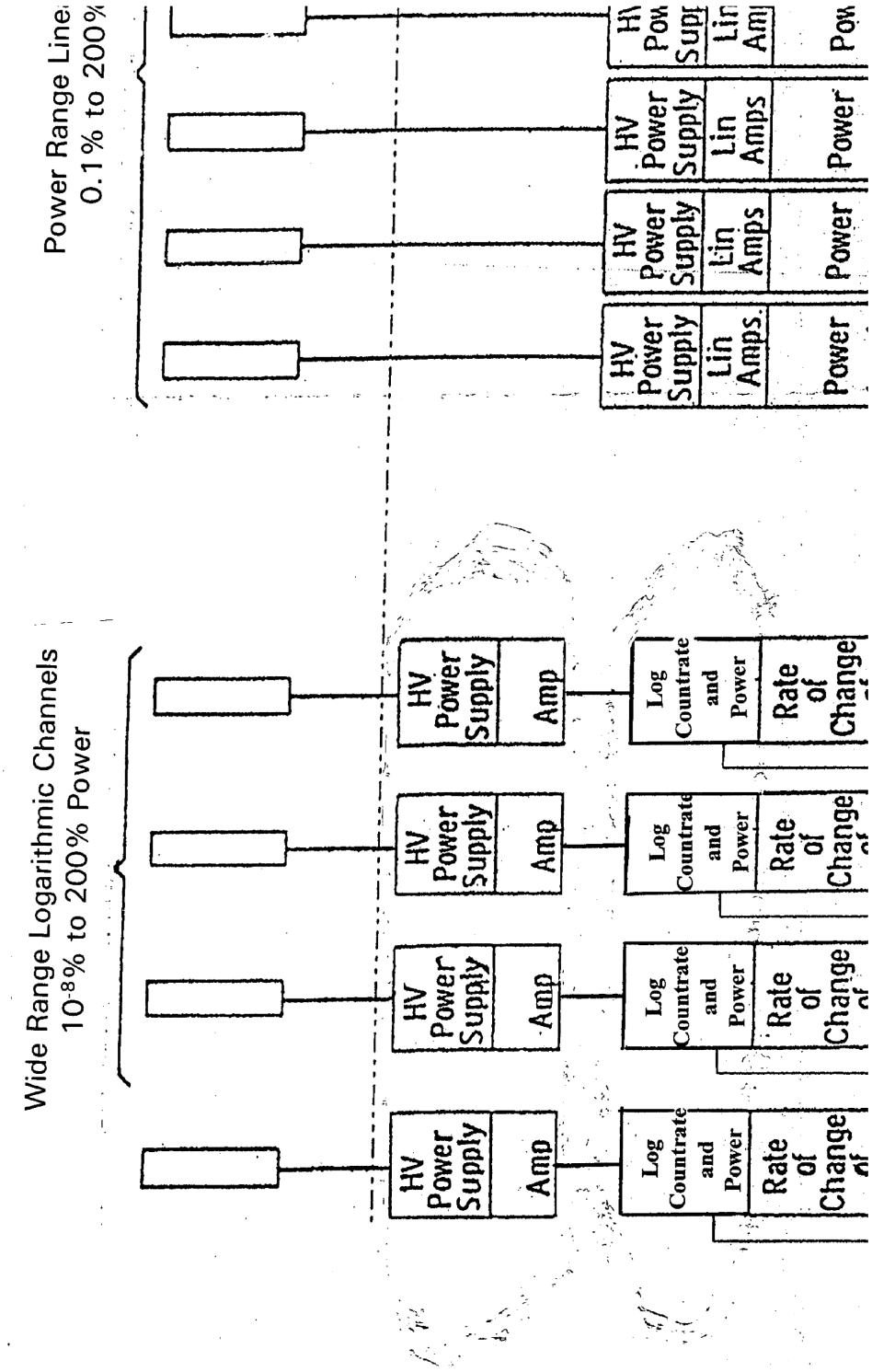


FIGURE 7.2–4 LOW FLOW PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

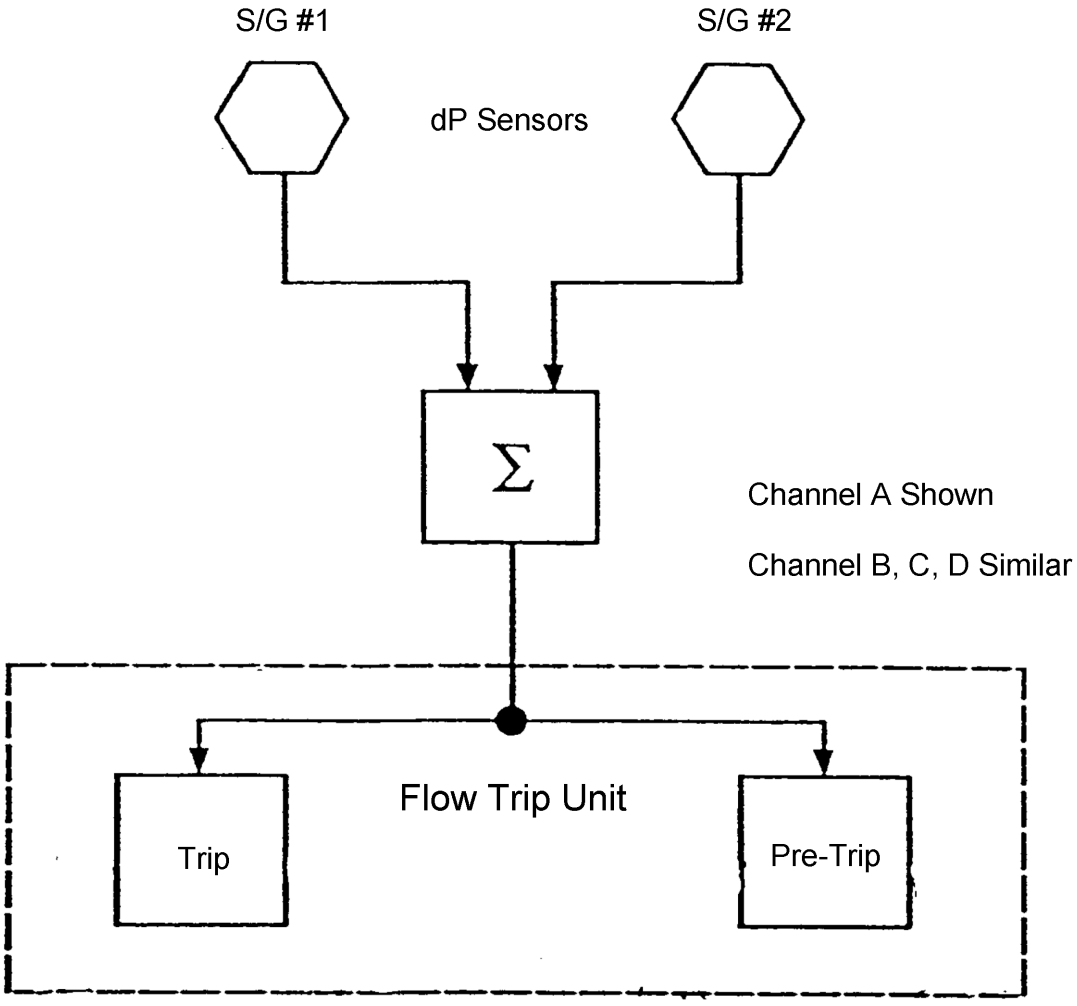
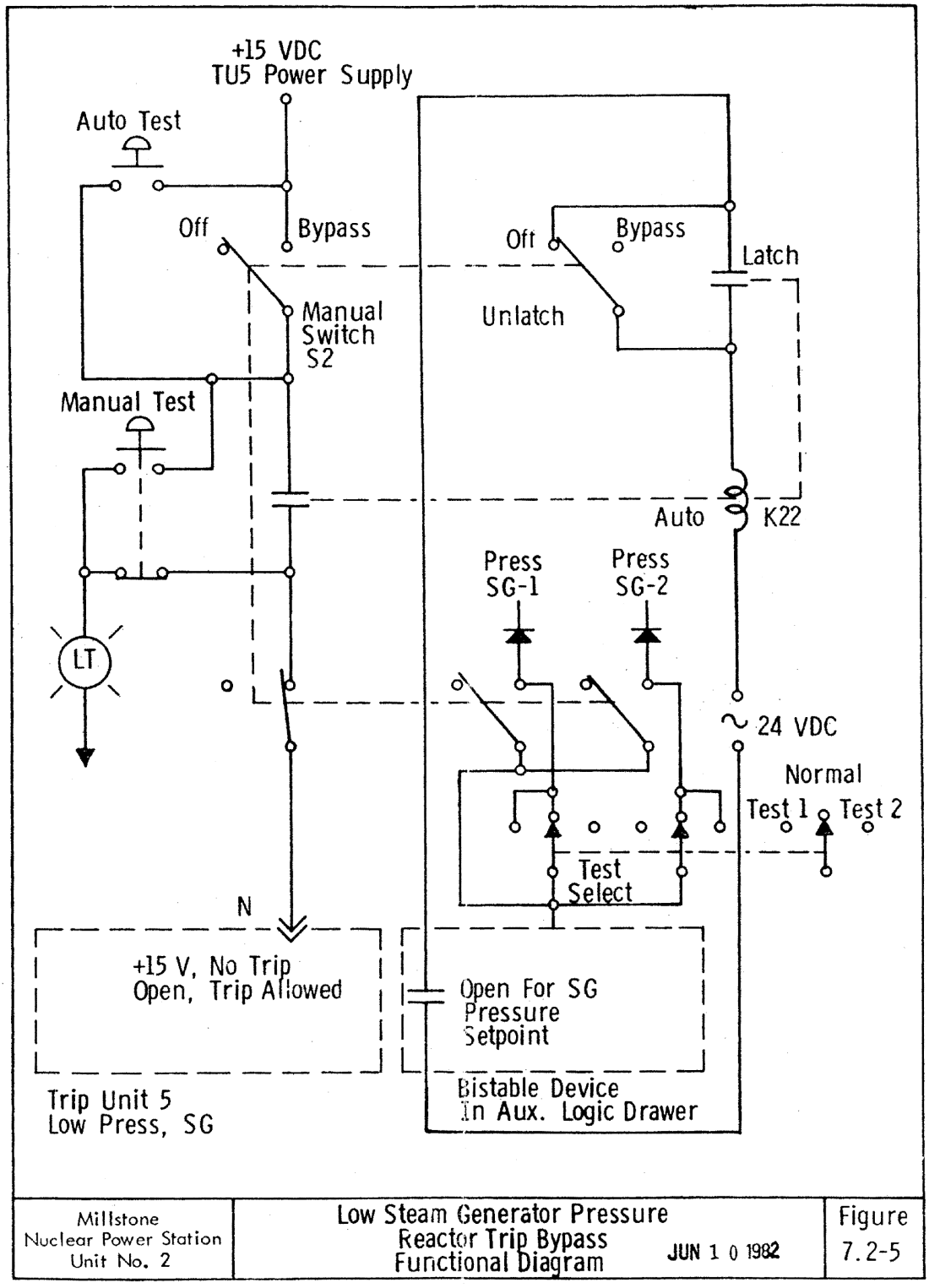


FIGURE 7.2–5 LOW STEAM GENERATOR PRESSURE REACTOR TRIP BYPASS FUNCTIONAL DIAGRAM



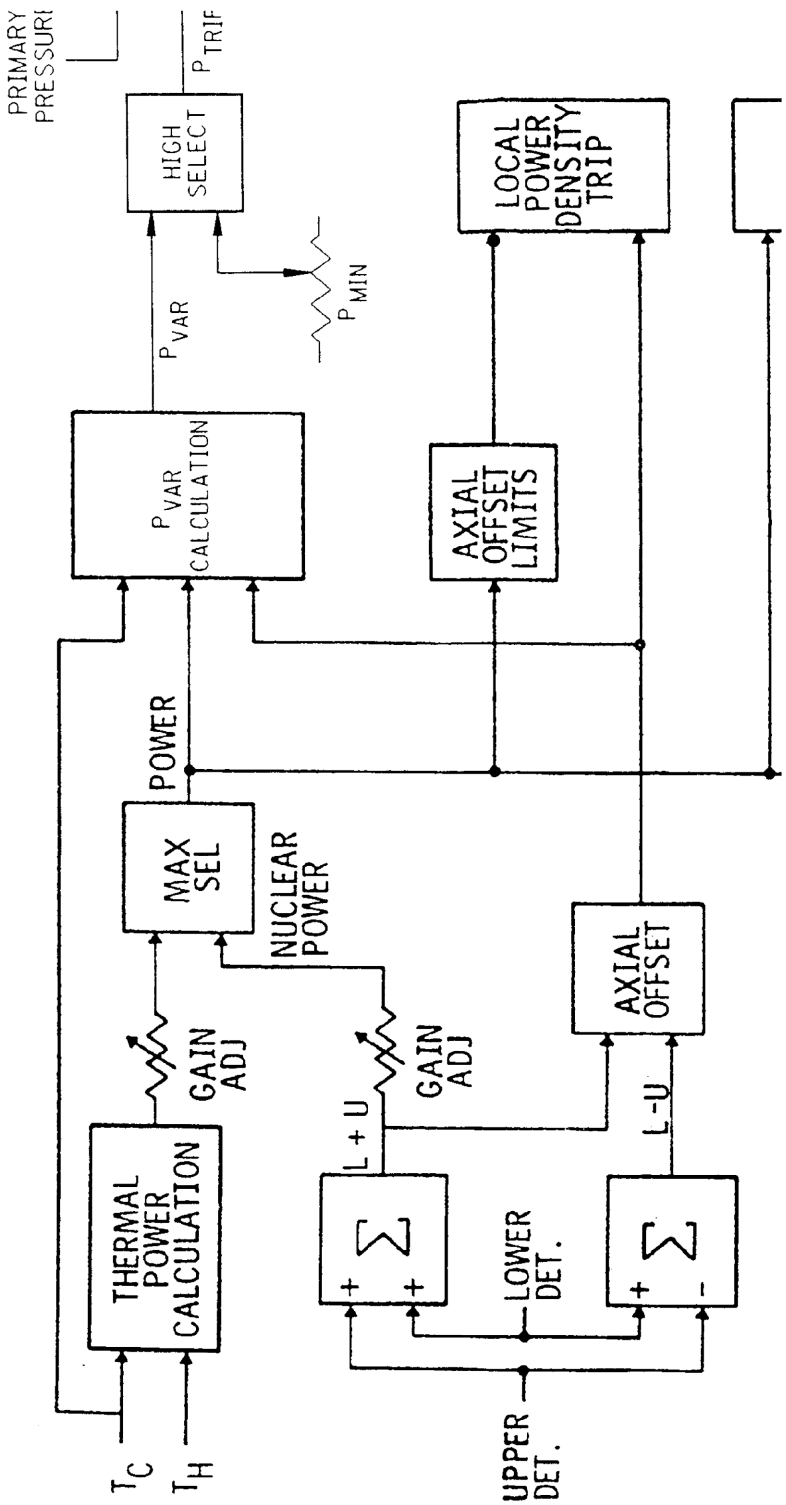


FIGURE 7.2–6 BLOCK DIAGRAM CORE PROTECTION TRIPS

FIGURE 7.2–7 NEUTRON FLUX MONITORING SYSTEM LOGARITHMIC RANGE CHANNELS

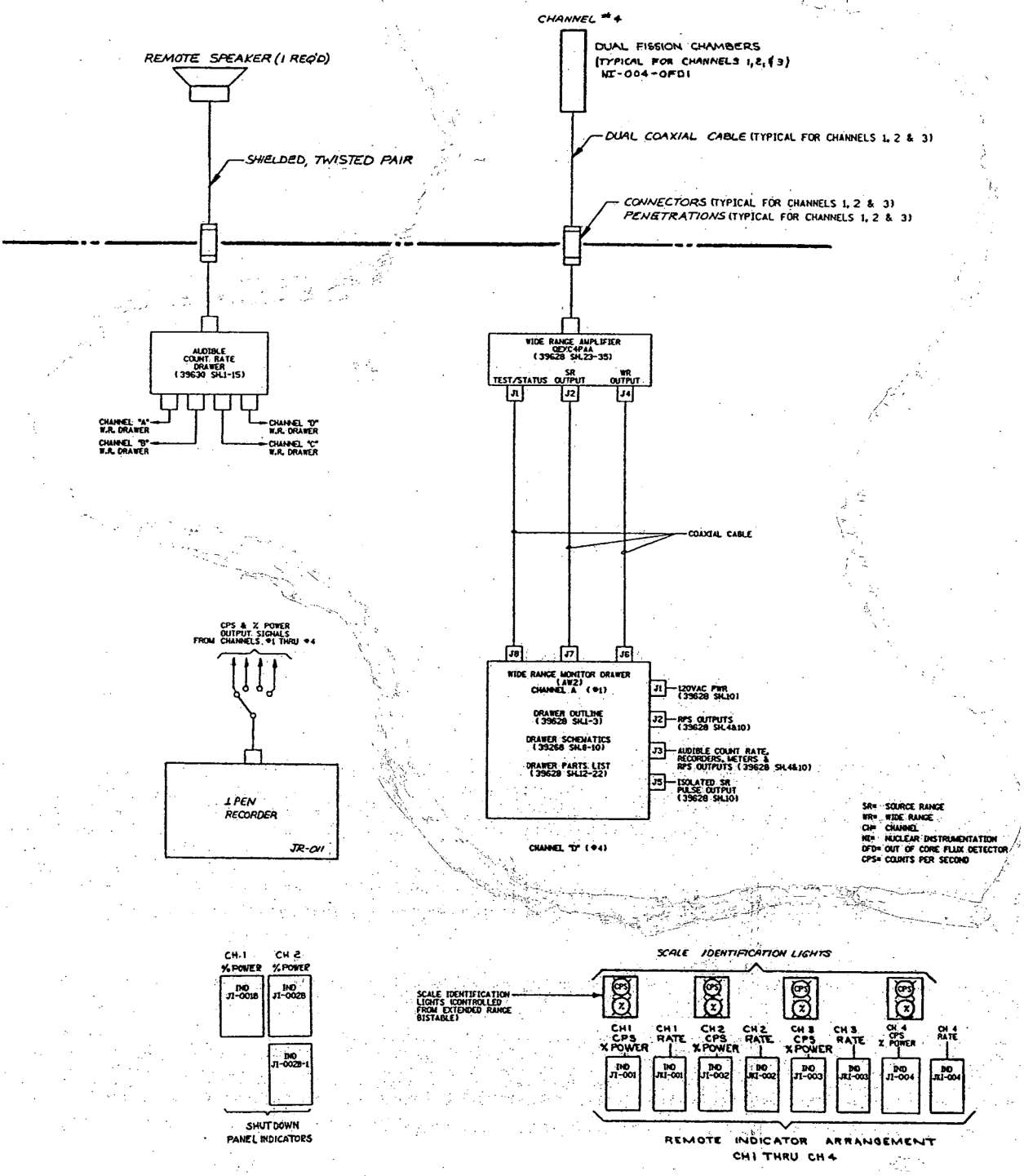
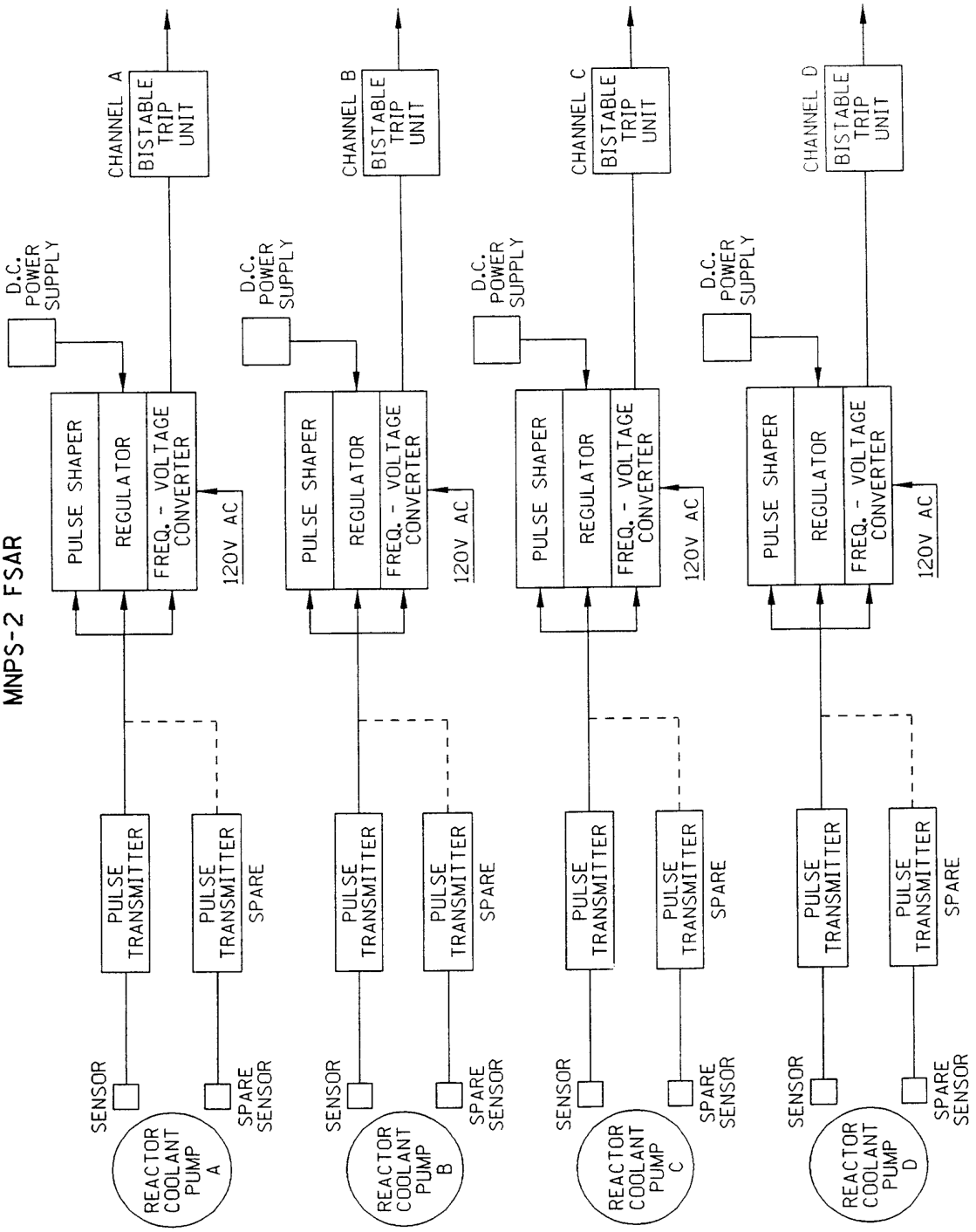


FIGURE 7.2–8 NEUTRON FLUX MONITORING SYSTEM POWER RANGE CHANNELS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.2–9 NOT USED

FIGURE 7.2–10 REACTOR COOLANT PUMP UNDERSPEED BLOCK DIAGRAM



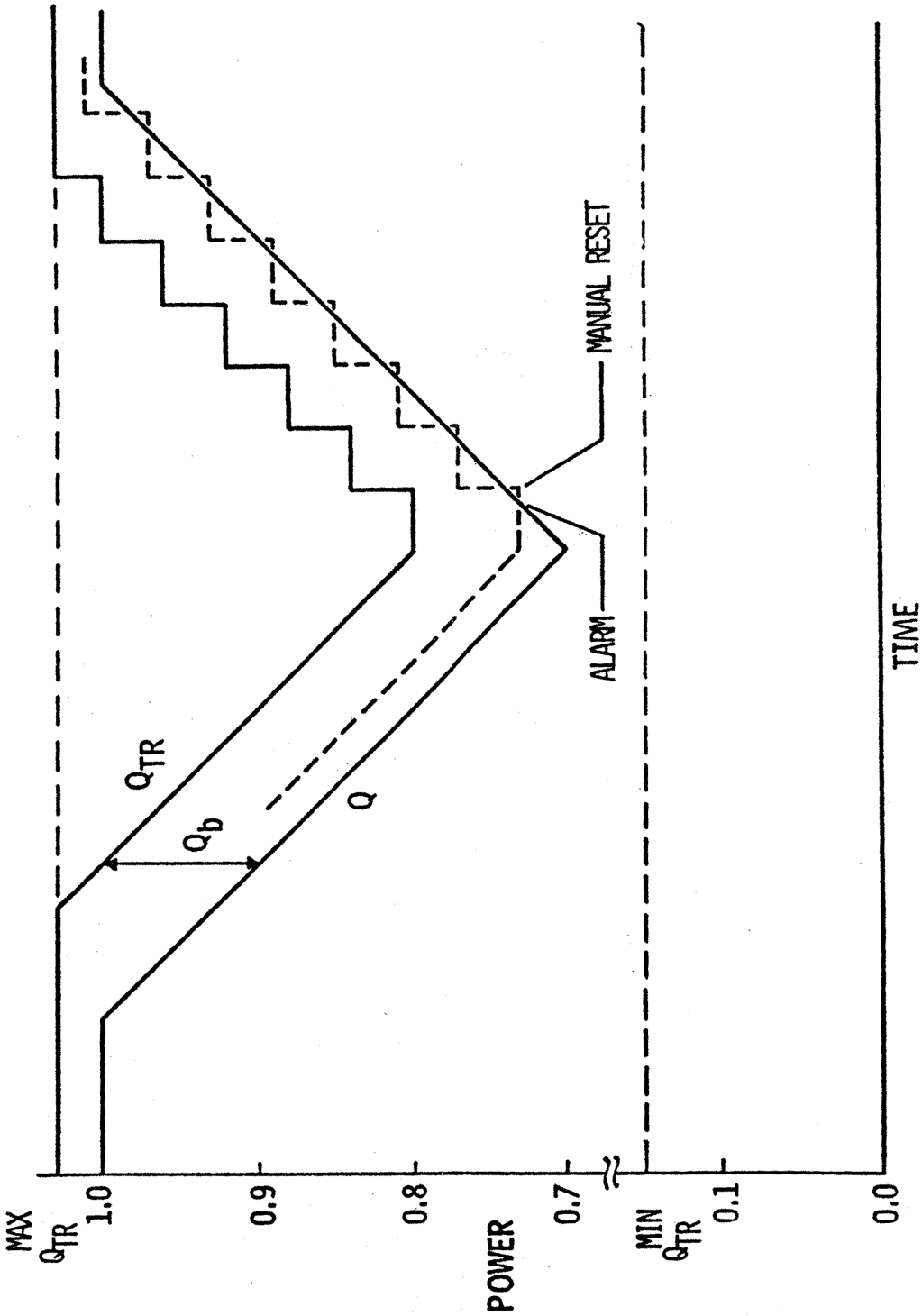


FIGURE 7.2–11 VARIABLE HIGH POWER TRIP OPERATION

FIGURE 7.2–12 THERMAL MARGIN TRIP

MNPS-2 FSAR

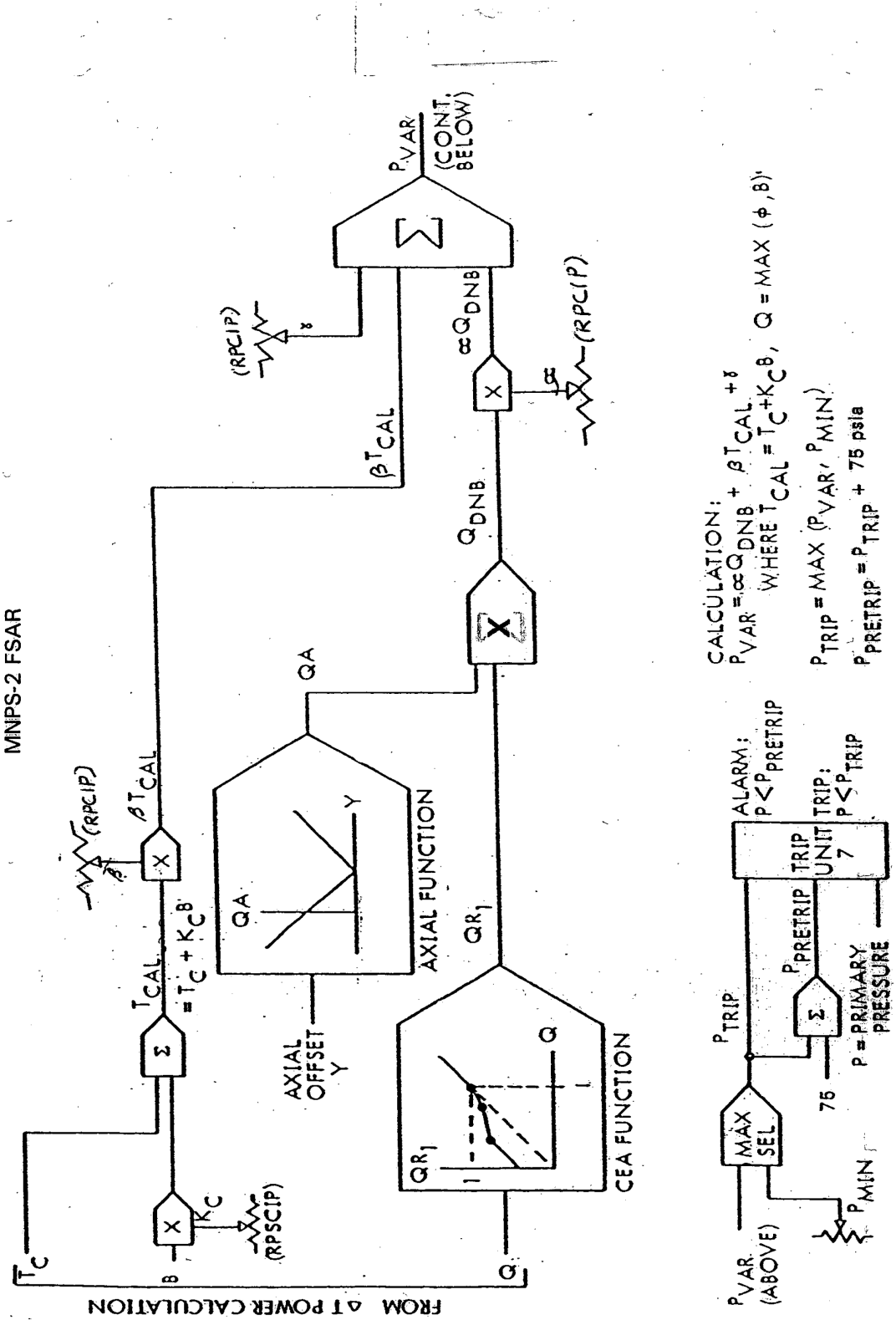


FIGURE 7.2–13 DT POWER CALCULATION

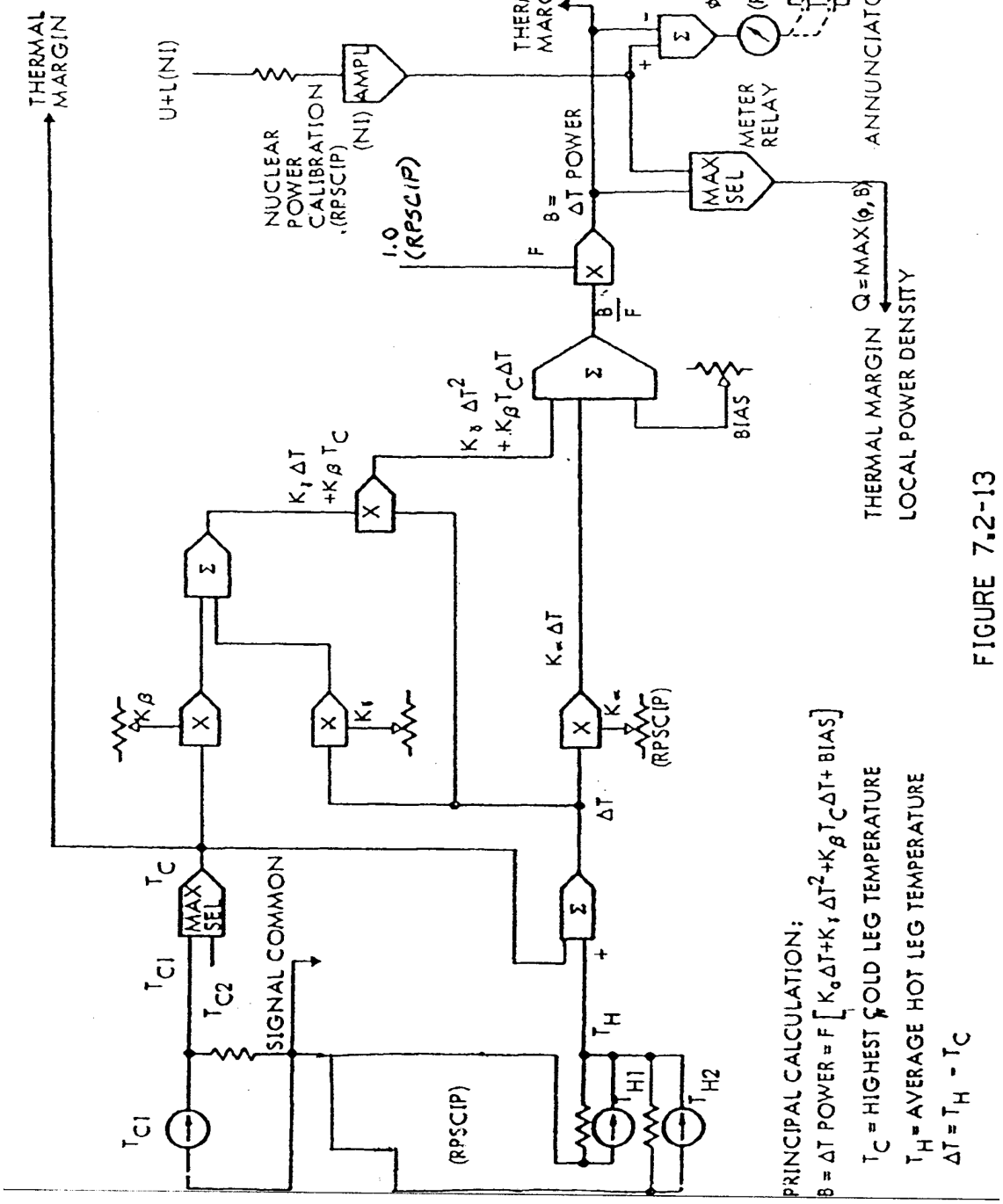


FIGURE 7.2-13

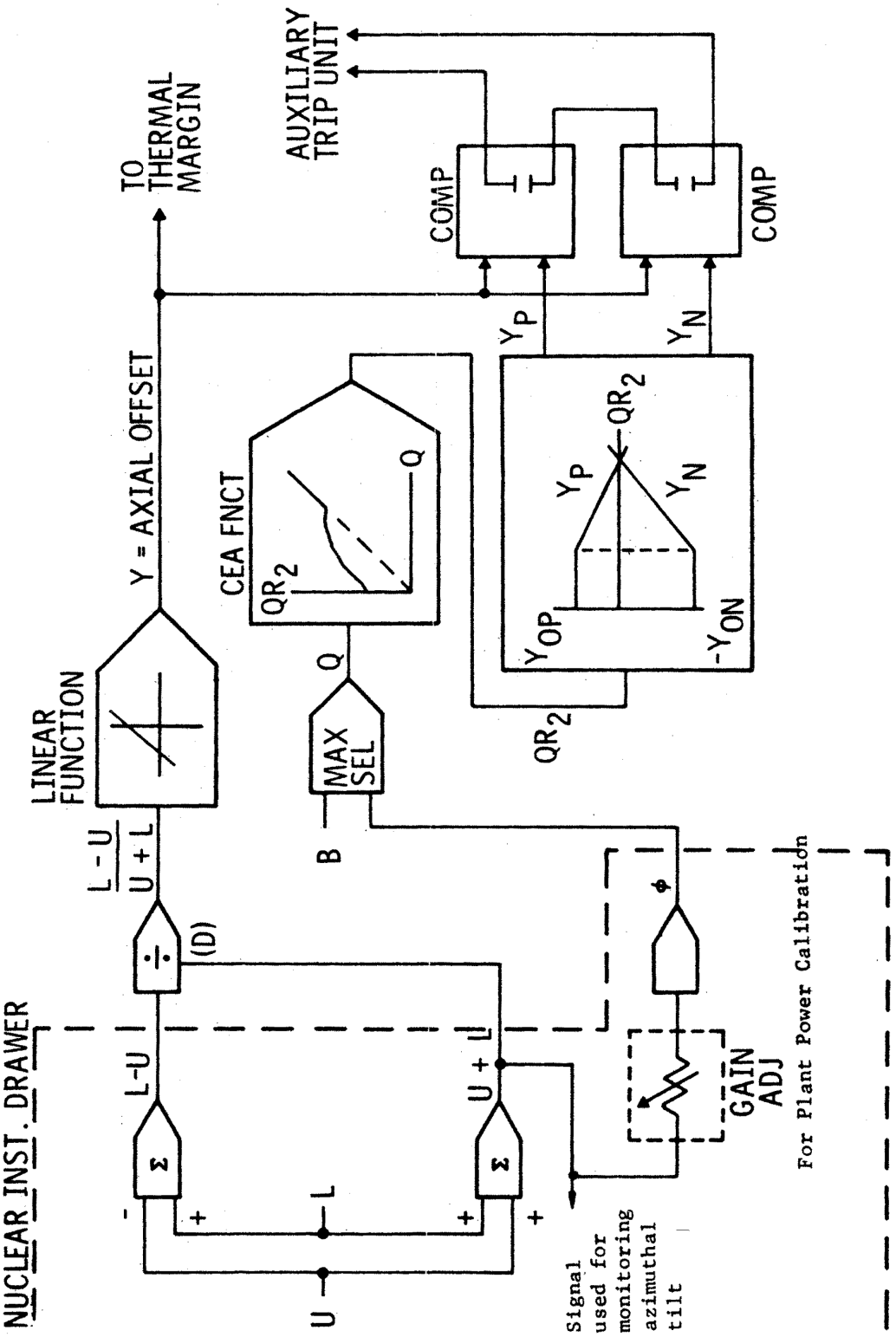


FIGURE 7.2–14 LOCAL POWER DENSITY TRIP

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

7.3.1 DESIGN BASES

The plant protection system consists of a sense and command design features involved in generating the signals used for reactor trip and engineered safety features actuations. The plant protection system consists of the Reactor Protection System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The ESFAS is that portion of the plant protection system used to automatically and manually initiate the operation of the engineered safety features (ESF) systems and essential auxiliary supporting (EAS).

At Millstone Unit 2, the ESFAS is comprised of: (1) the Engineered Safeguards Actuation System (ESAS) and (2) the Auxiliary Feedwater Automatic Initiation System (AFAIS).

7.3.1.1 Functional Requirements

The Engineered Safety Features Actuation System (ESFAS) is designed to detect accident conditions and initiate the operation of systems and components important to safety. The ESFAS is designed for high functional reliability and in-service testability commensurate with the safety functions it performs.

7.3.1.2 Design Criteria

7.3.1.2.1 Single Failure, Redundancy and Independence

The ESAS and AFAIS are designed with sufficient redundancy and independence to assure that (1) no single failure results in loss of protective function and (2) removal from service of any component of a channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

The ESFAS is designed and constructed in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 279, “Criteria for Nuclear Generating Station Protection Systems,” 1971. Emphasis has been placed on the single-failure criteria, which requires that no single failure shall prevent the system from initiating the safety feature functions when true initiating conditions exist.

The system functions are implemented by means of redundant sensors, instrument loops, logic and actuation devices.

Independence is provided between redundant elements to preclude any interactions between channels during maintenance or in the event of channel malfunction.

Redundant elements are electrically isolated from each other such that events affecting one element are not reflected in any other redundant element.

The electrical isolation provided is capable of withstanding the application of the maximum credible voltage or current transient without degradation below an acceptable level.

Physical separation is in accordance with Section 8.7.

7.3.1.2.2 Modular Design

Interchangeability without preselection is provided for all corresponding Engineered Safeguards Actuation System (ESAS) modules or components with the exception of keys. All ESAS items removable from the equipment such as assemblies, subassemblies, electrical parts, modules and hardware are replaceable physically and electrically with corresponding items. The replacement of parts, when accomplished in a manner prescribed by the ESAS manufacturer, will not cause the equipment to depart from the original specified performance.

The Auxiliary Feedwater Automatic Initiation System (ASAIS) utilizes modular Foxboro equipment for the signal conditioning, bistable, and coincidence logic functions.

7.3.1.2.3 Module Withdrawal and Bypass

Withdrawal of any ESAS subunit or module in any of the sensor channels results in a trip of the affected channel and annunciation of the tripped channel. Means are provided by noninterchangeable key-operated switches to manually bypass any single analog channel bistable trip module of a four-channel parameter. Annunciation of ESAS channel bypass is provided.

The removal of an AFAIS module may not result in a trip as AFAIS is an energize to actuate design. Each AFAIS actuation channel contains provisions to bypass the sensor channel input to that actuation channel’s coincidence logic by means of a key-operated switch. Annunciation of channel bypass is provided.

With the exception of auxiliary feedwater automatic initiation control, loss of power to a sensor channel of the ESFAS will result in a trip of the sensor channel affected. A trip of the ESAS channel is annunciated. A trip of an AFAIS sensor channel is indicated by status lamps on C517 and C518.

The system is designed such that routine servicing and preventive maintenance can be performed without affecting operation or availability. Normal maintenance does not reduce performance below the minimum safety level.

7.3.1.2.4 Environment

The ESAS cabinets are installed in the control room which is normally air conditioned and suitable for computers or computer-grade equipment. However, the equipment installed in the ESAS cabinets is designed to function continuously at temperatures ranging from 40°F to 140°F and at a relative humidity of 95 percent. Additionally, the following control room equipment in the SPEC 200 cabinets is environmentally qualified for a temperature range of 40°F to 120°F and a relative humidity of 10 percent to 95 percent (with an 86°F maximum wet bulb): (a) plant

process equipment that provides analog inputs to the ESAS and (b) plant process equipment that provides analog inputs and bistable and coincidence logic functions in the AFAIS.

7.3.1.2.5 Seismic Requirements

The ESFAS has been designed to function before, during and after a design basis earthquake (DBE). The ESFAS has been demonstrated to be seismically qualified by type tests and analysis. The original ESAS cabinet was evaluated by Consolidated Controls Corporation as documented in Engineering Reports No. 832 and NBR 863. The type tests and analysis on the remaining ESFAS equipment conform to the guidance of the IEEE standards governing seismic qualification described in Section 7.3.1.2.6 below.

7.3.1.2.6 Codes and Standards

The ESFAS and component parts conform to the requirements of the following IEEE standards and Nuclear Regulatory Commission (NRC) Regulatory Guide.

The auxiliary feedwater (AF) automatic initiation system and component parts and in containment mounted sensors conform to the 1974 edition of IEEE 323 and the 1975 edition of IEEE 344 and the standards listed below. The containment mounted sensors conform to the 1975 edition of IEEE 344 and the standards listed below.

a. IEEE 279	1971 Criteria for Protection Systems for Nuclear Power Generating Stations
b. IEEE 308	1970 IEEE Standard Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations
c. IEEE 323	1971 General Guide for Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations
d. IEEE 344	1971Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations
e. IEEE 336	1971 Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations
f. IEEE 338	1971 Periodic Testing of Nuclear Power Generating Station Protection Systems
g. Regulatory Guide 1.22	Periodic Testing of Protection System Actuation Functions
h. IE Bulletin No. 80-06	Engineered Safety Features Reset Controls

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 response time of the ESAS system from the sensor input to the trip device output terminals is less than or equal to 500 milliseconds. This response time is inclusive to the overall component response time requirements of ESFAS as described in the Millstone Unit 2 Technical Requirements Manual.

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.

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 Number 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 the actuation channels receives a signal from the following variables: (See Figures 7.3–1, 2, 7.3–3, 7.3–4.)

a. Pressurizer Pressure

Low pressurizer pressure during power operation is indicative of a loss-of-coolant accident (LOCA). It is measured with four redundant pressure transmitters. A pressure loss below 1714 psia on any two of four bistables in the ESF system will initiate a simultaneous safety injection actuation signal (SIAS), containment isolation activation signal (CIAS), and enclosure building filtration actuation signal (EBFAS). These signals will isolate all unnecessary lines at the containment penetration, initiate safety injection system (SIS) operation, and start the enclosure building filtration system (EBFS). The four pressure transmitters are also used for input signals to the RPS.

b. Containment Pressure

High containment pressure during power operation is indicative of a LOCA or main steam line break. It is measured with four pressure transmitters. An increase in containment pressure to 4.42 psig on any two of four bistables in the ESF system will initiate a simultaneous SIAS, CIAS, EBFAS, and MSIAS. Measurement of containment high pressure is a diverse means of sensing a loss of coolant condition. The transmitters are reverse acting type (increasing input gives a decreasing output signal) to permit fail safe operation.

With regard to power requirements, loss of instrument power results in a tripped bistable which would require only one more of the three remaining bistables to trip in order to get an actuation.

A failure mode analysis for loss of onsite instrument power is shown in Table 7.3.2.2-1.

A further increase in containment pressure to 9.48 psig will initiate a containment spray actuation signal (CSAS) which will start two containment spray pumps and open their respective discharge motor operated valves (MOV) to start spraying.

c. Containment Gaseous and Particulate Radiation

Two gaseous and two particulate monitors are used to detect the release of radioactive fission products to the containment atmosphere. If these monitors were to fail or were unavailable, grab samples are taken or portable continuous air monitoring equipment is used. The ESFAS logic will initiate containment purge

isolation should any one of the four detectors exceed its set point. However, the fuel handling area radiation system automatic functions are not credited in the fuel handling accident analyses.

d. Steam Generator Pressure

Each steam generator pressure is sensed by four pressure transmitters. A drop in pressure to 572 psia on any two out of the four sensor channels on either steam generator will actuate a main steam isolation actuation signal (MSIAS) which automatically closes both MSIVs. The four pressure transmitters are also used for input signals to the RPS.

e. Fuel Handling Area Radiation

Fuel handling area high radiation is sensed by four redundant area radiation monitors located on walls adjacent to the spent fuel pool. Upon detection of high radiation due to a fuel handling accident from any two of the four monitors, an auxiliary exhaust actuation signal (AEAS) is generated which stops the spent fuel pool area outside air supply fan and diverts the exhaust to the EBFS.

f. Refueling Water Storage Tank Level

The safety injection pumps initially take suction from the refueling water storage tank (RWST). After the tank level has decreased to 46 inches as measured by two of four level sensing channels, a sump recirculation actuation signal (SRAS) transfers the safety injection pump suction to the containment sump for long-term recirculation.

g. Emergency Bus Undervoltage

The undervoltage protection provided for the emergency buses consists of two independent schemes, one for each 4,160 volt emergency bus, 24C(A3) and 24D(A4). Each scheme employs redundant design features and consists of two levels of protection. The Level 1 undervoltage protection (loss of voltage) is designed to detect a loss of voltage on the emergency buses. The Level 1 undervoltage logic isolates the emergency buss from all sources, initiates automatic loading shedding, and provides a start signal for the diesel generator associated with the emergency bus. The Level 2 undervoltage protection (degraded voltage) is designed to protect the safety-related equipment from operation under sustained low (degraded) voltage conditions. The Level 2 undervoltage logic provides a trip signal to the reserve station service transformer (RSST) supply breaker of the associated bus. Figure 7.3–1 illustrates the emergency bus undervoltage protection logic (see also FSAR Section 8.2).

Potential transformers on 4,160-volt emergency buses 24C(A3) and 24D(A4) provide voltage inputs to the ESAS undervoltage sensor logic. In the ESAS sensor

cabinets, these analog inputs are compared to the trip setpoints for the Level 1 and Level 2 undervoltage protection. The Level 1 and Level 2 trip outputs associated with bus 24C(A3) are connected to the ESAS Facility 1 actuation logic cabinet. The Level 1 and Level 2 trip outputs associated with bus 24D(A4) are connected to the ESAS Facility 2 actuation logic cabinet.

The trip setpoints and associated time delays for the Level 1 and Level 2 undervoltage protection are defined in the Technical Specifications and are as follows:

Level 1	2,912 volts with a time delay of 2 seconds
Level 2	3,700 volts with a time delay of 8 seconds

The time delay for the Level 1 undervoltage setpoint was chosen to:

1. allow sufficient time for protective relaying schemes to detect and clear system and station faults that cause bus voltages to drop below the Level 1 undervoltage setpoint and

2. ensure that the Level 1 undervoltage actuation functions occur in a time frame that is consistent with the MP2 safety analysis for design basis accidents.

The time delay for the Level 2 undervoltage setpoint was chosen to be:

1. sufficiently long as to prevent unnecessary trips of the circuit breaker connecting the emergency bus with the RSST source during voltage transient conditions (e.g., the starting of large motors) and

2. sufficiently short as to (a) prevent damage to safety-related AC loads and (b) ensure that the Level 1 undervoltage actuation functions occur in a time frame that is consistent with the MP2 safety analysis for design basis accidents.

h. Steam-Generator Level

Level in each of the two steam generators is sensed by four common and redundant differential pressure transmitters. (Each steam generator level transmitter also provides an isolated signal to the RPS for input to the low steam generator water level reactor trip as described in Section 7.2.) AF is initiated automatically on low steam generator level by a 2 out of 4 logic matrix from the four independent bistable channels. The level input signals from both steam generators are auctioneered at the channel level, so that each auto AF bistable channel receives the lower of the input signals from two level transmitters (one from each steam generator). If any two of the four (total), measurement channels

detect a steam generator level decreasing below 26.8% (nominal), an automatic AF initiation signal is processed and the following occurs:

- each steam generator blowdown isolation valve receives a close signal
- a three (3) minute, twenty-five (25) second (nominal) time delay begins timing out
- and a control room alarm annunciates the automatic AF initiation signal.

Following the time delay, automatic AF start indication is provided on C05 and C21 and two auxililary relays per facility are energized. The auxiliary relays send a start signal to the respective facility motor driven AF pump and open signals to both AF regulating valves.

7.3.2.3 Actuation Channels

The two redundant and independent actuation channels monitor the sensor channel trips and by means of coincidence logic determine whether a protective action is required. The 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-out-of-4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2-out-of-3) or manual SIAS (main control board pushbutton).

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

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

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

e. Containment purge valves close signal (CPVIS) under the following conditions: high containment radiation 1-out-of-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-out-of-4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2/3) or manual MSIAS (main control board push buttons).

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

i. AEAS for the following conditions: high radiation in the fuel handling area 2-out-of-4(or 2-out-of-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-out-of-4 (or 2-out-of-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-out-of-4 (or 2-out-of-3) channels or on a SIAS. FSAR Section 8.3 discusses the automatic start of the emergency DGs.

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-out-of-4 (or 2-out-of-3 channels).

m. Auxiliary feedwater automatic initiation system (AFAIS) signal under the following conditions: Steam Generator water level low, ≤ 26.8%, 2-out-of-4 (or 2-out-of-3).

7.3.2.4 Actuation Interface

For each of the groups in each ESAS actuation channel, separate interfacing logic and coincident logic are provided such that any isolated actuation of one group will not produce actuation of any of the others. The ESAS actuation interface modules ensure that the signals provided by the actuation logic are distributed to the safety equipment. The ESAS actuation modules provide signals to the final ESAS relays for the control of safety equipment.

The ESAS actuation signals produced in the logic modules are separated and test groupings have been selected to permit equipment actuation without upsetting normal plant operation.

The AFAIS actuation is arranged in two redundant actuation channels. The actuation outputs are arranged such that actuation of either of the two redundant actuation channels will result in actuation of both auxiliary feedwater valves. Each of the two redundant channels will actuate its associated motor-driven auxiliary feedwater pump.

7.3.2.5 Channel Trips

When an actuation channel trips, due to unit conditions or manual initiation, its logic seals in its tripped state until it is manually reset.

7.3.2.6 Manual Initiation

Each actuation channel (1 or 2) of each of the following actuation signals can be initiated by a pushbutton located on the main control board: SIAS, CSAS, CIAS, EBFAS, SRAS, MSIAS, and AEAS. Each pushbutton initiates all actuation groups of the associated signal within its respective actuation channel.

The manual actuation switches located on the main control board are designated “HS” on Figures 7.3–1 and 7.3–4.

Each train (A or B) of the auxiliary feedwater system (AFWS) may be manually initiated utilizing normal system manual controls.

7.3.2.7 Manual Reset

Each actuation channel (1 or 2) of each of the actuation signals described in Section 7.3.2.3 has an independent momentary contact pushbutton located on the ESAS cabinets for manual reset of the actuation signals. The reset switch has no effect as long as a trip condition input to the logic matrix exists. Each pushbutton resets all actuation groups of the associated signal within actuation channels 1 or 2.

Manual reset of the AF automatic initiation circuitry is accomplished by positioning the respective train control switch to the reset (momentary) position. The reset switch has no effect as long as a trip condition input to the logic matrix exists.

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 blocks, 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-out-of-4 to 2-out-of-3 logic.

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-out-of-4 to a 2-out-of-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.

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.

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).

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.

b. The deadband or hysteresis of the bistable is internally adjustable.

c. The noise appearing in the system, such as power supply noise, switching transients, or noise included in wiring entering or leaving the cabinets, will not induce false outputs nor prevent true actuation.

d. The error of the bistable units will not be greater than 0.5 percent of span over a 30 day period for the combination of the effects of ambient temperature drift and supply voltage and frequency variations.

e. The bistables automatically reset when trip conditions no longer exist.

The AFAIS bistable is a Foxboro Spec 200 module with an adjustable setpoint.

7.3.2.12 Power Supply

The ESFAS and RC-30A, B, C, D, A-1 and B-1 cabinets are powered from four vital DC-AC inverters which are supplied from two vital 125 volt DC batteries. Each battery supplies two inverters. The four inverters supply power to the four sensor channels. Redundant power for the two actuation channels is supplied from the two redundant 125 volt DC batteries.

Each ESAS sensor cabinet is provided with a backup 120 volt vital AC source via isolation transformers internal to each sensor cabinet. Sensor Cabinet A is backed up by vital AC supplied from Sensor Cabinet D and vice versa. Sensor Cabinet B is backed up by vital AC supplied from Sensor Cabinet C and vice versa.

SPEC 200 Instrumentation Cabinets RC30A, B, C, and D, which provide most of the signal inputs to ESAS, are not provided with backup AC power. Loss of a vital DC bus with no credit of the non-vital Turbine battery results in loss of two inverters. The ESAS Sensor Cabinets would remain energized, and the associated SPEC 200 cabinets would de-energize. Loss of two inverters would result in all ESAS signals with the exception of LNP and SRAS.

A power supply interruption of up to 35 milliseconds or a voltage dip of 20 percent for up to 30 seconds will not produce actuation of a ESAS sensor channel bistable.

The availability of DC relay power for the ESAS is constantly monitored by the automatic test insertion (ATI) portion of the system. In the event of loss of DC power supply in either actuation channel, the ATI fault annunciation will sound. In addition, the power supply monitor lamp on the actuation cabinet front panel will extinguish.

Refer to Section 8.6 and Figure 8.5–1 for more detail on the vital instrument power supply system.

7.3.2.13 System Interface

The following redundant process variable signals are input to the ESFAS:

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 [1]
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)	174.5-42.3
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.

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1. Note: Due to physical instrument tap locations, RWST level cannot be monitored below approximately 4%.

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.

7.3.3 SYSTEM OPERATION

7.3.3.1 Operational Bypasses

Means are provided for the operator to block actuation of the SIAS while the RCS is undergoing depressurization. Blocking is effected as follows:

a. Permitted only when the pressurizer pressure is less than 1850 psia on three out of four analog sensor channels.

b. Initiated independently for channels 1 and 2 by momentary contact closure switches on the main control panel.

c. Removed automatically when the pressurizer pressure exceeds the 1850 psia block permit pressure on two out of four analog sensor channels.

d. Block only the output of the 1714 psia pressurizer pressure trip 2/4 matrices.

e. Permission to block and any block initiations are annunciated on the main control board.

Means are provided for the operator to block actuation of the MSIAS during startup or shutdown. Blocking is effected as follows:

a. Permitted only when the steam generator pressure is below 700 psia on three out of four analog sensor channels.

b. Initiated independently for channels 1 and 2 by momentary contact closure switches on the main control panel.

c. Removed automatically when the steam generator pressure exceeds the 700 psia block permit pressure on two out of four of the analog sensor channels.

d. Block only the output of the 572 psia steam generator pressure trip 2/4 matrices. The High Containment Pressure MSIAS trip function is not blocked.

e. Permission to block and any block initiations are annunciated on the main control board.

Means are provided for the operator to block actuation of the AFAIS at any time. Blocking is effected as follows:

a. Initiated independently for Trains A and B by placing the respective control selector switch in the block position, on the main control board or hot shutdown panel.

b. Any block initiations are annunciated on the main control board.

7.3.3.2 Bistable Trip Bypass

By means of a key lock switch, any one of four redundant ESAS channel bistables can be bypassed such that trip signals in two of the remaining three channels are required to initiate an actuation channel output. The key is retained when the switch is in the bypassed position. Each group of four sensor channel bistables has a single noninterchangeable key with a registered combination such that only one sensor channel of a group is capable of being bypassed at any one time. An alarm is annunciated if any bypass is actuated.

For the AFAIS, the coincidence logic inputs associated with any one of the four redundant channel bistables can be bypassed by means of two key lock switches; one for Train A and one for Train B. In this condition, trip signals in two of the remaining three channels are required to initiate an AFAIS actuation. The keys are retained when the switches are in the bypassed position. The bypass keys are administratively controlled such that only one AFAIS trip channel can be bypassed in both trains of actuation logic at any one time. Train-specific alarms are annunciated if any bypass is actuated.

7.3.4 AVAILABILITY AND RELIABILITY

7.3.4.1 Special Features

Means for continuously monitoring and indicating the ESFAS status are provided by indicating lights on the front of the cabinets. All indicating lights have features for manually checking bulb function.

Means for monitoring the status of the AFAIS is provided by indicating lights on the main control board, RC-517, 518 and hot shutdown control panel and annunciators on the main control board.

7.3.4.1.1 Sensor Channel Surveillance

The functions indicated by a lighted bulb are as follows:

a. Bistable tripped

b. Power supply failure

For the AFAIS, the functions indicated by a main control board annunciator are as follows:

a. Sensor channel bypassed

7.3.4.1.2 Actuation Channel Surveillance

The ESAS functions indicated by a lighted bulb are as follows:

a. Tripped actuation subchannel

b. Sequential actuation system blocking

c. Sequencer tripped

For the AFAIS, the functions indicated by a lighted bulb are as follows:

a. Blocked actuation train

b. Actuation train - power available

c. Actuation train - initiated

d. SPEC 200 micro failure

The functions indicated by main control board annunciation are as follows:

a. Actuation train initiated

b. Actuation train blocked

c. Sensor channel bypassed

7.3.4.1.3 System Test Surveillance

In addition to the indicating lights, a matrix of indicating lights is furnished as part of the ESAS ATI to locate failures.

7.3.4.2 Tests and Inspection

7.3.4.2.1 Testing

ESAS and AFAIS are tested in accordance with plant procedures and Technical Specification requirements. These tests confirm the operability of the ESFAS, final actuated equipment, and all supporting subsystems and power supplies.

7.3.4.2.2 Testing Features

The ESFAS incorporates the following testing features:

a. Bistable trip test

A variable test input signal with an adjustable range consistent with the bistable setpoint range is applied at the analog channel inputs to verify bistable setpoint and operation. Means are provided for indication of proper return to normal operation following completion of test.

b. Actuation channel trip test

Each ESAS coincidence 2-out-of-4 matrix is provided with independent test switches. Operation of the test switches will cause an output of the associated coincidence matrix and trips the related actuation channel logic. The overlapping testing procedure and the arrangement of the matrix ensure that a protective action will occur if any combination of two sensor channels simultaneously trip.

The AFAIS actuation channel trip test is performed by producing an output from the logic matrix which trips the related actuation channel logic.

c. Automatic Test Feature

The ESAS is provided with an automatic test feature which tests all combinations of two of four bistable trip conditions for each parameter. The automatic test feature automatically indicates and identifies logic faults and verifies bistable availability approximately every twenty-seven seconds. Each bistable is tested to check that the bistable trip setpoint is functioning properly, and will process a trip signal. This is accomplished by inserting two test pulses one after the other above and below the trip setpoints. After each test pulse, the output of the bistable is compared to the input by the ATI. A fault is indicated by lamps on the ATI panel and an annunciation on the main control board panel CO1. The time duration of the test pulses are sufficiently short to prevent picking up the output actuation relay. The ATI monitor is automatically turned off during any undervoltage trip condition.

7.3.4.2.3 System Reliability

Components and modules used in the manufacture of the actuation system exhibit a quality consistent with the nuclear power plant 40 year design life objective and with minimum maintenance requirements and low failure rates. ESAS reliability (failure rate) has been specified as equal to or better than 1×10^{-4} with manual testing on a 30 day schedule.

TABLE 7.3.2.2-1 ESAS FAILURE MODE ANALYSIS FOR CONTAINMENT PRESSURE CHANNELS

Postulated Failure	Failure Mode	Method of Detection	Effect on ESAS	Remarks
1. Loss of one containment pressure channel	Bistable goes to tripped state	Annunciation from tripped bistable	No detrimental effect on system	System logic becomes 1-out-of-3 mode.
2. Loss of one 125 volt DC battery with 120 volt regulated AC source operational	Two inverters' loads transfer to regulated AC source	Annunciation	Same as above	
3. Loss of one 125 volt DC. battery with 120 volt regulated AC source inoperative	Loss of power to two sensor channels and one actuation channel	Annunciation	With the exception of SRAs and LNP channels, all 2 out of 4 sensor channels are tripped. Final equipment on the other redundant channel is actuated to its safe state from remaining redundant power source.	Sensor Cabinets remain energized, but associated SPEC 200 Instrumentation Cabinets and radiation monitors de-energize.
4. Loss of one 125 volt DC battery with 120 volt regulated AC available coincident with a failure of a containment pressure channel	1 and 2 above	Separate annunciation of each condition.	1 above	1 above
5. Loss of one 125 volt DC battery with 120 volt regulated AC not available coincident with a failure of a containment pressure channel	1 and 3 above	Same as 4 above	3 above	3 above

FIGURE 7.3–1 ENGINEERED SAFETY LOGIC

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2A ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2B ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2C ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2D ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–3 ENGINEERED SAFETY LOGIC SEQUENCER & CHANNEL 5 OUTPUT

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–4 ENGINEERED SAFETY LOGIC

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.4 REGULATING SYSTEMS

The Reactor Regulating System (RRS), Pressure Regulating System, Pressurizer Level System, and Control Element Drive System (CEDS) are functionally identical to those in the Calvert Cliffs Plants (AEC Docket Numbers 50-317 and 50-318 through Amendment 19) with the exception of the Control Element Assemblies (CEA) motion inhibit which has been added to the CEDS, described in Section 7.4.2 and the RRS which does not incorporate the automatic CEA functions.

7.4.1 REACTOR REGULATING SYSTEM

7.4.1.1 Design Bases

The RRS does not have any automatic CEA control functions. The RRS consists of subsystems which provide the reactor operator with indication of T_{avg}, T_{ref}, and T_{ref}-T_{avg} deviation alarm. The RRS also provides signal inputs to the pressurizer pressure level program and steam dump program.

7.4.1.2 System Description

The RRS is comprised of two functional subsystems. The first includes two independent neutron flux measurement channels and the second subsystem is comprised of various I/O modules connected to a distributed control system (DCS) forming a single fault-tolerant control and indication channel. Two circuits of a non-vital instrument bus provide redundant power to the I/O modules. The system consists of:

a. Steam Dump Program Function Generator;

b. Pressure Level Set Point Function Generator;

c. T_{ref} - Function Generator;

d. T_{avg} - T_{ref} Calculation;

The system includes the following inputs:

a. Loop 1 T_{hot}, Loop 2 T_{hot} signals;

b. Loop 1 T_{cold}, Loop 2 T_{cold} signals;

c. Two first-stage turbine pressure signals.

The system develops the following outputs:

a. T_{ref} and T_{avg} signals to recorders;

b. Deviation alarms for T_{avg} - T_{ref}.

c. Pressurizer level setpoint

d. Condenser steam dump valve control signal (modulating)

e. Quick Open signal to condenser and atmospheric steam dump valves

7.4.1.3 System Operation

A temperature program calculation establishes the desired reactor coolant average temperature reference (T_{ref}) based on a power reference signal from first-stage turbine pressure. T_{ref} varies linearly with power from a nominal temperature of 532°F at hot standby to an adjustable limit of 520°F to 580°F at 100 percent power. T_{AVG} varies linearly with power from a nominal temperature of 532°F at hot standby to a nominal value of 568.9°F at 100 percent power. Operators may select T_{AVG} to be calculated from the Loop 1 temperature inputs, the Loop 2 temperature inputs, or the Loop 1 & Loop 2 temperature inputs combined. A deviation alarm will be generated if the difference between T_{ref} and T_{avg} exceeds the engineering established limit.

A steam dump valve position demand signal is calculated as a function of T_{AVG}. In addition, a quick opening binary signal is calculated as a function of this valve position demand signal; both of these signals are transmitted to the steam dump valve control system upon the initiation of a turbine trip.

A pressurizer level setpoint program is calculated as a function of T_{AVG} and transmitted to the pressurizer level controllers, external to the RRS.

7.4.2 CONTROL ELEMENT DRIVE SYSTEM

7.4.2.1 Design Basis

The reactor is controlled by reactivity adjustments with CEAs and with boric acid dissolved in the reactor coolant. Rapid changes in reactivity are compensated for or initiated by CEA movement. Long-term variations in reactivity due to fuel burnup and fission product concentration changes are controlled by adjusting the boric acid concentration. Since this rate of addition produces slow changes in the reactor power level, operator action suffices to control the boron concentration change. The shutdown CEA group provides a hot shutdown margin of at least 1 percent reactivity, even if the most reactive CEA is stuck out of the core. Prohibits require that the shutdown CEA group is in the full withdrawn position before other CEAs can be withdrawn, thereby assuring a shutdown margin equal to or greater than the required minimum. The CEA motion inhibit and alarm is provided when further insertion of the regulating group of CEAs would reduce the amount of effective shutdown reactivity in the CEAs below specified limits:

CEA movement is effected by the Control Element Drive Mechanisms (CEDM) (see Chapter 3). The CEDS transmits manual signals from the CEDS control panel to the Coil Power Programmers (CPP), which develop the pulses for magnetic jack operation.

CEA withdrawal will be prevented when a high power pretrip alarm or a thermal margin pretrip alarm is present. All CEA motion is inhibited when CEA malposition limits are reached:

7.4.2.2 Design Criteria

a. Racks, panels and associated equipment meet the Electronic Industries Associated Standard RS-310. Standard Nuclear Instrument Modules are utilized as adopted by the United States Nuclear Regulatory Commission (TID-20893).

b. Each switch on the CEDS panel is individually mounted in its own plug-in receptacle to permit fast replacement. The logic in the CEDS logic cabinet is modularized into group and individual CEA modules. All group modules are identical and all individual CEA modules are identical.

7.4.2.3 System Description

A block diagram of the CEDS is shown in Figure 7.4–3.

The CEDS control panel is a selection panel. Three types of selections are made by this panel: control mode, CEA group, and the individual CEA within each group. All selections are made by pressing the appropriate pushbutton switch. Upon selection the switch will light and remain lit and closed until another selection within its scope is made. Eleven selections will always be made: one mode selection, one group selection, and nine individual CEA selections (one in each of the nine groups). Electrical interlocks are incorporated in each of these eleven scopes of selection. This permits only one selection to be made in each scope. A new selection within any scope automatically cancels the previous selection.

There are three different modes of control of CEAs: Manual Individual, Manual Group and Manual Sequential. Two of these modes, Manual Individual and Manual Group apply to both Shutdown and Regulating CEAs. Manual Sequential Mode applies only to the regulating CEAs.

The following limits are provided by the CEA supervisory function of the plant computer to prevent the reactor from reaching undesirable conditions:

a. Upper CEA limit;

b. Lower CEA limit;

c. Upper CEA group stop;

d. Lower CEA group stop.

The CEDS and its associated interfaces with the plant computer and CEA Position Display System (CEA PDS) contain design features that ensure the following actions:

a. Insertion of the regulating CEAs before the shutdown CEAs are inserted;

b. Simultaneous withdrawal of no more than two groups of CEAs;

c. Proper sequential withdrawal of CEAs.

No single equipment or operator failure will cause the CEDS to improperly carry out the above listed actions to the extent that safety limits as defined in the Technical Specifications are reached.

Three lines of defense are utilized to ensure that safety limits are not exceeded. First, the reactor is operated under strict administrative controls which dictate the proper CEA movement. Second, alarms are provided to warn the operator if CEA movement is improper. The third line of defense is the functions and design features described below.

The plant computer CEA supervisory function operates CEA permissive contacts that feed the CEDM logic system. This design feature determines which group or groups of CEAs will be moved in the manual sequential mode. The computer also generates alarms if its logic detects an out of sequence movement of CEAs, more than two-groups movement of CEAs, improper insertion of shutdown CEAs, withdrawal of regulating CEAs prior to withdrawal of shutdown CEAs, or deviation of individual CEAs from their group.

The setpoints of these alarms are chosen such that the operator is given sufficient time to take corrective action before a safety limit is reached without alarming for normally occurring conditions.

If an equipment failure or operator error should cause any of these alarm conditions to be reached, another alarm is also received from the Reed Switch Position Indication System described in Section 7.5.3.3. This second system, diverse in nature from the plant computer CEA supervisory function, provides the operator with continuous indication of the position of each individual CEA and provides alarms redundant to those supplied by the plant computer. In addition, this system provides actuation signals to the CEA motion inhibit circuitry which stops all CEA motion when an alarm set point is reached.

The actuator signals are computed from CEA position information from the reed switch position transducers for each CEA and a reactor power signal (for power dependent insertion limit only). The actuation signals are independent of any control system function that would result in CEA motion. The actuation signals are contact openings that fail open upon loss of power. The actuation signals are banded together and the composite signal is sent to the lift coil power switch for each CEA from contact multiplying relays. The actuation signal to the lift coil power switch will stop any power from being applied to the lift coil, so that regardless of CEA control system motion demand, CEA motion is inhibited.

Actuation of the CEA motion inhibit upon violation of power dependent insertion limits is disabled below 10^{-4} percent power. Power dependent insertion limits are not needed below this power level. The motion inhibit is automatically re-enabled as power increases past 10^{-4} percent.

Equipment protective prohibits are also provided to prohibit regulating group withdrawal and prevent the reactor from reaching undesirable conditions. These interlocks are summarized in Table 7.4-1.

CEA speed is a function of the coil power programmer cycle speed, the CEA control system group speed setting and the mechanical limitations of the magnetic jack mechanism.

The coil power programmer that sequences power signals to the magnetic jack coils of an individual CEA has an upper operating setting of forty inches per minute for regulating CEAs and twenty inches per minute for shutdown CEAs. This maximum speed would result only from a continuous demand for withdrawal from the CEA control system.

A continuous withdrawal signal from the CEA control system would result only during abnormal operating circumstances. During normal operating conditions a continuous demand for an increase in reactor power will result in a sequenced withdrawal of CEAs within a group. The average speed of CEAs within the group is determined by the speed setting of the group programmer in the CEA control system. The upper bound of the speed setting of the group programmer is forty inches per minute, but the normal operating speed setting of the group programmer is the same as that of the individual CEA coil power programmer. Maximum CEA speed is determined by the setting of the individual CEA CPP, and this speed cannot be increased by any setting of the group programmer.

The absolute maximum speed of withdrawal of any CEA is determined by the magnetic jack latching mechanisms and coil current decay time constants. With the present coil power sequencing cycle, the mechanical restraints of latch operating time and coil drop out time allow a maximum possible withdrawal speed of 45-50 inches per minute.

7.4.2.4 System Operation

The CEAs are divided into the following groups:

a. Shutdown: two groups;

b. Regulating: seven groups.

Each CEA remains stationary except when a raise or lower signal is present. In response to a signal, the regulating CEAs move at a speed of up to 40 inches per minute. The shutdown CEAs move at a fixed speed of 20 inches per minute.

The CEA position setpoints are shown on Figure 7.4–2.

The shutdown CEAs may be moved with either manual individual or manual group movement. A selector switch prevents withdrawal of more than one shutdown group at any time. The shutdown groups must be withdrawn above a lower limit before regulating group withdrawal is possible. A limit prevents group insertion of shutdown CEAs unless the regulating groups are fully inserted. These limits may be bypassed to allow the CEA groups to be withdrawn or inserted out-of-

sequence during testing (e.g., rod drop time tests, rod worth measurements, etc.) or system maintenance using approved plant procedures.

Regulating CEA groups may be moved in manual group or manual sequential mode. Individual CEAs may be moved in the manual individual mode of control. Sequential group movement provides that when the moving group reaches a programmed low (high) position, the next group begins inserting (withdrawing); the initial group stops upon reaching its lower (upper) limit. This procedure, applied successively to all regulating groups, allows a smooth and continuous rate-of-change of reactivity.

Under sequential group control, when the regulating groups reach the “prepower dependent” insertion alarm point, this condition is annunciated. If sequential group insertion is continued, a “power dependent” alarm point limit is reached and a second alarm is initiated. These two programmed limits may be adjusted during the life of the plant and are provided to aid the operator in assuring adequate shutdown margin.

All CEAs are prevented from being withdrawn if either a high power or thermal margin/low-pressure (TM/LP) pretrip condition exists.

There is provision for manually bypassing the CEA motion inhibit circuitry at the CEDS operator’s console. The manual bypass requires a minimum of two operator actions to accomplish the bypass. The group(s) to be bypassed must be manually selected by the operator by depressing one pushbutton for each group. Actuation of any group bypass pushbutton results in an alarm at the Main Control Board alerting the operator of the bypass and in the bypass pushbutton illuminating red. All control pushbuttons on the console illuminate white with the exception of the “off” pushbutton that also inhibits all CEA motion and illuminates red. In addition to selecting the group to be bypassed, the operator must also depress the “CEA motion inhibit bypass” button to accomplish the bypass function. This pushbutton must be held depressed by the operator during the bypass operation, as release of this pushbutton will cause the bypass to be removed. Actuation of the “CEA motion inhibit bypass” pushbutton will also cause an alarm at the Main Control board, alerting the operator of the bypass, and in the pushbutton illuminating red.

7.4.2.5 Consequences of Single Failures

The Reactivity Control System contains two independent and diverse CEA position supervisory and control systems.

The pulse count system uses pulse counting techniques in the Plant Computer to determine and maintain a current record of the position of each CEA. These CEA positions are used to derive CEDS control inputs and CEA misalignment alarms. These alarms include regulating group sequencing and overlap violations, excessive CEA deviation within any group and excessive insertion of any regulating group below the prepower or power dependent limits. These alarms are outside the normal operation region of the CEDS and warn the operator that the CEAs are in an abnormal configuration while they are still well within fuel design limits. Each alarm is annunciated separately at the main control board.

The CEAPDS derives its CEA position information from a Reed Switch Position Transmitter (RSPT) which is mounted on each CEDM. The RSPT is a magnetically operated voltage divider which outputs an analog signal proportional to CEA position. This system provides a continuous display on the main control board of all CEA positions and monitors the positioning of CEAs to provide the same alarms as detailed in the description of the digital system. The CEAPDS only control function is to initiate a CEA motion inhibit if any one or combination of these alarm conditions are reached. As with the pulse count system, each alarm is annunciated separately at the main control board. The system is designed such that on a loss of electrical power each alarm is annunciated and CEA motion inhibit is initiated, (see Figure 7.4–9).

A single failure, for this response, has been defined as a component failure in either the active (energized) state or in the inactive (deenergized) state of the component, or a short circuit or open circuit in any signal line.

The Reactivity Control system is shown in block diagram form in Figure 7.4–8. The system has been separated into five functional blocks.

Functional block number 1 represents the integrated circuit logic elements and associated power supplies that determine motion demand for each of 61 CEDMs. The outputs of this block are raise and lower contact closure signals to the CPP and contact opening signals for the motion inhibit block circuit.

Single component failures within Figure 7.4–8 functional block number 1 may cause spurious raise or lower signals to the CPP timers. This spurious demand signal will cause CEA motion. This motion is monitored by the Control Element Assemblies Position Display System (CEAPDS) (functional block number 4). The CEAPDS monitors CEA deviations (highest to lowest CEA), out of sequence motion, violation of allowed CEA group overlap, power dependent insertion limits, shutdown group insertion prohibit and regulating group withdrawal permissives. When the CEAPDS senses motion that is in violation of the above conditions, the calculator will output a contact opening causing the motion inhibit relays to deenergize. When the motion inhibit relays deenergize they open a contact in the control circuit of the lift coil power switch of each CEDM, preventing the silicon controlled rectifiers from energizing. Deenergizing the lift coil power switch control prevents further motion, either insertion or withdrawal, regardless of demand from the CPP or CEDS logic. This inhibit does not affect the reactor trip capability of the CEAs as the reactor trip function will interrupt motive power input to all power switches, causing gravity insertion of all tripping CEDMs.

The setpoints of the CEAPDS are sufficiently conservative that the fuel design limits will not be reached for motion of any CEA or combinations of CEAs.

A single failure in functional block number 1 may also cause a spurious “off” signal to a lift coil power switch. This failure will cause the affected CEA to hold, regardless of motion demand. If motion is demanded of the group containing the affected CEA, the CEAPDS will sense the CEA deviation and cause all other CEA motion to be inhibited. The failure within functional block No. 1 cannot prevent the action of the inhibit circuitry since the circuitry of function block No. 1 is

isolated from the inhibit circuitry by the use of relays that provide inherent coil to contact isolation, as shown on Figure 7.4–9.

A single failure within functional block number. 2 may cause a single CEA to raise or lower or drop. This failure will be sensed by the CEAPDS as a CEA deviation and the CEAPDS will cause all CEA motion to be inhibited as detailed above, before fuel design limits are reached.

A single failure within functional block number. 3 may cause a CEA to be held if it occurs within the lift coil power switch or a single failure may cause a single CEA to drop or hold if it occurs in one of the other coil power switches. These failures will be sensed by the CEAPDS as a CEA deviation and the CEAPDS will cause all CEA motion to be inhibited before fuel design limits are reached.

A single failure within functional block number 4 may cause a spurious motion inhibit signal which will block all CEA motion. This failure will not cause fuel design limits to be reached since all CEA motion is prevented.

A single failure within functional block number 4 may also prevent actuation of the motion inhibit signal. This failure, however, will not cause fuel design limits to be approached as it cannot cause CEA motion.

The integrity of the CEAPDS will be assured by periodic testing.

A single failure within function block number 5 may cause the motion of some or all CEAs to be inhibited spuriously. The CEAPDS will sense this failure as a CEA deviation and cause the remainder of the motion inhibit relays to deenergize, inhibiting the motion of all CEAs before fuel design limits are reached.

A single failure within functional block number 5 may also prevent the actuation of the motion inhibit signal for some or all CEAs. This failure will not cause fuel design limits to be approached as it does not affect CEA motion. The integrity of the motion inhibit relays is assured by periodic testing.

Since functional block number 5 is implemented with relay logic, there are no single failures which can both affect CEA motion and prevent the actuation of the motion inhibit signal.

As shown in detail above, no single failure in any of the five functional sections of the reactivity control system will cause fuel design limits to be reached.

7.4.3 REACTOR COOLANT PRESSURE REGULATING SYSTEM

7.4.3.1 Design Bases

The reactor coolant pressure regulating system is functionally identical to that in the Calvert Cliffs Plant (NRC Docket Numbers 50-317 and 50-318).

The reactor coolant pressure regulating system maintains system pressure within specified limits by the use of pressurizer heaters and spray valves. Pressurizer pressure and level sensors provide input to the system.

7.4.3.2 System Description

A high pressurizer pressure functions to open the pressurizer spray valves on a proportional basis, thereby reducing pressure. A low pressurizer pressure functions to energize heaters on a proportional or group basis to increase pressure. A high pressurizer water level energizes the backup heaters in anticipation of a low-pressure transient; a low pressurizer water level de-energizes all heaters, for heater protection.

Two channels of control are provided and the controlling channel is selected by a switch. Manual control of the heaters and spray may be selected at any time.

7.4.3.3 System Operation

Two pressure channels independent of those in the Reactor Protective System (RPS) provide suppressed range (1500 to 2500 psia) signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function. During normal operation, a small group of heaters is proportionally controlled to maintain operating pressure. If the pressure falls below the proportional band all of the heaters are energized. Above the normal operating range the spray valves are proportionally opened to increase the spray flow rate as pressure rises. A small, continuous spray flow is maintained through the spray lines at all times to keep the lines warm to reduce thermal shock when the control valves open, and to aid in keeping the boric acid concentration in the coolant loops and pressurizer in equilibrium.

Outputs from the two pressure control channels are recorded in the control room and provide individual high and low alarms.

The control and alarm pressure setpoints are shown in Figure 7.4–4.

7.4.3.4 System Evaluation

Two individual channels are available for automatically regulating the pressurizer heaters and spray valves. Either channel may be used to control the pressure in the system, and the output from both channels is recorded in the control room. Individual high and low pressure alarms are provided.

7.4.4 PRESSURIZER LEVEL REGULATING SYSTEM

7.4.4.1 Design Bases

The pressurizer level regulating system is functionally identical to that in the Calvert Cliffs Plant (NRC Docket Numbers 50-317 and 50-318).

Pressurizer level is maintained by the action of the chemical and volume control system (see Section 9.2). The level set point is programmed as a function of coolant average temperature (T_{avg}). A low pressurizer level signal functions to reduce letdown flow proportionally and to start the available nonoperating charging pumps. A high level indication functions to increase letdown flow proportionally by opening the letdown control valves and stopping all but one charging pump. There are two independent automatic control channels with channel selection accomplished by means of a manual control switch. Automatic control is normally used during operation but manual control may be utilized at any time.

7.4.4.2 System Description

Two level channels provide pressurizer level signals for control of two specific functions:

a. A low level signal from either channel deenergizes all heaters;

b. A high level deviation signal from the controlling channel energizes the backup heaters and sets the proportional heater control to full power.

7.4.4.3 System Operation

The operating level in the pressurizer is programmed as a function of power to accommodate plant load changes and transients to minimize the changes in reactor coolant system (RCS) volume (see Figure 4.3–9).

The level programmer establishes a program level which is directly proportional to coolant average temperature, over the operating range of T_{avg}. The average temperature signal used by the level programmer is the signal used by the RRS.

The level controller compares the measured and programmed level signals and generates a proportional signal for regulating the letdown control valves. In addition, the level controller functions to start or stop additional charging pumps at low or high level set points. The outputs of either of two automatic control channels may be selected by the operator for level control in addition to manual control.

7.4.4.4 System Evaluation

Two level control systems are provided. The controllers are located in the control room. Both automatic and manual control of level is provided. Three charging pumps and two letdown control valves provide redundant means of increasing or decreasing reactor coolant inventory. The variable pressurizer level control program maintains the proper coolant inventory by means of discharge or addition as required during plant load changes.

7.4.5 STEAM DUMP AND TURBINE BYPASS SYSTEMS

7.4.5.1 Design Bases

7.4.5.1.1 Functional Requirements

a. The steam dump to atmosphere systems (one each for the two steam generators) provide means for unit cooldown in the event that the condenser is not available.

b. The turbine bypass system provides a means for removing decay heat and pump heat at no load conditions and a means for unit cooldown.

c. The steam dump to condenser, steam dump to atmosphere, and bypass systems combined provide a means of dissipating excess Nuclear Steam Supply System (NSSS) stored energy and sensible heat following a simultaneous reactor and turbine trip from full load without lifting the secondary safety valves.

7.4.5.1.2 Design Criteria

The steam dump to atmosphere systems will have a minimum combined capacity of 15 percent of full load in order to meet the functional requirements of a.

The bypass system will have a minimum capacity of 5 percent of full load in order to meet the requirements of b.

The total steam dump to condensers, steam dump to atmosphere and turbine bypass system will have a minimum capacity of 55 percent of full load in order to meet the functional requirements of c.

The steam dump to atmosphere control valves can be controlled from the main control board or the hot shutdown panel external to the control room.

The steam dump to condenser control system, when in automatic operation, shall be limited in reactor coolant cooldown capability.

The steam dump to atmosphere system for one steam generator shall be redundant and independent of that for the other steam generator.

7.4.5.2 System Description

A block diagram of the steam dump to the condenser and turbine bypass system is shown on Figure 7.4–5.

The steam dump to condenser and turbine bypass valves can be controlled at their respective control room stations in either automatic or manual mode.

The steam dump to atmosphere control valves can be controlled from the main or auxiliary panels. Automatic and manual modes are available on the main control panel and manual mode is available on the hot shutdown control panel. The steam dump to atmosphere system for steam generator number 1 and steam generator number 2 reside partially on a distributed control system utilizing common fault tolerant control processors. Pressure sensor and control feedback input signals and control output signals are independent of each other.

The total steam dump and turbine bypass is sufficient to prevent the lifting of the secondary steam safety valves following a simultaneous reactor and turbine trip at full power. The capacity of each of the steam dump to condenser and turbine bypass valves is approximately 10 percent of full load for a total of approximately 40 percent of full load. The capacity of the steam dump to each atmosphere valves is 7.5 percent of full load, each, for a total of 15 percent of full load. The capacity of the steam dump to the atmosphere system is sufficient for plant cooldown after a full load trip even if for any reason the condenser is not available.

Excessive cooldown of the RCS by the steam dump to condenser valves, when in automatic operation, is prevented by a narrow range temperature signal which has a minimum output corresponding to 532°F. At this point, the steam dump to condenser flow demand will be zero while the bypass valve remains in operation to control header pressure. The turbine bypass system will tend to control the steam pressure to 900 psia during hot standby when the condenser is available.

7.4.5.3 System Operation

Steam is discharged from the main steam lines to the condenser by way of the dump to condenser and bypass valves in response to T_{avg} and secondary pressure signals. Inputs to the system are the T_{avg} turbine trip signal, main steam line pressure and condenser loss of vacuum.

Upon receipt of a turbine trip signal the steam dump to condenser controller generates a suppressed range signal proportional to the quantity T_{avg} -532°F. When T_{avg} is within a predetermined range, the signal is sent to the steam dump to condenser valves and the turbine bypass auctioneering unit via the steam dump to condenser vacuum permissive contacts. The turbine bypass pressure controller generates a suppressed range signal proportional to secondary pressure over the range of 800 to 1000 psia. The turbine bypass valve receives the higher of the steam dump to condenser controller or turbine bypass controller signals through an auctioneering unit. When T_{avg} exceeds the predetermined range, a quick opening signal from the RRS opens the steam dump to condenser valves and simultaneously opens the turbine bypass valve. In either case, loss of condenser vacuum will prevent opening of the turbine bypass or steam dump to condenser valves.

If it is necessary to boost the capacity of the bypass system near the end of cooldown, the dump to condenser valves may be operated on manual control. The operator may control plant cooldown by combined manipulation of the steam dump to condenser and turbine bypass controllers, as required.

The steam dump to atmosphere system is set to a preselected dump pressure below safety valve setting and above bypass system setting on the main control panel controller. The controller is equipped with proportional mode only to give rapid response without reset windup, while maintaining its capability to modulate smoothly. Standard switching procedures on the local panel transfer station allow the operator to control the steam dump to atmosphere manually with an adjacent pressure indicator for guidance. Either the automatic controller setpoint or the local station on manual control may be used for cooldown.

7.4.6 TURBINE GENERATOR CONTROL SYSTEM

7.4.6.1 Design Basis

7.4.6.1.1 Functional Requirements

The turbine generator is supplied with an electrohydraulic control (EHC) system. The EHC system provides speed and acceleration control for startup, load control, load limiting, emergency trip, manual trip and valve testing functions.

7.4.6.1.2 Design Criteria

The EHC system is designed to provide dependable and accurate startup, load control and emergency trip functions. Redundant electronic speed sensing for control and high pressure hydraulic fluid supply systems are incorporated to increase system reliability. A primary electronic overspeed protection system is provided with an independent emergency electrical overspeed protection system.

7.4.6.2 System Description

7.4.6.2.1 System

The General Electric Company EHC system consists of electrohydraulically operated turbine control and stop valves operated from an electronic control cabinet and an operator’s panel located on the main control board. The operator’s panel contains controls for startup, load control, operational testing of valves and overspeed trip system and manual trip. System alarms and indicators are provided for monitoring system operation.

The EHC system includes the following features:

a. Full arc admission

b. Manual set load limit

c. Initial pressure regulator

d. Chest warming control

e. Load set capability

The control and stop valve operators receive signals from the EHC control cabinet and use fire resistant hydraulic fluid for positioning.

A high-pressure fluid power unit supplies fluid under 1600 psi pressure for operation of servo valve power actuators. The fluid power unit consists of two redundant pumps, coolers and filters arranged in parallel.

Electric power for the electronic logic and trip circuits is provided by two redundant battery-backed uninterruptible power supplies. All power supplies are monitored to provide indication of out of tolerance voltage.

Three redundant primary speed signals are provided to permit speed and acceleration control using the median of the three speed signal values. An independent set of three redundant emergency speed signals are used for emergency overspeed protection. Failure of a single speed signal will result in a system alarm. The turbine will be automatically tripped should a loss of two primary speed signals or a loss of two emergency speed signals occur. Tripping is initiated at nominal 109 percent of rated speed. An emergency electrical overspeed trip circuit is provided to trip the turbine in the unlikely event of a failure of the primary overspeed trip system or during an overspeed trip test. The emergency electrical overspeed trip system is independent from the primary speed control and overspeed protection system and is set to trip at nominal 109.5 percent of rated speed.

The primary overspeed trip and the emergency overspeed trip systems constitute two separate and independent means of protecting the turbine against an overspeed.

7.4.6.3 System Operation

7.4.6.3.1 Startup

The startup sequence followed by the operator consists mainly of the following steps which are controlled from the turbine generator control panel, CO7:

a. High-pressure turbine chest warming

b. Increase turbine speed to 1800 rpm in discrete steps

c. Perform overspeed trip test

d. Synchronize generator to the line and close the generator breaker

e. Establish initial loading rate

f. Set load limit and initial pressure limit

g. Increase unit loading to final desired value

7.4.6.3.2 Normal Operation

Under normal operating conditions, with no variables out of limit, the EHC system automatically maintains the desired unit load set by the operator. Also see Section 10.2.3.2.

7.4.6.3.3 Abnormal Operation

The turbine is automatically tripped on the following signals:

a. Turbine overspeed (primary or emergency systems)

b. Low condenser vacuum

c. Excessive thrust bearing wear

d. Deleted by FSARCR 05-MP2-023

e. Loss of generator stator coolant

f. Low bearing oil pressure

g. Loss of two primary OR two emergency speed signals

h. Deleted by FSARCR 06-MP2-007

i. High steam generator level

j. Loss of primary and secondary control power

k. Low hydraulic fluid pressure

l. Low shaft pump discharge pressure

m. Main generator and transformer protection system trip

n. Reactor trip

o. Power to Load Unbalance

Four mechanically and electrically isolated pressure switches are furnished on the EHC hydraulic trip system to provide four redundant channels to the RPS. The four pressure switches are combined into 2 out of 4 logic in the RPS to trip the reactor on a turbine trip.

7.4.6.4 Availability and Reliability

The EHC system is designed with highly reliable components. Maximum use is made of solid state components in measuring and logic circuits. The EHC control cabinet is completely factory

assembled, wired, tested and calibrated prior to shipment to the job site. Availability is increased through the use of redundant electrical and hydraulic power supplies.

7.4.7 FEEDWATER REGULATING SYSTEM AND FEEDWATER PUMP SPEED CONTROL

7.4.7.1 Design Bases

7.4.7.1.1 Functional Requirements

The feedwater regulating system (one for each steam generator) is designed to regulate the flow of feedwater to:

a. Maintain an inventory of water in the steam generator as required by the power production rate.

b. Assure a heat sink for the primary system.

The feedwater pump speed control is designed to maintain the speed of the feedwater pumps (2) as required by steam demand.

7.4.7.1.2 Design Criteria

Criteria for the design of the feedwater regulating system include the following:

a. Provide a means for both manual and automatic operating modes from the main control board.

b. Have proportional and reset control actions.

c. Provide automatic ramping to close the main feedwater regulating valves on turbine trip.

d. The control room portion of the system is to be factory wired and tested and shipped as a unit (one for each generator).

e. Provide recorder and computer display of steam generator level, steam flow and feedwater flow.

f. Provide a means for locking main the feedwater regulating valves in their last position on air and controller signal failure.

Criteria for the design of the feedwater pump speed control include the following:

a. Provide a means for both manual and automatic operating modes from the main control board on an individual pump basis.

b. Provide a redundant control architecture to minimize the effects of single point failures.

7.4.7.2 System Description

7.4.7.2.1 System

The feedwater regulating system inputs are steam generator level, steam flow, and feedwater flow. Each of these parameters is measured with differential pressure type instrumentation.

The Feedwater Control System for the Millstone Nuclear Generating Station - Unit 2 is based on utilizing SPEC 200 — a simplified package for electronic control. SPEC 200 provides a highly reliable complement of control room instrumentation.

In addition, the SPEC 200 control equipment is type-tested for qualification per IEEE Standards 323-1974 and IEEE 344-1975. These type-tests establish that the equipment can properly continue to perform its safety related functions before, during, and after specified design basis events. Such events include seismic disturbances that are considered by Foxboro to represent the most severe that would be anticipated at most nuclear power plant locations, as well as postulated changes to environmental conditions at the equipment site, both normal and abnormal.

The SPEC 200 Feedwater Control System has been configured to control the flow of feedwater into the steam generators, thus maintaining the water level within the desired range during all phases of plant operation. The control system includes both single and three element control to maintain steam generator level by positioning of the feedwater regulating valves.

The SPEC 200 analog instrumentation has been configured to meet the following functions:

- Single Element Control
- Three Element Control
- Main Feed Valve and Bypass Valve Control
- Process Measurement

A major consideration in meeting the above functions is to eliminate inadvertent plant trips due to the Feedwater Control System. Specifically, the effects due to transmitter failure, power supply disruption and operator error are taken into account and minimized.

All instrumentation is mounted in a total of two (2) SPEC 200 Racks, each rack will contain the same instrumentation with one (1) rack for each of the two Feedwater Control Elements.

All instrumentation is Class II (seismic only) as applicable to a Nuclear Power Plant.

SINGLE-ELEMENT LEVEL CONTROL

The single-element control consists of one control unit which receives the level signal and an external generated setpoint signal (from the “master controller”) and produces an output signal as a function of the difference between these signals. The rate and magnitude of the output signal change is determined by the control function. The single-element control also has an input from the RCS cold leg temperature to provide dynamic compensation for steam generator level changes at a low power level.

This control unit maintains steam generator level at a manually set value when steam flow is less than 15 percent.

The single-element control units’ output will track the output of the three element control whenever the system is on three element control.

This tracking feature is accomplished by forcing or switching both the high and low limits of the controller to be equal to the three element controller output.

THREE-ELEMENT FEEDWATER CONTROL

This is a three-element system in which the primary or level control unit functions to trim level and the secondary or feedwater control unit functions to maintain the balance between feedwater and steam flow.

Feedwater flow demand is set by steam flow with corrections added for changes in level. An increase in steam flow automatically increases the demand for feedwater flow. To compensate for changes in level due to shrink and swell, blow down or any other cause, the level control unit trims the feedwater setpoint signal in such a manner that the steam generator level is always returned to its setpoint.

The level control unit compares a measurement of level with its manually adjusted setpoint to develop an output signal. This signal is fed to one input of the computing unit where it is combined with 50 percent bias. When level is at the setpoint, output from the control unit will settle at about 50 percent to combine with the computing unit bias. Whenever the level is not at the control setpoint, the level control unit will alter the feedwater control unit setpoint to trim the flow of feedwater to the steam generator. Trimming action will continue until the level returns to the control setpoint.

The feedwater control unit compares a measurement of feedwater flow with the trimmed steam flow signal to develop an output signal to regulate the feedwater valve. Any change in steam or feedwater flow is immediately sensed as a discrepancy in the actual feedwater/steam flow relationship by the level control unit. This control unit, tuned to the response of the feedwater loop, will correct the feedwater flow to restore the feedwater/steam flow relationship.

The measurement input to the feedwater control unit is fed to the reset circuit of the level control unit to prevent reset windup of the primary control unit when the control station is in manual control, or when the level control unit cannot follow its setpoint.

Three element feedwater control is the most stable and dynamically responsive steam generator level control scheme that can be provided. This system maintains a mass balance between measured feedwater flow and steam generator level setpoint error compensated steam flow. The steam generator level setpoint error signal is added to the measured steam flow value in such a manner as to reduce steam generator level setpoint errors to a minimum amount. This system does not require an upset in steam generator level for the feedwater control to be modified. Note that whenever measured steam flow changes, there is an immediate change in feedwater flow, thereby inhibiting the steam generator level upset.

Note that measured steam flow is an input into the three element feedwater control system. Steam demands placed on the main feedwater system by steam generator blowdown, atmospheric dump valves (ADVs), and steam generator relief valves are not direct inputs into this mass balance equation. The slight steam generator level error caused by the continuous steam generator blowdown is compensated for by the steam generator level setpoint error signal. Large steam demands placed on the system by ADVs or relief valves opening cannot be adequately compensated for in this system. These conditions will result in significant steam generator level fluctuations and possible trips due to low steam generator water level. The main feedwater control system was designed to respond to routine plant transients and feedwater perturbations. This system was not designed to ameliorate accidents or respond to all off-normal events.

The transfer between three element and single element control will be accomplished automatically. The transfer from single-element to three-element occurs at 15 percent load, (900,000 lb/hr.) increasing. A 4 percent deadband is provided for stability which means the transfer back from three-element to single-element will occur at 11 percent load, (660,000 lb/hr.) decreasing. A tracking network is provided for bumpless/balanceless transfer between three-element and single-element control. The tracking is accomplished by switching the high and low limits to the output of the single-element controller.

As a security feature, the feedwater control system is provided with redundant differential pressure transmitters for feed and steam flows. One transmitter is arbitrarily designated as a “Main” and the other as a “ALT.” Failure of either redundant transmitter will be indicated on the main control board. Failure is a signal greater than 102 percent or less than minus 2 percent of full scale.

In order to minimize the impact on the feedwater control system of a transmitter failure, the average of the steam flow signal and the average of the feed flow signal will be calculated and used for control purposes. The operator has the ability to select either the “Main,” “Alt,” or “Both” transmitters on a main control board mounted selector switch. Selecting the “Both” position on the selector switch will average the signals.

As with steam and feedwater flow, the steam generator level measurement is redundant; a “Main” and “Alt.” When the selector switch is in the “Both” position the SPEC 200 control circuitry will automatically select the “low” steam generator level signal to provide the most conservative level measurement signal to the control system. A transmitter failure will be indicated on the main control board.

The main feedwater regulating valves are locked in their last position in the event of air failure to the valve positioner. The valves are also locked in the event of a low or no signal from the main control board controller to the valve positioner. These features are independent on the control systems for each of the steam generators.

A schematic and block diagram of the control system is shown in Figures 7.4–6 and 7.4–7.

The feedwater pump speed control system sets feedwater pump speed based on steam flow. The speed setpoint is high selected with a discharge pressure setpoint to ensure a minimum discharge pressure is maintained. The control system also features a high discharge pressure limiter.

7.4.7.2.2 Components

The feedwater regulating systems consists of differential pressure measuring devices for the steam generator level, steam flow and feedwater flow.

The main control room components, with the exception of display and switching devices, are contained in a factory wired, tested and packaged system. The system for each steam generator is independent from the system for the other generator.

The parameters of steam flow and feedwater flow are recorded on a recorder on the main control board. The steam generator level is recorded on a recorder on the main control board.

A control station for each generator includes provision for manual-automatic bumpless transfer switching and display of the generator level signal and is mounted on the main control board. This controller includes the steam generator level setpoint mechanism and displays a continuous indication of the output signal to the feedwater control valve.

The feedwater pump speed control system consists of measuring devices for pump speed, suction flow, suction pressure, and discharge pressure. Each controller is mounted in a separate cabinet which also houses a two-line display panel. The control panel for the feedwater pump speed control system is mounted on the main control board. It allows for auto/manual transfer, speed adjustment, and indication of major pump parameters. A hydraulic actuator is used to manipulate the SGFPT steam admission valves.

7.4.7.3 System Operation

The feedwater regulating system for each of the steam generators has a level control station titled steam generator level controller. The operator has provisions for automatic or manual level control with appropriate bumpless transfer switching and display of the generator level signal and the controller output signal to the respective feedwater control valve.

Each of the parameters of steam generator level, steam flow and feedwater flow is recorded on the main control board for operator guidance.

The feedwater pump speed control system incorporates features which allow the operator to control the system in manual mode. The speed of the feedwater pump(s) is adjusted by speed lower and speed raise buttons to the main control board.

7.4.7.4 Availability and Reliability

In the feedwater regulating system all equipment is designed with highly reliable components. Maximum use of solid state components in the electronic instruments and, where possible piston-operated valves are employed to fail locked (as-is) on loss of air or power.

All instrumentation and controls, where practical, are installed outside of the containment structure and in locations accessible for inspection and maintenance.

As mentioned, the feedwater valves can be placed on manual control from the main panel. Feedwater valve bypass valve can also be actuated from the main panel.

The feedwater pump speed control system uses three isolated control modules, each with its own power supply, processor, and input/output cards. The software coordinates data exchange between modules to support double and triple redundant processing. This design of the feedwater pump speed control system is highly reliable as it provides a high level of fault tolerance and detection.

With the provision for manual operation from the main control board, it is possible to remove the majority of the electronic circuitry for repair or replacement. Operation on manual control is equivalent to having fixed speed turbines in this service.

7.4.8 REACTOR COOLANT SYSTEM LOW TEMPERATURE OVERPRESSURIZATION PROTECTION (LTOP) SYSTEM

7.4.8.1 Design Bases

The LTOP system consists of two redundant relief trains each with a power operated relief valve (PORV) and associated relief piping. This system is controlled through a series of pressure and temperature actuated devices and hand controlled switches in the main control board and it is powered by independent and redundant power supplies. The LTOP system when operated as described below, ensures that the limiting mass addition transients due to an inadvertent start of a high pressure safety injection (HPSI) and/or a charging pump or the limiting energy addition transient associated with the start of a reactor coolant pump (RCP) do not result in peak pressurizer pressures in excess of the 10 CFR 50, Appendix G beltline limits (normal operation) developed for up to 54 EFPY. The LTOP system is required to be operable and capable of mitigating RCS pressure transients whenever the RCS cold leg temperature is ≤275°F, which exceeds the recommended temperature provided by Branch Technical Position RSB 5-2, unless the RCS is properly vented with a vent size ≥ 2.2 in^2. Under normal conditions, the LTOP system relies on the two pressurizer PORVs with a setpoint of ≤ 400 psig.

7.4.8.2 System Description

The LTOP system utilizes a combination of automatic activation devices, manual handswitches, and control board alarms to alert the operators and to automatically open the power operated relief valves in the event of an overpressure transient. An alarm which is set sufficiently lower than the PORV setpoint will provide operator warning of an ongoing RCS pressure increase prior to PORV opening.

The evaluations performed to demonstrate that the flow relieving capability of the PORVs is adequate to mitigate the corresponding design bases transients include system corrections. These corrections include the valve overshoot resulting from the PORV stroke time and system process delay time, fluid conditions at the valve inlet, RCS hydraulic effects resulting from RCS flow and elevation differences in addition to the PORV loop instrument uncertainties. These corrections ensure that the 10 CFR 50, Appendix G limits are not exceeded during the following pressure transients:

a. Energy Addition - The transient is caused by the start of an RCP with a maximum secondary to primary side temperature differential of 50°F. Since the expansion of the primary coolant resulting from the heat addition varies with the RCS pressure and temperature conditions, a maximum RCS pressure of 340 psia and a maximum RCS temperature of 275°F were chosen to ensure that the entire LTOP temperature range is bounded by the analysis.

   The first RCP is not allowed to start unless a pressurizer bubble of ≥ 900 cubic feet is established to ensure that adequate RCS volume is available to accommodate any potential primary coolant expansion without resulting in water 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 10 CFR 50 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 (T_c) is 190°F < T_c ≤ 275°F or when T_c 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 $T_c \leq 190°F$.

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.

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 deenergized 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.

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 reestablish the required SDC flow following the start of an RCP without exceeding the allowable pressure/temperature limits.

TABLE 7.4-1 PROHIBITS ON CEAS

Withdrawal Prohibit Condition

Pretrip Overpower

Thermal Margin / Low-Pressure Pretrip

Motion Inhibit condition

Group out of sequence movement

Individual CEA deviation

More than two Regulating group movement

Withdrawal of Regulating groups prior to complete withdrawal of all Shutdown groups

Insertion of Shutdown groups prior to insertion of all Regulating groups

Input signal sources for the withdrawal prohibits:

Reactor Protective System: Pretrip Overpower Thermal Margin /Low-Pressure Pretrip

Input signal source for CEA motion inhibit:

Reed switch CEA Position Display System (CEAPDS)

FIGURE 7.4–1 DELETED BY FSARCR 05-MP2-026

FIGURE 7.4–2 CEA POSITION SETPOINTS

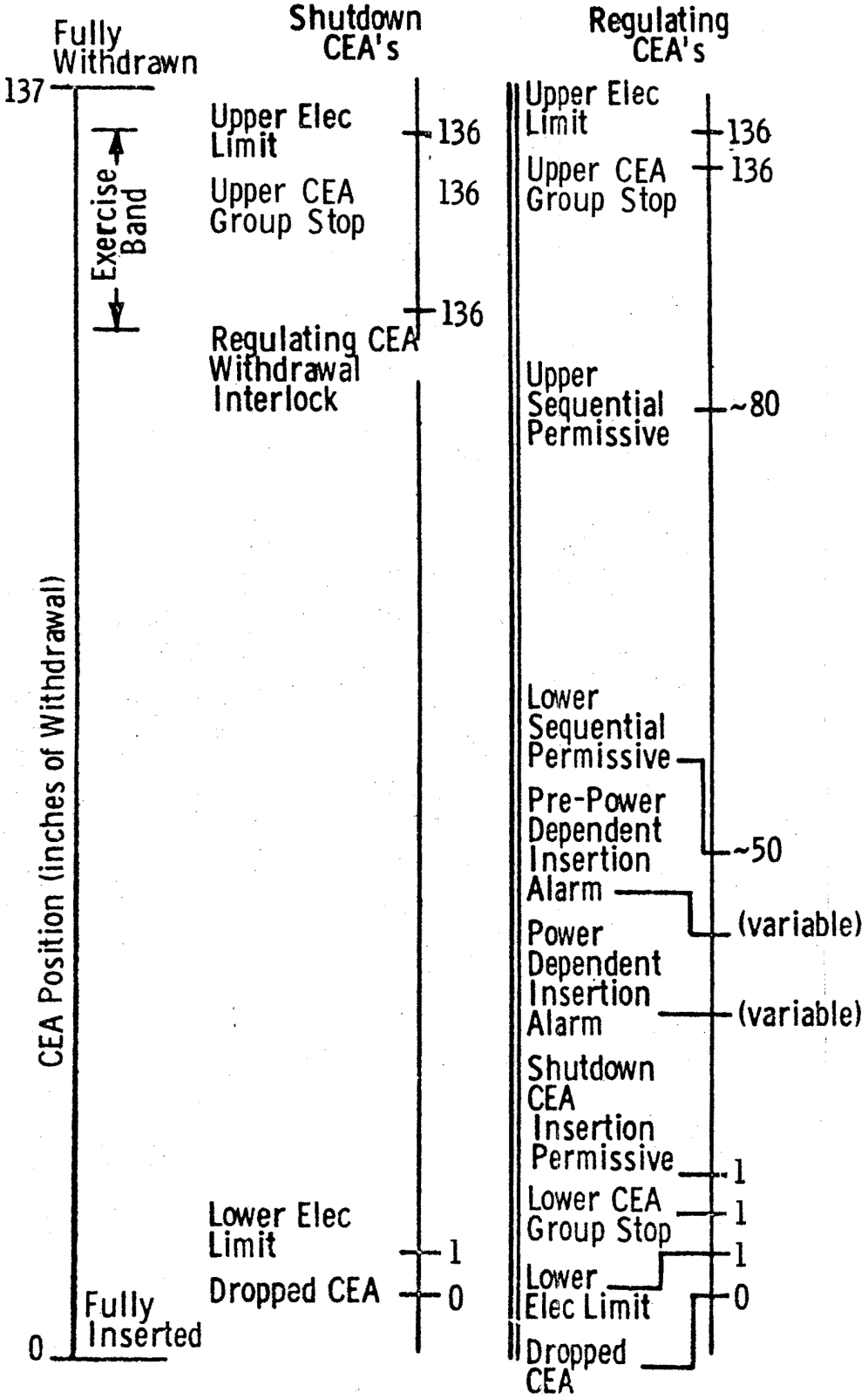
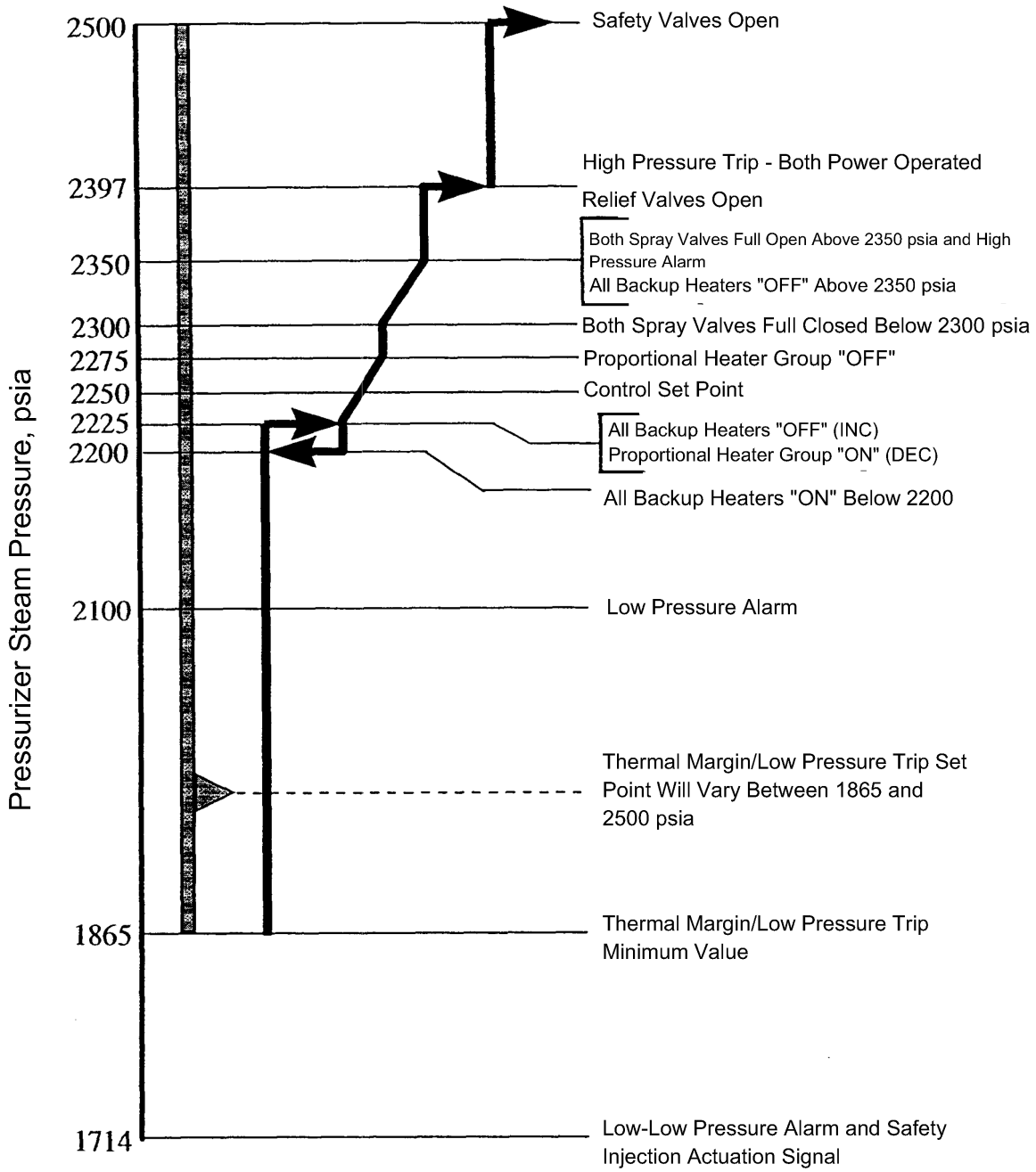
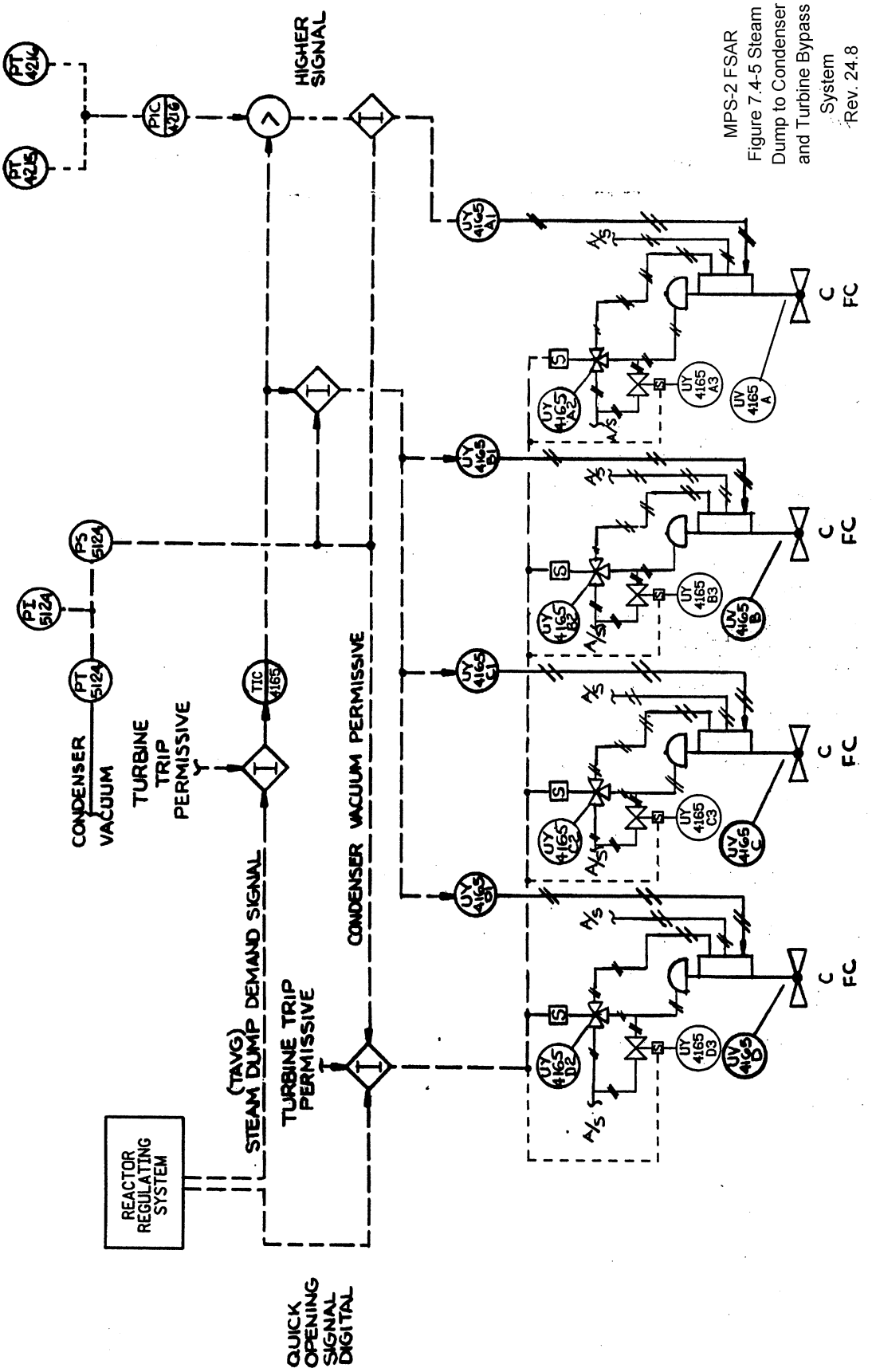


FIGURE 7.4–3 CEDS FUNCTIONAL BLOCK DIAGRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–4 PRESSURE CONTROL PROGRAM





MPS-2 FSAR
Figure 7.4-5 Steam Dump to Condenser and Turbine Bypass System
Rev. 24.8

FIGURE 7.4–5 STEAM DUMP TO CONDENSERS AND TURBINE BYPASS SYSTEM

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 5)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 6)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 7)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 8)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 9)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 10)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–7 FEEDWATER REGULATING SYSTEM BLOCK POWER DISTRIBUTION

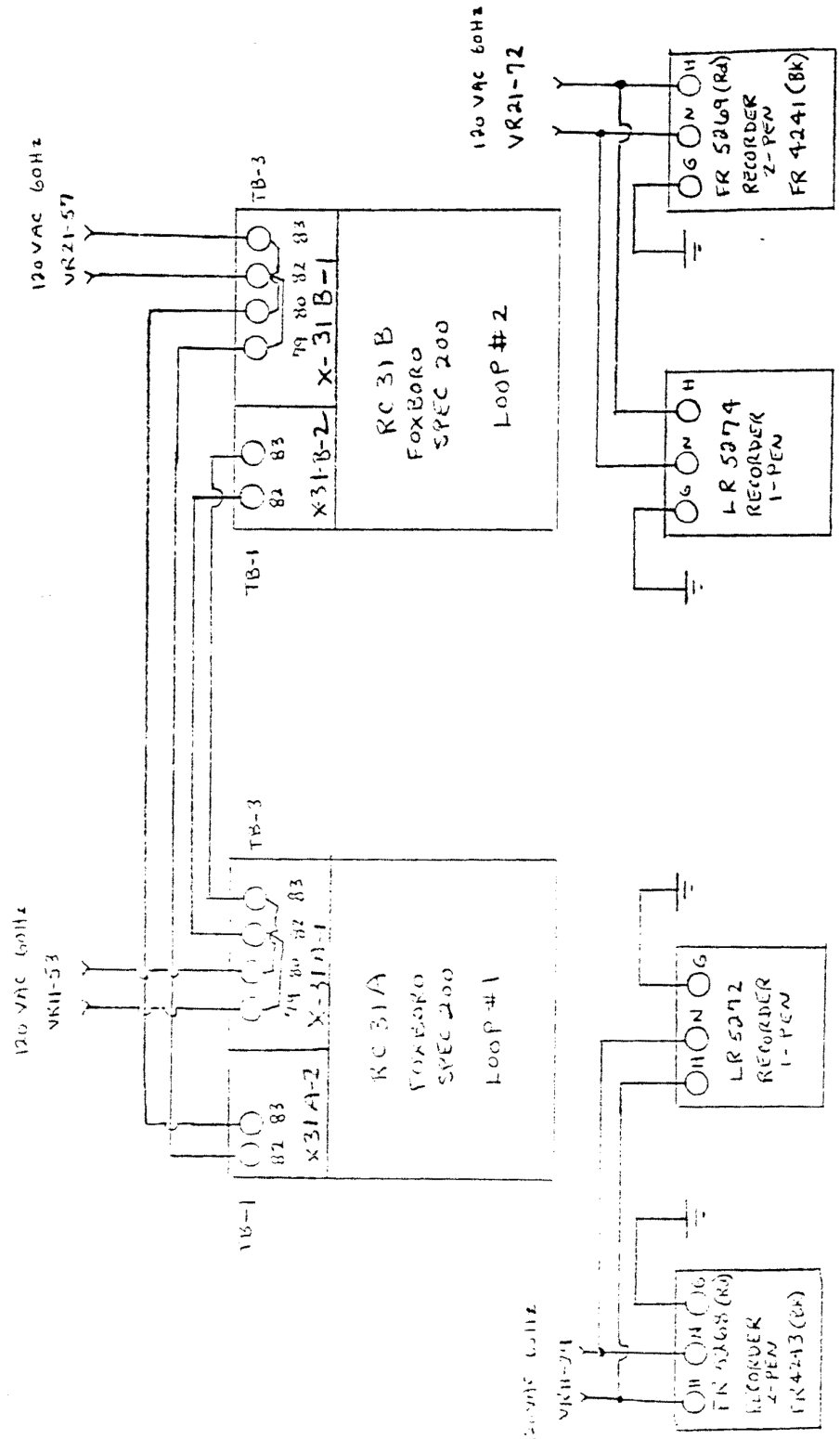


FIGURE 7.4-8 REACTIVITY CONTROL SYSTEM

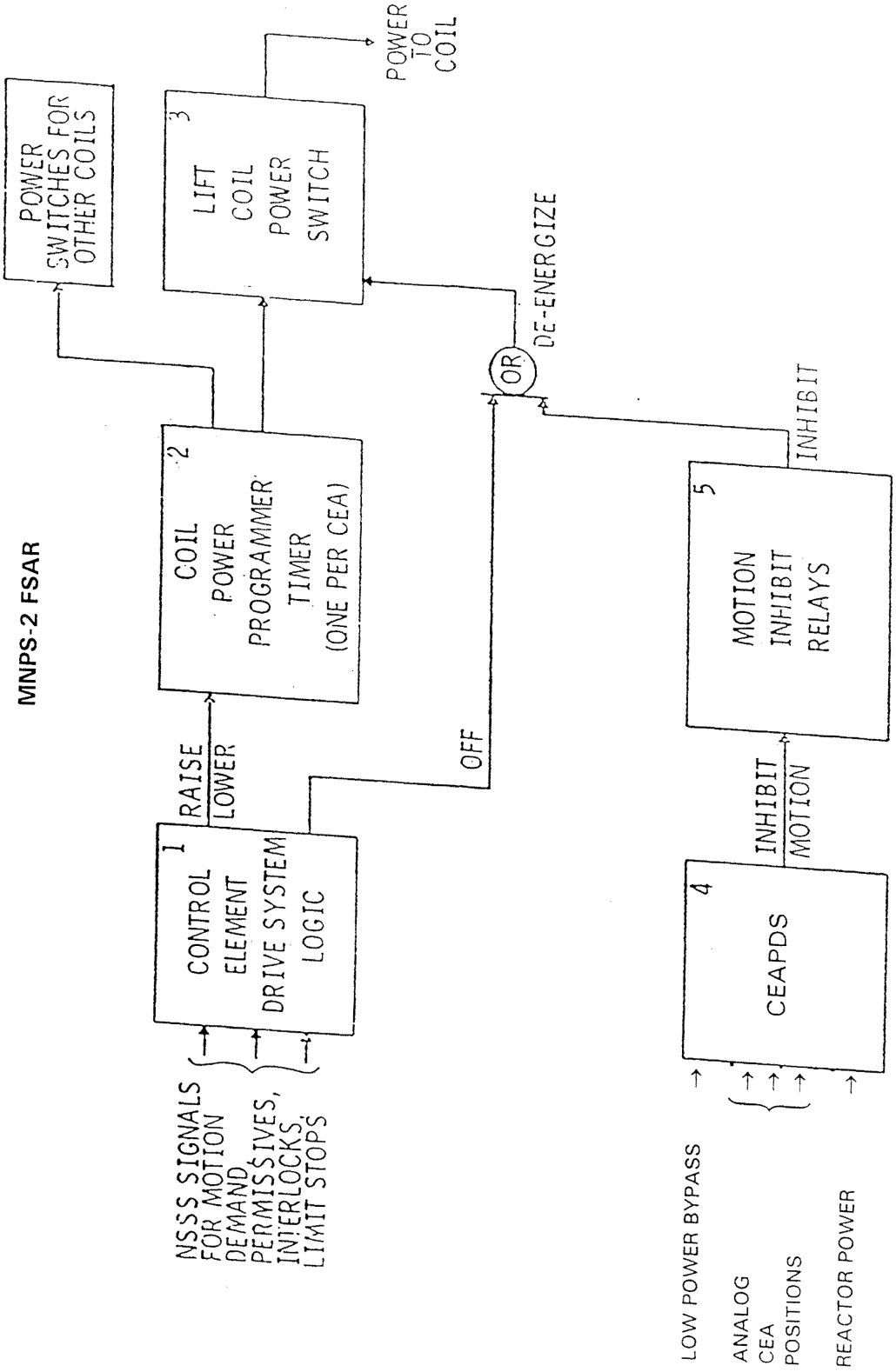


FIGURE 7.4–9 MOTION INHIBIT CIRCUIT INTERCONNECTION SIMPLIFIED DIAGRAM

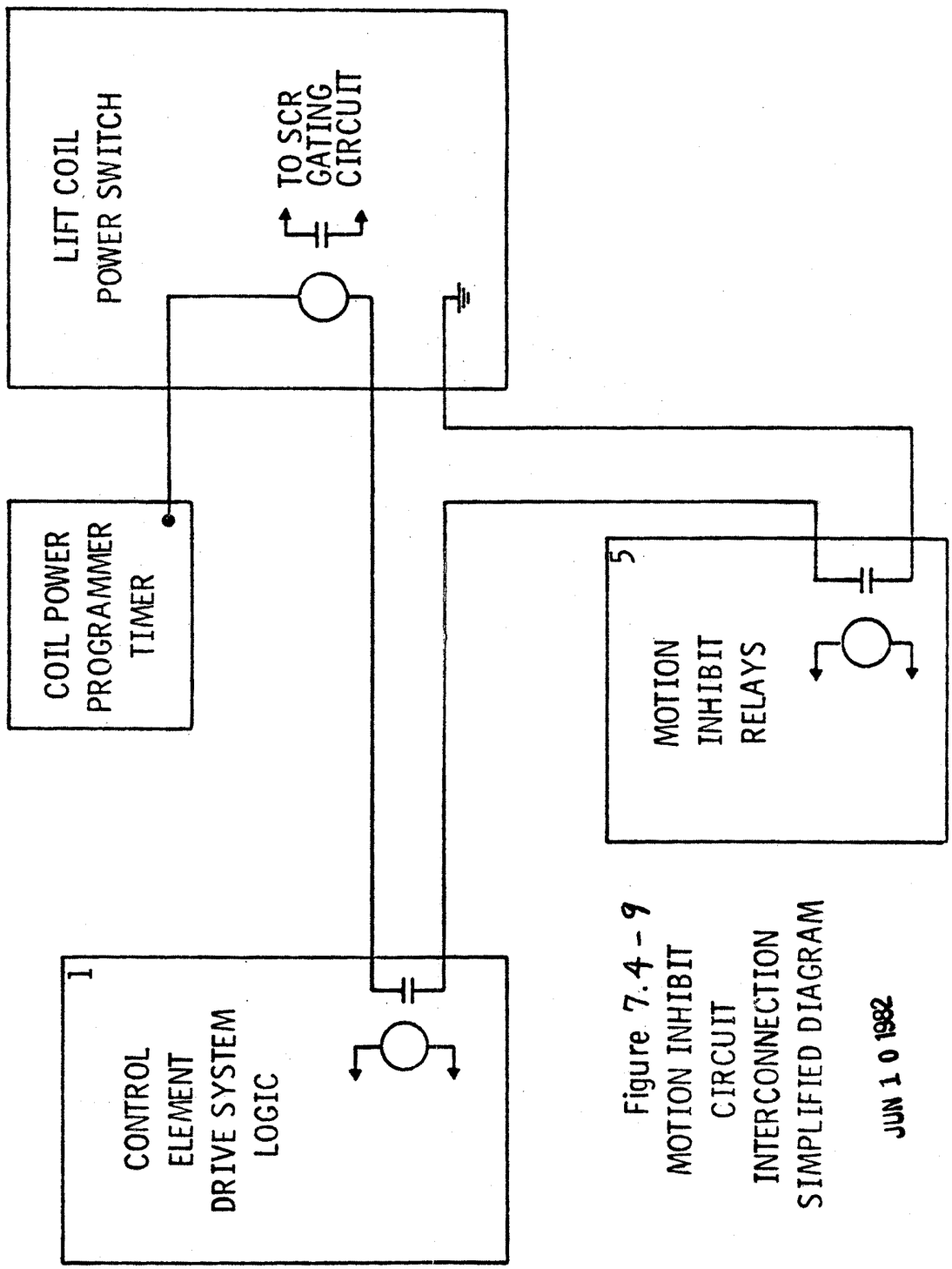


Figure 7.4-9
MOTION INHIBIT
CIRCUIT
INTERCONNECTION
SIMPLIFIED DIAGRAM

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7.5 INSTRUMENTATION SYSTEM

7.5.1 PROCESS INSTRUMENTATION

7.5.1.1 Design Bases

7.5.1.1.1 Functional Requirements

Non-nuclear process instrumentation devices are designed to perform one or more of the following functions:

Measurement

Display

Recording

Set Point Generation

Generation of Corrective Signal

Conditioning of Signal

These devices when connected together in a process instrument loop will monitor and control the process and alert the operator in the event the process variable exceeds beyond the allowable limits.

7.5.1.1.2 Design Criteria

The following criteria have been implemented in the design of the process instrumentation:

The process instrumentation shall measure and control temperature, pressure, flows, and level in all processes as required per 10 CFR 50, Appendix A, Criterion 13.

a. Alternate indicators and controls shall be located at other locations than the control room to allow reactor shutdown should the control room have to be evacuated. This provision is made in accordance with 10 CFR 50, Appendix A, Criterion 19.

b. Independent measurement channels shall be provided to monitor each process parameter required for the reactor protective system (RPS) and the engineered safety features actuation system (ESFAS) to meet the single-failure criterion per IEEE-279-1971.

7.5.1.2 System Description

7.5.1.2.1 System

Process instrumentation provided for reactor coolant temperature, pressure, level, and flow as well as the monitors for containment temperature, pressure, and radiation are described below. Major process instrumentation for the balance of the plant is tabulated in Table 7.5-1.

7.5.1.2.1.1 Reactor Coolant System

a. Temperature

The temperature measurements are made with precision resistance temperature detectors (RTD) which provide a signal to the remote temperature indicating control and safety devices. The following is a brief description of each of the temperature measurement channels.

Hot leg temperature: Each hot leg contains five temperature measurement channels. Four of these channels provide a hot leg temperature signal to the thermal margin/low-pressure (TM/LP) trip circuits. The other hot leg temperature measurement channel provides a signal to the loop T_{avg} computer in the reactor regulating system (RRS), to the Inadequate Core Cooling Monitoring System (ICCMS), and to a recorder. The five hot leg temperatures are indicated on the control panel.

Cold leg temperature: Each cold leg branch contains three temperature measurement channels. Two of the channels in each branch provide a cold leg temperature signal to the TM/LP trip circuits. These channels also provide cold leg temperature indications on the control panel. The third cold leg temperature measurement channel in two branches provides a signal to the loop T_{avg} computers and to the feedwater control system. This channel also provides a high alarm. The third channel in the other two branches provides an input to the ICCMS and Low Temperature Overpressurization Protection (LTOP) circuitry, and is recorded on the control panel.

Loop average temperature: The RRS receives a hot leg and cold leg temperature reading from both loops. The T_{avg} calculation receives hot and cold leg temperatures from Loop 1, Loop 2, or Loops 1 and 2 and provides average temperature outputs to a recorder. The temperature recorder is equipped with two channels. One channel records the average temperature and the other channel records the programmed reference temperature signal (T_{ref}), corresponding to turbine load (first-stage pressure).

b. Pressure

Pressure is measured by electromechanical pressure transmitters. The transmitter produces a DC current output that is proportional to the pressure sensed by the instrument. The DC current outputs are used to provide signals to the remote pressure-indicating, control and safety devices.

The following is a brief description of the pressure measurement channels:

Pressurizer pressure (protective action): Four pressurizer pressure transmitters provide independent, narrow range, pressure signals. These four independent pressure channels provide the signals for the RPS high pressure and TM/LP trips (see Section 7.2). The channels also provide the low-low pressure signal to initiate safety injection (see Section 7.3). All four pressure channels are indicated in the control room and RPS high pressure and TM/LP trip, associated RPS pretrip, and low-low trip alarms are annunciated. Figure 7.5–1 is a functional diagram of one of these channels.

Pressurizer pressure control (control action): Two pressure channels provide narrow-range signals for control of the pressurizer heaters and spray valves as described in Section 7.4. Outputs from the two pressure control channels are recorded in the control room and provide individual high and low alarms.

Pressurizer pressure (Low Range): Two pressure channels provide independent and redundant pressure signals to the shutdown cooling suction isolation valves and the power-operated relief valve (PORV). (See Sections 4.3.8.2.3 and 7.4.8.) These channels also provide an input to control room and hot shutdown panel indicators, computer and one signal to the subcooling margin monitor (see Section 9.3.4.1).

c. Pressurizer Level

Level is sensed by transmitters which measure the pressure difference between a reference column of water and the pressurizer water level. This pressure difference is converted to a DC current signal proportional to the level of water in the pressurizer. The DC current output of the level transmitters provides signals to the remote level indicating and control devices.

Two pressurizer level transmitters provide signals to the chemical and volume control charging and letdown system. In addition, signals are provided for pressurizer heater override control. These level transmitters are calibrated for steam and water densities existing at normal pressurizer operating conditions.

The selected pressurizer level control channel provides a signal for a level recorder in the control room. This recorder is a two channel recorder, with one channel recording actual level as sensed by the level control channel and the other channel recording the programmed level set point signal from the RRS. For additional details see Section 7.4.4.

d. Flow

An indication of reactor coolant flow is obtained by measuring the pressure drop between the hot leg piping and the outlet plenum of each steam generator. The pressure drop is sensed by differential pressure transmitters which convert the pressure difference to direct currents. The direct currents provide a signal to the remote flow indicating and safety devices.

Four independent differential pressure transmitters are provided in each reactor coolant loop to measure the pressure drop across the steam generators. The outputs of corresponding transmitters in each loop are summed by pairs to provide four independent signals representative of flow through the reactor core. These signals are indicated and supplied to the RPS for loss-of-flow determination. The differential pressure sensed by each transmitter is indicated in the control room. For additional details, see Section 7.2.

e. Subcooled Margin

See Section 7.5.4.4 for a description of the ICC System which provides subcooling information.

7.5.1.2.1.2 Containment

Containment temperature monitoring provisions include eight resistance temperature detectors in separate locations for measuring containment air temperatures. These RTDs have resistance-to-current transmitters which provide temperature signals to the plant computer for logging and for readout in the control room through an eight position selector switch. The RTD and cables are specified for continuous operation in the containment. The temperature transmitters are located outside the containment.

The capability of continuously monitoring pressure over the full range of postulated accidents is provided by four safeguards containment pressure transmitters which are located outside the containment. Dual redundant wide range containment pressure transmitters provide pressure monitoring beyond the range of the safeguards containment pressure transmitters. The wide range transmitters were procured and installed to the same criteria as the safeguards transmitters. These transmitters have sufficient range to measure between the normal operating conditions and the predicted peak pressure, due to either a LOCA or SLBA.

Dual-redundant containment level monitors are installed to measure water accumulation in the containment sump. These transmitters can monitor water depth up to seven feet above the sump (approximately $6x10^5$ gallons).

Containment radiation monitoring includes two gaseous and two particulate monitors of the types described in Section 7.5.6.3. The gaseous and particulate monitors are available except when the containment isolation actuation signal (CIAS) automatically closes valves in the sampling system.

This would preclude their use while the containment pressure is elevated during a LOCA or SLBA inside the containment.

Two containment high range area radiation monitors are located in the containment on the outside of the biological shield wall in the vicinity of the east and west electrical penetration areas. Each monitor has a range sufficiently wide to indicate activity levels following a serious accident. Each monitor is qualified to withstand the elevated pressures and temperatures and chemical spray associated with the LOCA or SLBA.

In addition, provisions for obtaining grab samples of containment atmosphere are included in the containment hydrogen monitoring sampling system.

7.5.1.2.2 Components

Standard commercially available instrumentation devices of high, proven quality have been used for all non-nuclear process loops. With the exception of a few multipoint recorders, all other display instruments are of the miniature type using the 10 to 50 ma, 4 to 20 ma, or 3 to 15 psig signals. All safety related instruments are designed to function in the most severe post-accident environmental condition to which they may be exposed.

7.5.1.3 System Operation

Each instrument loop is designed to display and/or control a certain process. Detailed description of major instrumentation systems are given in Section 7.5.1.2.1 and other appropriate sections where instrumentation is heavily involved in the processes.

In addition to the instrument channels listed in Table 7.5-1, the operator is provided with valve position lights, motor status lights, and alarms for the control of valves, circuit breakers and pumps required for the ESFASs. Additional ammeters are provided for motors of 250 horsepower and above.

7.5.1.4 Post-accident Monitoring

Instrumentation systems are provided for remote monitoring of system conditions during and following an accident to assure adequate public safety. While many parameters are available to the operator in the control room, those which would normally be used by the operator following an accident are those which: (1) aid in determining the nature of the accident, (2) can be used to follow the course of the accident and to aid in predicting its future course, (3) assure that the reactor trip systems (RTS) and engineered safety features (ESF) systems are functioning properly and that the plant is responding properly to these systems, (4) provide information for manual action by the operator.

Table 7.5-3 is a summary of the Regulatory Guide 1.97 (Rev. 2) information on file, that documents which instruments are credited with monitoring the associated variables. These indications of plant variables are required during accident situations to:

a. Provide information to permit the control room operator to take pre-planned action to accomplish safe plant shutdown.

b. Determine whether the RPS, ESF, and other systems important to safety are performing their intended function.

c. Provide information to the operator that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release, and to determine if a gross breach of a barrier has occurred.

By providing the instruments listed in this table, adequate instrumentation is provided for the operator to determine the nature of an accident, to follow the course of the accident, to determine what action is required to mitigate the consequences of an accident, and to determine the results produced by operator action.

7.5.2 NUCLEAR INSTRUMENTATION

7.5.2.1 Design Bases

The nuclear instrumentation monitors neutron flux over a range greater than ten decades with four independent and redundant channels.

Each of four wide range logarithmic channels provides: a) signals for power level and startup rate indication, b) a signal to the enabling circuitry of the RPS Zero Power Mode Bypass function, and c) a signal to the circuitry in the CEA Position Display System that blocks the CEA motion inhibit signals when reactor power is less than 10 E-4% power.

Four power range channels monitor neutron flux over the power range and provide four redundant, proportional signals to the RPS. These same four power range channels also are used to detect a dropped CEA by monitoring changes in reactor power.

Independent of the four wide range logarithmic channels and the four power range channels, the nuclear instrumentation system also includes two power-range channels that provide signals to the Reactivity Regulating System.

7.5.2.2 Design Criteria

The system is designed in accordance with the criteria of IEEE 279, 1971. In areas not covered or specifically identified by the criteria, the following criteria are used:

a. The nuclear instrumentation sensors are located so as to detect representative core flux conditions;

b. Four independent channels are used in each flux range;

c. The channel ranges overlap sufficiently to ensure that the flux is continually monitored from source range to 125 percent of rated power;

d. Power is supplied to the system from four separate AC buses. Loss of one bus deenergizes one power-range safety channel and one wide range logarithmic channel;

e. Loss of power to channel logic results in a channel trip;

f. All channel outputs are buffered so that accidental connection to 120 volts AC, or to channel supply voltage, or shorting individual outputs has no effect on any of the other outputs.

7.5.2.3 System Description

Ten channels of instrumentation are provided to monitor the neutron flux. The system consists of wide range logarithmic channels, power range safety and power range control channels. Each channel is complete with separate detectors, power supplies, amplifiers, and bistables to provide independent operation. The operating capability of the ten monitoring channels is greater than ten decades of neutron flux and is adequate to monitor the reactor power from shutdown through startup to greater than 125 percent of full power.

Four wide range logarithmic channels monitor the flux from source level to above full power. The flux signals, obtained from fission chambers, are conditioned and amplified in the cable vault amplifier assemblies and then transmitted to the signal processing drawers in the control room. The signal processing drawers further process the detector signal into signals that represent the source range logarithm of count rate and the rate of change of count rate, and the wide range logarithm of reactor power and the rate of change of reactor power. Audible count rate signals are available in the control room.

Four channels are designated as power range safety channels and provide signal outputs to the RPS. These channels operate from 0.1 to 200 percent of full power. Power level signals from these channels are supplied to the protective system. These four channels contain detectors composed of dual section ion chambers which monitor the full axial length of the reactor core at four circumferential positions equally spaced around the core. This arrangement enables detection of power tilts and imbalance.

Two separate power range control channels provide reactor power signals to the RRS cabinets. The channel output is a signal directly proportional to reactor power from 0.1 to 125 percent. The power signal is combined with the average coolant temperature, first stage turbine pressure, and pressurizer pressure signals as the control parameters to the RRS.

The gain of each channel is adjustable to provide a means for calibrating the output against a plant heat balance. Each control channel provides signals to remote indicators and the Anticipated Transients Without Scram (ATWS) Circuitry.

The comparator averager circuit is designed to provide the operator with an alarm upon failure of a linear power range safety channel or subchannel. In addition, this circuit can be used to alarm an off-normal azimuthal flux tilt condition. The comparator average is located in the rear of the RPS cabinet. It receives inputs from each of the four linear power range nuclear instrument channels and setpoints from the power ratio calculator. These setpoints are called High and High-High Deviation and are generated as a function of the average power sensed by all four power range safety channels. They represent a deviation band (absolute magnitude of deviation) that is allowed between the neutron flux measured by each channel and the average of all channels. The capability is provided for the operator to obtain two independently set alarms (High and High-High) representing different absolute magnitudes of deviation, in order to achieve a high degree of system flexibility to conform to changing operating conditions. The comparator averager measures the deviation between each channel and the average of the channels and then compares the measured deviations to the calculated allowable deviations represented by the setpoints. If the measured deviation exceeds the first setpoint, the High alarm is given. If the deviation continues to increase, the second setpoint may be reached, in which case the High-High alarm would be given. A High and High-High deviation alarm is provided for each channel.

Detector Cooling

Forced air cooling is provided for the eight neutron detectors and two spare detector wells which are located within the annular space between the reactor vessel and cavity wall. The remaining two detectors which are embedded in the cavity wall are not subjected to elevated temperatures and therefore do not require cooling.

Cooling air is provided by the containment air recirculation and cooling system (Section 6.5) to limit the maximum detector temperature below 200°F. Cooling air is supplied at 91°F to the base of the detector wells. The air is heated sensibly and discharged at 190°F through openings at the top of the detector wells.

If, however, a loss of air flow occurs, detector temperature would rise from 190°F to 260°F before attaining thermal equilibrium. The out of core detectors are designed to operate satisfactorily at a temperature of 300°F with no appreciable error. Therefore, even with loss of air flow, this value is not exceeded.

7.5.2.4 System Components

7.5.2.4.1 Wide-Range Logarithmic Channel Description

The wide range logarithmic channels are designed to provide to the operator the measure of the neutron flux level at the detector assembly and the measure of the rate of change of neutron flux level from source level (shutdown) to 200% of full power reactor operation.

The signal from the detector is composed of a series of charge pulses. The pulses result from alpha decay of the uranium coating in the detector, from gamma photon interaction with material in the electrodes of the detector, and from the fissioning of uranium atoms when a neutron is

absorbed. The pulse signal from alpha decay and from gamma radiation is unwanted signal and can be eliminated by amplitude discrimination because the neutron pulse signal is much larger.

Fission chambers are used to detect the neutron flux because of their proven high reliability in a harsh environment and because of their ability to operate in a high gamma flux without damage and without loss of sensitivity.

The cable and interconnection assemblies between the detector and the amplifier are designed to provide protection against electromagnetic and electrostatic noise interference, and to provide immunity to the postulated environment in the containment during a design basis event (main steam line break or loss of coolant accident).

The amplifier assembly contains the signal conditioning circuitry, DC power supplies, and the high voltage detector excitation supply. The signal processor assembly provides the circuitry to further process the detector signal into signals which are a measure of the logarithm of countrate, the rate of change of countrate, the logarithm of reactor power and the rate of change of reactor power, and it provides outputs for each of these signals. Full scale and bias scaling adjustments to align the outputs to the full reactor power are all made in the signal processor.

The signal processor outputs isolated source range pulses to the audible countrate drawer. The audible countrate drawer provides an audible sound corresponding to the source range pulse outputs. A switch on the front panel allows you to select each of the four channels. Another switch allows you to divide the signal by 1, 10, 100, 1000, or 10000, in order to hear individual tones at any shutdown countrate. There is also a volume control on the front panel for the local speaker as well as an on/off switch and volume control for a remote speaker. The audible countrate drawer is considered non-class 1E, however, it has been qualified not to damage adjacent class 1E equipment during a seismic event.

The number of neutron pulses per unit time from the detector is proportional to the magnitude of the neutron flux at the detector. The magnitude of the neutron flux in the reactor core is proportional to the fission power being generated in the reactor. If the magnitude of the neutron flux at the detector is proportional to the magnitude of the neutron flux in the reactor core, then the pulse rate from the detector is proportional to reactor power.

The neutron flux monitor measures the number of pulses per unit time from the detector over the range from source level to the level where the error from countrate loss due to coincident pulses becomes unacceptable. From about two decades below the upper end of the countrate range to full reactor power, the neutron flux monitor measures the mean square value of the time variant signal from the detector. This mean square value is proportional to the average rate of neutron pulses and is not dependent on the pulses being individually identifiable. It provides good discrimination against alpha and gamma signal.

The derivative of the logarithm of reactor power provides a measurement that is proportional to the change in reactor power per unit time. The signals are displayed on the RATE meters in decades per minute (DPM).

Four bistables are used in each wide range logarithmic channel. One bistable initiates an alarm when either the amplifier or the wide range drawer low voltage power supply is degraded or lost, the magnitude of the high voltage supply in the amplifier is low, or the channel in test (CH IN TEST) switch is in the ON position.

The second bistable is used by the RPS to remove the zero power mode bypass above 10^{-4} percent power. The zero power manually actuated bypass allows CEA drop testing, or CEA withdrawal for other tests during shutdown. The trips bypassed are low flow, Reactor Coolant Pump Underspeed and TM/LP. These trips are automatically reset by the wide range logarithmic channels prior to increasing reactor power to 10^{-4} percent power.

The third bistable is used by the CEA Position Display System to block the CEA Motion Inhibit for Power Dependent Insertion Limit Violation when reactor power is less than 10^{-4}% power. A fourth bistable enables the extended range mode of operation of the wide range Logarithmic Channel at very low power levels.

7.5.2.4.2 Power-Range Safety Channel Description

The four power-range safety channels are capable of measuring flux linearly over the range of 0.1 percent to 200 percent of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of approximately 12 feet. Integral shielded cable is used in the region of high neutron and gamma flux.

The DC current signals from both the upper and lower uncompensated ion chambers are fed directly to the rear of the linear power range monitors located in the RPS cabinets in the control room. The power range monitor houses the electronics that conditions the detector signals for front panel display, RPS inputs and bistable trips. Nuclear Power (upper + lower over 2) and subchannel deviation (upper-lower) signals are generated and output to the Reactor Protection System.

The Nuclear Power signal is also monitored by the rod drop circuit for a fast negative change in amplitude. Time delay and comparator circuits are used to generate a rod drop alarm whenever a negative change in the output signal level occurs in a corresponding period of time. This alarm condition is an indication that a rod had dropped from its proper position. Actuation of this alarm illuminates a LED on the front of the power range safety channel drawer, and actuates an alarm window on the main control board.

Each power range channel contains two bistables. One is used by the RPS to disable the Loss of Turbine and High Local Power Density trips below 15% power. The other bistable initiates an alarm when either the magnitude of the low voltage or high-voltage power supplies is degraded, or the TEST SELECT switch is not in the OFF position and the TEST ENABLE switch is in TEST ENABLE. The condition of each bistable is shown by a front panel LED.

7.5.2.4.3 Power-Range Control Channel Description

The two power range control channels are capable of measuring flux linearly over the range of 0.1 percent to 200 percent of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of approximately 12 feet. Integral shielded cable is used in the region of high neutron and gamma flux. The DC current signals from both the upper and lower uncompensated ion chambers are fed directly to the rear of the linear power range control monitors located in the RRS cabinets in the control room. The power range control channels are connected with the RRS and provide outputs to remote indicators and the ATWS circuitry.

7.5.2.4.4 System Component Location

The nuclear instrumentation safety signal processing equipment is located in the RPS cabinet in the control room. Four cabinets designated as A, B, C and D each house one channel and one wide range logarithmic channel. Mechanical and thermal barriers between the cabinets reduce the possibility of common event failure. The detector cables are routed separately from each other. This includes separation at the containment penetration areas. The location of the neutron detectors is shown in Figure 7.5–2.

The nuclear instrumentation control signal processing equipment is located in the RRS cabinets.

7.5.3 CONTROL ELEMENT ASSEMBLIES POSITION INSTRUMENTATION

7.5.3.1 Design Bases

The principal purpose of the CEA position indication system is to provide the operator with reliable, comprehensible and timely information on CEA position.

7.5.3.2 Design Criteria

a. Position readouts of all CEAs may be obtained;

b. Continuous position readouts of any selected CEA in a group are available;

c. A means of alerting the operator to deviation of CEAs within a group is provided;

d. A permanent record may be made of position of any or all CEAs. The operator may obtain a record at any other desired time;

e. Separate “full-in” and “full-out” indication is provided for each CEA;

f. Redundant and diverse means of indicating CEA position is provided.

7.5.3.3 System Description

Three different display systems of CEA position are provided for the operator on the main control board. The pulse-counting CEA position indication system is one of the outputs of the CEA supervisory function of the plant computer. The reed switch CEA position display system provides continuous indication to the operator of the position of all CEAs in a bar chart form on a video display and numeric information of selected groups. The core mimic display CEA position indicating system provides CEA travel limit information to the operator.

7.5.3.4 System Components

7.5.3.4.1 Pulse-Counting Control Element Assemblies Position Indication System Description

The pulse-counting CEA position indication system infers the position of each CEA maintaining a record of the raise and lower control pulses sent to each magnetic jack mechanism. The plant computer counts these pulses to determine changes in CEA position. The bottom most reed switch is input to the process computer as indication of a fully inserted or dropped rod independent of pulse counts. The resulting inferred position of CEA is available for display on any plant process computer workstation. A printout is available, on operator demand, of the inferred position of all CEAs or of those CEAs within a given group. The plant computer also provides deviation information. If the deviation in position between the highest and the lowest CEA in any group exceeds a preset amount, the computer provides an alarm. The plant computer provides position information for CEA group position alarms.

7.5.3.4.2 Reed Switch Control Element Assemblies Position Display System Description

The reed switch CEA position display system (CEAPDS) utilizes a series of magnetically actuated reed switches, spaced at 1.5 inch intervals along the CEA housing and arranged with precision resistors in a voltage divider network, to provide voltage signals proportional to CEA position. The signals are displayed in bar chart form on a touchscreen monitor on the main control board. The touchscreen allows navigation to various display page information. The display and interlock functions are generated by the CEAPDS software. The distributed control system logic provides redundant CEA malposition indication and alarm functions which are used as input to the CEA motion inhibit circuitry. A CEA position backup readout display is available to provide position of all CEAs.

7.5.3.4.3 Core Mimic Control Element Assemblies Position Indication

A group of 61 light displays, arranged in a shape corresponding to the CEA distribution, is located on the main control board. Each display, which represents one control element drive mechanism (CEDM), contains four colored lights providing the information listed in Table 7.5-7.

7.5.4 IN-CORE INSTRUMENTATION

7.5.4.1 Design Bases

The primary function of the in-core instrumentation is to provide measured data which may be used in evaluating the gross core power distribution in the reactor core as an aid to reactor operations. This data may be used to evaluate thermal margins and to estimate local fuel burnup. No credit is taken for this system in the accident analysis of Chapter 14. The in-core detectors will be used to periodically calibrate the out-of-core detectors as defined in the Technical Specifications.

7.5.4.2 Design Criteria

a. Detector assemblies are installed in the reactor core at selected locations to measure core neutron flux and coolant temperature information during reactor operation in the power range;

b. Flux detectors of the self-powered type, with proven capabilities for in-core service, are used;

c. The information obtained from the detector assemblies may be used for fuel management purposes and to assess the core performance. It will not be used for automatic protective or control functions;

d. The output signal of the flux detectors will be adjusted for changes in sensitivity due to emitter material burnup and for undesirable background signals;

e. Each detector assembly is comprised of four local neutron flux detectors stacked vertically for axial monitoring, and one thermocouple at the assembly outlet;

f. Axial spacing of the detectors in each assembly and radial spacing of the assemblies permit an evaluation of the gross core power distribution through the use of In-Core Analysis Computer Program.

g. In accordance with the guidance of NUREG-0737 and Regulatory Guide 1.97 the Core Exit Thermocouples from each train are distributed such that all four quadrants of the reactor vessel can be monitored during and after a design basis accident following the loss of a single train.

7.5.4.3 In-Core Instrumentation System Description

The in-core instrumentation (ICI) system consists of 43 fixed in-core detector assemblies inserted into selected fuel assemblies. Each assembly contains four 40 cm long rhodium detectors, and one Cr-Al thermocouple. The detector assembly is illustrated in Figure 7.5–3. Outputs are fed to the plant computer in the control room for processing and logging.

Assemblies are inserted into the core through instrumentation nozzles in the top closure head of the reactor vessel. Each assembly is guided into position in the center of the fuel assembly via a fixed guide tube and instrument thimble assembly. A Grayloc seal forms a pressure boundary for each assembly at the instrument nozzle. The locations of the in-core detectors in the core are shown in Figure 7.5–4.

The neutron detectors produce a current proportional to neutron flux by a neutron-beta reaction in the detector wire. The emitter, which is the central conductor in the coaxial detector, is made of rhodium and has a high thermal neutron capture cross section. The useful life of the rhodium detectors is expected to be about three years at full power, after which the detector assemblies will be replaced by new units.

The data from the detectors are read out by the plant computer which scans all assemblies, processes and prints out the data periodically or on demand. The computer periodically computes integrated flux at each detector to update detector sensitivity factors to compensate for detector burnout.

7.5.4.4 Inadequate Core Cooling (ICC) System Description

The Inadequate Core Cooling Monitoring System (ICCMS) integrates the processing and display of:

1. Subcooled/Superheat

2. Core Exit Thermocouples (CETs)

   Note: The CETs are part of the ICIs.

3. Reactor Vessel Level Monitoring System (RVLMS) Heated Junction Thermocouple (HJTC) System for inventory tracking.

The information provided by this system allows the plant operators to monitor reactor coolant status during normal and abnormal plant conditions. The operator uses this information to take corrective action as needed and/or confirm that actions taken produce the desired result. Thus, the approach to, existence of, and recovery from ICC conditions can be monitored consistent with the provisions of NUREG-0737, Section II.F.2. The Millstone Unit Number 2 ICC system is designed as Category I (Class 1E) with redundant trains (train A and train B). Each train contains stand-alone processing electronics and displays, which monitor and alarm ICC as shown in Figure 7.5–7.

The ICCMS CET channel assignments are shown on Figure 7.5–4. The core exit thermocouples from each train are distributed such that all four quadrants of the reactor core can be monitored during and after an accident following loss of a single train. Data received from both channels of ICCMS are combined and displayed on the non-Class 1E plant process computer (PPC).

Subcooled/Superheat calculations use reactor coolant system (RCS) temperatures and pressures to calculate the degree of subcooling or superheat in the reactor coolant either in terms of temperature or pressure. The calculations are based upon the most conservative input temperature and pressures.

Note: Temperatures are input from the CETs, Unheated Thermocouples (UHTCs), and RCS hot and cold leg RTDs.

CETs, HJTCs, and UHTCs are provided with required reference junction temperature thermocouple compensation. All CETs, HJTCs, and UHTCs can be displayed on a touchscreen graphic local display unit and the PPC.

The RVLM module monitors coolant inventory in the region above the core. Redundant strings of HJTCs are arranged in the reactor vessel head area to provide indication of conditions at eight discrete levels. The system is a two-channel system each consisting of a string of eight sensors. The detector assembly is illustrated in Figure 7.5–6.

The primary means of displaying the ICC information is provided via the non-Class 1E plant process computer (PPC). The PPC receives ICC data transmitted via optical isolation provided by the ICCMS.

The ICCMS local display provides backup IE display of ICC information. Each ICCMS cabinet (train A and B), which is located adjacent to the Control Room, has a qualified class 1E touchscreen graphic display that provides the following ICC information.

1. Subcooled/superheat in °F (300°F subcooling to 45°F superheat).*

2. CETs (200°F to 2300°F).

3. Percentage level (0 to 100%) in the vessel above the core.

* Minimum displayed range

This ICC information is provided on the local ICCMS displays by means of display pages which are selected via touchscreen. The hierarchy of the display pages is shown in Figure 7.5–9.

Alarms are provided on main control boards from the ICC cabinets. There are four alarms: Saturation Trouble alarm, CET High alarm, Reactor Level Low alarm, and ICCMS Trouble alarm. Alarm status touchscreen buttons also are provided on all ICCMS display pages.

7.5.4.4.1 Inadequate Core Cooling System Software

The objective of the ICC monitor is to provide the operator with a simple indication of core cooling conditions. This objective is achieved by monitoring a number of reactor system parameters, performing certain calculations using these parameters in a digital computer, and

providing the operator with the results of these calculations in real time. This section outlines the computer program that performs the required calculations.

Input Variables

The measured quantities used as input variables for the ICC monitor are RCS pressure, RCS hot and cold leg temperatures, CETs, HJTCs and UHTCs.

Calculations

A number of different calculations are performed using the input variables to generate output information for the operator's use. These calculations are characterized below:

1. All input values are converted to engineering units; °F for temperatures and psia for pressures. All engineering unit values are checked for sensor range limits and quality tags assigned before being used in calculations. CET, HJTC, and UHTC thermocouple processing, including reference junction temperature compensation, is also performed.

2. The HJTC Processing module calculates reactor vessel level based on eight incremental sensor pair locations, with each pair having heated and unheated thermocouples. A Reactor Level Low alarm will be generated if the calculated level exceeds the setpoint. HJTC heater control also is performed. Power to each of the eight heaters is individually controlled (on/off).

   The calculation includes provisions for the bypass of failed sensors and “substitution” of available, valid sensors for determining reactor level.

3. The CET processing module validates CET inputs used in the calculations and calculates the highest and second highest CET temperature (each quadrant and overall) from the validated set. This information is displayed on the applicable pages and is used in the Saturation Margin module. A CET High alarm is generated if the second highest (overall) CET temperature exceeds the setpoint.

4. The saturation margin module performs a number of calculations using all available, valid temperature and pressure inputs. The saturation temperature is calculated and then four temperature saturation margins are calculated. A saturation pressure margin is then calculated, and a Saturation Trouble alarm is generated if the saturation temperature margin exceeds the specified margin setpoint. (Adapted versions of the subroutines built by McClintock and Silverstri for the American Society of Mechanical Engineers (ASME) steam table are used for saturation calculations.)

5. Other calculations are performed which support various maintenance, diagnostic and test features of the ICCMS.

System Operation

The ICC monitor, once the initial startup is accomplished, is self sustaining. All program operations, from input through calculations to output, including periodic testing, are performed sequentially. Analog input points may be bypassed via touchscreen on the Signal and HJTC display pages, provided the Test Mode/Bypass keyswitch is enabled.

System Diagnostics

An online diagnostic function performs mathematical and Boolean logic checks and monitors the ICCMS application tasks for proper execution sequence. Resulting errors are logged, and a warm restart and ICCMS Trouble alarm will result if these checks fail.

Other on-line diagnostics include Input Processing range and quality checks, Plant Process computer and Remote Terminal data link health, cabinet temperature monitoring and Uniform Temperature Reference monitoring. Errors of this type are logged to the System Errors page, and an ICCMS Trouble alarm initiated. Cabinet temperature and reference temperature monitoring is performed as apart of the normal scan/alarm processing of the ICCMS.

System and Handler Software

C-based software performs the necessary ICC functions of data validation and conversion, selection of limiting pressures and temperatures, calculation of temperature and pressure margins, maintaining tabular and formatted data files, maintaining tabular and formatted system status files, and controlling communications.

System software is based on a real time, multitasking operating system. Applications programming uses the C language and utilities.

The applications program has two major divisions: one handling process calculations and the other handling input/output operations. Refer to Figure 7.5–8 for the following:

Data Acquisition

Data acquisition is performed by the Input Processing module. It provides the data validation process. The data is checked for bypassed, open, shorted, or out of range conditions. A different error code is stored in an array for each input depending upon the error. The data then is converted to engineering units according to standard temperature (or pressure) conversions.

Process Calculations

The process calculations include CET, HJTC, HJTC Heater Control, and Saturation Margin modules.

Communications

The Output Data Services module handles communication tasks with the PPC, the display units, and a maintenance diagnostic terminal connection.

Data is “broadcast” (one way) to the PPC via fiber optic modem approximately every 3 seconds. No “handshaking,” either hardware or software, occurs. This eliminates the potential for unacceptable interaction between the 1E ICCMS and the non-Class 1E PPC. All acquired and calculated data are transmitted.

The ICCMS display communicates with the ICCMS processor via a serial connection. Communication is two way; i.e., both the ICCMS processor and display send and receive data interactively. The display and processor are both class 1E.

A maintenance diagnostic terminal serial connection also is provided. Specific file data are provided in response to a specific file request by a requesting terminal. The terminal is connected only during maintenance.

7.5.5 PLANT COMPUTER SYSTEM

7.5.5.1 Summary Description

The plant computer is an online digital computer designed to perform a variety of data acquisition and information processing output functions. The computer presents information to the plant operational and technical personnel on printers, main control board color monitors, man-machine interface workstations, and recorders.

h. In fulfilling the overall data acquisition requirements of the computer, selected process instrumentation inputs are used.

i. In addition to the output of operational and technical information, the computer operates digital output latched contacts, some of which are used for control and supervision of CEAs.

7.5.5.2 Functional Objectives

The objectives of the plant computer system are as follows:

a. To assist the plant operational and technical personnel in monitoring the performance and operation of the plant equipment;

b. To provide for the acquisition and logging of process data which are available during unusual plant conditions;

c. To monitor all CEA movements and to control CEA group sequencing in the automatic and manual sequential modes and display CEA position;

d. To assist plant personnel in the analysis of reactor core operation during both steady state and transient conditions;

e. To assist plant personnel in performing certain periodic surveillance tests;

f. To relieve the plant operational personnel of routine plant logging functions.

7.5.5.3 System Description

The major equipment associated with the plant computer system is located in the computer room. This equipment consists of:

a. Processors, in a redundant host configuration.

b. Disk drive cabinets (Storage Area Network (SAN)).

c. Cabinets for process input/output interface equipment, consisting of digital input, digital output, analog input and analog output cards.

d. Cabinets for peripheral and communication equipment.

e. Network hubs for connections between the processor, MMI workstations, printers and digital plant equipment.

Equipment is also available in the computer room for specific reactor engineering and plant performance applications. This equipment includes MMI workstations and printers. A Data Diode exists between the Millstone Unit Number 2 PPC network and Millstone Site Data Network (MSDN) which isolates traffic and provides access to PPC data on the MSDN.

Output information from the computer is displayed in the main control room for use by the plant operating personnel. The central section of the main control board is designed to provide the operator with overall surveillance of all computer initiated data displays. Two two-channel recorders and three color monitors are provided on the vertical panels of this section.

A minimum of four MMI workstations are located in the Main Control Room and serve as the man/machine interface for the plant operators. These workstations provide the plant operators with access to numerous computer system data display functions.

Hardcopy printout capability is provided in the Main Control Room, readily accessible to plant operators. Print output is provided to more than one printer such that the capability to produce important reports is not lost if a single printer fails. Most reports can also be directed to the requesting workstation for local file viewing in lieu of printed output. Simulated printer output is also available at control room workstations for alarm messages.

The power to operate the computer equipment is normally supplied by a 75 kVA three phase inverter. The inverter is normally supplied by a 480 volt station service AC power source through

a Kirk-Key interlock scheme which allows for alignment to either emergency diesel generator. In the event of a loss of station service power, the inverter is automatically supplied by a battery. The battery, which has a 90 minute computer full load capacity, is separate from the station service batteries and is used only to supply the computer system loads. A second station service 480-120/208Y regulated supply provides for computer system loads during periods of inverter, battery charger or rectifier maintenance.

7.5.5.4 Functional Description

Functional objectives of the computer system are accomplished automatically, via computer programming and, on demand, via request through MMI workstations. Four basic types of functions are possible via the MMI workstations:

a. Printout request functions

b. Trend-display control functions

c. Data entry functions

d. Special operation functions

Printout requests enable the plant operator to demand a variety of printed logs such as a core map of CET temperatures, a post-trip review log, or a sequence of events log. Trend display control functions give the plant operator the ability to select and control the inputs to the recorders, and MMI workstations. Preset groupings of analog inputs or calculated values may be trended at interval multiples of one second using real-time data. Data entry functions give the operator the ability to modify stored data such as constants or assign temporary process alarm limit setpoints.

Computer functions which are performed automatically are as follows:

a. Analog and digital scanning

b. Sequence of events status checking and message logging

c. Alarming process out of limits and alarm message logging

d. Analog input averaging and engineering unit conversion

e. Selected nuclear and plant performance calculations

f. Video alarm message display

g. Video point display

h. CEA position pulse counting

i. Selected rod position digital display

j. Pre-post-trip review logging

k. Hourly plant data logging

l. Process variable pulse counting

m. CEA procedural limit alarming

n. CEA sequencing supervision

o. Hourly excore detector readout data logging

p. Daily excore detector readout data logging

q. Daily reactor coolant system (RCS) leakage data logging

r. Daily pump/fan motor run times

s. Updating incore correction factors

t. Incore detector reasonability checks

u. SPDS Screen (Section 7.5.5.5)

The following functions are automatic and are initiated by operator or reactor engineer MMI workstation requests.

a. Process value digital video display

b. Recorder trending

c. Archive recording

d. Certain periodic surveillance test logging

The foreground/background capability of the redundant processor pair will permit the use of the idle computers for online modification and/or addition of programs. This will increase system availability by greatly reducing the need to take the computer out of service for program maintenance.

RCS mid-loop level hot leg number 1 (L-112) and hot leg number 2 (L-122) are displayed on the process computer.

7.5.5.5 Safety Parameter Display System

The Millstone 2 Safety Parameter Display System (SPDS) is a computer based system that serves as an aid to the operating crew during normal and emergency operating conditions. The SPDS design addresses the guidelines of Supplement 1 to NUREG-0737, “Requirements for Emergency Response Capability” and is based on the Emergency Operating Procedure (EOP) Safety Function Status Checks (SFSCs). It is not a Category 1 system and is not a replacement or substitute for safety instrumentation.

The primary purpose of the SPDS is to help the operating crew monitor the safety status of the plant. It does this by providing a concentrated and organized source of plant data that allows the users to compare existing plant conditions with the safety function limits listed in the EOPs to determine if the limits are violated. The safety functions monitored by the SFSCs include Reactivity Control, Vital Auxiliaries, Reactor Coolant System (RCS) Inventory Control, RCS Pressure Control, Core Heat Removal, RCS Heat Removal, Containment Isolation, and Containment Temperature and Pressure Control. The SPDS performs signal validation on redundant sensor data and displays quality tags for sensor data and calculated values.

The SPDS displays are divided into three-tier hierarchies that readily support the evaluation of the SFSCs. The three-tier hierarchies consist of a single Overview Display (top level), one or more displays for each safety function (mid-level), and one or more sensor data displays for each safety function display (data level). In addition to the hierarchical displays, displays of trends and graphs that support the evaluation of the safety status of the plant are accessible from menus and/or command buttons.

One dedicated display terminal in the control room is designated to continuously display the Overview display of critical plant variables. The SPDS Overview provides a continuous concise display of the critical plant parameters and the parameter values are color coded. It has EOP specific acceptance criteria on the parameters to alert the operators that a safety function may not be met. The parameter values change color and blink when they exceed the EOP acceptance criteria. Selection of specific EOP zzcan be achieved manually by authorized operators and setpoints for each safety function parameters will automatically change according to the selected EOP. Colored boxes at the bottom of the display will indicate EOP Critical Safety Function (CSF) status, (green - limits not exceeded, red - limits exceeded) for specific EOP selected (with the exception being EOP 2525). Upon a reactor trip, the Overview is automatically displayed with EOP 2525’s, Standard Post Trip Action, setpoints.

SPDS displays can be accessed through the site’s computer network at various locations. Personnel outside of the control room cannot influence the analysis performed by the SPDS, that is, they cannot enter manual inputs, change setpoints, or change the EOP/mode that has been selected from within the control room.

7.5.6 RADIOACTIVITY MONITORING

7.5.6.1 Area Radiation Monitoring System

7.5.6.1.1 Design Bases

7.5.6.1.1.1 Functional Requirements

The basic purpose of the area radiation monitoring equipment is to provide information for the protection of plant personnel from radiation.

7.5.6.1.1.2 Design Criteria

The area radiation monitors are sensitive to gamma radiation. Audible radiation alarm signals are actuated on the main control board and at the detector location whenever an abnormal increase in radiation levels occur. Exception: The Containment High Range Radiation Monitors, RM-8240 and 8241, do not have audible alarms at the detector locations. The alarm setpoint for each area radiation monitor is variable and is set at a level sufficiently above the normal ambient background radiation level in the respective area to avoid spurious alarming. Table 7.5-4 shows the range, for each area monitor.

As a minimum, the area radiation monitors have a dynamic range of at least one decade above the normal expected level.

7.5.6.1.2 System Description

7.5.6.1.2.1 Location

Area radiation monitors are located as follows:

a. Containment personnel access hatch (elevation 38 feet, six inches)

b. Containment refueling machine service platform (elevation 38 feet, six inches)

c. Auxiliary building drumming and decontamination area (elevation 14 feet, six inches)

d. Control room (elevation 38 feet, six inches)

e. Auxiliary building - charging pump area (elevation 25 feet, six inches)

f. Sampling room - auxiliary building (elevation (+) 14 feet, six inches)

g. Radioactive waste processing area (elevation (-) 25 feet, six inches)

h. Radioactive waste processing area (elevation (-) 45 feet, six inches) (2)

i. Spent fuel pool area (elevation 38 feet, six inches) (4)

j. Containment high range (elevation 14 feet, six inches) (2)

7.5.6.1.2.2 Components

Each area radiation monitoring system consists of a gamma sensitive detector located in the area to be monitored. Most also consist of a local indicator/alarm module, located in or adjacent to the area being monitored (except the containment high range) and a remote readout module in the control room.

7.5.6.1.3 System Operation

Area radiation monitors are calibrated on a periodic basis.

Calibration of the area radiation monitor is accomplished by placing the detector assembly in a reproducible, fixed geometry configuration and exposing the detector to a calibrated radioactive source.

Power for the area radiation monitoring system is provided by either the 120 volt regulated or the vital AC distribution system.

A “loss of power/channel failure” and/or high radiation is monitored for each area radiation monitoring channel, with common annunciation in the control room.

Channel performance and test is available to the operator. An electronic signal is used to verify the performance of the control readout instrumentation.

Alarm settings are normally based on desired level above background.

Two containment high-range area radiation monitors are located in the containment on the outside of the biological shield wall in the vicinity of the electrical penetration area, elevation 14 feet, six inches. Each monitor has a range sufficiently wide to indicate activity levels following a serious accident. Each monitor is qualified to withstand the elevated temperatures and pressures and chemical spray associated with the LOCA or SLBA. Both monitors provide continuous indication and both channels provide signals for recording. Local indication and alarm functions are not included for these monitors.

In the event of high containment radiation, the high range monitors will actuate the automatic closure of the containment hydrogen purge isolation valves.

Four area monitors in the Spent Fuel Pool area furnish a -5 to -1 volt output signal to the engineered safeguards system to initiate operation of the auxiliary exhaust actuation system (AEAS).

7.5.6.1.4 Availability and Reliability

Area radiation monitoring equipment is designed for continuous operation during the 40 year life of the plant. High quality, commercial grade components are used throughout the system. Servicing of the electrical system is simplified by the use of plug-in components.

7.5.6.2 Liquid Radiation Monitoring System

7.5.6.2.1 Design Bases

7.5.6.2.1.1 Functional Requirement

Liquid radiation monitors are used with alarms and indicators to give warning of abnormal radioactivity in process piping and to prevent releases in excess of allowable limits.

In conjunction with control circuit components, liquid radiation monitors initiate valve action to stop the release of liquid waste, upon detection of high radioactivity.

7.5.6.2.1.2 Design Criteria

Liquid radiation monitors are designed to detect radioactivity in process lines.

Liquid radiation monitors are also designed to detect radioactivity in liquids prior to their release to the environment. In conjunction with control circuit components, selected radiation monitors initiate closure of valves to prevent the release of radioactivity to uncontrolled areas in excess of allowable limits.

7.5.6.2.2 System Description

7.5.6.2.2.1 Systems

Liquid radiation monitor systems are provided for the following:

a. Steam generator blowdown sample

b. Reactor building closed cooling water (RBCCW)

c. Clean liquid waste discharge

d. Aerated liquid waste discharge

e. Condensate recovery tank

f. Deleted

g. Condensate polishing facility - waste neutralization sump discharge

7.5.6.2.2.2 Components

The components used for each liquid monitor are as follows:

1. A liquid sample chamber.

2. A gamma sensitive scintillation detector positioned on the outside surface of the sample chamber.

3. Lead shielding for the sample chamber and detector to reduce the effect of background sources of radioactivity.

4. Local indication with audio and visual alarm.

5. Control room indication and alarm for the monitors. (Control room alarm annunciator only for condensate polishing facility-waste neutralization sump discharge monitor.)

6. Recorders for continuous records.

7.5.6.2.2.2.1 Steam-Generator Blowdown Sample Monitor

The steam-generator blowdown sample monitor supplies high radiation/instrument failure signal to the control logic to initiate closure of: blowdown line isolation valves from steam generators 1 and 2, blowdown quench tank discharge valve, blowdown tank discharge valve, and steam generators 1 and 2 sample valves. It serves as the final effluent monitor for liquid blowdown releases. It may provide indication of gross primary to secondary leakage.

7.5.6.2.2.2.2 Reactor Building Closed Cooling Water Monitor

The RBCCW system continuously circulates water in a closed loop which could become radioactive in the event of inleakage from the components handling radioactive materials. The discharge from the RBCCW pump is continuously monitored for high radioactivity. High activity or instrument failure is alarmed.

7.5.6.2.2.2.3 Clean Liquid and Aerated Liquid Waste Discharge Monitor

This monitor is provided to measure radioactivity in clean liquid waste and aerated liquid waste being discharged to the circulating water. It is a final effluent monitor. High radioactivity, low sample flow, or instrument failure will initiate closure of valves in the respective discharge line and will actuate alarms.

7.5.6.2.2.2.4 Condensate Recovery Tank Monitor

This monitor looks for the presence of radioactivity in the auxiliary condensate system by monitoring the discharge from the condensate recovery tank. On a high radiation or instrument failure alarm it will automatically terminate flow from the condensate recovery tank.

7.5.6.2.2.2.5 Failed Fuel Monitor

Deleted

7.5.6.2.2.2.6 Condensate Polishing Facility – Waste Neutralization Sump Discharge

This monitor measures the radioactivity in the effluent from the Condensate Polishing Facility - Waste Neutralization Sump. It is a final effluent monitor. High or low flow is alarmed. It automatically isolates discharge upon a high radiation or instrument failure alarm.

7.5.6.2.3 System Operation

Liquid radiation monitors will be checked and calibrated on a periodic schedule using radioactive sources.

7.5.6.2.4 Availability and Reliability

Liquid radiation monitoring equipment is designed for operation over long periods with minimum of service. Spare sample chambers are available to replace contaminated chambers.

High-quality, commercial-grade components are used throughout the system. Servicing of the electrical system is simplified by the use of plug-in components.

7.5.6.3 Airborne and Steam Radioactivity Monitoring

7.5.6.3.1 Design Basis

7.5.6.3.1.1 Functional Requirements

An airborne (including steam) radioactivity monitoring system is provided to detect and measure the levels of airborne radioactivity at various locations, both within the plant and in plant effluents, to satisfy the requirements of 10 CFR, Part 20 including Sections 1301 and 1302 and Appendix B, Table 2, and Part 50, including the general design criteria (GDC) of Appendix A.

The system will indicate the plant areas in which increases in radioactivity have occurred so that the cause can be determined and corrected to ensure on site and off site safety during all plant operations.

Instrumentation is also included to provide a record of the level of radioactivity for the points monitored. (Except Control Room Inlet Monitors)

7.5.6.3.1.2 Design Criteria

The criteria which determines the type of airborne radiation monitoring equipment selected for each application is as follows:

1. The nature of the radioactive release.

2. The sensitivity and range requirements to monitor normal operating or potential accident radioactive levels.

3. The response requirements to alert and warn personnel of the radioactive hazard so that protective measures may be taken.

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.

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. Each redundant system consists of a particulate detector and a gaseous detector, which share some common components such as power supplies and microprocessors. 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 iodine cartridge is removable for periodic laboratory analysis.

The redundant detection systems provide inputs to ESFAS from a total of four detectors, two particulate and two gaseous. High radiation and instrument failure alarm signals from any one of the four detectors will cause the ESFAS to initiate the containment purge isolation.

The containment isolation actuation signal (CIAS) closes the containment isolation valves for the sampling system, trips the sample fan and also closes local sample line isolation valves at each containment gaseous and particulate radiation monitor. The closed monitor sample line isolation valves prevent elevated pressure associated with occurrence of a LOCA or SLBA from affecting the containment gaseous and particulate radiation monitors once the containment isolation valves are reopened. Reopening the containment isolation valves under this condition will permit post-accident monitoring using the hydrogen analyzer and PASS which share the same sample lines with the containment gaseous and particulate radiation monitors.

Under post-accident conditions with the containment isolation valves reopened, the radiation monitor sample line piping system extending beyond the outermost containment isolation valve to and including the local monitor sample line isolation valves are considered part of an extended containment pressure boundary. The ANSI B31.1 (1967) piping system and associated valves which comprise this section of monitor sample line system are subject to testing and surveillance requirements typically applicable to ANSI B31.7 (1969) Class 2 piping systems and valve components. The local monitor sample line isolation valves do not provide a containment isolation function.

7.5.6.3.2.1.3 Radioactive Waste Ventilation Monitor

The radioactive waste airborne radioactivity monitoring system consists of four radiation monitors, each capable of continuously monitoring airborne radioactive particulates. One of the four also monitors for gaseous activity. In addition, each monitor contains a replaceable charcoal cartridge mounted in series and downstream of the particulate filter. The charcoal cartridge may be removed periodically for laboratory analysis of iodine activity.

The combined exhaust from the radioactive waste areas will be vented to the Unit Number 2 stack and will be monitored by the stack monitoring system (See Section 7.5.6.3.2.1.1).

7.5.6.3.2.1.4 Condenser Air Ejector Gaseous Monitor

The condenser air ejector system extracts the non-condensable gases from the condenser and exhausts to the Millstone stack. The presence of radioactivity in this line would indicate a primary-to-secondary tube leak in the steam generators. An adjacent-to-line gaseous monitoring system is provided to continuously monitor the air ejector vent line for the presence of radioactivity in the non-condensable gases. A high radiation or instrument failure alarm on this monitor will automatically isolate steam generator blowdown and effluents. This monitor may also provide indication of gross primary to secondary leakage and serves as a backup to the steam generator blowdown sample monitor.

7.5.6.3.2.1.5 Control Room Ventilation Gaseous Monitor

A fixed, GM, gaseous monitor is provided to continually monitor the control room atmosphere.

7.5.6.3.2.1.6 Filtered Waste Gas Monitor

When waste gases are to be released, the allowable radioactivity level of the release is established. An off line gaseous monitor is provided to monitor and record the radioactivity level of these releases. Should releases exceed the established level, the monitor will provide an alarm in the control room and initiate closure of valves (instrument failure alarm will also initiate valve closure) in the filtered waste gas discharge line.

7.5.6.3.2.1.7 Fuel Handling Exhaust Air Gaseous and Particulate Monitor

An off line airborne particulate and gaseous monitoring system is used to monitor the spent fuel pool ventilation exhaust air system.

The particulate and gaseous channels are monitored continuously. The monitor provides a charcoal sample for laboratory analysis of the iodine content.

7.5.6.3.2.1.8 Main Steam Line Monitors

Three monitors are provided to measure potential releases from the main steam lines following a steam generator tube rupture. These monitors each include an ion chamber mounted approximately twelve inches from the steam lines (one on each main steam line and one on an atmospheric dump line).

Remote equipment in the control room includes a six decade meter for each monitor, a multi channel recorder, alarm lamps for each monitor at the control room auxiliary cabinet, and a common alarm window at the main control board, which indicates equipment failure and high radioactivity.

The two unshielded gamma scintillation detectors, one mounted on each main steam line near the containment penetrations, measure N-16 activity in the main steam lines. A common radiation monitor provides main control board annunciation for N-16 equipment failure. The common

monitor also sends a signal to the plant process computer which calculates primary to secondary leak rate for each steam generator in gallons per day. The plant process computer records, displays, and provides main control board annunciation based on high leak rate. This monitoring system is Non-QA, and is designed to support Millstone's Primary to Secondary leak rate detection and monitoring program.

7.5.6.3.2.1.9 Control Room Inlet Monitors

Two redundant GM tubes are located on the inlet duct of the control room outside air supply. A high radiation or instrument failure alarm on either monitor will isolate normal control room ventilation and initiate use of the filtered recirculation system.

7.5.6.3.2.2 Components

When applicable, the various components used in the systems are as follows:

1. Isokinetic nozzles, designed in accordance with American National Standards Institute (ANSI) 13.1-1969, are used for extracting samples from process streams to assure representative sampling of particulates.

2. Sample chambers are fabricated of stainless steel to prevent corrosion and polished to reduce absorption (plate out) of radioactive materials.

3. The sample chambers and detection assemblies utilize lead shielding to prevent background radiation levels from affecting instrument sensitivity.

4. A radioactive check source, controlled from the readout module in the control room, provides a convenient operational check.

5. Readout instrumentation consists of log count ratemeters with wide range (five decade) capability. A failure alarm is provided to indicate loss of signal or power and an alarm is provided for high radiation level. On certain monitors an alert alarm is also provided.

6. Indication with visual and audible alarms are provided locally and on the radiation monitoring panel in the control room.

7. Multi channel recorders are used to record particulate and gaseous radiation levels continuously.

7.5.6.3.3 System Operation

Should the level of radioactivity exceed the alarm setpoint, an audible/visual alarm will be sounded at the local monitor, with the exception of the containment air radiation monitors which have no local alarm facility. In addition a visible and audible alarm will be annunciated on the

main control board. Should a loss of signal or loss of power occur on any monitor, a visible and audible alarm will also be annunciated on the main control board.

The gaseous and particulate monitors will be calibrated by using calibrated sources placed in a fixed geometry.

Administrative procedures established for the operators will ensure that releases will be kept to as low as practicable as specified in 10 CFR Part 50.34a and that releases will be less than the limits established by 10 CFR, Part 20 Sections 1301 and 1302 and Appendix B, Table 2.

In the event an audible alarm is sounded by an airborne radiation monitor, the operator will take appropriate action in accordance with plant procedures.

7.5.6.3.4 Availability and Reliability

Gaseous and particulate monitoring equipment is designed for operation over long periods with a minimum of maintenance. The sampling pumps are designed to be cleaned by a simple flushing operation. Filter replacement is a simple operation and the stainless steel sample chamber can be easily decontaminated by air purging and/or chemical flushing.

High-quality commercial-grade components are used throughout the system. Servicing of the electrical system is simplified by use of plug-in components.

The radiation monitoring panel was seismically qualified as documented in the report, “Technical Data Report, TDR-4174, Seismic Testing of Millstone Point Unit 2 Radiation Monitoring Instrumentation” dated April 1, 1974.

7.5.7 LOOSE PARTS MONITORING

7.5.7.1 Design Bases

7.5.7.1.1 Functional Requirements

The Loose Part Monitor (LPM) is designed to provide an indication of free and loose parts within the reactor coolant system (RCS). The LPM monitors the outputs of eight accelerometers: four on the reactor vessel and two on each of the steam generators. Metal-to-metal impacting within the RCS causes high-frequency, exponentially-decaying vibrations on the RCS shell which are detected and recorded by the LPM. The LPM provides an alarm in the control room to alert the operator so that further analysis and possible action can be undertaken.

7.5.7.2 Design Criteria

The LPM provides the means for detecting, recording, and evaluating metallic loose parts through analysis of transient acoustic signals produced by loose part impacts.

The LPM monitors the outputs of eight accelerometers:

a. Two on the lower reactor vessel.

b. Two on the upper reactor head.

c. Two on the inlet side of the channel head (one above and one below the tubesheet) of steam generator 1.

d. Two on the inlet side of the channel head (one above and one below the tubesheet) of steam generator 2.

The detection and recording circuitry are designed to capture impact data automatically with a minimum of operator intervention required.

The LPM is adjusted to achieve the maximum sensitivity commensurate with an acceptably low level of false alarms.

7.5.7.3 System Description

The detectors are accelerometers selected for operation in environments of high temperature and high radiation. Two types of cables are used in containment. Near the accelerometers, in areas of high temperature, mineral insulated, stainless steel jacketed cables are used. Beyond this, and to the containment penetrations, double insulated, twisted, low noise cables are used. The containment penetrations are low noise, coaxial type. Junction boxes are provided both inside and outside containment for ease of calibration and testing.

Outside the containment, eight differential charge amplifiers convert the output of each accelerometer to a proportional voltage signal which is routed to the control room electronics.

The control room electronics consists of signal conditioners, a data acquisition computer, real time displays and indicators, and interface circuits. A display screen and light emitting diodes are provided for monitoring and displaying the status of the incoming signals. Impact data is stored in local memory. Audio amplifiers and a headset permit the operator to monitor the sounds from each of the eight detectors.

7.5.7.4 System Operation

During normal operation, the LPM is in the automatic mode, ready to detect and record impact data from a loose part.

When an incoming signal(s) exceeds the preset trigger level, the impact data is collected, recored, and alarm(s) are generated. The waveforms recorded maintain a minimum 10 millisecond pre-trigger buffer and can capture pre-trigger, as well as post-trigger data. The alarm circuit notifies the operators that an event occurred. The recorded data is retained and can be downloaded to an external device for further analysis by Engineering and possible action.

7.5.7.5 Emergency Operation

In the event a loose part monitor alarm (annunciator) is sounded, the operator will check the LPM cabinet to verify the alarm in accordance with the applicable alarm response procedure. If the alarm can be attributed to valve testing, CEA movement or other known transient, the operator will reset the alarms and verify that they do not return immediately. The operator will then monitor all previously alarmed channels, using the headset, to verify that no unusual sounds are present.

If the alarm(s) cannot be attributed to known plant transient conditions, the operator will contact engineering department for evaluation. The alarms will then be reset, if possible. If the alarms reset, with no further alarms or unusual audible sounds, the operator's action is complete. If the alarms do not reset or if there are unusual audible sounds present, the operator will contact plant management for resolution.

Based on an evaluation of the nature of the loose part data by Engineering, the Engineering Department will recommend the appropriate action to protect the nuclear steam supply system.

7.5.7.6 Availability and Reliability

The components are conservatively rated for long life in this application. Accelerometers and cables are designed for the high radiation and high temperature environment to which they are exposed. Two installed spare accelerometers and cabling are provided for the lower reactor vessel to reduce future radiation exposure and speed repair in the event of a detector failure.

All other active components are located outside the containment for ease of accessibility during plant operation. Ready access to all electronic components, channel-to-channel parts interchangeability, and availability of a full complement of spare parts will result in high system availability during plant operation.

7.5.8 CHLORINE MONITORING SYSTEM (SYSTEM DELETED)

7.5.9 SECONDARY SIDE SAFETY RELIEF VALVE POSITION INDICATION

The Safety Relief Valve (SRV) position indication instrumentation complies with the requirements of Regulatory Guide 1.97, Rev. 2, D-18 variable.

The SRV instruments are designed to aid the operating crew in monitoring the status of the SRVs during normal and accident conditions.

The sensor/electronics which are located in a harsh environment comply with the Equipment Environment Qualification (EEQ) requirements of the Regulatory Guide 1.97 and are, therefore, treated as QA Category 1, Components.

The instrumentation provides signals to the Main Control Board annunciator and the plant computer to indicate when the SRVs are closed/not closed.

7.5.10 REACTOR COOLANT PUMP VIBRATION MONITORING SYSTEM

7.5.10.1 Design Basis

7.5.10.1.1 Functional Requirements

The reactor coolant pump (RCP) vibration monitoring system provides the capability to continuously monitor each RCP. The system has an alarm that is triggered if high vibration is detected. The system records vibration and shaft rotation data periodically and when an alarm is triggered.

7.5.10.2 Design Criteria

The RCP vibration monitoring system monitors the RCP pump, shaft and motor casing. Transducers are selected for vibration characteristics, ability to withstand the containment environment and maintainability.

System components are located outside of containment in normally accessible locations when required or feasible.

7.5.10.3 System Description

Motor vibration is monitored by two velomitor probes mounted near the upper motor bearing, 90° apart, that sense motor casing radial vibration.

Proximity probes monitor the RCP shaft. Two probes, mounted 90 degrees apart, sense shaft radial vibration. A third probe monitors shaft speed and provides a reference for vibration phase. The probes are mounted above the pump seal housing.

Cables connect the proximity probes to proximitors located in a junction box inside containment. Cables connect the velomitor probes to terminal strips in the junction box. Cables then carry the signals outside of containment to the RCP vibration monitoring system instrument rack, located in the west penetration room. Once in the instrument rack, the signals are digitized and transmitted to the RCP vibration monitoring system computer terminal. If high vibration is sensed, an alarm is sent to the main control board annunciator.

7.5.10.4 System Operation

During normal operation the instruments are continuously monitored and periodically recorded. When high vibration is detected, monitor alarm contacts actuate an alarm on the main board. “Acknowledge” and “Silence” options are available at the main board for the alarm. The alarm is cleared via the plant process computer.

7.5.10.5 Emergency Operation

An operating procedure provides instructions for operator response if an alarm is triggered.

7.5.10.6 Availability and Reliability

Components were selected and located for long term operability and maintainability. In addition, all active electronic components are readily accessible in the west penetration room. Spare parts are readily available for system components.

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION

Reactor Coolant System (see Figure 4.1–1)							
Component			**Protective Function ***	**Control Function ***	**Indicated**	**Recorded**	**Alarmed**
Reactor coolant loops							
	Flow		X		X		L
	Temperature (hot leg)		X	X	X	X	H
	Temperature (cold leg)		X	X	X	X	H
RCS mid-loop level							L
Pressurizer							
	Pressure		X	X	X	X	H, L, LL
	Level			X	X	X	H, L, LL
	Temperature						
		Water			X		
		Surge Line			X		L
		Spray Line			X		L
		Relief / Safety			X		H
		Discharge			X		H
		Steam			X		
Quench tank							
	Pressure				X		H
	Level				X		H, L
	Temperature				X		H

* Protective sensors and channels are independent of control sensors and channels.

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Engineered Safeguards and Containment (see Figure 6.1–1)				
Component		**Indicated**	**Recorded**	**Alarmed**
Low-pressure safety injection				
	Pressure	X		
	Flow	X		
	Temperature		X	
High-pressure safety injection				
	Pressure	X	H	
	Flow	X		
Safety injection tanks				
	Pressure	X	H,L	
	Level	X	H,L	
Containment spray header				
	Pressure			
	Flow	X		
	Temperature	X		

Component		Protective Function *	Control Function *	Indicated	Recorded	Alarmed
Refueling water storage tank						
	Level	X		X		H, L
	Temperature		X	X		H, L
Containment pressure		X		X		H, HH
Containment temperature				X		H **
Containment radiation		X		X	X	H

* Protective sensors and channels are independent of control sensors and channels.
** Plant computer alarm only.

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Chemical and Volume Control System (see Figure 9.2–2)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Letdown line Pressure	X	X		H, L
Flow		X		H
Temperature	X	X		H
Volume control tank				
Pressure		X		H, L
Level	X	X		H, L, LL
Temperature		X		H
Charging line				
Pressure		X		
Flow		X		L
Temperature		X		
Boric acid batching tank				
Temperature	X	X		
Boric acid tanks				
Level		X		H, L, LL
Temperature	X	X		H, L
Boric acid charging lines				
Pressure		X		
Flow	X	X	X	H, L
Miscellaneous				
Filter/strainer D/P		X		
Ion exchanger D/P		X		

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Sampling System (see Figure 9.6–1 and 9.6–2)	
Components	**Indicated**
Sampling lines	
Pressure	X
Flow	X
Temperature	X

Spent Fuel Pool Cooling and Purification System (see Figure 9.5–1)			
Component	**Indicated**	**Recorded**	**Alarmed**
Fuel pool			
Level	X [1]		H, L
Temperature		X	H
Heat exchanger flow path			
Pressure	X		
Temperature			H
Flow	X		L
Purification flow path			
Pressure	X		
Flow	X		
Filter D/P	X		H
Ion exchanger D/P	X		H
Ion exchanger strainer D/P	X		H

Notes: 1. Continuous Nonsafety augmented quality wide range level indication is provided remotely in the Cable Vault and East 480V Switchgear Rooms of the Auxiliary Building.

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Shutdown Cooling System (see Figure 9.3–1)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Shutdown heat exchangers				
Temperature (out)		X		
Discharge lines				
Pressure		X		
Flow	X	X		
Temperature		X	X	
Return lines				
Temperature			X	
Makeup water flow	X	X	X	H, L
Reactor coolant pump controlled bleedoff pressure	X			H, HH

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Steam and Feedwater System (see Figure 10.3–1 and 10.4–2)						
Component		Protective Function *	Control Function *	Indicated	Recorded	Alarmed
Steam generator						
Level (N.R.)		X	X	X	X	H, L, HH
Feedwater inlet temperature				X		
Feedwater flow			X		X	
Steam flow			X		X	
Outlet Pressure		X	X	X	X	L
Level (W.R.)				X		
Condenser						
Hot well level			X	X	X	H, L
Absolute pressure						
	Narrow range				X	H
	Wide range		X	X		
	Conductivity			X	X	H, HH
Turbine						
Throttle steam pressure				X	X	
First stage pressure			X	X		
Feedwater heaters level			X	X		H, L
Steam generator feedwater pumps						
Suction pressure				X		L
Discharge pressure			X	X		H
Speed			X			
Condensate pumps discharge pressure			X	X		
Condensate pump discharge total flow			X	X		
Auxiliary feedwater pumps						
Discharge pressure					X	
Feedwater regulator						

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION (CONTINUED)

Steam and Feedwater System (see Figure 10.3–1 and 10.4–2)					
Component	**Protective Function ***	**Control Function ***	**Indicated**	**Recorded**	**Alarmed**
Differential pressure		X	X		
Auxiliary feedwater supply					
Flow			X		
Valve position			X		
Moisture separator drain tanks - Level		X			H, L
Reheater drain tank - Level		X		X	H, L
Condensate storage tank - Level			X		H, L, LL
Miscellaneous					
Main steam pressure		X	X	X	
Main steam conductivity			X	X	H
Auxiliary steam pressure			X		
Condensate makeup conductivity		X	X		

* Protective sensors and channels are independent of control sensors and channels.

Circulating Water System (see Figure 9.7–1)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Condenser				
Inlet temperature		X		
Outlet temperature		X		
Service Water Pumps				
Discharge pressure	X	X		L

TABLE 7.5-2 OMITTED

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
A-01	Pressurizer Pressure		
	LR: P-103, 103-1	LR: 0-1600 psia	
	HR: P-102A, B	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 inches (20 inches above tube sheet to top of moisture separators)	
A-06	Deleted		
A-07	Deleted		
	AE-8152, 8154		
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)	10^{-8} to 200% FP	
	JI-001, JI-004 (MCB)	10^{-8} to 100% FP	
B-02	Control Rod Position	Full in or not full in	
B-03	RCS Soluble Boron Concentration	None	See Note 5

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
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	LR: 0-1600 psia	
	HR: P-102A, B	HR: 1500-2500 psia	
B-07B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 psig	
B-08	Core Exit Temperature 21 Sensors per Channel “A”, 22 Sensors per Channel “B”	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	LR: 0-1600 psia	
	HR: P-102A, B	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 490,000 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	

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
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 21 Sensors per Channel “A”, 22 Sensors per Channel “B”	200°-2300°F	
C-02	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	None	See Note 5
C-03	Analysis of Primary Coolant (Gamma Spectrum)	None	See Note 5
C-04A	RCS Pressure		
	Low range: P-103, 103-1	Low range: 0-1600 psia	
	High range: P-102A, B	High range: 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 feet to 7 feet, 0 to 490,000 gallons	See Note 2
C-06B	Containment Sump Water Level (Narrow Range) L-9155 (L-9155A Backup)	0-100%	See Note 3

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
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		
	Low range: P-103, 103-1	Low range: 0-1600 psia	
	High range: P-102A, B	High range: 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%	
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 Regulatory Guide 1.97, Rev. 3	See Note 6
C-14	Effluent Radioactivity-Noble Gases RM-8168 RM-8132B RM-8169	1×10^{-7} to 1×10^5 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	

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
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	
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	
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	

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-16	Steam Generator Level		
	L-1113A, B	0-100% Top of tube bundles to separators	
	B L-1123A, B		
	L-1114A, B	0-460 inches (20 inches above tube sheet to separator)	
	B L-1124A, B		
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 10^5 lbs/hr	
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 (CONTINUED)

Regulatory Guide 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, 8003B, 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	

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
E-03B	Plant Vent Flow		
	MP2: F-8412	10 to 10^5 cfm (MP2)	
	MP1: F-20-34	0 to 223,000 cfm (MP1)	
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	Sampler Particulate & Iodine filters are used for laboratory analysis	See Note 11
	RM-8132A/B	1 x 10^{-3} to 1 x 102 μCi/cc	
	RM-8168	1 x 10^{-7} to 1 x 105 μCi/cc	
	RM-8169		
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	0-360°	See Note 13
	Speed	0-100 mph	
	Temperature	(-10)-(+18) °F	
	Various		

NOTES

NOTE (1) Deleted

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. Contingency plans to obtain and analyze samples of primary coolant are contained within Chemistry Department implementing procedures.

NOTE (6) Deleted on Regulatory Guide 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-202, 2-HV-203A,2-HV-203B, 2-HV-206A, 2-HV-206B, 2-HV-207, 2-HV-208, 2-HV-210, 2-HV-211, 2-HV-212A, 2-HV-212B, ZS-8000, ZS-8001, ZS-8002, ZS-8003B, ZS-8004, 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 Regulatory Guide1.97, Rev. 2.

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

NOTE (14) Deleted.

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

TABLE 7.5-4 AREA RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.)		CONTROL LOGIC
				CONTROL ROOM	LOCAL	
RM-7890	Containment Personnel Access Match	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7891	Containment Refueling Machine Service Platform	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7892	Auxiliary Building Drumming and Decontamination Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7894	Charging Pump Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7895	Sampling Room	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7896	Radioactive Waste Processing Area (Elevation (-)25 feet 6 inches)	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7897	Radioactive Waste Processing Area (Elevation (-)45 feet 6 inches)	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A

TABLE 7.5-4 AREA RADIATION MONITORS (CONTINUED)

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.)		CONTROL LOGIC
				CONTROL ROOM	LOCAL	
RM-7899	Control Room	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-8139	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8142	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8156	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8157	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System

TABLE 7.5-4 AREA RADIATION MONITORS (CONTINUED)

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.)		CONTROL LOGIC
				CONTROL ROOM	LOCAL	
RM-9813	Drumming Area (Elevation (-)45 feet 6 inches)	10 to 10^6	Gamma-Scintillation	N/A	A, V	N/A
RM-8240	Containment High Range Radiation Monitor	10^3 to 10^{11}	Ionization Chamber	A, V	N/A	1-out-of-2 H2 Purge Valves Closure
RM-8241	Containment High Range Radiation Monitor	10^3 to 10^{11}	Ionization Chamber	A, V	N/A	1-out-of-2 H2 Purge Valves Closure

TABLE 7.5-5 LIQUID PROCESS/EFFLUENT RADIATION MONITORS

CHANNEL NO.	DESCRIPTION *	DECADE RANGE CPM	AUTOMATIC ISOLATION FUNCTION
RM-4262 *	Steam Generator Blowdown Sample Monitor	10 to 10^6	Steam Generator Blowdown, Blowdown Tank Discharges & Blowdown Sample Valves
RM-6038	Reactor Building Closed Cooling Water Monitor	10 to 10^6	None
RM-9049	Clean Liquid Waste Monitor	10 to 10^6	Clean Liquid Waste Discharge
RM-9116	Aerated Liquid Waste Monitor	10 to 10^6	Aerated Liquid Waste Discharge
RM-9327	Condensate Recovery Tank Monitor	10 to 10^6	Condensate Recovery Tank Discharge
CND-RE245	Condensate Polishing Waste - Neutralizing Sump Monitor	10 to 10^6	Condensate Polishing Facility Waste - Neutralizing Sump Discharge

* All monitors include an off-line, gamma scintillation detector.

TABLE 7.5-6 AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

Channel Number	Description	Detector Type	Sampler Type	Decade Range
RM-8132A *	Unit 2 Stack Monitor Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8132B	Unit 2 Stack Monitor Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
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-8169	Unit 2 Millstone Stack Wide Range	GM Tubes	Off-Line Gas & Part.	10 x E -7 to 10 x E+5 μCi/cc
RM-8123A *	Containment Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8123B	Containment Air Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
RM-8262A *	Containment Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8262B	Containment Recirculating Air Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
RM-8434A *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8434B	Radwaste Vent Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-5099	Condenser Air Ejector Discharge Monitor - Gaseous	Gamma Scintillation	Adjacent to Line Gas	10 - 10^7 cpm
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-8145A *	Fuel Handling Exhaust Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm

TABLE 7.5-6 AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

Channel Number	Description	Detector Type	Sampler Type	Decade Range
RM-8145B	Fuel Handling Exhaust Air Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-8997 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8998 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM 8999 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-9095	Filtered Waste Gas to Millstone Stack Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm

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

TABLE 7.5-6A PROCESS RADIATION MONITORS (STEAM)

Channel Number	Description	Detector Type	Sampler Type	Decade Range
RM4299A	Number 1 Steam Generator Main Steam Line Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4299B	Number 1 Steam Generator Atmospheric Dump Steam Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4299C	Number 2 Steam Generator Main Steam Line Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4296A	Number 1 Steam Generator N-16 Monitor	Gamma Scintillator	On Line Proximity	10^{-7} - 10^{-1} μci/ml
RM4296B	Number 2 Steam Generator N-16 Monitor	Gamma Scintillator	On Line Proximity	10^{-7} - 10^{-1} μci/ml

TABLE 7.5-7 CEA POSITION LIGHT MATRIX

Light Color	Regulating CEAs	Shutdown CEAs
Red	Upper electrical limit	Upper electrical limit
Green	Lower electrical limit	Lower electrical limit
Amber	Dropped CEA	Dropped CEA
White	Between upper and lower limits	(Not Applicable)
Blue	(Not Applicable)	Exercise limit

FIGURE 7.5-1 PRESSURIZER PRESSURE MEASUREMENT CHANNEL FUNCTIONAL DIAGRAM

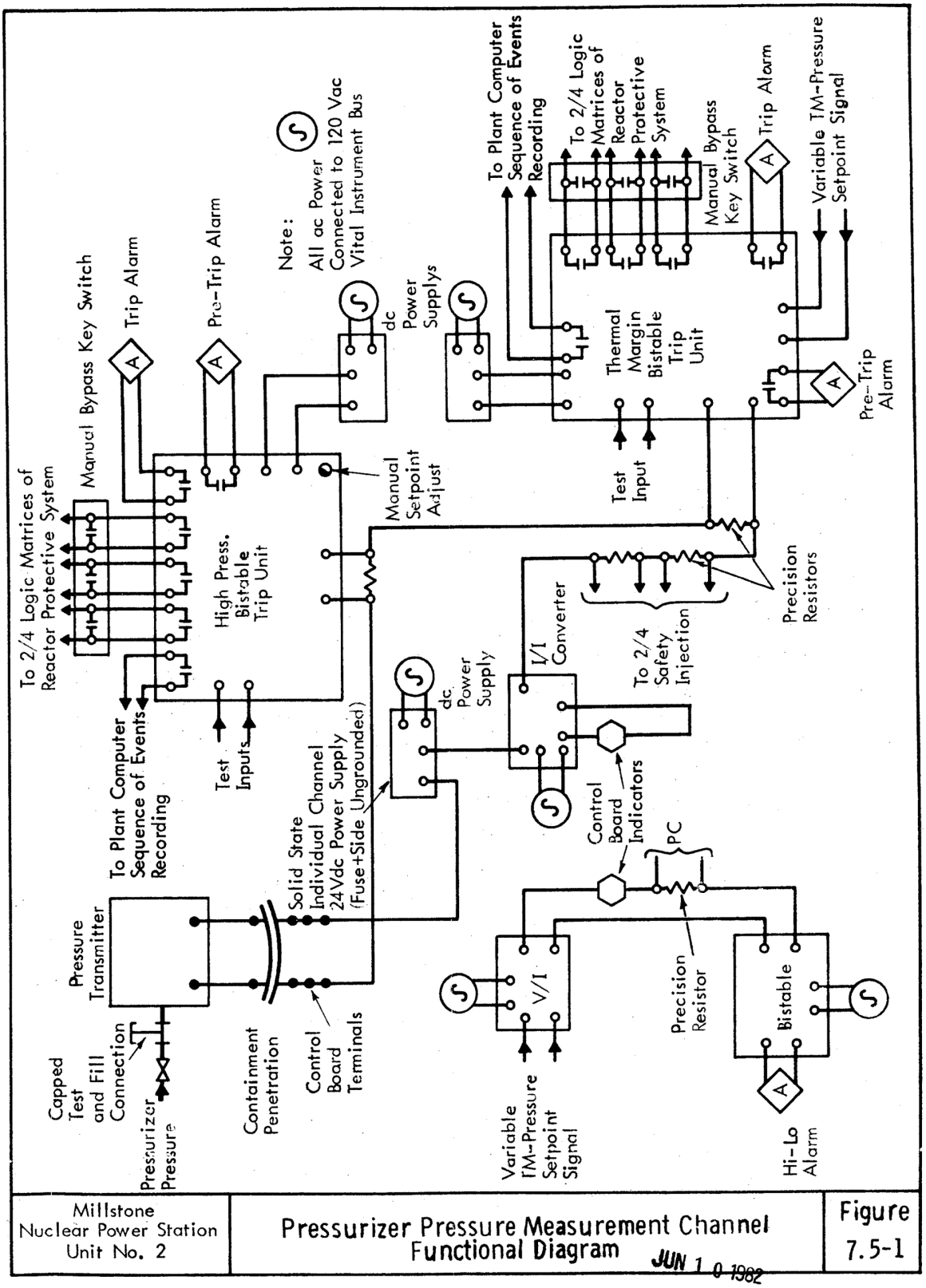
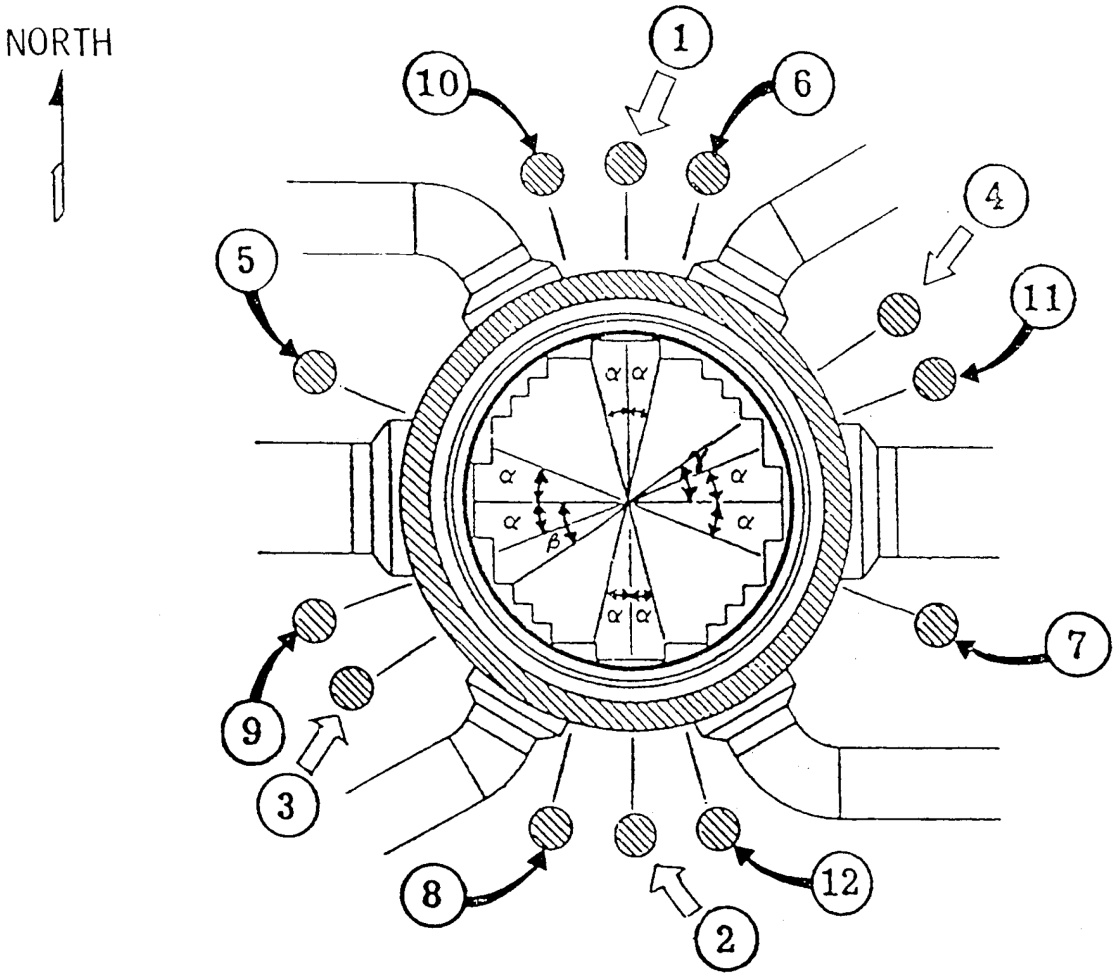


FIGURE 7.5–2 OUT-OF-CORE NUCLEAR DETECTOR LOCATION



α = 17°
β = 37½°
γ = 25°

WIDE RANGE LOGARITHMIC CHANNELS IDENTIFIED BY ⟹

CHANNELS	NUMBERS
WIDE RANGE LOGARITHMIC	1 - 4
CONTROL	10 & 12
SAFETY	5 - 8
SPARES	9 & 11

FIGURE 7.5–3 IN-CORE NUCLEAR DETECTOR ASSEMBLY

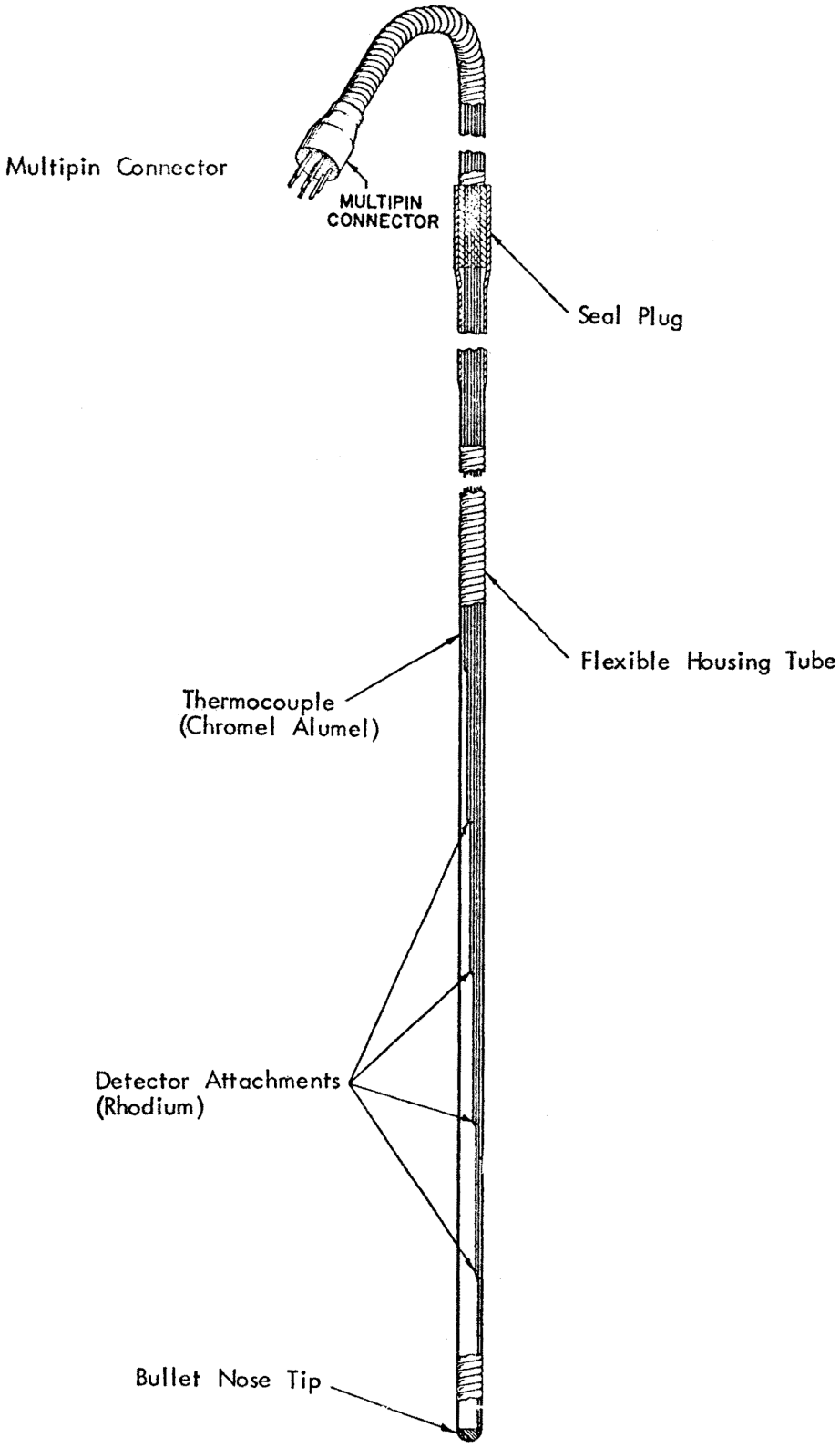


FIGURE 7.5–4 IN-CORE DETECTOR/CORE EXIT THERMOCOUPLE AND RVLM SENSOR LOCATIONS

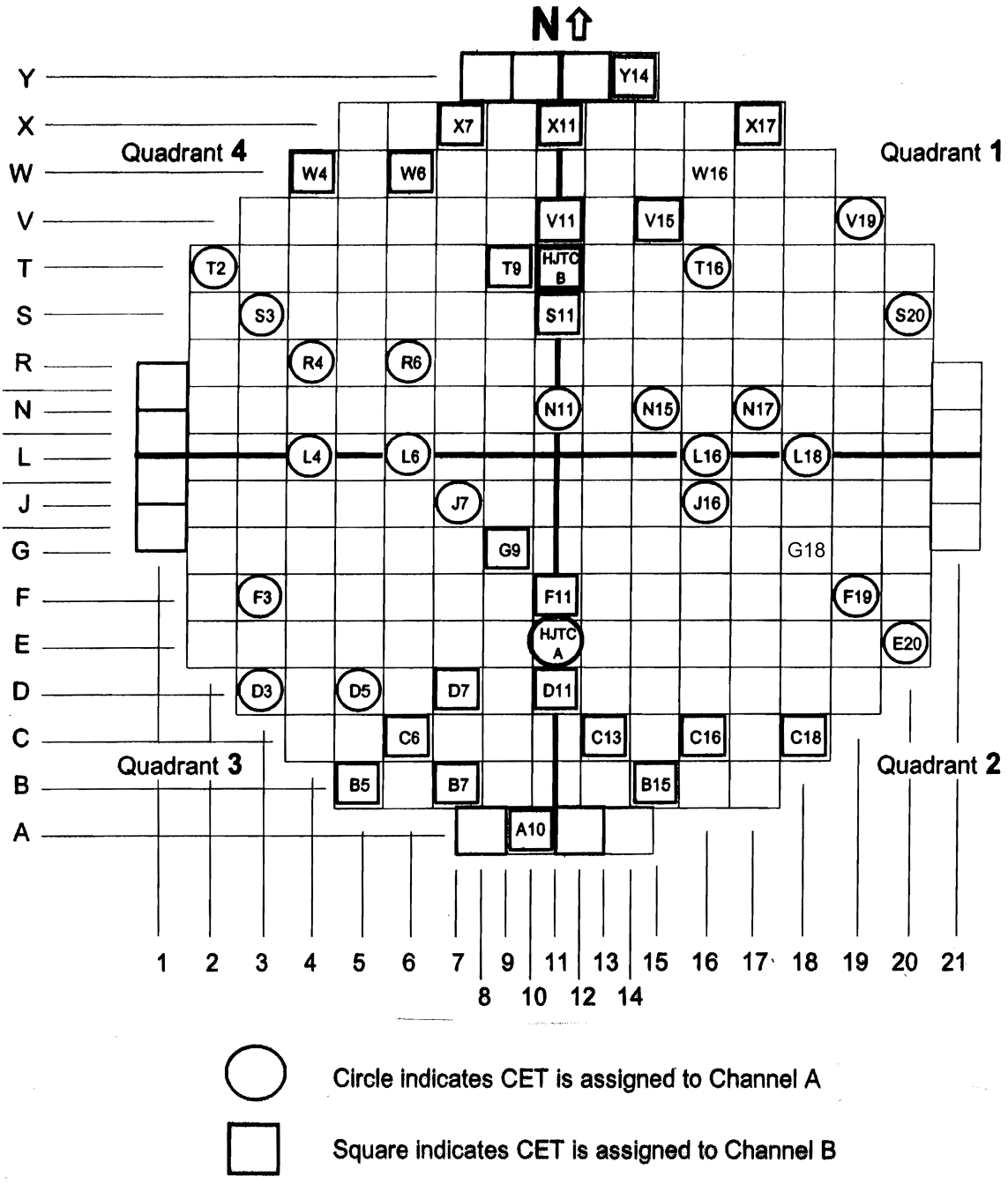


FIGURE 7.5–5 PROCESS RADIATION MONITOR SCHEMATIC

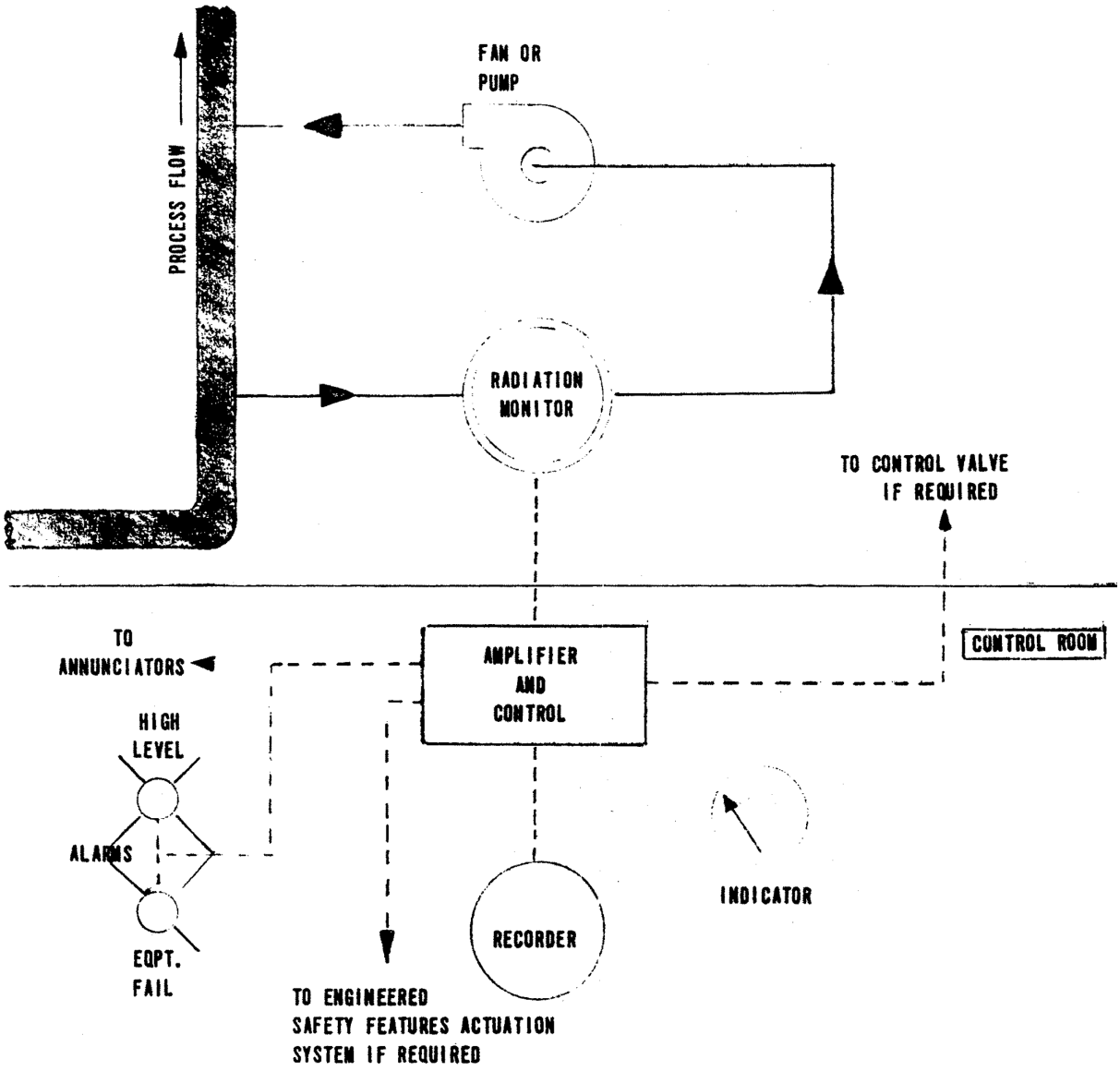


FIGURE 7.5–6 HEATED JUNCTION THERMOCOUPLE PROBE ASSEMBLY

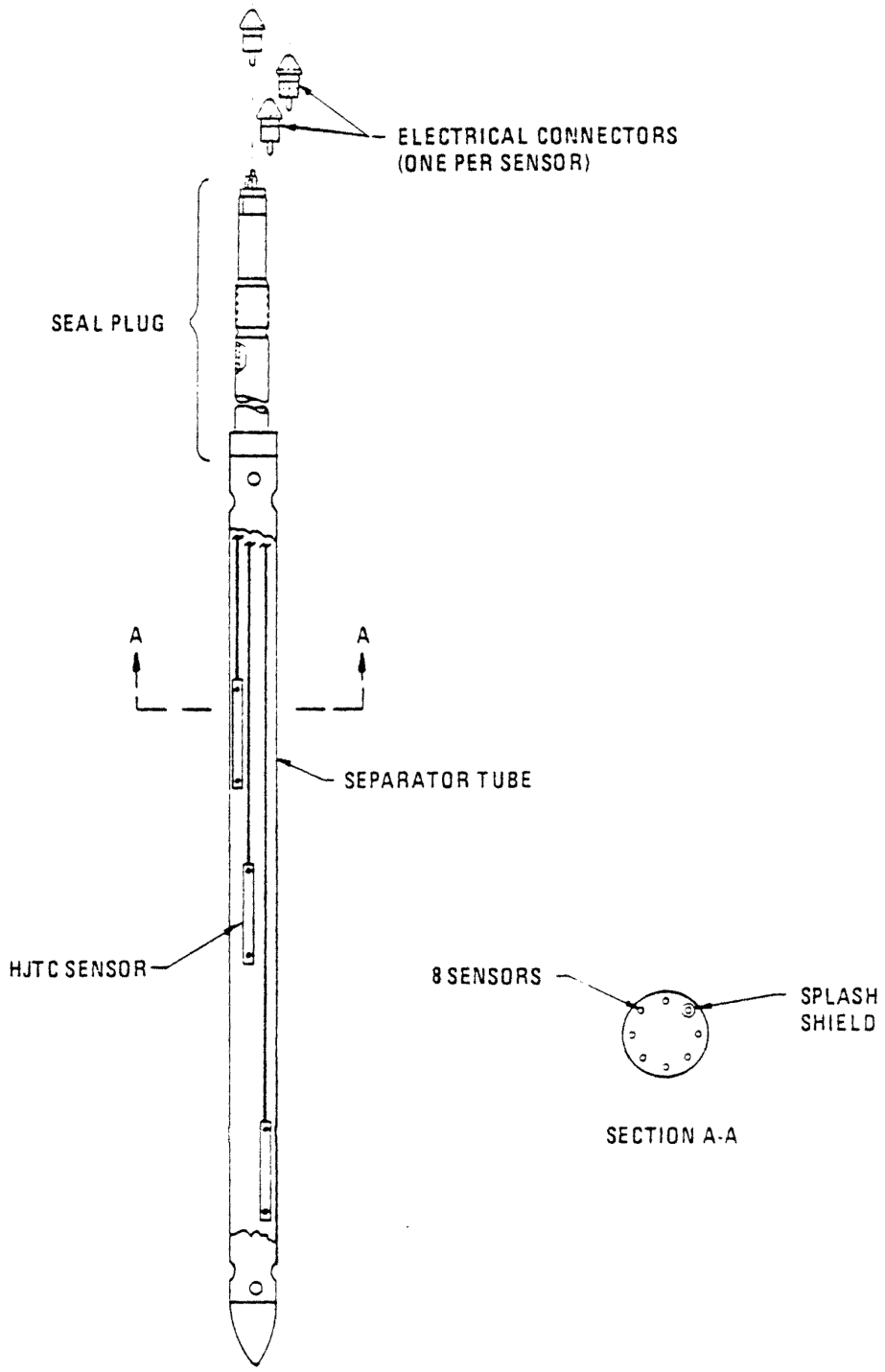


FIGURE 7.5–7 ICCMS FUNCTIONAL BLOCK DIAGRAM

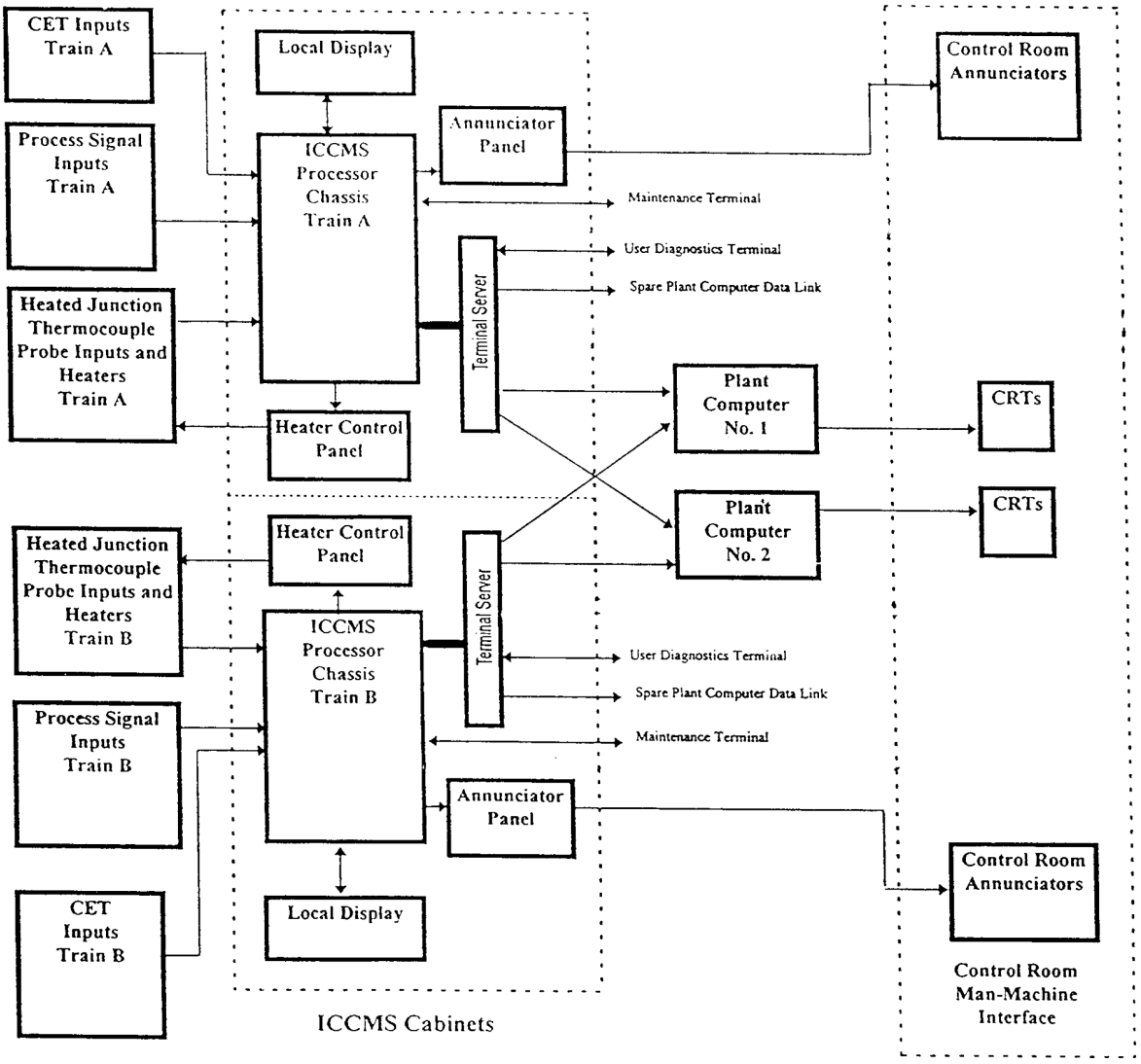


FIGURE 7.5–8 ICCMS DATA PROCESSING

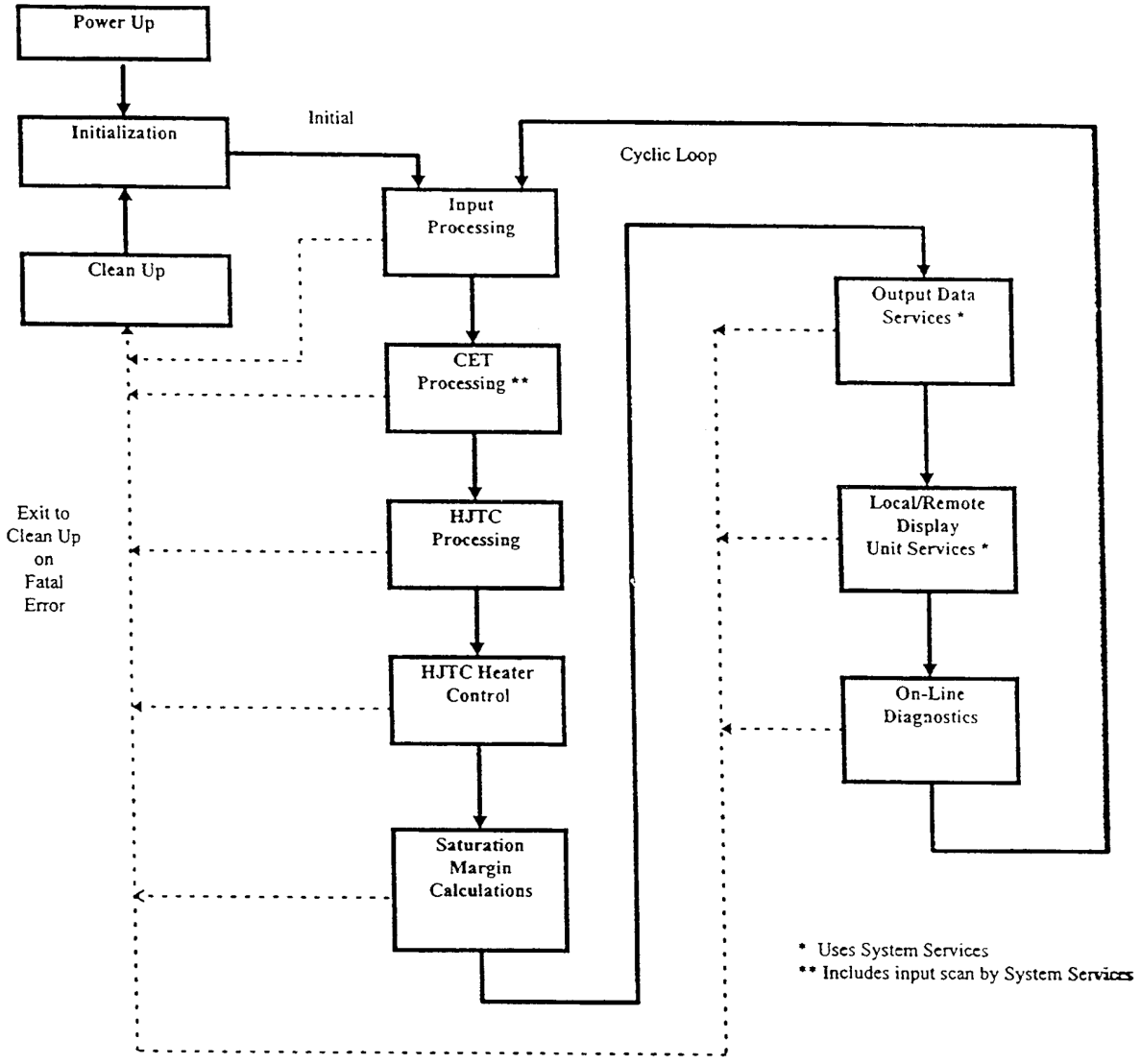
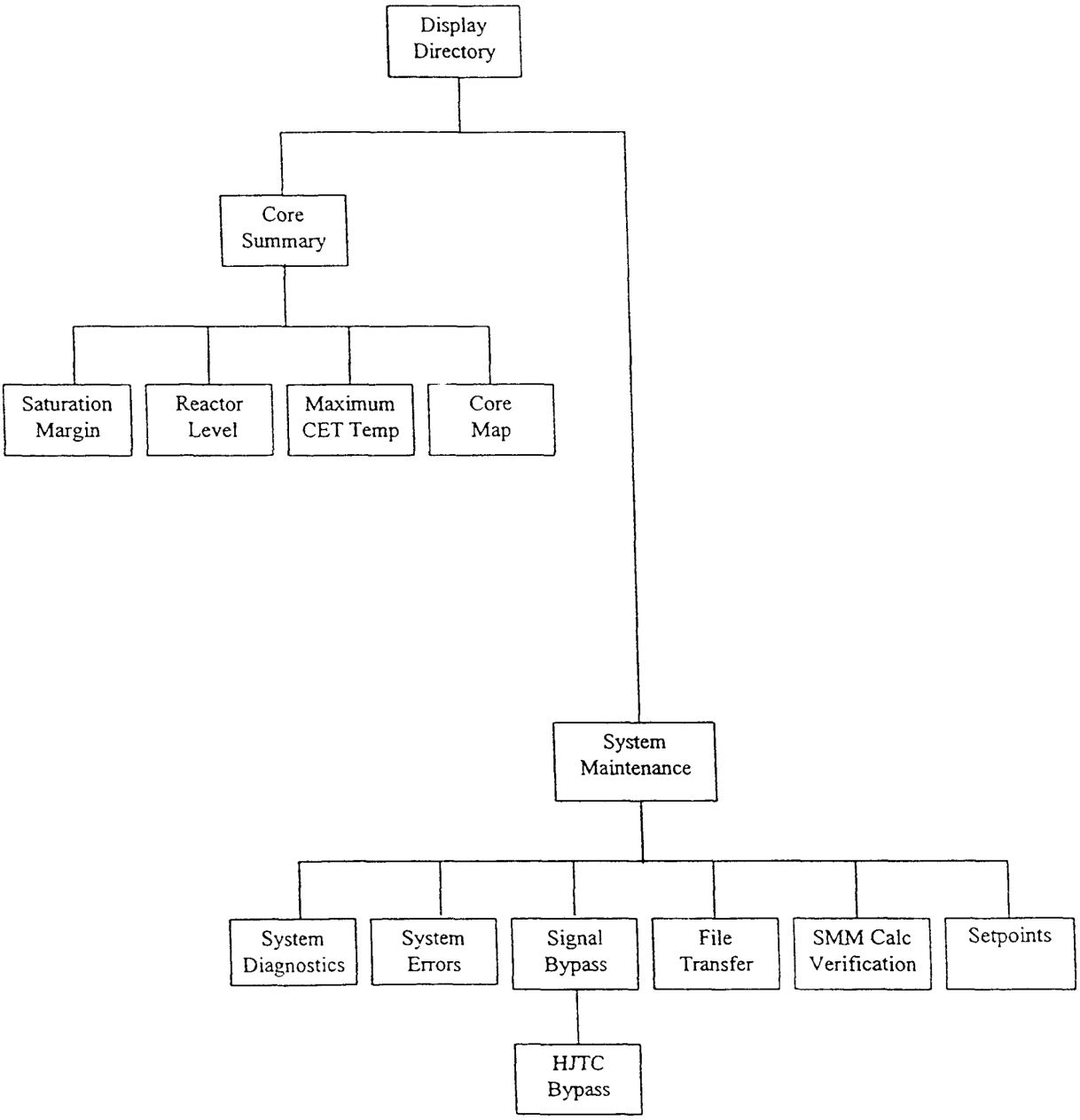


FIGURE 7.5–9 ICCMS DISPLAY HIERARCHY



7.6 OPERATING CONTROL STATIONS

7.6.1 CONTROL ROOM

7.6.1.1 Design Bases

7.6.1.1.1 Functional Requirements

The control room accommodates controls, alarms, indications, and instrumentation necessary to operate the nuclear power unit. This includes instrumentation for startup, normal operation, shutdown, and maintaining the plant in a safe condition under abnormal situations including loss-of-coolant accidents (LOCA).

7.6.1.1.2 Design Criteria

The following criteria have been implemented in the design of the control room:

a. The control room shall be equipped with adequate radiation protection to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 rem to the whole body, or its equivalent to any part of the body, for the duration of the accident as required by 10 CFR 50, Criterion 19.

b. Ventilation shall be provided to allow occupancy during and after a design basis accident (DBA).

c. Section 5.4.3 contains the structural design criteria for the auxiliary building in which the control room is located.

d. For Missile Protection, see Section 5.2.5.1.

7.6.1.2 Description

The Unit 2 control room is adjacent to the Unit 1 control room and is accessible from both the auxiliary and turbine buildings. It houses the enclosed walk-in duplex type main control boards, with integral consoles and miscellaneous instrument panels and racks, as well as its own air-conditioning and fire protection panels. All control boards which are safety related are designed to Seismic Class I requirements. In addition, the control room is equipped with separate, enclosed supervisor’s offices. Figure 7.6–1 shows the general arrangement of the control room.

The Unit 2 control room can be occupied under all credible accident conditions and is provided with redundant air conditioning systems, redundant filtration systems, an airborne radioactivity detector in the fresh air intake ductwork, and fresh air isolation dampers. A high radiation signal automatically switches the air conditioning system to the recirculation mode by closing the fresh air dampers, starting both filtration trains, and closing the exhaust dampers. The recirculation mode can also be manually actuated from the control room. Makeup outside air can be drawn

through the high efficiency particulate air (HEPA) charcoal filter assembly at operator discretion if needed. An area radiation monitor is provided to indicate and alarm when a high radiation level occurs inside the control room.

The materials used in the construction of the control room will not support combustion. Electrical wiring is flame resistant. Portable CO_2 fire extinguishers are placed in readily accessible stations in the control room, and respiratory protective equipment is available to the operators at all times.

7.6.2 MAIN CONTROL BOARDS

7.6.2.1 Design Bases

7.6.2.1.1 Functional Requirements

The main control boards are designed for plant control during startup, normal operation, shutdown, and emergency operation.

7.6.2.1.2 Design Criteria

The following criteria are used in the design of the main control boards:

a. Protective systems are separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, independence requirements of the protection system.

b. Wherever redundant protective or safeguards channels of a system are provided, devices and related wiring including the incoming terminal blocks of one channel are isolated from other channels to ensure that failure of any one channel will not affect any of the remaining channels.

A Control Room Design Review (CRDR) was performed in response to Nuclear Regulation (NUREG) 0737, Supplement 1. The control panels were modified to conform to the NUREG 0700 design criteria. All devices are grouped according to their functions. Circuits are channelized and physically separated in accordance with facility codes assigned to each loop and device.

Each circuit and raceway within the board is given a unique identification and each wire is color coded for easy channel identification. For separation criteria details, see Section 8.7.3.1.

Control Room panel-mounted indicators and valve-position indicating lamps that are used to satisfy Regulatory Guide 1.97 (Rev. 2) requirements have been specifically encoded by marking the instrument labels to identify them as Regulatory Guide 1.97 indications. See Table 7.5-3 for a list of Regulatory Guide 1.97 (Rev. 2) instruments.

7.6.2.2 System Description

7.6.2.2.1 System

The main control boards are comprised of the following eight sections.

Boards Function	Designation
a. Engineered safeguards	CO 1
b. Chemical and volume control (CVCS)	CO 2
c. Reactor coolant system (RCS)	CO 3
d. Reactivity control	CO 4
e. Steam generator (SG) and feedwater control	CO 5
f. Plant auxiliaries	CO 6
g. Turbine-generator-exciter control	CO 7
h. Station service electrical	CO 8

The eight sections are arranged in an “L” shaped array from left to right in the listed order with the reactivity control boards section in the center corner (see Figure 7.6–1). The boards are enclosed walk-in duplex switchboard type with an integral bench-type control console in the front. All sections are built and analyzed to meet Seismic Class I specifications. (See Section 5.8.5 for analysis procedure.) The control board, including all mounted equipment, will remain structurally intact such that no equipment will become loose, separated, or dislocated when subjected to a design basis earthquake (DBE). Furthermore, those devices which are safety related will function during and after a DBE. Figures 7.6–2 through 7.6–8 show the control board arrangement.

Control and indicating instruments, switches and indicating lights are installed on the front face of the control boards and on console tops. Instrument power supplies, protection devices, isolation amplifiers, and miscellaneous blind instruments as well as some of the secondary instruments are mounted on both the rear face of the control boards, and the instrument racks located adjacent to the control panels. Annunciator panels are installed across the top of each of the vertical section of each board. (The annunciator is discussed in Section 7.7.)

7.6.2.2.2 Components

With the exception of a few multi channel recorders, most process instruments on the main control boards are miniature electronic type using the 10 to 50 ma and 4 to 20 ma signal levels.

All miniature recorders are mounted on racks which can be pulled out on chassis tracks for ease of maintenance.

The turbine electrohydraulic control (EHC) panel is located on the front face of CO 7 board section and is furnished as part of the General Electric (GE) turbine-generator package. Arrangement of controls and monitoring instrumentation are shown in Figure 7.6–7, Sheet 1.

Control element assembly (CEA) position indication and controls are located on the front face of CO 4 board section. A detailed description is given in Section 7.5.3.

In general, control switches for circuit breakers, pump motors, and fans are equipped with pistol-grip handles and rectangular escutcheon plates. Control switches that operate valves and dampers are typically equipped with knob operators and legend plates. Switches for voltmeters and ammeters typically have round knurled handles. Some control switches are key operated.

Switches that operate the drain valves for the main and auxiliary steam turbine systems are back-lighted push-button type. These switches are grouped together on their respective board section for ease of identification.

Indicating lights for valve position, motor operation, etc., are generally the filament type with covered lense. The colors of the lenses, as shown in the listing, indicate the status of equipment operation:

Red: Utilized to denote a component in its active state, (i.e., equipment running, valve open, breaker closed).

Green: Utilized to denote a component in its passive state, (i.e., equipment off, valve closed, breaker open).

Amber: Utilized to denote an off-normal condition, (i.e., relay trip or off-normal status of equipment or parameter).

White: Utilized to denote power available or automatic operational mode.

Blue: Utilized to denote permissive conditions for operation fulfilled.

Wire terminations, except coaxial, triaxial and certain other detector and plug-connected cables, are made with ring-tongue solderless compression-type connectors which are securely bolted to barrier-type terminal blocks.

Incoming and outgoing 10 to 50 ma and 4 to 20 ma current loop signals are terminated at Class 1E, Test/Disconnect Terminal Block Assemblies.

Each control board is equipped with two copper ground busses, one on the front and one on the rear side of the duplex control board extending the entire length with the steel structure and connected to the bus so as to effectively ground the entire structure. Circuits requiring grounding are separately and directly connected to the ground bus. Device cases not otherwise grounded are grounded through the device enclosures and the steel structure of the control board.

7.6.2.3 System Operation

The main control boards are designed to allow control of the plant during all modes of operation. This includes normal plant operation, startup and shutdown, as well as emergency operation.

7.6.2.4 Availability and Reliability

7.6.2.4.1 Special Features

For the convenience of operating personnel, all major processes are presented graphically. Instruments are either located on process mimic lines or are shown connected to certain processes by influence lines. SGs, turbines, tanks, and etc., are shown schematically in the process. Pumps and valves in most cases are represented by their respective control switches and indicating lights. All process lines and symbols are color coded.

For those pumps and valves associated with safety actuation signals that are provided with a manual override capability, override is possible only subsequent to the safety actuation signal. The switch must first be turned to the "safe" condition before it can override the safety command signal. This operating procedure is designed to prevent an operator from overriding a safety signal unintentionally. When a safety-actuated device is overridden, an indication is provided by the status panel. Override capability for the AFW System actuated components is as described in Sections 7.3.3.1 and 10.4.5.3.

Two independent sets of manual safety signal actuation switches and safety signal block switches are provided on the CO 1 engineered safeguard section. The switches are guarded so as to prevent inadvertent operation. Details of the manual actuation switch functions are given in Sections 7.3.2.3 and 7.3.2.6.

Two pairs of reactor trip pushbutton switches are provided on the CO 4 board section. Details are given in Section 7.2.3.3.12.

In addition to the position indicating lights or running lights for valves, pumps, fans and dampers, each safety-related equipment, which is automatically initiated to satisfy safety functions, is provided with a white and blue status light and a set of Engineering Safety Feature (ESF) annunciators. These windows will alarm during an ESF actuation if any safety-related component fails to relocate to its accident position.

An ESF Status Light Panel designated as COIX, is designed to provide continuous indicating of the status of ESF equipment under all normal plant operation. The position of safety-related control valve RB-402 is also continuously indicated on Panel C01X because it responds to the same CIAS closure signal as isolation valve CH-089. No other ESF position monitoring attribute is provided for valve RB-402. Also, the position of containment air radiation monitor sample line isolation valves AC-527, -528, -529 and -530 are continuously indicated on Panel C01X using a single blue and white status light for each pair of isolation valves, sample supply and return, serving each of the two redundant monitor safety trains.

The status panel is designed to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971 for removing ESF equipment from operation and indication of bypasses.

The ESF Status Panel is a free standing enclosed fabricated steel panel with rear access doors as shown in Figure 7.6–11. The status panel is located adjacent to the Engineered Safeguards Panel CO 1 (see Figure 7.6–1). The panel is divided into three separate sections with barrier plates to provide adequate separation between channels. Each piece of ESF equipment is provided with a white and a blue indicating light on the front of the panel.

a. Blue Status Light - The blue status light will light when its respective valve, pump or fan is in its "safe" position, i.e., the state of the equipment after actuation by an ESF signal. Examples would be closure of a containment isolation valve on a containment isolation actuation signal (CIAS) or starting a safety injection pump on a safety injection actuation signal (SIAS). The blue light will be continuously lit when the equipment is in its "safe" state during normal plant operation.

b. White Status Light - The white status light is located adjacent to the blue status light for each piece of ESF equipment. The white light is normally off. It is arranged to go on when power is not available to the actuating circuit due to a blown fuse, tripped or racked out circuit breaker or loss of power, or when the equipment is administratively bypassed for maintenance. In addition, "LOCK-OUT" position of key operated and safety-related switches are indicated by a white light on the status panel.

   The white lights are powered from redundant 125-volt DC vital battery circuits.

c. Status Panel Light Grouping - The white and blue status lights are grouped on the panel according to each ESF actuation signal such as CIAS, SIAS, enclosure building filtration actuation signal (EBFAS), etc. This method of display provides ease in operator verification after an ESF actuation signal that all equipment actuated by a particular ESF signal has gone to its safe state since all blue lights within that grouping should go on.

Provisions are made so that it is possible to conduct online testing of the main steam isolation trip valves to ensure that the valves are capable of performing their functions when a trip signal is initiated.

Computer inputs from instrument loops that are connected to the control boards are obtained through precision resistors connected in series in the loops. These resistors are located at the terminal blocks at each applicable board section.

7.6.2.4.2 Tests and Inspections

Continuity and dielectric tests will be conducted on each wire prior to plant operation. Additionally, each instrument channel will be given a functional test using simulated signals at the

input terminals to prove the correct operation and proper polarity of all connected components in each channel.

7.6.3 RADIOACTIVE WASTE PROCESSING SYSTEM PANELS

7.6.3.1 Design Bases

7.6.3.1.1 Functional Requirements

The radioactive waste disposal system control panels are required to provide the controls, instrumentation, and alarms required to operate and monitor the waste process systems.

7.6.3.1.2 Design Criteria

The following criteria have been implemented in the design of the radioactive waste disposal system control panels.

a. Appropriate systems shall be provided for the radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to enable the operator to initiate appropriate control actions.

b. Means shall be provided for monitoring effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated incidents.

c. Same as Section 7.6.2.1.2.b.

7.6.3.2 System Description

7.6.3.2.1 System

Four radioactive waste processing control panels are provided, one for each type of radioactive waste.

Board Function	Designation
Clean liquid radioactive waste	C63
Gaseous radioactive waste	C61
Aerated liquid radioactive waste	C60
Condensate demineralizer waste	CDX

These control panels are located in the general vicinity of the respective equipment. The panels are free standing cubicles with instruments, switches and annunciators located on the front and

full-height double doors in the rear. All devices are grouped according to the function and are channelized in accordance with facility codes assigned to them. Separation requirements are essentially identical to those stipulated in Section 8.7.3.1. Figures 7.6–9, 7.6–10 and 7.6–12 through 7.6-15 show the panel layout arrangements.

7.6.3.2.2 Components

The radioactive waste control panels have a mixture of pneumatic and electronic instruments, and some capillary type temperature indicating devices. With the exception of one large case recorder, all instruments are the miniature type.

The electronic instruments utilize 10 to 50 ma signals whereas the 3 to 15 psig signal is standard for the pneumatic instruments.

Each of the radioactive waste control panels is equipped with an annunciator system installed across the top of the panel. The annunciator is of solid-state design and is complete with logic modules, flasher, horn, power supply, and pushbutton switches. All alarms are annunciated at their local panel with a master alarm provided in the control room. Annunciator details are found in Section 8.7.3.1.b.

Control switches, indicating lights, terminals, terminal blocks and wiring are identical to those used for the main control boards. All instrument tubings are seamless, soft-annealed copper, one-quarter inch OD. All fittings are brass flareless compression types.

7.6.3.3 System Operation

The radioactive waste processing control panels are designed to control and monitor the disposal of radioactive wastes in a safe and efficient manner.

7.6.3.4 Availability and Reliability

7.6.3.4.1 Test and Inspection

Continuity and dielectric tests were conducted on all wires. Additionally, each instrument channel was given a functional test using simulated input signals. Pneumatic systems were also given a leakage test.

7.6.4 HOT SHUTDOWN PANEL

7.6.4.1 Design Bases

7.6.4.1.1 Functional Requirements

Numerous design features are provided to make control room inaccessibility a highly unlikely event. However, in the event the operator is forced to abandon the control room, the hot shutdown

panel provides the instrumentation and controls (I&C) necessary to maintain the unit in the hot shutdown condition.

7.6.4.1.2 Design Criteria

The following criteria have been implemented in the design of the hot shutdown panel:

a. Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls, to maintain the unit in a safe condition during hot shutdown as required by Design Criterion 19.

b. Circuits in the control room shall be maintained intact.

c. The reactor shall be tripped.

7.6.4.2 System Description

7.6.4.2.1 System

The Hot Shutdown Panel is located in the West 480 Volt Switchgear Room on the 36 feet 6 inches elevation. This panel, which is designated as C21, is normally not in use. All plant operation including emergency shutdown can be accomplished at the main control boards in the control room. However, in the event the operator is forced to abandon the control room and the reactor is tripped, it is possible for the operator to maintain the unit in the hot shutdown condition by controls and instrumentation provided on this panel. This panel is built and analyzed to meet seismic Class I specifications. (See Section 5.8.5 for analysis procedure.) The panel, including all mounted equipment, will remain structurally intact such that no equipment will become loose, separated, or dislocated when subjected to a DBE.

Major equipment normally required for hot shutdown are shown in Table 7.6-1.

The following indications and controls are provided on the hot shutdown panel:

Indications:
Pressurizer level
Pressurizer pressure
Steam generator level
Steam generator pressure
Steam generator auxiliary feedwater (AF)
Condensate storage tank (CST) level
Cold leg temperature
Wide range neutron flux

Letdown temperature
Letdown pressure
Controls:
Steam dump to atmosphere
Letdown flow
Pressurizer spray
Charging pump
Pressurizer backup heater
AF valve
AF pump
AF pump crossover valve
AF pump turbine speed
Main Steam to AF
Pump turbine stop valve

All controls and instrumentation are compatible with those provided on the main control board. The panel arrangement is shown on Figure 7.6–10.

Subsequent to a hot shutdown, it is possible to bring the unit to the cold shutdown condition safely (external to the control room) with the following additional provisions and procedures:

Boric acid gravity feed valves can be manually operated to effect boric acid flow to the charging pump suction.

Low-pressure safety injection (LPSI) pumps can be controlled by control switches provided on the associated 4,160 volt emergency switchgear cubicles.

Major equipment normally used for cold shutdown is shown in Table 7.6-2.

7.6.4.2.2 Components

Instruments on this panel are generally the miniature electronic type using the 10 to 50 ma signal. Control switches, indicating lights, terminals, terminal blocks, and wiring are similar to those used for the main control boards.

7.6.4.3 System Operation

The hot shutdown panel is provided for emergency operation only. In an event which forces evacuation of the main control room, the operators will be able to bring the plant safely to the hot shutdown condition by controls provided on this panel. It includes controls and instrumentation for the pressurizer heaters and sprays, charging pumps, and auxiliary feedwater system (AFWS). Normally controlling instruments on this panel are set in the “By Pass” position, i.e., the main

control board has direct control of the final elements. However, the system is connected in such a manner that it is possible to override the main board instruments and take control at this panel.

7.6.4.4 Availability and Reliability

7.6.4.4.1 Special Features

Since the hot shutdown panel is never used except in case of an emergency, full-height doors are provided to close off the panel front. Doors are normally closed and locked. An open door is alarmed in the control room.

To ensure maximum availability, two channels of controls and instrumentation are provided on this panel. One channel is capable of performing its function to maintain hot shutdown. Separation requirements are given in Section 8.7.3.1.

7.6.4.4.2 Tests and Inspections

To ensure the integrity and availability of the system in case of an emergency, the controls and instrumentation are inspected and functionally tested in accordance with the Technical Specification.

7.6.5 FIRE SHUTDOWN SYSTEM PANELS

7.6.5.1 Design Basis

7.6.5.1.1 Functional Requirements

The Fire Shutdown System is comprised of I&C panels which provide the means to achieve hot shutdown in the event of a fire in any single fire area. This capability is achieved through three distinct design control measures. The first control requires that the Fire Shutdown System panels are located in a different fire zone from the main control room. This will ensure that both control stations will not fail because of fire in the main control room. The second control method is to ensure that all I&Cs used for the Fire Shutdown System are electrically isolated from the I&Cs used in the main control board. This will ensure that I&Cs used at both locations will not be harmed by a control room fire. The third method is to control cable routing so that a fire in any plant fire zone cannot simultaneously cause a loss of vital indication and control at both the Fire Shutdown System panels and at the main control board.

7.6.5.1.2 Design Criteria

The Design Criteria for the Fire Shutdown System is contained in 10 CFR 50 Appendix R. A summary of the applicable criteria is given below:

- A single train of I&Cs must be available to achieve hot shutdown following a fire in a single fire zone.

- Equipment will be qualified to maintain qualification when interfacing to existing safety systems.

- The system must be electrically isolated to prevent electrical faults from affecting the equipment.

- The system shall accommodate postfire conditions where offsite power is not available for 72 hours.

- I&Cs shall functionally provide:

    • Reactivity control

    • Reactor cooling makeup

    • Decay Heat Removal

7.6.5.2 System Description

The Fire Shutdown System is divided into two distinct systems: the first system, referred to as the “Bottle-Up Panels”, provide the means to remove power from all valves which can cause a loss of primary system water inventory and secondary system steam; the second system, referred to as the “Fire Shutdown Panel”, provides the vital indication and control for critical shutdown systems.

7.6.5.2.1 Bottle-Up Panels (C70A, C70B)

The Bottle-Up Panels are located in the East 480 V Switchgear Room along the exit route from the main control room into the turbine hall. Bottle-Up Panel C70A is used to isolate Z1 control schemes and Bottle-Up Panel C70B is used to isolate Z2 control schemes.

Each Bottle-Up Panel contains five (5) isolation switches. In the “normal” position, the isolation switches connect field cables directly to the control room. This position permits control from the control room. In the “isolate” position, the isolation switches open all the field cable wires which removes all voltage sources from the valve schemes. This action forces the affected valves to their closed (failsafe) position.

The following isolation valves may be closed/isolated at the Bottle-Up Panels:

C70A (Z1 Schemes)	C70B (Z2 Schemes)
(2 HS's) Main Steam Isolation Valves (MSIV) (MS-64A & MS-64B)	MSIV'S (MS-64A & MS-64B) - (2 HS’s)

C70A (Z1 Schemes)	C70B (Z2 Schemes)
(1 HS) SG #1 Blowdown Isolation (MS-220A)	SG #2 Blowdown Isolation (MS-220B) -(1 HS)
(1 HS) Pressurizer Power Operated Relief Valve (PORV)* (RC-402)	Pressurizer PORV* (RC-404) - (1 HS)
(1 HS) SG #1 Atmospheric Dump Quick Open Permissive and Normal Control Operation Isolation	SG #2 Atmospheric Dump Quick Open Permissive Isolation - (1 HS)

7.6.5.2.2 Fire Shutdown Panels (C9, C10)

The Fire Shutdown Panels (C9, C10) are located in the turbine building at the fifty-four (54') foot level. Panel C9 contains instrumentation signal processing electronics, which condition field transducer signals, process the signals for meter display at Panel C10 and isolates and retransmits the signals for use in the control room. Panel C10 contains meters to display the C9 instrumentation signals and contains control switches and isolation switches. C10 switches both isolate circuits from the control room, and permit local manual control.

The Control Room Panels use facility 1 and Z1 I&C schemes to ensure that the plant can achieve hot shutdown for a fire at panels C9 and C10. For this scenario, hot shutdown can be achieved from the main control room by using the facility 1 and Z1 I&Cs.

Field I&C cables are routed directly from the instrumentation or control device to panels C9 and C10. Routing is controlled so that most of the facility 2 and Z2 cabling does not pass through the same fire zone with functionally redundant facility 1 and Z1 channels. Where the same fire zone must be shared, the Z2 cabling is wrapped in a three hour fire barrier.

At panel C10, the field signal is isolated by switches prior to cable routing for use in the main control room. These switches are two position maintained contact switches. In the “remote” position, the field cable wiring is directly connected to the control room cable which permits control scheme operation remotely from the control room or hot shutdown panel. In the “local” position, the control room connection is broken and the field cable wires are connected to a local control handswitch, a local power source and local fuses. This position permits manual control at panel C10 only.

Faults in the control scheme cable runs from panel C10 to the control room are removed by placing the scheme isolation switch into the “local” position. In the local position, local fuses and local power sources are used to assure power availability independent of any control room fusing or power sources.

The following indications and controls are provided at the Fire Shutdown Panels:

Indication:

Steam Generator number 2 Pressure

Steam Generator number 2 Level
Pressurizer Pressure
Pressurizer Level
Hot Leg Temp
Cold Leg Temp
AF Flow *
CST Level
Wide Range Nuclear Instrument *
Boric Acid Tank LVL T8A *
Boric Acid Tank LVL T8B *
Steam Generator number 2 Atmospheric Dump Pressure *
Controls:
Controller for Steam Generator number 2 Atmospheric Dump Valve (ADV) HIC 4224A *
Controller for AF Flow FW - 43B *
Charging Pump P18C Control
Charging Pump P18B Control
Charging Pump Header Isolation Valve CH - 429 *
Letdown Isolation Valve CH - 089
Charging Line Distribution Valve CH - 519 *
Auxiliary Spray Valve CH - 517
Terry Turbine Steam Supply Valve (SV - 4188)
Terry Turbine Speed Control (HS - 4192C)

* Indicator or control may not be available for an R-1 fire. Alternate methods of compliance are provided in the Millstone Unit 2, 10 CFR 50 Appendix R Compliance Report Components

I&C switches on panel C10 are generally the miniature electronic type. Indicating lights, terminals, terminal blocks, and wiring on panels C9, C10, C70A and C70B are similar to those used for the main control boards.

7.6.5.3 System Operation

The Fire Shutdown System Panels (C9, C10, C70A and C70B) can be utilized for any emergency event which requires control room evacuation. In an event which forces evacuation of the main control room, the operators will be able to bring the plant safely to the hot shutdown condition by controls provided on these panels. The fire shutdown system panels can be used for any emergency event, unlike the hot shutdown panel which would lose control features if fire damages circuits in the Control Room.

7.6.5.4 Availability and Reliability

Always available and highly reliable.

7.6.5.4.1 Special Features

Since the fire shutdown system panels are never used except in case of an emergency, full height doors are provided to close off the panel fronts. Doors are normally closed. An open door is alarmed in the control room.

7.6.5.4.2 Tests and Inspections

To ensure the integrity and availability of the system in case of an emergency, the controls and instrumentation are inspected and functionally tested periodically.

7.6.6 MISCELLANEOUS LOCAL CONTROL PANELS

7.6.6.1 Design Bases

7.6.6.1.1 Functional Requirements

Local control panels for noncritical systems are located throughout the plant. Most of these local panels are part of packages furnished by mechanical and electrical manufacturers. Each panel contains the indications, controls, and alarms required for safe operation of the system.

TABLE 7.6-1 MAJOR EQUIPMENT NORMALLY USED FOR HOT SHUTDOWN

Pumps	
Auxiliary feedwater	10.4.5.4.4, 10.4.5.4.5, Table 10.4-1
Charging	9.2.2.2, Table 9.2-9
RBCCW	9.4.2.2, Table 9.4-1
Service water	9.7.2.2.2, Table 9.7-2
Reactor Coolant	4.3.3, Table 4.3-4
Valves	
Letdown	9.2.2.2, Table 9.2-4
Auxiliary feedwater regulating	10.4.5.3
Atmospheric dump	10.3.2.2, Table 10.3-1, Table 10.3-3
Pressurizer spray	4.3.7, Table 4.3-8
Storage Tanks	
Condensate storage	10.4.5.3
Heat Exchangers	
RBCCW	9.4.2.2, Table 9.4-1
Miscellaneous	
Pressurizer heaters	4.3.5
Containment cooling units	6.5
Diesel generator	8.3
Switchgear, 4160 volts	8.2
480 V emergency unit substation	8.4
480 V emergency motor control center	8.4
125 DC battery	8.5
125 VDC switchgear and distribution panels	8.5
AC/DC inverters and 120 VAC vital instrumentation buses	8.6
Reactor protection instrumentation	7.2

TABLE 7.6-2 MAJOR EQUIPMENT NORMALLY USED FOR COLD SHUTDOWN

Pumps	
Auxiliary feedwater	10.4.5.4.4, 10.4.5.4.5, Table 10.4-1
Charging	9.2.2.2, Table 9.2-9
RBCCW	9.4.2.2, Table 9.4-1
Service water	9.7.2.2.2, Table 9.7-2
Boric acid transfer	9.2.2.2, Table 9.2-11
Low pressure safety injection	6.3.2.2, Table 6.3-2
Reactor Coolant	4.3.3, Table 4.3-4
Valves	
Letdown	9.2.2.2, Table 9.2-4
Auxiliary feedwater regulating	10.4.5.3
Atmospheric dump	10.3.2.2, Table 10.3-1, Table 10.3-3
Pressurizer spray	4.3.7, Table 4.3-8
Storage Tanks	
Condensate storage	10.4.5.3
Boric acid storage	9.2.2.2, Table 9.2-10
Heat Exchangers	
RBCCW	9.4.2.2, Table 9.4-1
Shutdown	6.3.2.2, Table 6.3-3
Miscellaneous	
Pressurizer heaters	4.3.5
Containment cooling units	6.5
Diesel generator	8.3
Switchgear, 4160 volts	8.2
480 V emergency unit substation	8.4
480 V emergency motor control center	8.4
125 VDC battery	8.5
125 VDC switchgear and distribution panels	8.5
AC/DC inverters and 120 VAC vital instrumentation buses	8.6
Reactor protection instrumentation	7.2

FIGURE 7.6–1 GENERAL ARRANGEMENT CONTROL ROOM PLAN ELEVATION 35 FEET 6 INCHES

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 1)



FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–6 C06, CRP. FRONT VIEW ARRANGEMENT PLANT AUXILIARY (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–6 C06, CRP. FRONT VIEW ARRANGEMENT PLANT AUXILIARY (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–9 LOCAL CONTROL PANELS (C60) & (C61)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–10 LOCAL CONTROL PANELS (C63) & (C21)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–11 ENGINEERED SAFETY EQUIPMENT STATUS PANEL LAYOUT (C01X)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–12 CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX

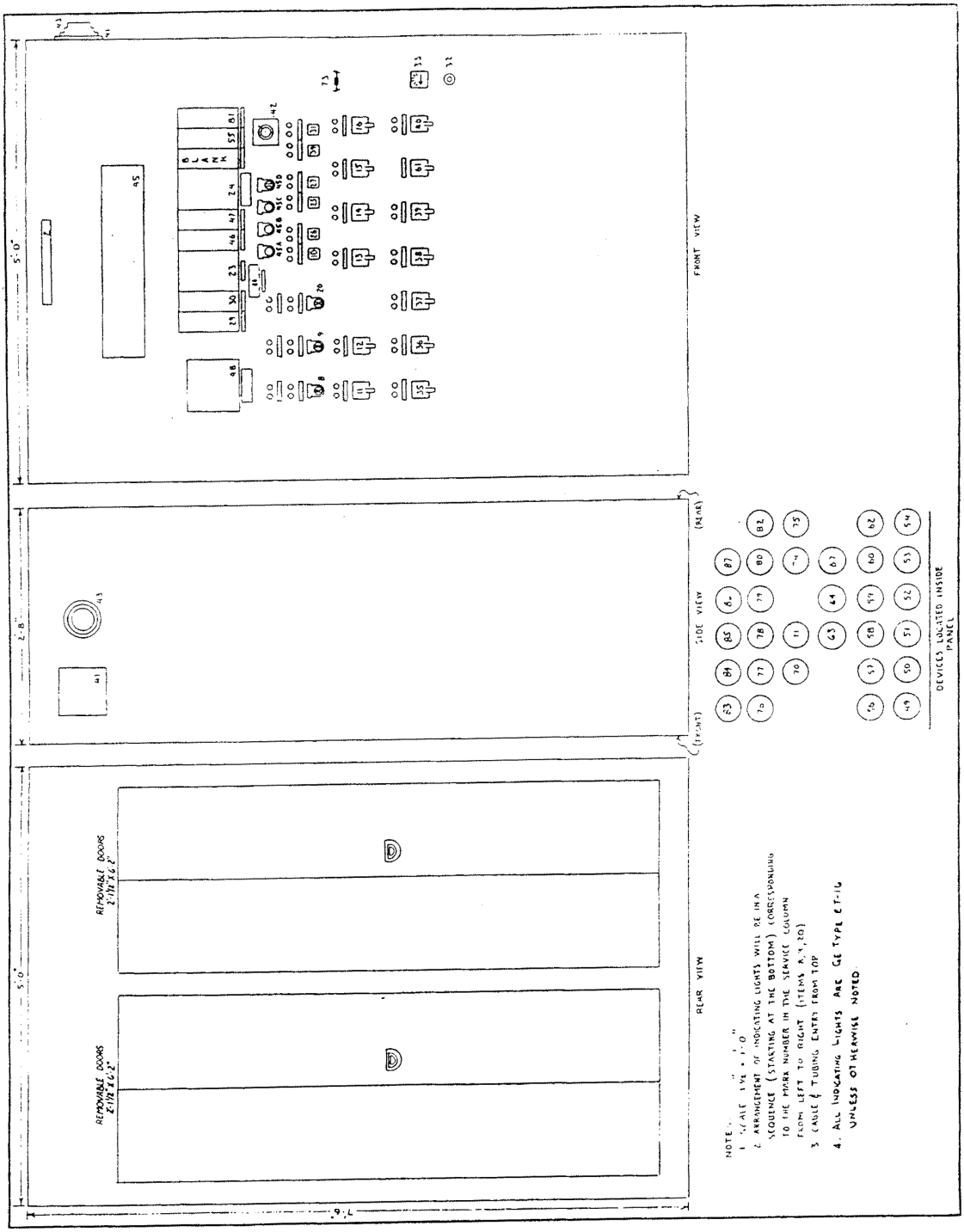


FIGURE 7.6–13 CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX

ITEM NO.	MARK NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
1	2CND-PNLCDX	S	STEEL PANEL 5'-0"W X 2'-8"D X 7'-8"H		
2		S	NAMEPLATE - CONDENSATE DEMINERALIZER WASTE TREATING PNL. 2CND-PNLCDX		
3					
3A					
3B					
3C					
3D					
4					
5					
6					
7					
8	43-2CNDK02	S	SELECTOR SWITCH ESK-3G DET AUG GE-TYPE CR2940 W/2 SETS RAG LIGHTS	ACID NEUTRALIZATION FEED PUMP DISCHARGE VALVES 2CND-AOV242B,A	ESK-7JU
9	43-2CNDK03	S	SELECTOR SWITCH ESK-3G DET AUG GE-TYPE CR2940 W/2 SETS RAG LIGHTS	CAUSTIC NEUTRALIZATION FEED PUMP DISCHARGE VALVES 2CND-AOV243B,A	ESK-7JV
10	PB-2CND-AOV244A	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/ 1 SET RAG LIGHTS	WASTE NEUTRALIZING SUMP PUMP DISCHARGE VALVE AOV244A	ESK-7KT
11	1-2CND-P8	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	ACID NEUTRALIZATION FEED PUMP P8	ESK-6GQ
12	1-2CND-P10	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	CAUSTIC NEUTRALIZATION FEED PUMP P10	ESK-6GO
13	1-2CND-P6A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6A	ESK-6GK
14	1-2CND-P6B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6B	ESK-6GN
15	1-2CND-P6C	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6C	ESK-6GP
16	1-2CND-P6D	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION PUMP PUMP P6D	ESK-6GP
17					
18					
19					
20	43-2CNDK01	S	SELECTOR SWITCH ESK-3G, DET AUG, GE-TYPE CR2940 W/2 SETS RAG LIGHTS	URC DISCHARGE VALVES TO WASTE NEUTRALIZATION PUMP 2CND-AOV241B,A	ESK-7JU
21					
22					UNUSED
23	2CND-FR246	S	ELECTRONIC 1 PEN FLOW RECORDER	WASTE NEUTRALIZATION SUMP PUMP DISCHARGE TO SEAL WELL	LOOP 2CND-246
24	2CND-AR297	S	ELECTRONIC 2 PEN PH RECORDER	WASTE NEUTRALIZATION SUMP PUMPS DISCHARGE	LOOP 2CND-297
25	PB-2CND-AOV245	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE VALVE TO SEAL WELL 2CND-AOV245	ESK-7JW
26	PB-2CND-AOV298A	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE RECIRC. VALVE 2CND-AOV298A	ESK- 7JX
27	PB-2CND-AOV299	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PP TO CONDENSATE DEMINERALIZER LIQUID WASTE	ESK- 7JW
28					
29	2CND-L1237A	S	ELECTRONIC LEVEL INDICATOR	WASTE NEUTRALIZATION SUMP (TK 10)	LOOP 2CND-237
30	2CND-L1237B	S	ELECTRONIC LEVEL INDICATOR	WASTE NEUTRALIZATION SUMP (TK 11)	LOOP 2CND-237
31	PB-2CND-AOV298B	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE RECIRC. VALVE 2CND-AOV298B	ESK-7JX
32		S	RECEPTACLE, CANNON TYPE XLR-4-13N	MAINTENANCE COMMUNICATION STATION	2570.000-267-008A
33		S	SELECTOR SWITCH, GAI-TRONICS SERIES 100, 4P, 6POS	MAINTENANCE COMMUNICATION STATION	2570.000-267-008A
34	PB-2CND-AOV244B	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET RAG LIGHTS	WASTE NEUTRALIZATION SUMP PUMP DISCHARGE VALVE AOV244B	ESK-7KT
35	1-2CND-P11A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	ACID FEED PUMP P11A	ESK-6MG
36	1-2CND-P11B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	ACID FEED PUMP P11B	ESK-6MG
37	1-2CND-P12A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	CAUSTIC FEED PUMP P12A	ESK-6MH
38	1-2CND-P12B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RAG LIGHTS	CAUSTIC FEED PUMP P12B	ESK-6MH

LEGEND

- △ CHANGE. ISSUE NUMBER INSIDE
- Ⓐ COMPLETE ADDITION
- Ⓓ COMPLETE DELETION
- S FURNISHED BY SELLER
- E FURNISHED BY ENGINEERS
- * EQUIPMENT PROVIDED BY INFILCO DEGREMONT INC.

NOTES:

1. ALL ELECTRONIC BISTABLES WILL HAVE A CONTACT OUTPUT ISOLATOR. AN ITE J10 RELAY WILL ALSO BE USED IN CASES WHERE THE ELECTRONIC BISTABLE IS USED FOR A CONTROL FUNCTION. REFER TO SKETCH 12552-I-1.

FIGURE 7.6–14 CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX

ITEM NO.	MARK NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
39	1-2CND-P7A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RED LIGHTS	WATER RECOVERY SUMP PUMP P7A	ESK-6HF
40	1-2CND-P7B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SB1 W/1 SET RED LIGHTS	WATER RECOVERY SUMP PUMP P7B	ESK-6HF
41		S	HORN, EDWARDS FLUSH-TYPE CAT. NO. 871-P1	ANNUNCIATOR	
42		S	TIMER, EAGLE MODEL 1D1-08-A6-00	40 MIN RANGE, STANDARD; PUSHBUTTON RESET TIMER	
43		S	BUZZER, EDWARDS ADAPTABUZZER CAT.NO. 340A-N6	TIMER	
45	2CND-ANN01	S	ANNUNCIATOR BETA SERIES 1211-A		2 X 10 X 8 WIDE
45A	PB	S	PUSHBUTTON; ESK-2K, DET BQ GE-TYPE CR2940	ANN "SILENCE"	
45B	PB	S	PUSHBUTTON; ESK-2K, DET BQ1 GE-TYPE CR2940	ANN "ACKNOWLEDGE"	
45C	PB	S	PUSHBUTTON; ESK-2K, DET BQ2 GE-TYPE CR2940	ANN "RESET"	
45D	1	S	CONTROL SWITCH; ESK-3P, DET CS1 GE-TYPE CR2940	ANN "LAMP TEST-GROUND TEST"	
46	2CND-AI297A	S	ELECTRONIC PH INDICATOR	WASTE NEUTRALIZATION SUMP (TK10) PUMP DISCHARGE	LOOP 2CND-297
47	2CND-AI297B	S	ELECTRONIC PH INDICATOR	WASTE NEUTRALIZATION SUMP (TK11) PUMP DISCHARGE	LOOP 2CND-297
48	2CND-LR237	M	ELECTRONIC 8 PEN RECORDER; L&N SPEEDOMAX M MARK III	WASTE NEUTRALIZATION AND WATER RECOVERY SUMPS	LOOP 2CND-237
49	2CND-LY237A	S	SIGNAL CONDITIONER I/E	2CND-LS237A1, A2 & 2CND-LT237A	LOOP 2CND-237
50	2CND-LY237B	S	SIGNAL CONDITIONER I/E	2CND-LS237B1, B2 & 2CND-LT 237B	LOOP 2CND-237
51	2CND-LS237A1	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) LEVEL ALARM LO	LOOP 2CND-237
52	2CND-LS237A2	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) LEVEL ALARM HI	LOOP 2CND-237
53	2CND-LS237B1	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) LEVEL ALARM LO	LOOP 2CND-237
54	2CND-LS237B2	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) LEVEL ALARM HI	LOOP 2CND-237
55	2CND-LT300	S	ELECTRONIC LEVEL INDICATOR	WATER RECOVERY SUMP	LOOP 2CND-300
56	2CND-LY300	S	SIGNAL CONDITIONER I/E	2CND-LS300A, B, & 2CND-LT300	LOOP 2CND-300
57	2CND-LS300A	S	ELECTRONIC BISTABLE	WATER RECOVERY SUMP LEVEL ALARM HI	LOOP 2CND-300
58	2CND-LS300B	S	ELECTRONIC BISTABLE	WATER RECOVERY SUMP LEVEL ALARM LO	LOOP 2CND-300
59	2CND-AS297A	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) PH LEVEL HI ALARM	LOOP 2CND-297
60	2CND-AS297B	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) PH LEVEL HI ALARM	LOOP 2CND-297
61	43-2CND-P7A	S	SELECTOR SWITCH; ESK-3H DET BH GE TYPE SB1	WATER RECOVERY SUMP PUMP P7A / P7B	ESK-6HF
62	74-2CNDN15	S	ALARM RELAY: ITE J13PA3012, 125VDC (3 NO. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT LOSS OF CONTROL POWER	ESK-7NV
63	86-AOVE45		AUXILIARY RELAY; ITE J13P6012; 125VDC (6 NO. CONTACTS)	WASTE NEUTRALIZATION SUMP DISCHARGE VALVE	-32010 SW30 (ESK-3JJ)
64	86-AOVE99		AUXILIARY RELAY; ITE J13P6012; 125VDC (6 NO CONTACTS)	WASTE NEUTRALIZATION SUMP TO LIQUID WASTE SYSTEM	-32010 SW30 (ESK-3JW)
65					
66					
67	71X5-2CNDN15	S	AUXILIARY RELAY; ITE J13P3012, 125VDC (3 NO. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NV
68					
69					
70	71X8-2CNDN15	S	AUXILIARY RELAY: ITE J13P3012, 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NV
71	71X9-2CNDN15	S	AUXILIARY RELAY; ITE J13P3017, 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NV
72					
73		S	HANDSET HANGUP HOOK GTA-TRONICS PART NO 5884	MAINTENANCE COMMUNICATION STATION	
74	2CND-FY246	S	SIGNAL CONDITIONER I/E	2CND-FR246	LOOP 2CND-246
75	2CND-LS318A	S	ELECTRONIC BISTABLE	CAUSTIC STORAGE TK LEVEL LOW-LOW INTLK P12A-B	LOOP 2CND-318
76	2CND-LS318B	S	ELECTRONIC BISTABLE	CAUSTIC STORAGE TK LEVEL ALARM LOW	LOOP 2CND-318
77	2CND-LS322A	S	ELECTRONIC BISTABLE	ACID STORAGE TK LEVEL LOW-LOW INTLK P11A,B	LOOP 2CND-322

FIGURE 7.6–15 CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX

ITEM NO.	MARK NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
78	2CND-LS322D	S	ELECTRONIC BISTABLE	ACID STORAGE TK LEVEL ALARM LOW	LOOP 2CND-322
79	2CND-LY318	S	SIGNAL CONDITIONER I/E	2CND-LS318A,B	LOOP 2CND-318
80	2CND-LY322	S	SIGNAL CONDITIONER I/E	2CND-LS322A-B	LOOP 2CND-322
81	2CND-LI318A	E	ELECTRONIC 2 POINT LEVEL INDICATOR	ACID & CAUSTIC STORAGE TANKS	LOOP 2CND-318
82	74-2CND-PNLCDX	J	AUXILIARY RELAY: ITE J13PA3012, 125VDC (3 N.O CONTACTS)	2CND-PNLCDX - LOSS OF 125VDC TO PANEL ALARM	ESK-1NV
83	71X0-2CND18	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-1NV
84	71X1-2CND18	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-1NV
85	71X2-2CND18	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-1NV
86	2CND-FY846A	J	SIGNAL CONDITIONER (SQ. ROOT INTEGRATOR)	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-846
87	2CND-FY846B	J	SIGNAL CONDITIONER (PULSE CONVERTER)	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-846
88	2CND-FQI846	J	FLOW TOTALIZER	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-846

7.7 CONTROL ROOM ANNUNCIATION

7.7.1 DESIGN BASES

7.7.1.1 Functional Requirements

The Main Control Board Annunciator provides the operator with visual and audible indications from external contacts if a limiting condition is being approached or abnormal conditions exist for any system so annunciated.

7.7.1.2 Design Criteria

The Main Control Board Annunciation System has been designed to meet the following criteria:

1. Respond to a permanent, fixed time momentary, or time dependent momentary alarm condition.

2. Via visual and audible devices (lights and electronic horns) indicate this response on the front panels of the Control Board.

3. By visual and audible devices indicate a return to normal condition for the point in alarm.

4. Provide means to silence the initial alarm horn and still maintain a bona fide alarm indication via lights.

5. Provide means of testing alarm lights without interfering with the normal status of the various systems in the plant.

6. Provide auxiliary contact outputs to the plant computer.

7. Provide means to silence audible alarms after a reactor trip via a Master Silence Switch.

7.7.2 SYSTEMS DESCRIPTION

7.7.2.1 System

The annunciator windows are installed in panels across the top front of each board section. The annunciators in each panel are grouped in accordance with the function of each board section, in general, i.e., alarms for Engineered Safeguards on CO1, Chemical & Volume Control Systems alarms on CO2, etc. Board sections CO1, CO4, CO5, and CO8 have individual electronic horns and alarm push-button controls and horn.

The annunciator logic is of solid-state design, and housed in free-standing cabinets located external to the Main Control Board. The system is powered by a primary AC to DC power supply

fed from IAC VR11 and a backup AC to DC power supply fed from IAC VR21, which are both backed by separate UPS batteries capable of carrying their respective loads for a minimum of one hour. IAC VR11 and IAC VR21 will automatically swap to their respective diesel generator backed AC source in the event of a failure of the UPS. This ensures the annunciator system with a reliable power source. The voltage across the field contacts is 125 VDC except for parts of nine circuits which go through the RPS cabinets. Interposing relays were installed in those circuits to lower the voltage to 28 VDC in the RPS cabinets and to reduce potential common mode noise. The annunciator's DC voltage is isolated from the IAC power supplies through the two redundant AC to DC converters.

Ground detectors are provided to alarm whenever a ground exists on any annunciator contact input circuit or any annunciator power supply. Portions of nine circuits which go through the RPS cabinets are not monitored by ground detectors. These circuits are protected by fuses. The assembled annunciators are tested per NEMA Standard ICS-2.42.

The Master Silence Switch is located in the Control Room at the SCO's desk (C17C).

7.7.3 OPERATION

The annunciator is equipped with six sets of push buttons; their function and operation are as follows:

"Acknowledge" push button — Upon receipt of an alarm, i.e., field contact off norm, depressing the "acknowledge" push button on the corresponding control panel changes the flashing window to steady and silences the audible alarm.

"Silence" push button — Depressing any one of the "silence" push buttons silences all alarm horns should the operator choose to do this prior to acknowledging an alarm. "Silence" push buttons do not operate to silence the audible tone that sounds when the field contacts return to normal.

"Reset" push button — After an alarm has been acknowledged and the field contacts have returned to normal, depressing the "reset" push button on the corresponding control panel section causes the slowly flashing bright window to change to steady dim lighted and silences the audible tone.

"Test" push button — Depressing the "test" push button on the corresponding control panel section shall perform a functional test of all annunciator alarm points and the audible alarm on that panel.

The alarm sequence is as follows:

Condition	Visual	Audible
Field contact normal	Dim lighted	Off
Field contact off-normal	Fast flashing bright	On
Field contact off-normal Silence Button depressed	Fast flashing bright	Off
Field contact return to normal before acknowledge	Fast flashing bright	On
Alarm acknowledged	Steady bright if field contact off-normal	Off
Field contact returned to normal after acknowledge	Slow flashing bright	On *
Field contact returned to normal, alarm reset	Dim lighted	Off

* Different Tone

Master Audible Silence Switch will turn off all audible alarms in “silence” mode.

Major system alarms are described with systems in their appropriate section.

7.8 COMMUNICATION SYSTEMS

7.8.1 DESIGN BASES

7.8.1.1 Functional Requirements

Communication systems are provided to meet the requirements for operation and maintenance of this generating unit. Further provision is made for routine and emergency communications between the unit operator and outside locations such as the system operator and public authorities.

7.8.1.2 Design Criteria

Industry standards for communication are observed, and precautions are taken so no failure of these systems will compromise the proper functioning of any protective system. Redundancy and separation are provided between the dial telephone system, the public address system, and the "walkie-talkie" radios. The power supply for all communication systems is from a dependable AC or battery source.

7.8.2 SYSTEM DESCRIPTION

7.8.2.1 Systems

a. Public Switched Network (off-site Dial Telephone System)

b. Intraplant Private Branch Exchange (PBX) telephones

c. Radio facilities

d. Maintenance system

e. Fuel handling system

f. Public address system

g. Evacuation alarm

h. Millstone 2 Simulator

7.8.2.2 Components

a. Public Switched Network

The public switched network is operated by various telephone companies and connects various outside agencies.

b. Intraplant Private Branch Exchange (PBX) Telephones

The intraplant switching network or private branch (PBX) is a telephone system consisting of standard telephones, multiline telephones, pico cellular phones, and a digital PBX.

c. Radio facilities consist of multiple separable systems available to the unit operator.

d. Equipment for the maintenance system consists of directly connected amplified outlet jacks wired to cover each of several working areas, with five channel selector switches at each station. Portable instruments consist of headsets with boom mounted transmitter, each with a plug to match the outlet jacks.

e. The fuel handling system provides a noise-canceling, dynamic type headset with amplifier mounted in enclosures located throughout the fuel handling area inside and outside the containment structure. This system includes an instrument on the reactor polar crane.

f. The public address system, manufactured by Gai-Tronics, consists of permanently installed loudspeakers throughout the unit. Cone type speakers are used indoors, and weather proof re-entrant horn type speakers are used outdoors. Each loudspeaker has its own integral amplifier. An amplifier failure would affect only its associated loudspeaker, permitting normal use of all others.

g. The evacuation alarm consists of audio frequency oscillators that supply a distinctive tone signal to the public address loudspeakers located throughout the unit. An ambient noise level device assures that the output sound level is sufficiently high for each location.

h. Millstone 2 Simulator Trunked Radio System consists of a Radio Control Station, a Desktop Remote Controller and an 800MHz antenna installed on the Simulator Building roof.

7.8.3 SYSTEM OPERATION

7.8.3.1 Normal Operation

a.& b. The intraplant switching network is directly coupled to the public switched network. Separate telephone lines are dedicated to the NRC and include the following:

- Emergency Notification System (ENS) - Unit Control Room, Technical Support Center (TSC), and Emergency Operating Facility (EOF)
- Reactor Safety Counterpart Link (RSCL) - EOF and TSC
- Management Counterpart Line (MCL) - EOF and TSC

- Local Area Network (LAN) - EOF and TSC
- Protective Measures Counterpart Link (PMCL) - EOF and Health Physics Network (HPN) - EOF

This network allows telephone calls to be made to the NRC, both in Bethesda and Region One.

c. Multiple Radio Systems

The multiple radio systems include the following communication systems:

- O&M/security/trunked radio system
- Connecticut Valley Electric Exchange (CONVEX) Command Control Network (CCN);
- Waterford Police System;
- State Police system;
- Very High Frequency (VHF) radio paging system.

A dedicated Desktop Radio Remote Control Console installed in the MP2 Control Room powered from a local receptacle receiving power from Panel LPC-1. Connected via a Control Area Network (CAN) cable to a dedicated Radio Control Station, installed in the Millstone 1 Control Room, powered from Panel L50. A wideband antenna installed on the Roof of Building 118 provide the Millstone 2 Control Room Operators tri-band (800MHz, UHF, and VHF) radio communications with associated onsite as well as offsite radio facilities (as outlined above). The identical dedicated standalone radio system is installed in the Millstone Control Room.

The Radio Control Station is installed outside the required EMI boundary (Unit 1 Control Room) to ensure that unintentional actuation of vital components or false indications are not caused by keying of the radio control stations in the Control Rooms.

A dedicated talk group established to provide direct communications between Control Room Operators and Security is provided. The Desktop radio Remote Controller will be tuned to this talk group, at a desirable volume, but will be quiet in the Control Room environment at all times, until Security supervisors feel the need to page the Control Room Operators and vise versa for either emergency communication or procedural testing of the radio system. Following a page, Operators can tune the Desktop Radio Remote Controller to any available talk

group for normal or emergency communications, than return to the dedicated quiet talk group afterwards.

Radio communications with the required outside agencies, Waterford and State Police as well as the Connecticut Valley exchange (CONVEX) will be made available to operators through the Trunked Radio System via the interoperability Gateway (IOG) installed in the Network Switching Cabinet (NSC), located at the Secondary Radio Site, Main Communications Room in Building 475 first floor. The IOG will provide the interface between the existing legacy Radio Control Stations, and associated antennas installed in the CPF analog Microwave Room and the (Bldg. 212) roof and the Trunked Radio System, making radio communications with outside agencies available to any talk group that is programmed to have access to them.

O&M/Security/Trunked Radio System

This is a digital Harris tri-band Trunked Radio System. The system is a redundant two site design with a Distributed Control Point (DCP) Simulcast, Multiband (800 MHz, UHF and VHF), VIDA Foundation Core, P25 multiband Radio Frequency (RF) communication system that does not require user intervention.

Each RF site consists of six (6) 800 MHz Repeaters, two (2) UHF (450 MHZ) Repeaters and two (2) VHF (150 MHZ) Repeaters. The Trunked Radio system broadcast information simultaneously (i.e. the same information, at the same frequency, at the same time) from redundant repeater locations to achieve high reliability and availability. Single band radios will be used by non-security personnel on the 800 MHz channels only. If jamming of the 800 MHz system were to occur, only multiband radios available to Security and Operations personnel will automatically move to another frequency (UHF or VHF).

Switching to the UHF or VHF channels is done seamlessly and the users are not aware that it is being done. Once the 800 MHz channels are available, the system automatically returns to the standard 800 MHz channels.

The Trunked Radio system consists of two redundant equipment sites; The Primary Site is located in Building 505/1 and the Secondary Site is located in Building 475/1 (Main Communications Room). The Primary Site is located in a non-essential MP2 power source through security distribution equipment and backed up by security emergency power supplies. The Secondary Site is powered from and independent source (i.e. Waterford/Flanders Line) and backed up by a dedicated Diesel Generator and Uninterruptible Power Supply (UPS) providing continuous power with enough run time for the generator to start. A Network Switching Center (NSC), referred to as the Voice Interoperability Data and Access (VIDA) Foundation Core, is installed at the Secondary equipment site (475/1). The NSC provides the capability for users to configure system settings. The NSC is

powered by the same source as the Secondary Site equipment (Flanders Line) and backed up by the same UPS and Diesel Generator.

Two (2) each Harris Symphony Dispatch Consoles are installed in CAS and SAS. The Dispatch Consoles allow Security personnel to access NSC configuration applications to modify talk groups, tracking modules to review calls received, and allows for the configuration of monitoring modules. The Dispatch Consoles receive backup power from a non-vital source with battery backup. If the NSC fails or losses power, the Dispatch Consoles will lose communication with the RF sites, however handheld radio communications will continue to operate and provide backup communications capabilities until the NSC is restored.

Two (2) 800 MHz (746 - 960 MHz) Panel antennas, a dual array VHF (138 - 174 MHz) and a dual port UHF (406 - 470 MHz) antenna are installed on the Millstone Stack at the 135’ Elevation to provide radio communications for the Primary Radio equipment site, designed to withstand windspeeds up to and including 145 mph.

Two (2) 800 MHz (806 - 900 MHz) V-Pol Omni antenna, a dual array VHF (138 - 174 MHz) and a dual port UHF (406 - 470 MHz) antenna are installed on the roof of Building 475 to provide radio communications to the Secondary Radio Equipment Site, designed to withstand windspeeds up to and including 145 mph.

Five (5) GPS antennas are installed on the roof of Building 405 (2) and the roof of Building 475 (3) to provide simulcast timing between the Primary and Secondary Site radio equipment.

Wideband Discone antenna (118 MHz - 3 GHz) are installed for the MP2 Control Room (Building 118) and MP3 Control Room (Building 317) to provide tri-band Trunking Radio Communications to Security and Operations, designed to withstand windspeeds up to and including 145 mph. See section 7.8.3.1c for Control Room radio equipment details.

Two (2) additional Wideband Discone antennas (118 MHz - 3 GHz) are installed on Building 201 (SAS) and Building 405 (CAS), designed to withstand windspeeds up to and including 145 mph. A Radio Control Station and Desktop Radio Remote Controller in each location will provide backup multi-band radio communications to CAS and SAS.

Station Operators and Security personnel communicate via the Trunked Radio system using hand-held portable radios. Each user group has radios and battery charging stations.

A one channel, analog based 900MHz radio system is provided to the Site Security in the event of a failure of the primary 800MHz radio system. The 900MHz repeater is located in the CPF Building. The repeater is capable of 100 watts of RF

power output with continuous duty operation. The primary power source is 120 VAC and the backup power source is battery. The individual portable radio is also equipped with a small antenna which provides “portable-to-portable” feature between the radios. An omnidirectional antenna is mounted on the roof of the CPF Building and the installation design is such to withstand windspeeds up to and including 100 mph.

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 operating facilities.

This system is controlled by the radio console in Units 2 and 3 via an Inter-operability gateway (IOG) through the Trunked Radio System. 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 13.8 VDC 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 one-half 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 three-quarter inch diameter tubing and reinforced with seven-eighths 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.

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 2 and 3 via an Inter-operability gateway (IOG) through the Trunked Radio System. The base station is located in the CPF telecommunications room.

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

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 2 and 3 via an Inter-operability gateway (IOG) through the Trunked Radio System. 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 VAC, and it is protected from overcurrent conditions.

The base station is connected to the antenna via a jacketed one-half 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 one-half 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.

Dominion Energy Emergency Notification System (DEENS)

The Dominion Energy Emergency Notification System (DEENS) is web-based emergency notification and callback verification software. This software is designed to meet the needs of nuclear power facilities, including the requirements of notifications in accordance with 10 CFR 50 Appendix E and NUREG-0654. The software utilizes multiple distributed data centers, with flexible capacity, and full stack redundancy with multiple SMS and voice providers vetted to ensure no downstream inter-dependencies. When activated, the system will contact the Millstone Emergency Response Organization and State and Local agencies via email, SMS text messaging, cell phone and LAN line. Emergency event information is included in the emailed Emergency Notification (ENF) sent to the State and local agencies.

d. 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.

e. 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. All stations are "common talk," and ringing facilities are not included.

f. 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.

g. 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.

h. Millstone 2 Simulator

A new Radio Control Station is installed just outside of the MP2 Simulator Control Room with a Desktop Radio Remote Controller inside for Security and Operations training purposes. An 800 MHz antenna is installed on the Simulator Building roof.

7.8.4 AVAILABILITY AND RELIABILITY

7.8.4.1 Special Features

“Walkie-talkie” radios and pico cellular phones 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.

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 that is equipped with a rectifier and backup battery. The normal and emergency power supply systems for the Public Switched Network are located at the telephone company operating facilities. Both Trunked Radio equipment sites Primary (405) and Secondary (475) are powered from reliable power sources and backed up by Uninterruptible Power Supplies (UPS) and Diesel Generators.

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.

7.9 ANTICIPATED TRANSIENTS WITHOUT SCRAM CIRCUITRY

7.9.1 DESIGN BASES

7.9.1.1 Functional Requirements

On July 26, 1984, the Code of Federal Regulations (CFR) was amended to include the "ATWS Rule" (Section 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without SCRAM [ATWS] Events for Light-Water-Cooled Nuclear Power Plants"). An ATWS is an expected operational transient (such as loss of main feedwater, loss of condenser vacuum, or loss of offsite power), which is accompanied by a failure of the Reactor Trip System (RTS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

The 10 CFR 50.62 requirements applicable to pressurized water reactors manufactured by Combustion Engineering (CE), are:

1. Each pressurized water reactor must have equipment from sensor output to final actuation device that is diverse from the RTS, which will automatically initiate the Auxiliary (or emergency) Feedwater System (AFWS) and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing RTS.

2. Each pressurized water reactor manufactured by CE must have a diverse SCRAM system from the sensor output to interruption of power to the control rods. This SCRAM system must be designed to perform its function in a reliable manner and be independent from the existing RTS (from sensor output to interruption of power to the control rods).

7.9.2 DISCUSSION

An ATWS is an anticipated operational occurrence (e.g., loss of main feedwater, or turbine trip) which is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor. The limiting ATWS events are typified by a rapid Reactor Coolant System (RCS) heatup and pressurization to above 3200 psia before moderator reactivity feedback substantially reduces reactor power.

Historically, CE has performed an analysis of selected transients which provide sufficient characterization of the CE Nuclear Steam Supply System (NSSS) design to ATWS events. For assessing the results of the ATWS analysis, CE used the following generalized criteria:

1. Radiological release within 10 CFR 100 Guidelines

2. RCS pressure less than emergency limits (3200 psia)

3. Reactor fuel - no melting

4. Fuel cladding - no degradation

5. Containment Pressure - within design pressure

Based on CE's analysis, the consequence of a failure to SCRAM would lead to an RCS overpressurization, a violation to Criterion 2. Those ATWS events which would cause RCS overpressurization are:

1. Zero Power Control Element Assembly (CEA) Withdrawal

2. Loss of load

3. Loss of Main Feedwater (complete or partial)

The Millstone Unit 2 Diverse SCRAM System (DSS) and ATWS Mitigating System Actuating Circuitry (AMSAC) fulfills the NRC requirement addressed in 10 CFR 50.62. The DSS is diverse and electrically independent from the RTS and provides a redundant path of reactor and turbine trip by a high-pressure setpoint. The AMSAC is modified from the existing Auxiliary Feedwater Actuation System (AFAS) to mitigate an ATWS event by redundant Auxiliary Feedwater (AF) initiation from DSS.

The DSS reduces the ATWS probability and AMSAC provides some limited mitigation. However, DSS/AMSAC is not expected to fully mitigate all ATWS events. Therefore, the ATWS system is not a direct response to all accidents analyzed in Chapter 14.

Reference:

1. Analysis of Anticipated Transients Without Reactor SCRAM in Combustion Engineering's NSSS's, May 1976, Combustion Engineering, CENPD-158, Revision 1.

7.9.3 DESIGN CRITERIA

7.9.3.1 General

The systems and equipment required by 10 CFR 50.62 do not have to meet all of the stringent requirements normally applied to safety related equipment. However, this equipment is part of the broader class of structures, systems, and components important to safety defined in the introduction of 10 CFR 50, Appendix A (General Design Criteria [GDC]). Although the ATWS mitigation system is not required to meet all of the stringent requirements normally applied to safety related equipment per 10 CFR 50.62, the DSS/AMSAC is designed to Quality Assurance Category 1 requirements in accordance with the Quality Assurance Program (QAP).

The ATWS system is designed to the following bases:

a. No single component failure can prevent the performance of a safety function.

b. Channel independence is assured by separate connection of the sensors to the process systems and of the channels to vital instrument buses.

c. The four measurement channels provide trip signals to six independent logic matrices, arranged to effect a two-out-of-four coincidence logic having outputs to two independent trip paths.

d. A trip signal from any two-out-of-four protective channels on the same parameter causes an ATWS trip.

e. When one of the four channels is taken out of service, the protective system logic may be changed to two-out-of-three coincidence for an ATWS trip by bypassing the removed channel.

f. Independence is provided between redundant elements to preclude any interactions between channels during maintenance or in the event of channel malfunction.

g. Redundant elements are electrically isolated from each other such that events affecting one element are not reflected in any other redundant element.

h. The tripping function is accomplished via an energize to trip logic path to preclude inadvertent plant trips due to component failure.

7.9.3.2 Electrical Independence

The ATWS rule requires that the DSS be electrically independent from the RPS. Both the RPS and DSS share four pressurizer pressure sensor channels. Each channel output is isolated between output to the RPS and output to the DSS. The isolation design is consistent with the present licensing basis for Millstone Unit 2. Subsequent DSS processing is totally electrically independent from the RPS.

7.9.3.3 Environmental Qualification

The DSS is designed to operate for Anticipated Operational Occurrence environment which means a normal containment environment and a mild control room environment. There is no requirement to qualify the DSS/AMSAC to Loss-of-Coolant Accident (LOCA) or High Energy Line Break (HELB) environment since these are not considered anticipated operational occurrences. The DSS electronics which share the same instrument rack and same power supply as isolation electronics for the RPS are qualified to IEEE-323-1983.

7.9.3.4 Seismic Qualification

The DSS need not work during or after a seismic event. However, all components that interface with Category 1E systems will be seismically restrained and electrically certified not to degrade the Category 1E equipment. The DSS electronics which share the same instrument rack and same power supply as isolation electronics for the RPS are qualified to IEEE-344-1975.

7.9.3.5 Annunciation and Display

The DSS status and bypass indications will be provided in the control room. Trip alarm and bypass will be annunciated on the main control board.

7.9.3.6 Testability

The DSS is designed as a four-channel system with individual bypass switch for each channel. The bypass switch will permit individual channel testing while the reactor is operating at power.

7.9.3.7 Diversity

The Millstone Unit 2 DSS design uses the existing RPS pressurizer pressure sensors to generate both the RPS and DSS actuation signals. Even though the DSS and RPS use common sensors, the DSS uses qualified, Category 1E electronics that have been analyzed to demonstrate isolation from the RPS. This minimizes the potential for adverse electrical interactions between the two systems.

Diversity of manufacturer exists for the DSS and RPS bistables, power supplies and matrix relays. The DSS and RPS initiation relays and final actuation devices are both manufactured by the same vendor; however, diversity of model/design principle exists.

The Millstone Unit 2, Diverse Turbine Trip (DTT) design is such that the DTT shares all circuit components with the DSS, up to, but not including, the final trip device. Thus, all of the information that is applicable to DSS components discussed previously are applicable to the DTT components, up to, but not including the final trip device. When the DSS causes a reactor trip, it also causes a turbine trip as the DSS interrupts power to the turbine trip undervoltage coils. The diverse SCRAM relays provide isolation between the Category 1E and Noncategory 1E portions of the circuits. Thus, with the implementation of the DSS, the existing turbine trip becomes a DTT due to the diversity between the DSS and the existing RPS.

The existing AFWS actuation circuitry, when installed at Millstone Unit 2, contained significant diversity from the RPS.

The AFWS design at Millstone Unit 2, was upgraded following the TMI-2 accident in accordance with the TMI Action Plan Items II.E.1.1, “Auxiliary Feedwater System Evaluation,” and II.E.1.2, “Auxiliary Feedwater System Automatic Initiation and Flow Indication,” of NUREG-0737, “Clarification of TMI Action Plant Requirements.” Diversity of manufacturer exists for the Diverse Auxiliary Feedwater (DAFW) bistables, matrix relays, initiation relays and power

supplies. The RPS and DAFW initiation relays and final actuation devices are both manufactured by the same vendor; however, diversity of model/design principle exists.

7.9.4 SYSTEM DESCRIPTION

The Millstone Unit 2 scheme for mitigating ATWS events consists of a DSS, an AMSAC system, and a DTT system. The DSS provides a redundant path of reactor and turbine trip by a high pressurizer pressure setpoint. The AMSAC uses the existing AFAS to mitigate an ATWS event by redundant AF initiation from DSS.

The ATWS is designed to be highly reliable, resistant to inadvertent actuation, and easily maintained. Reliability is assured through the use of internal redundancy and industry proven system components. Inadvertent actuations are minimized through the use of internal redundancy, energize to trip design, and good human factors practices. The time delay on low steam generator level and the coincidence logic used also minimize inadvertent actuations. Figure 7.9–1 details the ATWS system.

7.9.4.1 Diverse Scram System

The DSS receives input from the existing RPS pressurizer pressure sensors. The pressurizer pressure signals are routed to signal processing instrumentation consisting of bistables and logic circuitry arranged in a two-out-of-four energize-to-actuate logic to trip the RPS Motor Generator (MG) set output contactors upon detection of conditions indicative of an ATWS event.

Visual and audible alarms provide the control room operator with indication of DSS initiation.

Provisions have been included to allow the bypass of any one sensor input thus converting the logic to two-out-of-three to allow for maintenance and testing of the sensors and associated signal processing electronics. All bypasses are annunciated on the Main Control Board.

7.9.4.2 Diverse Auxiliary Feedwater Actuating System

The existing AFAS has been modified to satisfy the ATWS requirement. The present AFAS is initiated upon low steam generator level (two-out-of-four logic) following a 3 minute, 25 second time delay. An ATWS event with reactor power greater than 20 percent will initiate automatic AF following a 10-second time delay. An ATWS event with reactor power less than 20 percent will initiate automatic AF following a 3 minute, 25 second time delay.

Visual and audible alarms provide the Control Room operator with indication of AFAS initiation.

7.9.4.3 Diverse Turbine Trip

The Millstone Unit 2 DTT design is such that the DTT shares all circuit components with the DSS, up to, but not including, the final trip device.

The existing turbine trip is actuated by the undervoltage relays on the Control Element Drive Mechanism (CEDM) power cables from the MG output. The DSS relays interrupt the MG output thus providing a DTT.

7.9.5 SYSTEM COMPONENTS

7.9.5.1 System Hardware and Interface

Figure 7.9–2 provides an overview of the DSS/AMSAC hardware and interconnections between components. Four pressurizer pressure channels each containing logic for high-pressure trip, provide the inputs to DSS matrix located in panel C100. Two Nuclear Instrument (NI) channels, which are indicative of reactor power level, interface with the AF initiation facilities Z1 and Z2 to provide a redundant AF initiation.

The DSS matrix combines four channel trip contacts in a two-out-of-four voting matrix. The Matrix is arranged so that two channels will issue a DSS signal. If a channel is removed out of service, the voting matrix automatically converts to a two-out-of-three voting configuration.

The output from the DSS matrix directly drives the DSS relays, 94A/DSS and 94B/DSS. Both relays provide for the redundant AF initiation and the tripping of the MG output contactors.

7.9.5.2 Pressurizer Pressure Channels

The Millstone Unit 2, design uses the existing RPS pressurizer pressure sensors (PT-102A-D) to generate both the RPS and DSS actuation signals. Even though the DSS and RPS use common sensors, the DSS uses qualified, Category 1E electronics that have been analyzed to demonstrate isolation from the RPS. This minimizes the potential for adverse electrical interactions between the two systems.

The High Alarm contact drives four relays on a SPEC 200 N-2A0-L2C-R relay isolator card which provides train isolation and interconnections for use in the DSS matrix. The High Alarm setpoint is 2400 psia.

7.9.5.3 Neutron Monitoring Channels

Two independent analog signals corresponding to 0 - 100 percent reactor power are routed from the Reactor Regulating System (RRS) Channel “X” and “Y” to qualified Category 1E electronics. A 20 percent reactor power alarm contact drives a relay on a SPEC 200 N-2A0-L2C-R relay isolator card which provides train isolation and interconnections to the AF initiation circuitry.

7.9.5.4 Diverse SCRAM Matrix

Figure 7.9–3 shows the scheme arrangement for the Diverse SCRAM (DS) Matrix and its logical interconnections.

There are six possible combinations to implement a two-out-of-four voting scheme. These six combinations are represented by the six vertical contact strings, each representing a voted combination. Each contact in a string is associated with a disable/enable contact from the channel keylock switch (HS-102).

The matrix is arranged to activate the DSS relays 94A/DSS and 94B/DSS only when power is available. A High Alarm on any two channels will close two contacts on a vertical string. If the HS-102 switch is closed for both contacts, then both relays will activate.

Both relays 94A/DSS and 94B/DSS are used to provide isolation between the DSS matrix, the AF initiation facilities and the MG contactors and to ensure a DS in the event of a single relay failure. Interposing (auxiliary) relays 94X/MG1 and 94X/MG2 provide the auxiliary contacts to operate the MG contactors and to provide additional isolation between the 480V MG control circuit and the DSS control circuit.

7.9.5.5 Auxiliary Feedwater Initiation

Two contacts from the DSS relay are used in the AF initiation scheme. The first contact parallels the steam generator low level - AF initiation relay contact and functions to start the 3 minute, 25 second timer to initiate AF. The second contact starts a 10 second timer which will be used to conditionally start AF initiation 10 seconds after receiving a DS signal with Reactor Power level ≥20 percent.

7.9.5.6 Motor Generator Contactors

The existing contactor schematic circuit is tripped by loss of motor power for longer than 2 seconds, high or low current through the exciter, synchronization errors or bus overvoltage.

Each MG contactor can be operated from either redundant DSS relay via auxiliary relay schemes. Auxiliary relays 94X/MG1 and 94X/MG2 are activated separately, by circuits containing parallel, normally open, contacts from the 94A/DSS and 94B/DSS relays. When activated by a DSS relay(s), both auxiliary relays provide signals to open each MG contactor via normally closed contacts connected in series within each contactor control circuit. For a postulated single failure in a DSS, auxiliary relay or associated contact, this control scheme ensures that the redundant DSS scheme will trip both contactors.

7.9.5.7 Power Supply

The DSS and AMSAC circuits use common AC power sources from the 120 VAC vital buses, VA10, VA20, VA30, and VA40, for all components from the sensors to the initiation relays.

The C100 matrix tripping relays (94A/DSS and 94B/DSS) are powered from 125 VDC power supplies (via VA10 and VA20 respectively). Each power supply is rated for 100 percent capacity and their outputs are diode auctioneered such that a single failure will not impact system operation.

7.9.6 SYSTEM OPERATION

The DSS/AMSAC provides a backup to the RTS for initiating a reactor and turbine trip and AF flow in the event of an anticipated transient.

An ATWS trip will occur when a high pressurizer pressure signal exceeds the setpoint of 2400 psia on two of the four inputs channels. Initiation of the ATWS will result in the following system response:

1. A reactor (and subsequent turbine) trip will be processed by opening of series DSS auxiliary relay contacts in the RPS MG Set outputs.

2. AF initiation will occur after a 10 second time delay if reactor power remains ≥20 percent, otherwise, AF initiation will occur following a 3 minute, 25 second time delay.

7.9.6.1 Bypasses

An individual bypass (keylock) switch is available for each of the four pressure channels. This bypass will allow on-line maintenance and testing capabilities of the DSS/AMSAC circuitry.

7.9.6.2 Annunciation and Display

Indication of individual pressure channel trip is available at the C100 status panel. An “Amber” lamp will be illuminated on the C100 status panel indicating a channel trip.

Annunciation alarms on main control board C04F are provided to alert the operator should an “ATWS” or “AAFWIS” (Automatic Auxiliary Feedwater Initiation System) initiation occur.

ATWS channel bypasses are annunciated on Main Control Board C04F.

7.9.6.3 Inadvertent Actuation

The DSS and AMSAC systems have been designed such that the frequency of inadvertent actuations is minimized. Reliability of this system is ensured through the use of redundancy, majority voting logic, bypasses and energize to trip circuitry.

The change slightly increases the likelihood of spurious reactor trip (i.e., a trip when no protection setpoints have been exceeded) because of the addition of electrical components which can interrupt power from the MG set to the CEDMs.

The inadvertent actuation of the DSS or malfunction of the MG output contactors would result in the interruption of power to the CEDM coils causing the rods to drop by gravity into the core. The undervoltage condition will also cause a turbine trip signal to be generated. Therefore, inadvertent DSS actuation would result in simultaneous turbine and reactor trip. An inadvertent DSS Actuation would resemble the interruption of power to or from the MG set as in the existing

design. Therefore, a spurious actuation is neither a Design Basis Accident (DBA) nor a new accident.

The change could also result in automatic AFWS actuation, regardless of steam generator level (as in current design), if the DSS is inadvertently actuated and a high neutron flux signal (> 20 percent) is sustained for 10 seconds. This could increase the challenges to the Engineered Safeguards Features (ESF), and increase the potential for feedwater nozzle thermal shock, steam generator overfill, and so on. However, for this to be a concern, two low-probability events must occur. Inadvertent DSS actuation is of low probability. Given that DSS actuation has occurred, reactor SCRAM will occur and neutron flux will be below 20 percent within 10 seconds. Therefore, a second (low probability failure) must occur, that is, a high neutron flux signal because of miscalibration error or electrical component failure. Given that the plant was operating with neutron flux measurement calibrated in the first place, these failure modes are low probability. The conclusion is that inadvertent AFWS actuation, due to the change, is of insignificant probability compared to all other mechanisms for inadvertent actuation.

FIGURE 7.9–1 LOGIC DIAGRAM - ANTICIPATED TRANSIENT WITHOUT SCRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.9–2 BLOCK DIAGRAM DIVERSE SCRAM SYSTEM (ATWS)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.9–3 SCHEMATIC DIAGRAM DIVERSE SCRAM SYSTEM ATWS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.10 AUXILIARY STEAM LINE BREAK DETECTION/ISOLATION SYSTEM

7.10.1 DESIGN BASES

7.10.1.1 Functional Requirements

The auxiliary steam line break detection/isolation functions to isolate steam supply to specific safety related areas if steam leakage is detected due to an auxiliary steam line (or other) high energy line break.

7.10.2 DISCUSSION

Auxiliary Steam System

The auxiliary steam system is provided for building heating, freeze protection for outdoor water storage tanks, and radwaste process requirements. The system is nonnuclear safety (NNS). Steam is normally provided by the Unit 3 auxiliary steam system via a crosstie between the Units. Because the Unit 3 auxiliary steam system operates at 150 psig and the Unit 2 auxiliary steam system operates at 50 psig, a pressure reducing valve station, including isolation and relief valves are installed. In addition a condensate return line from the Unit 2 auxiliary feedwater surge tank is routed to the Unit 3 condensate system. Condensate is routed to the auxiliary boiler deaerator, when the Unit 3 auxiliary boilers are supplying auxiliary steam, or to the Unit 3 condensate surge tank when Unit 3 main steam is supplying auxiliary steam.

During Unit 2 plant shutdown, the Unit 3 auxiliary steam system provides steam to the Unit Number 2 auxiliary steam system. In the event both Units are shutdown the Unit 3 auxiliary steam boilers maintains the capability to provide house heating steam to both Units.

Technical Specification Numbers 3/4.1.2 (Boration Systems - Modes 5 & 6) and 3/4.5.4 (Refueling Water Storage Tank (Modes 1, 2, 3 and 4) establish minimum temperature requirements for safety related tanks that are heated by the auxiliary steam system.

10 CFR 50, Appendix A, Design Criterion 4, requires structures, systems, and equipment important to safety to be designed for effects and be compatible with environmental conditions associated with effects of postulated pipe ruptures.

The auxiliary steam system is classified as a high energy fluid system. The auxiliary steam temperature detection system is installed to detect an auxiliary steam leak or line break in specific safety related areas shown on Figure 7.10–1. The detection system, if activated, will isolate the steam supply to these areas to ensure operability of safety related equipment needed to safely shut down the plant.

Pipe break analysis criteria and guidance are discussed in Section 6.1.4.

Guidance concerning equipment qualification related to high energy line break were provided by the NRC in November 1979 DOR Guidelines “Guidelines for Evaluating Environmental

Qualification of Class 1E Electrical Equipment in Operating Reactors," and NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment."

Regulations for environmental qualification of electrical equipment are specified in 10 CFR 50.49.

7.10.3 DESIGN CRITERIA

7.10.3.1 General

The following criteria have been used in the design of the auxiliary steam detection/isolation (ASDI) system:

a. The system shall have redundant, independent subsystems.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system function with onsite power (assuming offsite power is not available) or with offsite electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as temperature detectors and automatic isolation valves to assure the integrity and capability of the system.

f. The ASDI system shall be designed to permit appropriate periodic pressure and functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the detection system.

g. The components of the detection system shall be designed to operate in the most severe environment to which it is exposed during an auxiliary steam line break.

7.10.3.2 Electrical Independence

The ASDI system is designed with two electrically independent channels in accordance with IEEE 279, Criteria for Protection Systems for Nuclear Power Generating Stations (1971).

7.10.3.3 Environmental Qualification

Temperature detectors associated with the ASDI system are qualified to withstand the predicted environment resulting from an auxiliary steam line break in the auxiliary and enclosure buildings. Detectors and associated components are qualified in accordance with IEEE 323, Qualifying Class 1E Equipment for Nuclear Power Generating Stations (1974). Isolation valves, associated controls, and other active components are designed to fail in the safe direction and are not required to be environmentally qualified.

7.10.3.4 Seismic Qualification

The ASDI system is seismically qualified to ensure detection of a steam line break in the required areas, transmission of the signal, and closure of isolation valves for the auxiliary steam supply to the Auxiliary Building. Components are seismically qualified to IEEE 344, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Systems (1974).

7.10.3.5 Annunciation and Display

The ASDI system status and bypass indications are provided in the control room. Detection alarm and bypass are annunciated on the main control board.

7.10.3.6 Testability

The ASDI System is designed as a two channel system. Each channel is provided with a bypass switch to allow periodic test of each channel without causing isolation of the auxiliary steam supply to the Auxiliary Building.

7.10.4 SYSTEM DESCRIPTION

The ASDI System is shown schematically in Figure 7.10–1. The ASDI System consists of eight pairs of RTDs located in the areas of the auxiliary building as shown.

Temperature detectors (RTDs) are installed at locations easily accessible from floor level for maintenance and calibration. The temperature detector electronics are located in control cabinets C502 and C503 in the East 480 volt Loadcenter Room. Associated control relays and hand switches are located in panel C-80 in the main Control Room.

RTDs are resistance temperature detector type. The detection/isolation are set at a maximum threshold set point value of approximately 115°F to rapidly detect a potential steam line break or leak. The exact set point is determined based on maximum design temperature applicable to the respective area (see FSAR Section 9 for this basis) with a suitable margin included as needed to avoid spurious actuation and alarms. Actuation of any alarm bistable causes the respective steam isolation valve to close. Status indicating lights (red) are provided at C502/C503 panels for each temperature switch. A computer point and main control board alarm is provided for each Auxiliary Steam Detection Isolation System Actuation.

Each RTD of the pair in each environmental (EEQ) zone is independently powered and redundant to each other (single failure protected).

Each train of alarm switches has their control contacts wired in series. Reset of the system after an alarm switch actuates is accomplished via hand switch in the control room (panel C-80). The handswitch will reopen the auxiliary steam isolation valves after the affected RTD returns to normal ambient conditions. The RTDs are seismic and environmentally qualified.

Each RTD will actuate one of (2) automatic isolation valves to close. The position of the auxiliary steam isolation valves is indicated by red-green indicating lamps on the C80 panel in the control room. Maintenance/Bypass switch engagement is indicated by an amber lamp on C-80 panel and also annunciated on a common alarm window on panel CO6.

After an actuation of an isolation valve(s) and the alarm bistable resets, the annunciator will reset, but the auxiliary steam isolation valves will remain closed. The operator must take deliberate action by operating hand switch(es) on the C80 panel to re-open the closed valve(s).

The control scheme has a built-in maintenance/bypass keylock switch on the C80 panel which allows calibration of the bistables without closing the auxiliary steam isolation valves. This function is a “key locked” hand switch. The bypass condition is annunciated on panel CO6 when in the bypass position. An amber light on C80 panel indicates when the alarm bistable trip function has been bypassed. Procedural controls have been established and will be maintained to effect compensatory measures during periods when the trip function is bypassed.

The automatic isolation valves are located in the Turbine Building prior to the Auxiliary Steam penetration into the Auxiliary Building at the Turbine/Auxiliary building wall. The valves are independent and redundant to each other to meet single failure criteria. The valves are high performance (very low leakage), butterfly type with an offset disc and metallic seat.

The valves are air-operated, air to open, vent to close. They fail closed with spring assist on loss of instrument air or power. Each valve is provided with a high capacity, rapid discharge solenoid vent valve to ensure closure time of less than five seconds. The valve and air operator are seismically qualified. The solenoid vent valves and valve position limit switches are purchased EEQ and seismically qualified. However, these items do not have to be maintained in the MP2 EEQ master list since the isolation valves are not required to isolate if a steam line breaks in the Turbine Building. Each isolation valve has an air pressure regulator based on design pressure limitations of the air operator diaphragm. The regulator is located upstream of the solenoid vent valve and is nonsafety related.

The piping system is designed in accordance with ANSI B31.1 design criteria.

FIGURE 7.10–1 PIPING AND INSTRUMENTATION DIAGRAM AUXILIARY STEAM AND CONDENSATE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.11 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT IMPORTANT TO SAFETY

Since environmental conditions vary for different areas of the plant, there are several environmental zones. Safety related equipment and components are qualified to meet their performance requirements under normal, abnormal, and accident operating conditions based on the environmental zone in which the equipment is exposed.

The Millstone Electrical Equipment Qualification (EEQ) Program is a process that ensures the continued qualification of equipment which must function during and following the design conditions postulated for design basis accidents and the post-accident duration. This program has been developed to ensure that environmental qualification criteria are applied to electrical equipment important to safety in accordance with 10 CFR 50.49 and to document the process through which this qualification is demonstrated. It incorporates NRC Regulation, Regulatory Guides, and other positions and guidelines, as well as Institute of Electrical and Electronic Engineers (IEEE) Standards and sound engineering practices.

The Design Basis area of the EEQ program consists of the Environmental Qualification Master List (EQML) and the plant normal and accident environmental conditions. The EQML is the set of equipment required to be environmentally qualified. Environmental Parameters include temperature, pressure, relative humidity, chemicals, spray potential, submergence, accident duration, and gamma/beta radiation dose (where applicable). These parameters are given in terms of a time-based profile. Changes in equipment function or operating mode, substitution of new model numbers, and the addition or deletion of equipment in the plant, shall require a revision of the EEQ program documentation. Changes to the environmental parameters such as revisions to accident analyses, rerouting or modifying the high energy lines, or changes in the functional operation of the HVAC system may result in revisions to the EEQ program documentation.

Test Report Assessments (TRA), Equipment Qualification Records (EQR), and referenced design documents provide an auditable proof of the equipment qualification. The qualification process is based on the testing and/or analysis of same or similar equipment or material such that this equipment performance becomes the model or proof of how the installed equipment is anticipated to behave when exposed to design basis accident environmental conditions. Emulation of the tested equipment’s internal, external, and maintained configuration is necessary to enable the test to represent the plant installed equipment. Products of the qualification verification include installation, maintenance, and procurement requirements that must be implemented to ensure that the installed equipment is the same as the equipment tested.

Chapter 12.10 describes alternate requirements for RISC-3 SSCs.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 8: Electrical Systems

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25203-30052	3	4	
25203-30053	11	0	
25203-30053	8	1	
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25203-30053	4	4	
25203-32002	13	0	
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25203-32002	8	2	
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25203-32002	5	6	
25203-32002	5	7	
25203-32002	5	8	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32002	5	9	
25203-32002	5	10	
25203-32002	5	11	
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25203-32002	6	19	
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25203-32002	3	21	
25203-32002	2	22	
25203-32003	1	A	
25203-32003	2	B	
25203-32003	6	27	
25203-32003	6	28	
25203-32003	2	29	
25203-32003	3	30	
25203-32003	4	31	
25203-32003	2	32	
25203-32003	2	33	
25203-32003	2	34	
25203-32003	5	35	
25203-32003	6	37	
25203-32003	8	38	
25203-32003	2	39	
25203-32003	3	40	
25203-32003	4	41	
25203-32003	2	42	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32003	2	43	
25203-32003	2	44	
25203-32004	2	A	
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25203-32004	3	3	
25203-32004	3	4	
25203-32004	3	5	
25203-32004	2	6	
25203-32004	2	7	
25203-32004	3	8	
25203-32004	5	9	
25203-32004	5	11	
25203-32004	3	12	
25203-32004	3	13	
25203-32004	3	14	
25203-32004	2	15	
25203-32004	2	16	
25203-32004	3	17	
25203-32008	7	A	
25203-32008	0	B	
25203-32008	0	C	
25203-32008	4	1	
25203-32008	4	2	
25203-32008	5	3	
25203-32008	7	4	
25203-32008	8	5	
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25203-32008	5	8	
25203-32008	3	9	
25203-32008	4	10	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32008	5	11	
25203-32008	7	12	
25203-32008	6	13	
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25203-32008	5	17	
25203-32008	5	18	
25203-32008	5	19	
25203-32008	5	20	
25203-32008	5	21	
25203-32008	5	22	
25203-32008	5	23	
25203-32008	5	24	
25203-32008	6	25	
25203-32008	5	26	
25203-32008	5	27	
25203-32008	5	28	
25203-32008	5	29	
25203-32008	5	30	
25203-32008	4	31	
25203-32008	4	32	
25203-32008	3	33	
25203-32008	5	34	
25203-32008	4	35	
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25203-32008	9	40	
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25203-32008	3	54	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32008	3	55	
25203-32008	2	56	
25203-32008	2	57	
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25203-32008	2	59	
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25203-32008	4	61	
25203-32008	2	64	
25203-32008	3	69	
25203-32009	5	A	
25203-32009	1	B	
25203-32009	4	C	
25203-32009	6	4	
25203-32009	7	5	
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25203-32009	2	36	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32009	1	37	
25203-32009	3	38	
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25203-32009	4	43	
25203-32009	1	44	
25203-32009	2	45	
25203-32009	1	46	
25203-32009	3	47	
25203-32009	2	48	
25203-32012	4	B	
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25203-32012	3	13	
25203-32012	1	21	
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25203-32012	0	28	
25203-32013	5	A	
25203-32013	0	B	
25203-32013	5	C	
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25203-32013	7	7	
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Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32013	3	17	
25203-32013	3	18	
25203-32013	7	19	
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25203-32013	5	21	
25203-32013	5	22	
25203-32013	5	37	
25203-32013	5	38	
25203-32013	3	39	8.4-2
25203-32013	5	40	
25203-32013	5	41	
25203-32013	5	42	
25203-32013	5	43	
25203-32015	6	A	
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25203-32015	8	2	
25203-32015	8	3	
25203-32015	5	8	
25203-32015	6	9	
25203-32015	6	10	
25203-32015	5	11	
25203-32015	7	12	
25203-32015	7	13	
25203-32015	4	14	
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Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32015	3	24	
25203-32015	4	25	
25203-32015	2	26	
25203-32015	2	27	
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25203-32015	7	35	
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25203-32015	3	40	
25203-32015	2	41	
25203-32015	2	42	
25203-32015	2	43	
25203-32015	2	44	
25203-32015	2	45	
25203-32015	2	46	
25203-32015	2	47	
25203-32015	1	49	
25203-32017	4	A	
25203-32017	1	B	
25203-32017	3	7	
25203-32020	6	A	
25203-32020	1	B	
25203-32020	8	1	
25203-32020	9	2	
25203-32020	2	7	
25203-32020	2	8	
25203-32020	3	9	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32020	4	11	
25203-32020	4	12	
25203-32020	2	13	
25203-32020	4	14	
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25203-32020	0	27	
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25203-32021	0	B	
25203-32021	5	6	
25203-32021	4	8	
25203-32021	3	9	
25203-32021	3	10	
25203-32021	4	12	
25203-32021	4	13	
25203-32021	4	14	
25203-32021	4	15	
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25203-32022	24	0	
25203-32022	24	00	
25203-32022	24	000	
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25203-32022	2	D	
25203-32022	8	1	
25203-32022	8	2	
25203-32022	8	3	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32022	8	4	
25203-32022	5	6	
25203-32022	2	8	
25203-32022	2	9	
25203-32022	6	10	
25203-32022	6	11	
25203-32022	6	12	
25203-32022	6	13	
25203-32022	6	14	
25203-32022	6	15	
25203-32022	7	16	
25203-32022	8	17	
25203-32022	8	18	
25203-32022	4	19	
25203-32022	4	20	
25203-32022	6	21	
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25203-32022	8	24	
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25203-32022	5	27	
25203-32022	4	28	
25203-32022	4	29	
25203-32022	6	30	
25203-32022	5	31	
25203-32022	3	34	
25203-32022	5	35	
25203-32022	4	36	
25203-32022	5	37	
25203-32022	4	38	
25203-32022	4	49	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32022	2	40	
25203-32022	3	41	
25203-32022	3	42	
25203-32022	4	43	
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25203-32022	5	50	
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25203-32022	4	52	
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25203-32022	3	54	
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25203-32022	3	56	
25203-32022	3	57	
25203-32022	3	58	
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25203-32022	4	60	
25203-32022	5	61	
25203-32022	4	62	
25203-32022	5	63	
25203-32022	5	64	
25203-32022	3	65	
25203-32022	3	68	
25203-32022	2	69	
25203-32022	0	70	
25203-32022	2	71	
25203-32022	1	72	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32022	3	73	
25203-32022	1	74	
25203-32022	2	75	
25203-32022	2	77	
25203-32023	9	A	
25203-32023	1	B	
25203-32023	4	1	
25203-32023	6	2	
25203-32023	6	3	
25203-32023	6	4	
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25203-32023	7	6	
25203-32023	8	7	
25203-32023	8	8	
25203-32023	6	37	
25203-32023	5	38	
25203-32023	8	39	
25203-32023	8	40	
25203-32023	7	42	
25203-32023	7	43	
25203-32025	3	A	
25203-32025	0	C	
25203-32025	1	10	
25203-32025	1	11	
25203-32025	2	17	
25203-32026	2	A	
25203-32026	0	C	
25203-32026	3	1	
25203-32026	3	2	
25203-32027	3	A	
25203-32027	0	B	

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32027	3	3	
25203-32030	1	A	
25203-32030	0	B	
25203-32030	4	3	
25203-32030	4	4	
25203-32041	2	A	
25203-32041	9	1	
25203-32041	9	2	
25203-32041	7	3	
25203-32041	5	4	
25203-32041	11	5	8.3-2, Sheet 5
25203-32041	7	6	8.3-2, Sheet 6
25203-32041	7	7	8.3-2, Sheet 7
25203-32041	3	8	8.3-2, Sheet 8
25203-32041	2	9	8.3-2, Sheet 9
25203-32041	5	10	8.3-2, Sheet 10
25203-32041	6	11	8.3-2, Sheet 11
25203-32041	7	14	8.3-2, Sheet 14
25203-32041	6	15	8.3-2, Sheet 15
25203-32041	5	16	8.3-2, Sheet 16
25203-32041	10	17	8.3-2, Sheet 17
25203-32041	6	18	8.3-2, Sheet 18
25203-32041	7	19	8.3-2, Sheet 19
25203-32041	3	20	8.3-2, Sheet 20
25203-32041	2	21	8.3-2, Sheet 21
25203-32041	4	22	8.3-2, Sheet 22
25203-32041	6	23	8.3-2, Sheet 23
25203-32041	7	26	8.3-2, Sheet 26
25203-32041	3	27	8.3-2, Sheet 27
25203-32041	1	28	8.3-2, Sheet 28
25203-32041	3	29	8.3-2, Sheet 29

Drawing Number	Revision Number	Sheet Number	Figure Number
25203-32041	1	30	8.3-2, Sheet 30
25203-32043	4	1	
25203-32043	4	2	
25203-32043	5	3	
25203-32043	4	4	
25203-32045	4	1	8.3-2, Sheet 1
25203-32045	4	2	8.5-2, Sheet 2
25203-32045	4	3	8.5-2, Sheet 3
25203-32045	4	4	8.5-2, Sheet 4
25203-32045	4	5	8.5-2, Sheet 5
25203-32045	4	6	8.5-2, Sheet 6
25203-32045	4	7	8.5-2, Sheet 7
25203-32045	4	8	8.5-2, Sheet 8
25203-32045	4	9	8.5-2, Sheet 9
25203-32045	4	10A	
25203-32045	3	10B	
25203-32045	3	10D	
25203-32045	4	11	8.5-2, Sheet 11
25203-32045	1	12	

2. Combustion Engineering Corporation Electrical Drawings:

Drawing Number	Revision Number	Sheet Number	Figure Number
D-18767-411-022	05		
D-18767-411-029	02		
D-18767-411-030	00		
D-18767-411-031	05		
D-18767-411-035	02		
D-18767-411-037	02		
D-18767-411-036	02		
D-18767-411-038	03		

Drawing Number	Revision Number	Sheet Number	Figure Number
D-18767-411-366	03		
D-18767-414-460	NA		
D-18767-414-461	NA		
D-18767-414-469	03		
D-18767-416-103	02		
D-18767-416-104	02		
D-18767-416-111	02		
D-18767-416-112	03		
D-18767-416-131	05		
D-18767-416-401	03		
D-18767-416-402	03		
D-18767-416-451	00		
D-18767-416-470	03	1	
	04	2	
	04	3	
	01	4	
D-18767-416-472	03		
D-18767-416-481	00		
E-18767-411-003	04		
E-18767-411-011	03		
E-18767-411-012	05	1	
	07	2	
	07	3	
	06	4	
E-18767-411-013	04	1	
	03	2	
	04	3	
	02	4	
E-18767-411-018	02		
E-18767-411-021	05		
E-18767-411-024	04		

Drawing Number	Revision Number	Sheet Number	Figure Number
E-18767-411-025	02		
E-18767-411-033	03		
E-18767-411-034	03		
E-18767-411- 039	05		
E-18767-411-040	03		
E-18767-411-043	02		
E-18767-411-071	05		
E-18767-411-072	06		
E-18767-411-084	03		
E-18767-411-102	02		
E-18767-411-103	02		
E-18767-411-302	03		
E-18767-411-310	03		
E-18767-411-323	03	1	
E-18767-411-323	03	2	
E-18767-411-324	04		
E-18767-411-325	03		
E-18767-411-350	03		
E-18767-411-376	02		
E-18767-411-400	04		
E-18767-411-401	03		
E-18767-413-012			

8.1 ELECTRIC POWER

8.1.1 INTRODUCTION

This chapter describes the utility grid and its interconnections to other grids and to the Millstone Nuclear Power Station 345 kV switchyard. The on site electric system is also described. Definitions used in this chapter are given below.

Transmission System

The transmission system includes all transmission lines coming to the Millstone Nuclear Power Station complex up to, but not including, the point of connection to the 345 kV switchyard.

Off Site System

The off-site system includes the transmission system and the 345 kV switchyard and extends up to, but does not include, the main transformer bank. Included in the off site system are the reserve station service transformers, Millstone Units 2 and 3.

On Site System

The on-site system includes the Millstone Unit 2 electric power systems out to, and including, the main transformer bank (Figure 8.2–3); this includes the normal station service transformer.

Portable generator connection points have been provided on several plant electrical buses. These connections are defense-in-depth features available for coping with an extended loss of AC power (ELAP) event. The connections are shown on Figure 8.2–3, “Main Single Line Diagram” and Figure 8.5–1, “Single Line Diagram”.

Normal Operation

Normal operation is when the main generator is transmitting electrical power through the main transformer bank and when plant auxiliaries are being supplied from the normal station service transformer.

Normal Power System

The normal power system includes that equipment required to support the main turbine generator, plant systems, and equipment associated with the reactor.

Emergency Power System

The emergency power system includes that electrical distribution equipment required to support the safe shutdown and post-accident operations of Millstone Unit 2. Included in the emergency power system are the emergency 4160 V switchgear and all extensions except those going to the

normal switchgear and the reserve station service transformer and Unit 3 (Figure 8.2–3). The emergency power system and equipment is Class 1E and safety related.

Standby Power System

The standby power system includes the Class 1E emergency diesel generators, which are referred to as the on-site emergency power supply.

Preferred Power System

The preferred off site power supply is from the 345 kV switchyard and the reserve station service transformer.

Alternate Off Site Source

The alternate off site source is the 4160-V tie to Millstone Unit 3 via bus 34A or 34B.

8.1.1.1 Design Criteria

The 345 kV switchyard and the associated transmission lines provide the off site sources that are the preferred power supplies, as outlined in Section 5.2.3 of IEEE Standard 308-1971 and Criterion 17 of Appendix A of 10 CFR Part 50. The conventional and accepted design of these facilities has been shown to be conservative and reliable.

Transmission facilities connecting the Millstone generating units to the main transmission grid are designed in accordance with the “Design and Operation of the Bulk Power System,” developed by the Northeast Power Coordinating Council (NPCC).

8.1.1.2 Utility Grid

The utility electrical system consists of interconnected diverse energy sources including fossil fueled, hydro-electric and nuclear fueled plants supplying electric energy over a 345/115 kV transmission system (Figure 8.1–1C).

ISO-New England is the regional transmission organization which has authority over the operation of the transmission system in Connecticut. The main transmission system fed by Millstone Power Station is part of the New England power system. The Connecticut Valley Electric Exchange (CONVEX) is one of the local control centers in New England and assists ISO-New England in running the power system in Connecticut.

The electrical output of Millstone Unit 2 is delivered to the 345 kV switchyard (Figure 8.1–1D). Four 345 kV transmission lines feed power to the 345 kV system. Two of these lines feed the eastern part of Connecticut by connecting respectively to the Card and Montville substations. The remaining two lines feed the central part of Connecticut by connecting to the Haddam and Manchester substations.

8.1.1.3 Interconnections

Millstone Power Station is connected to the Eversource Energy, Inc. transmission system which is closely integrated with transmission systems of several other utilities and operating companies. The New England power system is part of the larger northeast interconnection power grid and is tied through various connections points throughout New England. These interconnections include 345kV, 230kV, 138kV, 115kV, 69kV, and DC lines. The New England power system is also tied to neighboring grids such as New York, Hydro Quebec and New Brunswick, which are under the control of other reliability coordinators within the NPCC region.

8.1.1.4 345-kV Switchyard System at Site

The 345 kV switchyard is designed in an arrangement as shown on Figure 8.1–1D. The switchyard consists of ten 345 kV breakers, four 345 kV transmission Units 2 and 3 lines, two 345 kV tie lines to the generator step-up transformers, and two 345 kV tie lines to the reserve station service transformers. The Millstone 1 generator step-up transformer and reserve station service transformer are no longer in service.

The breakers and motor-operated disconnect switches are controlled primarily from CONVEX via the Supervisory Control and Data Acquisition System (SCADA) and from the Millstone Unit 1 Control Room. Millstone 2 Operations is responsible for the switching and tagging of equipment located in the Millstone 1 Control Room. Via the Millstone Unit 1 Control Room Millstone 2 Operations has primary control of breakers 8T, and 9T, as well as indication only of the remaining breakers and motor-operated disconnect switches. The Millstone Units 2 and 3 Control Rooms are equipped with remote panels that show the status only of the breakers and motor-operated disconnect switches in the switchyard. Through the operation of control switches, all breakers can be operated at the switchyard, if necessary.

Each element of the Millstone bus and associated line terminations are protected by redundant sets of primary and backup relays. The primary and backup relays are supplied from separate DC sources, separate current transformers, separate coupling capacitor voltage transformers, and communication channels.

The DC power is supplied by two independent batteries, one primary and one backup. Each battery is equipped with its own charger and distribution panel. A manual transfer scheme is provided to allow one battery and charger to carry the DC load upon the failure of the other battery and charger.

8.1.2 OFF SITE POWER SYSTEM

8.1.2.1 Description

The off site power system is designed to provide reliable sources of power to the on site AC power distribution system adequate for the safe shutdown of the unit. Details of the 345 kV switchyard are shown on Figure 8.1–1D.

The switchyard, which is configured in a combination breaker-and-a-half and double breaker-double bus arrangement, buses together four 345 kV transmission line circuits, two generator circuits and two station service circuits. The Millstone 1 generator and station service circuits are no longer in service.

The four transmission line circuits terminated at the switchyard are:

a. Millstone to Haddam (Line Number 348, this line includes Line 3252)

b. Millstone to Card (Line Number 383)

c. Millstone to Montville (Line Number 371, this line includes Line 364)

d. Millstone to Manchester (Line Number 310)

These circuits connect the station to the 345 kV system transmission grid and follow a common right-of-way from Millstone to Hunts Brook Junction (9.0 miles).

These four circuits are individually mounted on separate structures which are installed across a 415-500 foot wide Right of Way to provide adequate physical independence of the transmission lines. The transmission towers, which support the four lines, consist of a combination of steel and wooden mono-pole structures, and steel and wooden H-frame structures. The towers are designed to the National Electric Safety Code Part C2, and Eversource Overhead Transmission Line Standards, which have both strength and overload design factors to provide for conservative designs. The towers for all four transmission lines are periodically inspected for proper physical condition.

With four lines feeding the Millstone Switching Station in service, the offsite power source complies with GDC-17 with no reasonable failure that can affect all circuits in such a way that none of the four circuits can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. In particular, a sequence of cascading events from a particular tower falling in a specific manner, at one of only a few specific locations, or a line falling at Hunts Brook Junction, the worst case would be the loss of two circuits.

All four of the 345 kV lines leaving Millstone cross over two 115 kV circuits which supply the Waterford Substation, and constitute the off site source for Millstone Unit 1. However, the mechanical failure of a single 345 kV line, and the consequential failure of the 115 kV circuits will not affect the preferred source of off site power to Millstone Units 2 and 3.

At Hunts Brook Junction, the four transmission line circuits diverge along three separate rights-of-way (Figure 8.1–1B). The 348 line turns west to the Haddam Substation, the 383 and 310 lines continue north to the Card Street and Manchester Substations, respectively, and the 371 line turns east to Montville Substation. At this junction, aerial crossover of lines exist (line 383 and line 310 cross over line 371/364); however, at worst, only two of the four circuits from the Millstone Switching Station would be removed from service should a structure collapse or a conductor drop.

The main transformer bank and reserve station service transformer circuits, connecting the generating station to the 345 kV switchyard, are 0.35 mile long and each circuit is installed on separate structures. These circuits are supplied from different bus positions in the switchyard, so placed that no single equipment or component failure would remove both circuits from service at one time.

The inspection and testing of the 345 kV circuit breakers and the transmission line protective relaying are done on a routine basis, without removing the transmission lines from service. The insulating oil for the transformers (main step up, normal station service transformer (NSST), reserve station service transformer (RSST)) is sampled and tested on a routine basis. During these routine inspections and tests, the operability and functional performance of the electric systems are verified.

8.1.2.2 Analysis

The possibility of power failure due to contingencies in the connections to the system and the associated switchyard is minimized by the following arrangements:

a. The connections to the system have been designed to comply with the NPCC "Design and Operation of the Bulk Power System" and the ISO New England "Reliability Standards for the New England Area Bulk Power Supply System." Compliance with these criteria ensure that the supply of off-site power will not be lost following contingencies in the interconnected transmission system. Transient stability studies have been performed to verify that widespread or cascading interruptions to service will not result from these contingencies. In addition, the loss of Millstone Unit 2 or the loss of any other unit in the system will not result in cascading system outages and thus will not cause loss of off-site power to units 2 and 3.

   The 345 kV circuit breakers are SF6 puffer type and are pneumatically operated. Electrical controls are provided for both local and remote Millstone 1 control room or CONVEX operation. Each power circuit breaker has a separate pneumatic supply unit capable of operating the breaker for a minimum of three close-open operations after the loss of its pneumatic supply unit. The essential AC station service for the power circuit breaker pneumatic supply units and the other switchyard requirements is supplied by an off site 23 kV line which has transfer capability to a source from Millstone Unit 3. The circuit breakers are equipped with a closing solenoid and two independent trip solenoids. A standard anti-pump and trip-free control scheme is used.

   Two 125 VDC batteries are located in the switchyard control and relay enclosure for switchyard relaying and control. Each battery has its own charger and DC distribution panel. The redundant batteries and protective relaying systems are physically and electrically separate.

b. The 345 kV system is protected from lightning and switching surges by overhead electrostatic shield wires, earth grounding at most structures, surge arrestors on the switchyard main buses, surge arrestors at the transformer high voltage bushings, and rod gaps on the line terminals.

c. Each 345 kV transmission line is protected from phase to phase and phase to ground faults by two sets of diverse protective relays, one primary and one back-up, both of which are high speed schemes.

The primary line protection consists of a distance relaying package in a directional comparison blocking scheme communicating with the remote terminal over a carrier current channel.

The backup line protection consists of step-distance relays operating independently from the remote terminal. This equipment is used with a transfer trip channel to provide a high speed permissive over-reaching scheme. A second transfer trip scheme provides tripping of the breakers at the remote terminal in the event of a stuck breaker at Millstone as well as tripping Millstone breakers following the reception of a trip signal from the remote end.

Pilot wire relaying is used for primary protection of the 345 kV tie lines between the switchyard and the main step-up transformers. Backup protection consists of directional distance, single zone and directional ground overcurrent relays located in the switchyard and for transfer tripping to the plant dual channel digital teleprotection equipment is used.

Tripping of the switchyard breakers following the operation of the generator or main step-up transformer bank primary and backup relays is accomplished by the means of the transfer trip via pilot wire and digital teleprotection.

The impact of open phase conditions (OPCs) on the capability of the generator step-up transformers was evaluated. The conditions analyzed consisted of single (i.e., one of three) and double (two of three) open phase conductors on the high voltage (345 kV) side of the generator step-up transformers. The analysis considered open phase conditions with and without a ground. Open phase detection (OPD) systems for the transformers were install in response to NRC Bulletin 2012-01, “Design Vulnerability in Electric Power System,” and in accordance with the NEI OPC Initiative. Upon detection of an OPC, a Main XFMR Open Phase alarm occurs within the main control room utilizing a coincidence logic scheme.

A risk-informed assessment utilizing the Millstone Unit 2 specific electrical design configuration was performed in accordance with the guidance in NEI 19-02, “Guidance for Assessing Open Phase Condition Implementation Using Risk Insights.” The assessment demonstrated that in the event of an OPC on the generator step-up transformers, the risk associated with an OPD system that is

reliant on manual operator action versus the automatic actuation of an open phase isolation system is considered “very small” in accordance with Regulatory Guide 1.174. Based on the results of the risk-informed assessment, Millstone Unit 2 has opted to utilize the OPD system and operator manual actions to address OPCs on the generator step-up transformers.

The Millstone 2 RSST 345 kV tie line is protected by two sets of protective relays, one primary and one backup. The directional distance relays detect phase-faults, and the directional ground overcurrent relays detect ground-faults. Operation of these relays will trip the appropriate 345 kV circuit breakers on the A switchyard bus, and send a transfer trip signal via digital teleprotection system with fiber optics communications medium to trip the transformer’s low side circuit breaker at the plant. The operation of any of the RSST transformer protective relays at the plant will trip the low side breaker, and send a transfer trip signal via digital teleprotection system with fiber optics communications medium to trip the appropriate switchyard breakers.

The impact of open phase conditions (OPCs) on the capability of the RSST was evaluated. The conditions analyzed consisted of single (i.e., one of three) and double (two of three) open phase conductors on the high voltage (345 kV) side of the RSST. The analysis considered open phase conditions with and without a ground. Open phase detection (OPD) systems for the transformers were installed in response to NRC Bulletin 2012-01, “Design Vulnerability in Electric Power System,” and in accordance with the NEI OPC Initiative. Upon detection of an OPC, an RSST Open Phase alarm occurs within the main control room utilizing a coincidence logic scheme.

A risk-informed assessment utilizing the Millstone Unit 2 specific electrical design configuration was performed in accordance with the guidance in NEI 19-02, “Guidance for Assessing Open Phase Condition Implementation Using Risk Insights,” The assessment demonstrated that in the event of an OPC on the RSST, the risk associated with an OPD system that is reliant on manual operator action versus the automatic actuation of an open phase isolation system is considered “very small” in accordance with Regulatory Guide 1.174. Based on the results of the risk-informed assessment, Millstone Unit 2 has opted to utilize the OPD system and operator manual actions to address OPCs on the RSST.

Primary and backup breaker failure relays are provided for each of the 345 kV circuit breakers to trip adjacent breakers in the event that the primary breaker fails to trip. The DC power for breaker failure operation is supplied from the primary and backup battery systems, respectively.

A breaker failure timing scheme is initiated every time a main breaker is signaled to trip by primary or backup relays. If the breaker has not tripped before the time period has expired, tripping of the adjacent breaker will take place.

Phase angle sensitive impedance relays are also included in the backup protection of the main step-up transformer bank tie lines to protect the generator.

Automatic reclosing of 345 kV breakers is allowed following the protective relay tripping of the 345 kV transmission lines. The reclosing is designed for time delay reclosure from the remote ends only. At the switchyard, the breakers will close via synch-check scheme.

A synch-check scheme is also provided for each circuit breaker to supervise both manual closing and automatic reclosing of the breaker.

d. Primary and backup relaying is provided for each circuit breaker along with breaker failure protection. These provisions permit the following:

1. Any circuit can be switched under normal or fault conditions without affecting another circuit.

2. Any single circuit breaker can be isolated for maintenance without interrupting power or protection of any circuit.

3. Short circuits on any section of a bus are isolated without interrupting service to any element other than those connected to the faulty bus section.

4. The failure of any circuit breaker to trip within a set time initiates the automatic tripping of the adjacent breakers and thus may result in the loss of a line or generator for this contingency condition; however, power can be restored to the good element in less than eight hours by manually isolating the fault with appropriate disconnect switches.

Complete battery failure is considered highly unlikely since two independent 125 VDC battery systems are provided. Failure of a single battery system results only in a momentary loss of one set of protective relays until the DC is manually transferred to the other battery. Therefore, no single failure could negate the effectiveness of the relaying to clear a fault.

The Millstone design provides two off site circuits between the switchyard and the 4.16 kV Class 1E buses. The immediately available off site supply is the Millstone Unit 2 RSST while the alternate supply is the Millstone Unit 3 bus 34A or 34B.

The normal supply to the plant with the plant on line is the NSST. If this source is lost due to a plant trip, a fast bus transfer scheme connects the plant electrical system (6.9 kV and 4.16 kV) to the RSST. The second or alternate source of off site power is available by manual controls to Millstone Unit 3 bus 34A or 34B for 4.16 kV power.

Physical separation of the off site power sources, switchyard protection, redundancy, and the transmission system design based on load flow and stability analyses minimize the possibility of

simultaneous failure of all power sources (reserve station service supply, standby AC emergency generators, and Millstone Unit 3 bus 34A or 34B.

The 345 kV transmission system supplying off site power to Millstone is normally operated at 357 kV at Millstone. This system voltage is controlled by varying the reactive power generation on the Millstone Power Station units. The Millstone Units 2 and 3 operators control the unit excitation as specified by CONVEX Operation Instruction Number 6913. The unit operators are required to balance the reactive power output of the units.

The CONVEX system operator supervises the system reactive power dispatch. The CONVEX operator directs the loading of all the reactive power sources in CONVEX to balance the reactive supply. The CONVEX operator keeps the Millstone Power Station reactive power generation in balance with the Eastern Area requirements so that the effect on the system of voltage variations is minimized when a unit is lost.

One objective of the reactive power dispatch is to prevent the voltage at the Millstone Power Station from going below the minimum required to support actuation of the Engineered Safety Features equipment. A switchyard voltage of 345 kV will assure successful actuation and operation of all necessary safeguards loads in the unlikely event Millstone Unit 2 experiences a Loss-of-Coolant Accident and trips off the transmission system. CONVEX operates the system to assure that this minimum voltage requirement will be met, following the loss of the unit. When in Reactor Modes 5 or 6, with the auxiliary electrical system lightly loaded, Millstone Unit 2 can assure successful actuation and operation of all necessary safeguards loads with a switchyard voltage of 335 kV. The maximum allowable voltage at Millstone Station is 362 kV based on equipment ratings.

If abnormal system conditions result in voltages approaching minimum levels, system operating instructions and procedures direct the CONVEX operator to take specific corrective actions to restore voltage.

Actual experience and system simulations show that the CONVEX operator is able to control the system voltages within the desired limits.

The Millstone plant is connected to the transmission system by four 345 kV circuits (described in Section 8.1.2.1). Transmission operating procedures are in place to ensure that no more than the minimal number of circuits would ever intentionally be taken out of service, except in an emergency, when both Millstone generating units are on-line.

If both Millstone units 2 and 3 are on line at full output, certain contingencies on the transmission system as determined by CONVEX result in procedural restrictions on the station’s net output, in order to assure that system synchronous and voltage stability will be maintained.

By careful design of the switchyard and protective relays, the possibility of the simultaneous loss of both units 2 and 3 at Millstone has been significantly reduced. The system has been computer modeled for both light load and heavy load conditions. The stability analysis indicates that the rest of the system remains in synchronism after the loss of any one Millstone unit. The probability of

losing both on-line units simultaneously is extremely small because of the preventive measures discussed in the following paragraphs. Accordingly, the Licensee believes it is reasonable to count upon on site power sources to supply the necessary station service power requirements in the very remote event that both Millstone units 2 and 3 should be lost at once accompanied by the total loss of the transmission supply to the station.

A primary objective in designing the connection of the Millstone Nuclear Power Station to the 345 kV transmission network in Connecticut has been to prevent the loss of the entire station output. The reliability criteria of NPCC and ISO New England are a fundamental part of this design process. The most severe outage which the system has been designed to survive in order to minimize the possibility of a total plant outage is as follows:

With any one of the four Millstone 345 kV transmission circuits out of service the plant remains stable for any three-phase fault normally cleared (four cycles) or any one-phase fault with the delayed clearing (eight cycles).

The Millstone units are connected to the large interconnected transmission system in the eastern half of the United States. The interconnected system frequency is maintained at 60 ± 0.03 Hz in accordance with NPCC standards for the bulk power system. The system is designed and operated such that the loss of the largest single supply to the grid does not result in the complete loss of preferred power. The system design considers the loss, through a single event, of the largest capacity being supplied to the grid, removal of the largest load from the grid, or loss of the most critical transmission line. This could be the total output of a single Millstone reactor unit, the largest generating unit on the grid, or possibly multiple generators as a result of the loss of a common transmission tower, transformer, or a breaker in a switchyard or substation.

In order to ensure the interconnected system will remain stable and offsite power circuits meet GDC-17 requirements, the following technical requirement actions and generation output restrictions will be implemented when both Millstone Power Station Unit 2 and Unit 3 are at power:

With any of the 345 kV offsite transmission lines (310, 348 (includes 3252 line), 371 (includes 364 line), and 383) out of service or nonfunctional, the nonfunctional transmission line shall be restored to functional status within 72 hours or total station output shall be reduced to $\leq$ 1650 MWe net within the next 6 hours; or, alternatively, within 7 days for Lines 310, 348/3252, and 383 or 14 days for Line 371/364 with the following action requirements in place:

a. Once per shift, verify the remaining lines are functional,

b. Once per shift, perform a weather assessment,

c. Once per 24 hours, verify the EDGs are operable and the SBO diesel is available.

If any of the above actions cannot be met or if a weather assessment predicts adverse or inclement weather will exist while a transmission line is nonfunctional (i.e., out of service), total station

output shall be reduced to ≤ 1650 MWe net within the next 6 hours to ensure the stability and availability of the electrical grid is maintained.

With two 345 kV offsite transmission lines nonfunctional, total output shall be reduced to ≤ 1650 MWe net within the next 30 minutes.

The allowed outage times (AOT) for Lines 310, 348/3252, 371/364, and 383 are based on the configuration of the transmission lines at Hunts Brook Junction where Lines 383 and 310 cross over Line 371/364 and Line 348/3252 runs to the west of the crossover. With Line 348/3252, 310, or 383 nonfunctional, the possibility exists that either Line 383 or 310 could drop on Line 371/364 and result in three lines nonfunctional. This condition would impact grid stability and therefore, a 7-day AOT is allowed with the specified action requirements in place. When Line 371/364 is nonfunctional, if either Line 310 or 383 drops, two transmission lines remain functional. Therefore, a 14-day AOT is allowed with the specified action requirements in place.

8.1.2.3 Extreme Contingency Events

The design of the switchyard protective relay schemes and circuit breaker installations is such that at most only one pole or phase of a three-phase circuit breaker will fail to clear a fault. Breakers which are designed to meet this criteria are classified as having independent pole tripping. Independent pole tripping is ensured by installing breakers with mechanically independent poles and two separate methods of tripping the circuit breaker. These installations include two sets of relays, trip coils, and two sets of current and potential transformers. The wiring for the relay packages are installed in separate duct banks, the relay packages are physically separated in the control house and two separate DC supplies are provided.

The breaker failure schemes are physically separated in accordance with NPCC Regional Reliability Reference Directory #4, “Bulk Power System Protection Criteria”, with the exception of limited legacy wire routing as permitted by Directory #4 allowance.

The 345 kV switchyard at Millstone is designed so that the loss of more than one transmission circuit due to a failure of a breaker to trip requires at least two circuit breakers to simultaneously fail to operate. The failure of even one circuit breaker is very unusual. At least three circuit breakers would have to fail before three transmission lines would be lost due to malfunctions in the switchyard.

8.1.3 STATION BLACKOUT (SBO)

On July 21, 1988, the Code of Federal Regulations 10 CFR Part 50, was amended to include Section 50.63 entitled “Loss of All Alternating Current Power,” (Station Blackout [SBO]). The SBO rule requires that each light-water-cooled nuclear power plant be able to withstand and recover from a SBO event of specified duration, requires licensees to submit information as defined in 10 CFR Part 50.63, and requires licensees to provide a plan and schedule for conformance to the SBO rule. The SBO rule further requires that the baseline assumptions, analyses and related information be available for NRC review. Guidance for conformance to the rule is provided by (1) Regulatory Guide (RG) 1.155, Station Blackout; (2) NUMARC 87-00,

Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors; and (3) NUMARC 87-00 Supplemental Questions/Answers and Major Assumptions dated December 27, 1989 (issued to the industry by Nuclear Management and Resources Council, Inc. [NUMARC], dated January 4, 1990).

8.1.3.1 Station Blackout Duration

Millstone Unit 2 has the ability to cope with a loss of preferred and emergency on-site AC power sources for up to eight hours. Unit 2 must provide decay heat removal during the event. It is assumed that there will be no AC power available during the first hour of the SBO except for battery backed power supplies (i.e., vital 120V AC). Within the first hour, Millstone Unit 2 will have an alternate AC power source available from Millstone Unit 3 Alternate AC (SBO) diesel generator.

Millstone Units 2 and 3 each have two emergency diesel generators in addition to the Unit 3 Alternate AC (SBO) diesel generator. In accordance with the SBO Rule and NUMARC 87-00, one of the four emergency diesel generators would be available and a Station Blackout is postulated to occur at one unit only at any one time. In the event of a Unit 2 station blackout, the Unit 3 AAC diesel will be made available within one hour by Unit 3 operator action and connected to Unit 2 Bus 24E by Unit 2 operator action.

8.1.3.2 Ability to Cope with a Station Blackout

Eight-hour coping assessments were performed for the 1) condensate inventory available for decay heat removal, 2) Class 1E battery capacity, 3) compressed air capability, 4) effects of loss of ventilation, 5) containment isolation, 6) emergency lighting, 7) communications, and 8) heat tracing. The results of these assessments are summarized below.

Condensate Inventory Available for Decay Heat Removal

A calculation was performed to determine the available volume in the Condensate Storage Tank (CST) for decay heat removal during SBO. The necessary condensate inventory required for decay heat removal plus cooldown is less than the Technical Specification minimum requirement for the CST, including consideration of required turbine driven auxiliary feedwater pump NPSH. Therefore, the Millstone Unit 2 condensate inventory would be adequate for decay heat removal during an 8 hour station blackout.

Class 1E Battery Capacity

An assessment was performed to ensure that the unit has adequate battery capacity to support required DC loads during a SBO. It is assumed that one battery charger will be returned to service within one hour when the alternate AC source is available. The DC power requirements for a station blackout were calculated using the methodology of IEEE-STD-485, “Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations.” This standard incorporates design margins for aging and temperature correction that are addressed in various other industry standards such as IEEE-STD-450, “Recommended Practice for

Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries." This methodology calculates battery load requirements for various sections of time. The magnitude of DC loads for each section of time is referred to as the section size. Various section sizes are calculated in order to construct a battery duty cycle.

The assessment concluded that the Class 1E batteries have adequate capacity to supply the required DC loads for eight hours during a station blackout.

Compressed Air Capability

No Millstone Unit 2 air operated valves are relied on to cope with a station blackout for eight hours. Long term decay heat removal will be accomplished by manual operation of the atmospheric dump valves. The auxiliary feedwater regulating valves will fail open on loss of air, and auxiliary feedwater flow will be controlled by varying the speed on the auxiliary feedwater pump turbine.

The Effects of Loss of Ventilation

Detailed room heat-up calculations were performed for different areas of the unit containing station blackout equipment which would have post-SBO 8 hour steady state ambient air temperatures greater than 120°F. An area containing SBO equipment with a final temperature in excess of 120°F would be considered a dominant area of concern.

NUMARC 87-00 calculation methodologies were utilized to the maximum extent possible to facilitate compliance with the SBO requirements. However, in most cases different methods had to be used. These methods used either a computer generated calculation of room/area temperatures over time, or a manual calculation of steady state temperatures. Either method is more conservative than the NUMARC methodology.

For dominant areas of concern, the SBO equipment contained in the area is either qualified for the environment, or a reasonable assurance of operability has been provided.

Containment Isolation

Containment isolation valves have been reviewed to verify that valves which must be capable of being closed or that must be operated (cycled) under SBO conditions can be positioned (with indication) independent of the Millstone Unit 2 preferred or Class 1E AC power supplies. No plant modifications nor associated procedure changes were required to ensure that appropriate containment integrity can be provided under SBO conditions.

Emergency Lighting

Emergency lighting was evaluated to ensure that the unit has sufficient emergency lighting for personnel to safely perform the operations required for an orderly safe shutdown of the unit during SBO. The Station Blackout Safe Shutdown Scenario credits the Appendix R lighting fixtures (self-contained battery units and the security lighting) for providing eight-hour lighting to

areas where operators maintain safe plant operations; areas where operators maintain the plant in a safe shutdown conditions; and access/egress routes to accomplish these functions. The 1.5-hour rated, life safety battery units provide additional portable emergency lighting for ingress / egress into plant areas during a SBO Event.

Communications

An assessment was performed to ensure that the unit has sufficient reliable effective communications systems available to plant personnel during SBO.

Heat Tracing

An evaluation was performed to determine the effects of a loss of electric heat tracing to systems essential to plant shutdown during SBO and to systems that are returned to service when power is restored. Existing electric heat tracing is adequate for coping with the eight hour SBO for systems required for safe shutdown excluding the local level instrumentation on the CST. However, an alternative method is available for determining CST level.

FIGURE 8.1–1A DELETED BY FSARCR MP2-UCR-2016-007

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Withhold under 10 CFR 2.390 (d) (1)

FIGURE 8.1–1B HUNTS BROOK JUNCTION

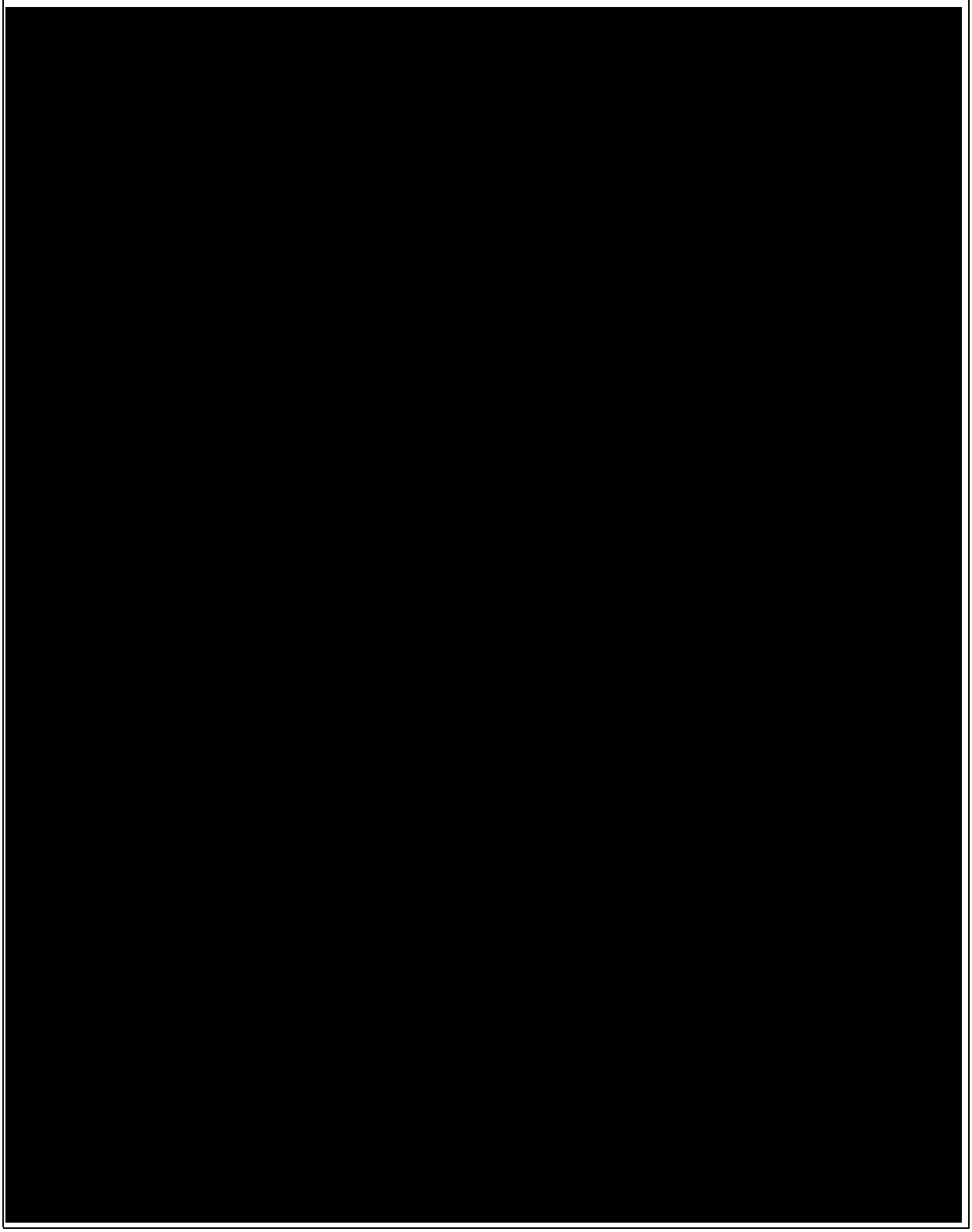
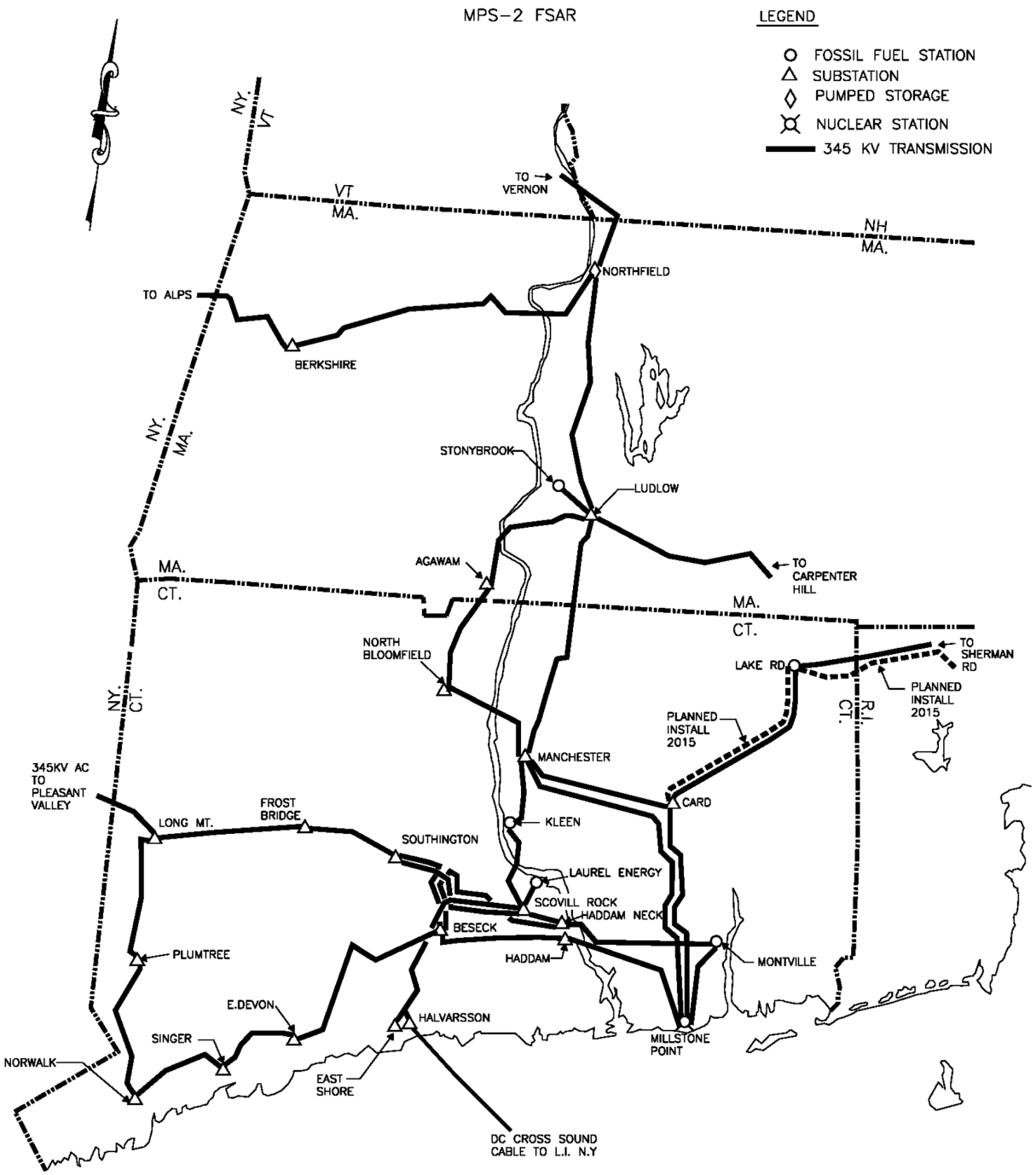


FIGURE 8.1–1C 345 KV TRANSMISSION MAP OF CONNECTICUT AND WESTERN MASSACHUSETTS



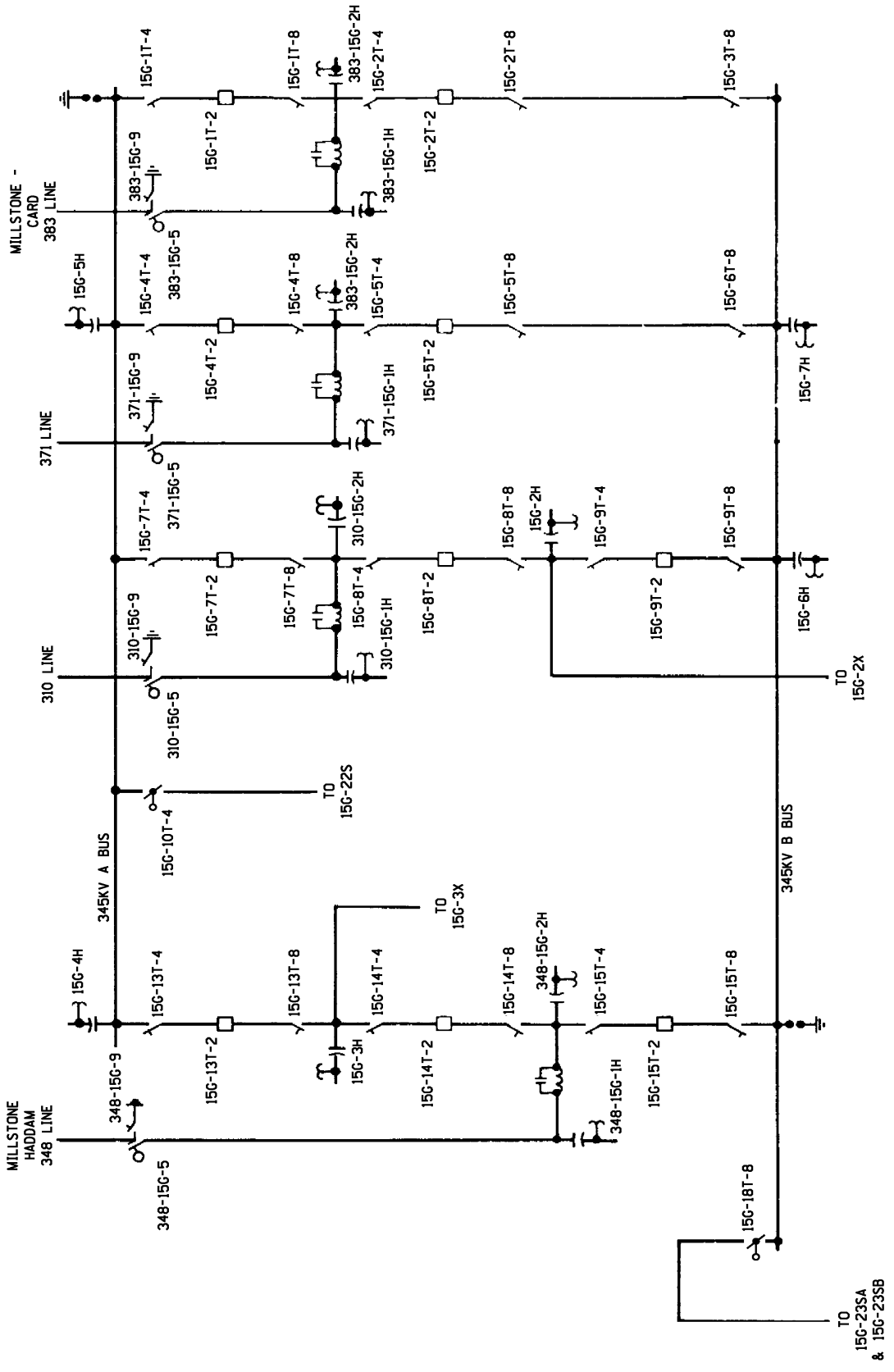


FIGURE 8.1–1D 345 KV SWITCHYARD

FIGURE 8.1–2 DELETED BY PKG FSC MP2-UCR-2011-010

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8.2 4160-VOLT AND 6900-VOLT SYSTEMS

8.2.1 DESIGN BASES

8.2.1.1 Functional Requirements

From either of two full capacity station service transformers, power is supplied to two 6900 volt buses and to six 4160 volt station service buses. The 6900 volt switchgear feeds motors of 3000 horsepower and larger. The 4160 volt switchgear feeds large (250 to less than 3000 hp) motors and supplies power to 480 volt load center transformers. Additional sources of 4160 volt power are from the emergency diesel generators, and from the Unit 3 reserve station service transformer or Unit 3 Normal Station service transformer (backfeeding).

8.2.1.2 Design Criteria

None of the 6900-volt equipment is required for safety related services, but three of the 4160-volt switchgear groups are part of Class 1E systems as defined in IEEE Standard 308, 1971. The Class 1E switchgear has been designed, built and tested in accordance with Sections 4 and 5.2 of IEEE Standard 308, 1971, Criteria 1, 2, 3, 17 and 18 of Appendix A of 10 CFR Part 50, seismic criteria as defined in Sections 5.8.1 and 5.8.1.1 of this report, and Safety Guide 6.

8.2.2 SYSTEM DESCRIPTION

8.2.2.1 System

The 6900 volt system, shown in Figure 8.2–3, is a reliable source of power for the reactor coolant and condensate pumps. The system consists of two buses, 25A (H1) and 25B (H2) each capable of being fed from the 6900 volt winding of either the normal or the reserve station service transformer. The 6900 volt winding of each transformer is sized to supply the full-load requirements of both buses.

The 4160-volt system, shown in Figure 8.2–3, consists of five buses, 24A (A1), 24B (A2), 24C (A3), 24D (A4), and 24E (A5), each consisting of a metal-clad switchgear assembly with vertical lift air circuit breakers. The 4160-volt system provides a reliable source of power to large AC motors and to 480-volt load centers. During plant operation, power is supplied to buses 24A (A1) and 24B (A2) from the normal station service transformer. Bus ties connect buses 24A (A1) and 24B (A2) to buses 24C (A3) and 24D (A4), respectively. During other periods, such as startup and shutdown when the normal station service transformer is not used, power is supplied from the reserve station service transformer directly to buses 24C (A3) and 24D (A4) and via the bus ties to buses 24A (A1) and 24B (A2). The 24E (A5) bus may be fed from either bus 24C (A3) or 24D (A4), or from Unit 3 reserve station service transformer or Unit 3 normal station service transformer (backfeeding). All auxiliary loads (other than the 6.9 kV motors detailed above) are supplied from the 4.16-kV buses. Motors of 250 horsepower and larger are connected directly to the 4.16 kV system; other loads are supplied at 480 volts from 4160-480 volt unit substations; 120/208 volt sources are available by stepping down from 480 volts.

8.2.2.2 Components

The 6900 volt switchgear for buses 25A (H1) and 25B (H2), and 4160 volt switchgear for buses 24A (A1) through 24E (A5) and Unit 3 buses 34A and 34B consist of indoor, free standing, metal clad units containing vertical lift air circuit breakers and necessary auxiliaries; all located in a Class I structure. Overcurrent and motor overload protection is provided by an overcurrent relay with instantaneous attachment, and a ground fault sensor relay on each feeder.

The equipment is all of General Electric Company manufacture, and its principal components are rated as follows:

6900 Volts

Component	Amperes Continuous	MVA Interrupting
Bus	2000	
Feeder breakers, main	2000	500
Feeder breakers, motor	1200	500

4160 Volts

Component	Buses 24A (A1) - 24D (A4)	Bus 24E (A5)	Buses 34A, B, C, D
Bus (amp cont)	2000	1200	3000/2000
Main feeder breakers (amp cont)	2000	1200	3000
Other feeder breakers (amp cont)	1200	1200	1200
All circuit breakers (MVA inter)	250	250	350

Buses 24C (A3), 24D (A4), and 24E (A5) are emergency buses that supply power to equipment required for a LOCA or other transient and conform to the requirements for Class 1E equipment.

Where means exist for manually connecting together redundant load groups, interlocks are provided as directed by Safety Guide 6 to prevent an operator error that would parallel the standby power sources. In addition to electrical interlocks, a Kirk Key Interlock System is utilized on the feeder breakers to bus 24E (A5). This operates on the principle that the key required to close one breaker can be removed from the second breaker only when its locking bolt is in a predetermined (breaker open) position. The key must be removed from one lock before another lock can be operated.

A test report, dated June 14, 1971, from General Electric on the 4160 volt emergency switchgear contains the following summary of seismic test results:

a. The stresses developed from combined seismic and normal loads do not exceed normal design stresses and are less than the allowable values specified in the American Institute of Steel Construction, Steel Construction Manual.

b. The deflections in structural members under combined horizontal and vertical acceleration forces are not large enough to introduce the possibility of breaker dropout.

c. The lowest natural frequency in the breaker support system is 19 hertz.

d. The breaker operating mechanism will withstand acceleration forces up to 44 g without false tripping or closing.

e. When vibrated in the frequency range of 0 to 37 hertz, door mounted LAC 66 relays tested on the factory floor were found to function normally; that is:

The contacts did not close due to vibration when the relay was not energized, and

The relays operated correctly when energized above their pickup setting.

8.2.3 SYSTEM OPERATION

8.2.3.1 Normal Operations

During normal operation, the source of electric power for plant auxiliaries is from the normal station service transformer. This transformer is a forced oil/air-cooled three-winding unit rated 45 MVA, with its primary connected to the generator isolated phase bus. One secondary winding feeds the 4160 volt system and the other supplies the 6900 volt system.

During a normal startup the reserve station service transformer is used. After the main generator is synchronized with the system, the operator will manually initiate live transfer of the auxiliary load from the reserve station service transformer to the normal station service transformer. The circuit is interlocked and arranged to permit only a momentary parallel. Prior to shutdown, the operator will reverse the procedure before unloading and disconnecting the main generator from the system.

8.2.3.2 Abnormal Operation

The reserve station service transformer is a similar 60 MVA three-winding unit with its primary fed directly from the 345 kV station switchyard. It is the preferred source during startup and periods when the main turbine generator is off-load.

An automatic transfer scheme is provided to ensure continuity of auxiliary power from the preferred source. The transfer scheme will sense conditions resulting in, or leading to, a loss of voltage from the normal station service transformer and will initiate a transfer for conditions such as a turbine or generator trip.

The transfer is high speed and considered simultaneous. It is not a supervised scheme. That is, the secondary breakers for both the normal station service transformer and reserve station service transformer receive essentially simultaneous signals to open and close, respectively. It should be noted there exists an interposing relay that provide a close permissive in the reserve station service transformer secondary breaker closing circuit, and starts an 8 cycle timer. The timer's function is to verify that all breaker close permissives have been accomplished within the specified 8 cycle time frame. In the unlikely event that these permissives have not been met within the allowed 8 cycle time frame, the fast transfer scheme will NOT be permitted. In the unlikely case that a phase displacement in excess of a predetermined value exists between the normal and alternate source voltages, PRIOR to the fast transfer signal being generated, the automatic transfer is also prohibited. The transfer to the reserve station service transformer is further prohibited if there is no voltage on its secondary side, or if the source is faulted. The fast transfer scheme has been demonstrated by test to result in a total dead bus time of less than six cycles.

The RSST has an on load tap changer (OLTC) that allows for changing the 4.16kV winding tap without taking the transformer out of service. When the RSST connects to the plant electrical system, this allows for changing the voltage on the 4.16kV bus and maintaining design system bus voltage.

Undervoltage protection for the emergency buses is provided via the Engineered Safety Features Actuation System (ESAS), which is discussed in FSAR Section 7.3. Undervoltage protection consists of two independent schemes (one for each 4160 Volt emergency bus 24C (A3) and 24D (A4)). Relay contacts from the ESAS undervoltage actuation logic provide outputs to control circuits for automatic bus load shedding, and start of emergency diesel generators.

A Level 1 undervoltage actuation (loss of voltage) provides a trip signal to the RSST supply breaker to the emergency bus and the normal to emergency bus tie breaker, initiates breaker trips for automatic load shedding, and provides a start signal for the emergency diesel generator associated with that bus.

A Level 2 undervoltage actuation (degraded voltage) provides a trip signal to the RSST supply breaker to the bus. When a 4160 Volt emergency bus is fed from the RSST during degraded voltage conditions, this breaker trip results in a loss of power to the bus and a subsequent Level 1 undervoltage actuation.

A Level 2 undervoltage actuation does not initiate a trip of the normal to emergency supply breaker and, as such, does not isolate the power supply from the NSST. When a 4160 Volt emergency bus is fed from the NSST during degraded voltage conditions, operator actions are used to prevent damage to safety-related equipment, in accordance with operating procedures.

Both transformers have their low-voltage windings grounded, the 6.9 kV winding has its wye neutral grounded through a resistance and the delta 4160 volt winding through a grounding transformer and resistor. Both are effective low-impedance grounds which will limit ground fault currents to less than 400 and 200 amperes, respectively. All load feeders and bus feeders are provided with ground fault trip protection.

8.2.3.3 Emergency Conditions

When off site power is not available, redundant emergency diesel generators are available to supply power to the emergency 4160 volt buses. A full description of this power source is given in Section 8.3.

It could be postulated that on site emergency power is being used following an accident because of the lack of availability of power from the Unit 2 reserve station service transformer. To relieve the diesel generator of continued post-incident operation in such a case, it is possible (by operator control) to bring off site power to Unit 2 via Unit 3 bus 34A or 34B through the Unit 3 reserve station service transformer, or the Unit 3 normal station service transformer (backfeeding).

A 4160V crosstie from Unit 3 is provided to the 24E (A5) bus from the Unit 3 reserve station service transformer or Unit 3 normal station service transformer (backfeeding). This feeder is sized to provide sufficient power to place the unit in a safe shutdown condition or to provide required minimum post accident power requirements. The 24E (A5) bus also serves as a transferable power source for spare units of emergency equipment. It supplies power for the following components:

a. Service water pump P5B

b. Reactor building closed cooling water pump P11B

c. High-pressure safety injection pump P41B

The 24E (A5) bus is connected to either the 24C (A3) or 24D (A4) bus. However, both electrical and mechanically operated key interlocks prevent connecting bus 24E (A5) to both bus 24C (A3) and 24D (A4) simultaneously. The same interlock scheme is arranged to transfer the source of control power so that the control power for bus 24E (A5) is from the same redundant system as the control power for the bus to which it is connected. When a piece of equipment is out of service for maintenance, such as the service water pump P5A, its control switch on the main control board will be in the “LOCK-OUT” position (pull-to-lock), and the 24E (A5) bus will be connected to the 24C (A3) bus to allow the spare pump to be energized. The diesel generator load sequencer is so enabled through interlocks with the bus 24E (A5) tie breakers that pump P5B will start on a safety injection actuation signal (SIAS) in place of pump P5A.

8.2.3.4 Startup, Shutdown and Refueling

For conditions of normal startup, shutdown, or refueling of the unit, all station auxiliary power is supplied from the 345 kV network through the Unit 2 reserve station service transformer or through the NSST via the Main Generator Step up Transformer with the generator links removed.

8.2.4 AVAILABILITY AND RELIABILITY

8.2.4.1 Special Features

Safety related components are duplicated and their power supplies and distribution systems are arranged to ensure that neither a failure of a bus, nor the failure of equipment connected to a bus (including the diesel generator), will prevent proper operation of the safety related systems.

The normal and reserve station service transformers are physically separated. The Unit 3 reserve transformer and Unit 3 normal station service transformer are also physically isolated from the other two and are fed from different positions in the switchyard than the Unit 2 reserve transformer. When the Unit 3 RSST is out of service, the Unit 3 NSST connection must be credited as the Unit 2 alternate off site source. A single failure of breaker 13T in the 345 kV switchyard would cause simultaneous loss of both Unit 2 off site sources, and therefore breaker 13T and associated disconnect switches must be maintained “open” when this situation exists. The capacity and capability of the Unit 2 immediate off site source is unaffected by this action. An interconnecting 4160 volt crosstie between Unit 3 and Unit 2 bus 24E (A5) provides an alternate power source from Unit 3 reserve station service transformer or the Unit 3 normal station service transformer (backfeeding).

The 24A (A1) and 24B (A2) 4160 volt buses supply loads required for normal operations while 24C (A3) and 24D (A4) provide power to vital equipment including vital 480 volt AC load centers necessary for emergency shutdown or for incident conditions. Buses 24C (A3) and 24D (A4) serve redundant systems. The vital auxiliaries supplied from either bus are sufficient under all conditions to provide safety feature action or to safely shut down the plant.

8.2.4.2 Tests and Inspections

The 6900 and 4160 volt circuits and associated devices are given operational checks while individual equipment is shut down or not in service. Circuit breakers are withdrawn individually to the test position and their functions tested. The preventive maintenance program for switchgear breakers and protective relays is performed in accordance with Maintenance Procedures, based on industry, regulatory and vendor recommendations.

FIGURE 8.2–3 MAIN SINGLE LINE DIAGRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

8.3 EMERGENCY GENERATORS

8.3.1 DESIGN BASES

8.3.1.1 Functional Requirements

To provide a reliable onsite source of auxiliary power if the preferred source is lost, the unit has two onsite emergency generators. They are redundant, independent and separate, and are used for no purpose other than that described. Each is connected to a 4160 volt emergency bus as shown in Figure 8.2–3.

8.3.1.2 Design Criteria

The emergency generators and their associated devices are designed, built, and tested in accordance with Section 5.2.4 of IEEE Standard 308 1971, Safety Guides 6 and 9, and Criteria 1, 2, 3, 17 and 18 of Appendix A of 10 CFR Part 50. Seismic criteria are defined in Sections 5.8.1 and 5.8.1.1 of this report.

8.3.2 SYSTEM DESCRIPTION

8.3.2.1 System

Two physically and electrically separate, quick starting, skid-mounted diesel generators are provided. Each diesel generator set has the capability to initiate the engineered safety features (ESF) in rapid succession, and to supply continuously the sum of the loads needed to be powered at any one time for a loss-of-coolant accident (LOCA). Each diesel generator is rated as follows:

2750 kw	Continuous
3000 kw	2000 hours
3250 kw	300 hours

The predicted loads listed in Tables 8.3-2 and 8.3-3 indicate that the continuous rating is not exceeded.

At no time during the loading sequence will the frequency and voltage decrease to less than 95 percent of nominal and 75 percent of nominal, respectively. During recovery from transients caused by step load increases or resulting from the disconnection of the largest single load, the speed of the diesel generator will not exceed rated speed plus 75 percent of the difference between rated speed (900 rpm) and the overspeed trip setpoint (value in the range of 1008 rpm to 1053 rpm). Voltage is restored to within 10 percent of rated voltage and frequency is restored to within two percent of 60 hertz in less than 40 percent of each load sequence time interval.

The adequacy of each diesel generator to perform its functions within the guidelines of Safety Guide 9 has been demonstrated by preoperational tests.

The recommendations of Safety Guide 6 are met as noted below:

a. Loss of one redundant load group (AC or DC) will not prevent the minimum safety functions from being performed.

b. Each vital AC load group can be connected to the preferred (off site) power source, or to a standby (on site) power source consisting of a diesel generator. Neither standby source can be automatically connected to any load group except the one it is normally associated with.

c. The two standby power sources cannot be automatically paralleled with each other nor with the power system.

d. One load group cannot be automatically connected to another load group.

e. Loads cannot be automatically transferred between the redundant standby power sources.

f. 4160 Volt bus A5 (see Figure 8.2–3) can be manually tied to either standby power source but only under the restrictions described in Sections 8.2.2.2 and 8.2.3.3.

8.3.2.2 Components

Each unit consists of a 12 cylinder 900 rpm opposed piston Fairbanks Morse diesel engine, a frame 966-40 generator, a high-speed solid state exciter-regulator unit, and associated control and auxiliary equipment. The unit has a continuous rating of 2750 kW, defined as 8760 hours of operation a year with an availability equal to or greater than 95 percent. This assures the opportunity for maintenance of the offload unit during post-accident operation.

The units can be started by injecting compressed air into every other cylinder between the opposed pistons, but for increased reliability in starting, compressed air is injected into every cylinder. The starting air system is shown on Figure 8.3–8. To further ensure starting reliability, the following features are incorporated in each diesel generator unit:

a. Two air flasks are provided, each having sufficient capacity for a minimum of three starts. Their characteristics are as follows:

Normal operating pressure: 150 psi (min) to 230 psi (max)

Tank design pressure: 250 psi

Design Code: American Society of Mechanical Engineers (ASME) VIII (1971)

b. The air flask valving is so designed that a rupture of one flask will not depressurize the other nor affect its capability to start the engine.

c. Both the lube oil and jacket water systems are provided with “stay warm” heaters and circulating pumps to maintain each unit at an elevated temperature to facilitate starting. An alarm alerts the operator if the heat fails.

d. A 480 volt AC motor-driven air compressor, supplied from an emergency bus and controlled by air flask pressure, is provided for each starting unit and is capable of recharging depleted flasks in one-half hour. The depleted flask pressure is considered at 150 psig since this corresponds to the pressure in the flask after three successive starts.

e. A DC motor-driven backup starting air compressor is provided for each diesel unit, manually operated or automatically controlled on flask pressure and supplied from the 125 volt DC vital system. It is annunciated when started.

f. An activated alumina desiccant air dryer, an oil coalescing prefilter, and a particulate after filter are provided in each line between the starting air compressor header discharge and the starting air flasks to provide clean dry air. The dryer is sized to accommodate the flow of both air compressors operating simultaneously.

g. The diesel engine lube oil system is so designed that the unit may be started from the main control board without priming the lube oil system.

A jacket water cooler, Tubular Exchange Manufacturing Association (TEMA) "R" shell and tube type, is provided for each emergency diesel generator. Cooling water is circulated in a closed loop through the diesel engine cooling water passages and the shell side of the cooler by an engine-driven jacket coolant pump. An electric motor-driven standby jacket coolant pump and a jacket coolant heater are provided to keep the cooling water heated while the diesel engine is not operating. The jacket cooling water system is shown on Figure 8.3–8. The cooling medium flowing through the tube side of the jacket water cooler is supplied by the service water system and is shown on Figure 9.7–1. The engines are capable of operating for approximately three minutes at full load without cooling water supply. This provides sufficient time for the initiation of flow in the service water system by pumps connected to the emergency bus. In an emergency, when service water is not available, one emergency diesel may be cooled with fire water cross-connected to the service water system from valve 2-FIRE-258.

Each diesel generator, its piping, and its auxiliaries are housed in its own Class I structure within the auxiliary building. Ventilation, emergency lighting, and fire protection are provided. The ventilating system is described in Section 9.9.12, and the fire protection is covered in Section 9.10.

Diesel fuel oil for the emergency diesel generators (EDGs) is stored in a 25,000 gallon (approximate capacity) above ground diesel oil storage tank (T - 148). The fuel oil is transferred from the storage tank by two 25 gpm diesel fuel oil transfer pumps to the two 13,500 gallon (approximate capacity), Class I, diesel oil supply tanks (T - 48A and T - 48B) which are located in the structure that houses the associate EDG. A minimum of 12,000 gallons of fuel oil is stored in each above ground diesel oil supply tank. The piping is arranged such that normally one diesel fuel oil transfer pump supplies diesel fuel oil to one diesel oil supply tank. However, there is an interconnection in the supply piping with a "locked closed" valve so that each transfer pump can supply diesel fuel oil to either supply tank. Diesel fuel oil is transferred by gravity from the supply tank to the engine-driven diesel fuel oil pump which supplies fuel oil to the diesel engine. The piping is arranged such that normally one supply tank provides diesel fuel oil to one EDG.

However, there is an interconnection in the piping with a “locked closed” valve so that one EDG can be provided with diesel fuel oil from either or both supply tanks. The volume of fuel oil in each diesel oil supply tank is maintained at a level above the required Technical Specification volume by operation of the respective diesel oil transfer pump. Both diesel oil transfer pumps take a suction from the above ground diesel oil storage tank.

The volume of fuel oil is not expected to vary during normal plant operation because the diesel generators are in standby. When the EDGs are running, diesel oil supply tank level will go down until automatic makeup from the diesel oil storage tank occurs. Each diesel oil transfer pump is controlled by a level switch which is set above the required Technical Specification volume.

The required Technical Specification volume in the diesel oil supply tanks (T-48A and T-48B) will provide sufficient fuel oil for two EDGs to operate at 2750 KW for 24 hours, and then one EDG to continue operation at 2750 KW for a total of approximately 3.5 days, assuming the two tanks are cross-connected after securing one EDG. An EDG run time of approximately 3.5 days provides a significant margin of time for replenishment of EDG fuel oil. Replenishment can be accomplished from the non-safety related above ground diesel oil storage tank or offsite sources. The nonseismic above ground diesel oil storage tank provides the normal makeup path for fuel oil to the fuel oil supply tanks. It contains fuel oil which is fully qualified and tested regularly.

At least one diesel oil transfer pump will be available to supply fuel oil from the storage tank to the diesel oil supply tanks, since the pumps are supplied from vital power sources. If the volume of fuel oil in the above ground diesel oil storage tank is taken into consideration, the one remaining EDG will be able to continue operation at 2750 KW for an additional 3.5 days. This will increase the total time one EDG will be able to operate to approximately 7 days.

If a seismic event occurs, the fuel oil in the above ground diesel oil storage tank cannot be relied upon, because the above ground diesel oil storage tank is not seismically qualified. However, replenishment of fuel oil could be accomplished via an offsite source.

The required Technical Specification volume of 12,000 gallons in each diesel oil supply tank (T-48A and T-48B) is verified by the associated Technical Specification Surveillance Requirement. The required volume of fuel oil in the above ground diesel oil storage tank is 18,106 gallons (which include EDG frequency variation effects) plus a small volume increase to compensate for unusable fuel due to instrument inaccuracy and suction pipe stub height, is not required by Technical Specifications. It is required by the Technical Requirements Manual and is verified by a similar surveillance requirement. Also, the above ground diesel oil storage tank low level alarm is set above this required volume. The associated alarm response procedure provides the necessary guidance to restore the required above ground tank volume.

It is only necessary to maintain the required volume in the above ground diesel oil storage tank when the plant is operating in Modes 1 through 4. When the plant is in Mode 5 or below, the expected EDG loading will be significantly below rated load. Therefore, the Technical Specification requirement for diesel fuel oil will provide reasonable assurance that sufficient time will be available to obtain diesel fuel oil from an offsite source, if the above ground tank is not available.

In addition to the total volume of fuel oil maintained onsite, Millstone Unit Number 2 has a fuel oil supply contract for delivery of fuel oil to the unit. Emergency Plan procedures require an evaluation of the need to order additional fuel from offsite sources within 4 hours following a LOCA and LNP. The Emergency Plan procedures also require the Technical Support Center staff to provide load shedding recommendations within 24 hours of a LOCA and LNP. These load shedding recommendations may include securing one EDG, cross connecting the two diesel oil supply tanks, and securing any electrical loads not needed to support plant recovery. The specific recommendations will vary depending on the situation, and will be developed within the 24 hour time period. This time is sufficient to allow the Technical Support Center staff to develop the specific load shedding recommendations, and will allow an evaluation of the likelihood of fuel oil delivery to Millstone Unit Number 2. If fuel oil delivery is imminent, load shedding will not be necessary.

The EDG fuel oil system is shown on Figure 8.3–7. A failure mode analysis of the EDG fuel oil system is given in Table 8.3-1.

Each diesel engine is provided with a lubrication system which circulates lube oil through a filter, a cooler, a strainer, and then through the diesel engine by means of an engine-driven lube oil pump. An electric motor driven pre-lube pump is provided to be used for lubricating the engine before it is manually started.

An electric motor-driven standby lube oil pump and lube oil heater are provided to keep the lube oil heated while the diesel engine is not operating. The diesel lube oil system is shown on Figure 8.3–8.

Each diesel generator is tied to a 4160 volt emergency bus through a circuit breaker having provision for tripping as noted in Section 8.3.4.1. When the 4160 volt system is fed exclusively from the emergency diesel generators, the system ground is provided by a resistor loaded distribution transformer between each diesel generator neutral and ground. This grounding device limits ground current to one (1) ampere.

Seismic calculations for the diesel generator components were completed by the Fairbanks Morse Power Systems Division of Colt Industries on July 26, 1972. These calculations were made for the following eight items that were judged essential for continuous operation of the diesel generator:

Engine-generator skid assembly

Heat exchanger stack assembly

Lube oil filter

Lube oil strainer

Jacket water expansion tank

Air compressor skid assembly

Air receiver tank

Electrical control cabinet, plus relay test certification

Certified calculations have been submitted and independently verified, showing that the diesel generating unit and its accessories meet the seismic requirements as elsewhere defined for this project.

At a later date, a mathematical analysis was made of the overspeed governor and shutdown system, similar to the calculations for the above listed assemblies. This analysis showed that this assembly meets the seismic requirements with a safety factor of 2.3.

8.3.3 SYSTEM OPERATION

8.3.3.1 Emergency Operation

As discussed in Section 7.3, the diesel generator associated with a 4160 volt AC emergency bus receives an automatic start signal under either of the following conditions:

a. ESAS Level 1 Undervoltage Actuation from the associated 4160 volt AC emergency bus

b. An ESAS Safety Injection Actuation Signal (SIAS)

The generator field is automatically flashed by DC from the 125 volt DC vital system when the unit reaches 250 rpm. Within 15 seconds of the start signal, the diesel generator accelerates to 90 percent or greater of rated speed and 97 percent or greater of rated voltage.

If a loss of voltage occurs on a 4160 volt AC emergency bus (with or without a concurrent SIAS), the following items occur:

a. The tie between the applicable buses (bus 24A (A1) - bus 24C (A3) or bus 24B (A2) - bus 24D (A4)) and the feeder applicable breaker from the reserve station service transformer are tripped.

b. All load feeder breakers on the emergency bus except small permanently connected loads are tripped.

c. The diesel generator breaker is automatically closed.

d. The closure of the diesel generator breaker starts a load sequencer that closes load feeder breakers in discrete steps. During the application of load steps, the minimum generator voltage and frequency and the recovery time to within 10% of nominal voltage and 2% of nominal frequency are within Safety Guide 9 requirements.

If a SIAS occurs with no concurrent loss of voltage on a 4160 volt AC emergency bus, then the corresponding diesel generator starts but does not load. Consequently, if SIAS occurs with no

concurrent loss of voltage to either 4160 volt AC emergency bus, both emergency diesel generators start but do not load.

A logic diagram outlining the operation of the diesel generator and its auxiliaries is shown in Figure 8.3–4, and the details of the control circuitry are shown in Figure 8.3–5. The load sequencers are located in the ESAS actuation cabinet, and their operation is fully described in Section 7.3.

Tables 8.3-2 and 8.3-3 provide a breakdown of the automatic loading sequence for EDG A and B, respectively. The "B" pumps on the A5 swing bus are swing pumps and can be powered from Facility Z1 (bus A3) or Facility Z2 (bus A4). The A5 swing bus can only be aligned to one facility at a time. When the A5 bus is aligned to a facility, that facility's ESF signal will sequence the appropriate "B" pump into operation if the pump breaker is not in "pull-to-lock" or the breaker is not "racked-down." By procedure, the "B" pump(s) are made available for service only if their corresponding "A" or "C" pump(s) are not available. Additional loads, such as the instrument air compressor or auxiliary feedwater (AF) pump, may be added by operator control.

Total required loads for a LOCA with only one diesel operating, as shown in Tables 8.3-2 and 8.3-3, have been conservatively estimated. The values of brake horsepower (bhp) for large motors (over 100 hp) are based on the maximum operating requirements of the driven equipment for all modes of operation. When converting horsepower to kilowatts for these motors, the actual motor efficiencies are used. Equipment loads for motors rated 100 hp and less are based on the motor full load rating and motor efficiency values which are typical for the motor size and type are used. Battery charger load requirements are based on the maximum calculated output requirement. Transformer loads for supplying low voltage, instrumentation, and emergency lighting loads are 80 percent of the transformer full load rating, without application of a factor for diversity. Demand factors are applied for specific loads in cases where operation is controlled by process conditions, such as temperature, pressure or level.

8.3.3.2 Abnormal Operation

The usual mode of operation of the onsite emergency power system is for both diesels to operate, independently supplying emergency buses A3 and A4. However, if only one diesel will start, its load sequencer will operate as described in the preceding section.

8.3.4 AVAILABILITY AND RELIABILITY

8.3.4.1 Special Features

The diesel generator units are physically isolated and their feeders are run separately. The engines require no outside power for starting other than 125 volt DC for control logic, field flashing, and breaker control. One of the two station vital batteries supplies 125 volt DC to one diesel with the other battery supplying the second diesel via separated cable routings.

The starting air solenoid valves are fail-safe. That is, a loss of DC voltage or other de-energization of either of the solenoids admits starting air to the engine.

The fail-to-start relay functions to interrupt the starting of the diesel engine if a speed of 250 rpm is not reached or lube oil pressure is not established within 12 seconds of the start initiation. Operating experience has shown that a diesel which does not start within ten seconds is not likely to start at all. Therefore, the start is interrupted to conserve the air supply for later start attempts, subsequent to a system checkout and correction.

Each of the following alarm conditions of an emergency diesel generator (EDG) have a corresponding annunciator in the main control room but does not have an annunciator on the local control panel of the applicable EDG:

Diesel generator breaker trouble

Diesel automatic start

Diesel SIAS start

Diesel generator ready to load

Exciter not reset

Diesel control switch in maintenance position

Diesel generator auto voltage regulator not set at 4160

The following alarm condition of an EDG has a specific annunciator in the main control room and a duplicate annunciator on the applicable local control panel:

* Generator differential current

Each of the following alarm conditions of an EDG has: (1) either a specific or shared annunciator on the applicable local control panel, and (2) a shared common annunciator in the main control room, “Diesel Generator U12 (or U13) Trouble:”

* Lube oil pressure - low

* Engine overspeed

Loss of unit 4160 volt auxiliary power

Lube oil temperature - high - low

Lube oil level - low

Jacket water temperature - high - low

Service water coolant - low flow

Starting air pressure - low

DC air compressor - start

Engine start failure

Fuel oil supply tank - low

Generator underfrequency

Generator undervoltage

Loss of excitation

Generator neutral ground fault

Generator overvoltage

Generator bearing temperature - high

Jacket water level - low

Fuel oil supply valve - shut

Generator stator temperature - high

Loss of 480 volt auxiliary power

Loss of 125 volt DC control power

Pre-lube pump running

Key lock switch in slow start

Jacket water pressure - low (results in trip only when on test)

Crankcase pressure - high (results in trip only when on test)

Fuel oil pressure - low (results in trip only when on test)

Reverse power flow (results in trip only when on test)

Instantaneous overcurrent (results in trip only when on test)

Core balance ground fault (results in trip only when on test)

Any auxiliary control switch not in automatic position

The three devices marked * above are the only shutdown functions permitted during an emergency start. These three conditions, if allowed to continue, would rapidly result in the loss of the diesel generator. Hence, it is advantageous to remove it from service when the condition is first sensed and thereby minimize damage to the engine or the generator. These trip devices are all considered essential as noted below:

a. Low Lube Oil Pressure

This shutdown functions upon coincidence of two out of three signals when the normal pressure of over 20 psi has dropped to 16 psi or two out of two signals when normal pressure has dropped to 18 psi. The engine manufacturer considers this to be an essential feature.

b. Engine Overspeed

A centrifugal governor shuts down the engine at about 116% of rate speed. Full load rejection can be made without actuating this trip.

c. Generator Differential Current

This feature protects the generator in the event of an internal fault.

The Voltage Restraint Overcurrent relay protects the EDG generator from excessive current in the event of a 4160 VAC bus or load fault. This relay has the capability to trip open the EDG breaker during surveillance testing and during emergency operation. The operator has the facility to reset and close the EDG breaker should the fault clear or be isolated by a load breaker.

No single failure can prevent both diesel generators from functioning. Surveillance instrumentation will warn the operator during normal station operation of detectable inadequacies or failures which could lead to loss of function of the diesel generator and its power supply system. Components whose correct functioning can be verified only during operation of the diesel generator system will be tested periodically. An analysis of the effects of a single failure within the instrumentation or controls of one of the diesel generators under an emergency condition is given in the following:

Signal or Component	Malfunction	Comments and Consequences
Undervoltage signal or potential transformer on 4160 volt bus	Failure to detect undervoltage	Undervoltage sensors are redundant and arranged in two-out-of-four logic so single failure will have no consequences on operation of system.
Bus clearing circuitry	Failure to operate	Failure of bus clearing circuitry will lock out corresponding diesel generator feeder breaker. However, other diesel generator will not be affected; sufficient emergency equipment will remain in operation for safe shutdown.
Diesel engine or generator control circuitry	Failure to start engine or to close generator breaker	Failure of one diesel engine to start or generator breaker to close will leave one completely redundant emergency system available for operation.
Differential protection relay Engine overspeed Low lube oil pressure	Relay and/or contact failure	Will result in tripping diesel generator; other diesel generator will carry redundant load, so will not affect safety.

Signal or Component	Malfunction	Comments and Consequences
Voltage Restraint Overcurrent Relay	Relay and/or contact failure	Will result in tripping the diesel breaker. Indication of the fault will be provided by the “DIESEL GEN 12U 913U) BKR CLOSING CIRCUIT BLOCKED” alarm and DIESL GEN 12U (13U) BKR TRIP. If the fault has cleared or has been isolated the operator can attempt to reset the condition and close the breaker, otherwise, the other diesel will carry the redundant load and so safety will not be affected.
Ground fault relay	Relay and/or contact failure	Indication of fault will be provided on local annunciator with common (“diesel generator trouble”) alarm in main control room. Relay and/or contact failure will initiate false alarm but will have no effect on diesel generator operation.
Load sequencer	Failure to operate in proper sequence	Worst case will result in tripping corresponding diesel generator. Other diesel generator will not be affected, and sufficient emergency equipment will remain in operation for safe shutdown or other emergency action.
Emergency bus tie breakers	Failure of electrical interlocks to prevent closure of tie breaker	Mechanical interlocks will prevent paralleling of two main 4160 volt emergency buses.
DC supply	Loss of one station battery or its distribution system concurrent with loss of off site power	Could not open supply breakers nor shed loads; diesel generator voltage would not be established due to lack of field flashing; generator breaker would not close. Because of complete separation of redundant DC supplies, other diesel generator would function as required for safe shutdown or other emergency action.

8.3.4.2 Tests and Inspections

Diesel generator tests are designed to demonstrate that the diesel generators will provide adequate power for the operation of the equipment. They also assure that the emergency generator control

systems for safety related equipment will function automatically in the event of a loss of all normal 4160 volt power, in accordance with Criterion 18.

Frequent tests are made to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and faults are annunciated. An abnormal condition in these systems will be signaled without having to place the diesel generators on test (see Figure 8.3–5).

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

a. EDG load testing, load rejection testing and start time testing is performed in accordance with Technical Specifications requirements.

b. Demonstration of the automatic sequencing equipment during normal unit operation. For details, see Section 7.3.

Monthly Surveillance Testing

To minimize the mechanical stress and wear on the diesel engine, each diesel engine will be started in slow speed using the Mechanical Governor.

During the slow start procedure, a key lock switch is used to block certain full speed function (i.e., excitation, jacket water low pressure) from occurring during reduced speed.

The protective devices that function to shut down the diesel generator are checked periodically for their trip setpoints. For the overspeed device, the diesel can be forced to increase in speed until trip occurs. Crankcase and lube oil pressure switches are disconnected and an outside pressure source applied until the trips occur. Periodically, the protective relays are inspected and their setpoints checked.

A sample of the fuel oil in the above ground storage tank will be taken manually and analyzed periodically.

TABLE 8.3-1 DIESEL FUEL OIL SYSTEM MODE ANALYSIS

COMPONENT	FAILURE MODE	MONITOR	DETRIMENTAL EFFECT	CORRECTIVE ACTION	RESULTANT STATUS
Diesel fuel oil supply tanks T-48A & T-48B	Rupture of one tank	Low level alarm	Loss of fuel oil	Isolate supply tank; open valve between lines to engines	Redundant diesel supply tank available
Supply header to diesel engine	Pipe break	Low level alarm	Loss of fuel oil and one diesel generator	Isolate break and repair	Redundant diesel generator in service (1)
Engine-driven fuel oil pump (one per engine)	Mechanical failure	Low pressure alarm	Loss of one diesel generator	Isolate pump and repair	Redundant diesel generator in service (1)
Diesel generators H7A & H7B	Loss of fuel oil for other reason	Low pressure alarm	Loss of one diesel generator	Isolate diesel generator	Redundant diesel generator in service (1)

Notes:

(1) One diesel generator has adequate capacity for emergency conditions.

TABLE 8.3-2 "A" EMERGENCY DIESEL GENERATOR LOADING

CONNECTED LOADS

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Battery Charger			X					X
TRF and Cable Losses			X					X
Motor Operated Valves			X					
Emergency Lighting			X					X
Reg Transformers			X					X
Computer UPS			X					X
Computer Room Air Conditioning			X					X
Miscellaneous Connected			X					X

SEQUENCE 1

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Service Water	450	426		365				365
High Pressure Safety Injection	400	460		388				388
Containment Air Recirculation Fans (2)		Scenario		X				X

TABLE 8.3-2 "A" EMERGENCY DIESEL GENERATOR LOADING (CONTINUED)

SEQUENCE 2

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
RBCCW	350	315			265			265
Charging (2)	200	Scenario		388	182			

SEQUENCE 3

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Low Pressure Safety Injection	400	380				325		
Containment Spray (2)	250	230				195		195
Safety Related HVAC						X		X

SEQUENCE 4

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Auxiliary Feedwater	350	360					306	268
Safety Related HVAC							X	X

TABLE 8.3-2 "A" EMERGENCY DIESEL GENERATOR LOADING (CONTINUED)

MANUAL LOADS

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG A for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Spent Fuel Pool Cooling								X
Hydrogen Recombiner								
Instrument Air Compressor								X
Containment Sample								X
Post Incident Recirculation Fan								X

Emergency Diesel Generator (EDG) running load is less than 2750 KW at all steps, this is the continuous rating of the EDG.

Notes:

(1) At < 3 minutes following auto sequencing, Motor Operated Valve (MOV) loads will have reached their required position.

(2) At < 10 minutes following auto sequence, auxiliary feedwater pump load will be at normal LOCA flow.

(3) Auxiliary feedwater pump BHP value is different for pump run-out (360 BHP) and required flow (315 BHP).

(4) Load modeled with EDG at Technical Specifications maximum Hz frequency value.

TABLE 8.3-3 "B" EMERGENCY DIESEL GENERATOR LOADING

CONNECTED LOADS

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Battery Charger			X					X
480 Volt Load Center Transformer Losses			X					X
Motor Operated Valves			X					
Emergency Lighting			X					X
Reg Transformers			X					X
Computer UPS			X					X
Computer Room Air Conditioning			X					X
Miscellaneous Connected			X					X

SEQUENCE 1

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Service Water	450	426		365				365
High Pressure Safety Injection	400	460		388				388
Containment Air Recirculation Fans (2)	75	Scenario		X				X

TABLE 8.3-3 "B" EMERGENCY DIESEL GENERATOR LOADING (CONTINUED)

SEQUENCE 2

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
RBCCW	350	315			265			265
Charging (2)	200	Scenario		388	182			
Boric Acid (2)	50				X			X

SEQUENCE 3

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Low Pressure Safety Injection	400	380				325		
Containment Spray (2)	250	230				195		195
Safety Related HVAC						X		X

SEQUENCE 4

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Auxiliary Feedwater	350	360					306	268
Safety Related HVAC							X	X

TABLE 8.3-3 "B" EMERGENCY DIESEL GENERATOR LOADING (CONTINUED)

MANUAL LOADS

Description	Motor Rated HP	Actual Load BHP	Running KW for Loads Powered from EDG B for LOCA with LNP					
			Automatically Started Loads					Post LOCA Loads
			0 Sec	2 Sec	8 Sec	14 Sec	20 Sec	
Spent Fuel Pool Cooling								X
Hydrogen Recombiner								
Instrument Air Compressor								X
Turbine Auxiliaries								X
Containment Sample								X
Post Incident Recirculation Fan								X

Emergency Diesel Generator (EDG) load is less than 2750 KW, this is the continuous rating of the EDG.

Notes:

(1) At < 3 minutes following auto sequencing, Motor Operated Valve (MOV) loads will have reached their required position.
(2) At < 10 minutes following auto sequence, auxiliary feedwater pump load will be at normal LOCA flow.
(3) Auxiliary feedwater pump BHP value is different for pump run-out (360 BHP) and required flow (315 BHP).

FIGURE 8.3–4 LOGIC DIAGRAMS DIESEL GENERATORS (SHEET A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–4 LOGIC DIAGRAMS DIESEL GENERATORS (SHEET B)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 0A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 1A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 2A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 2A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 5)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 6)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 7)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 8)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 9)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 10)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 11)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 12)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 13)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 14)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 15)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 16)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 17)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 18)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 19)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 20)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 21)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 22)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 23)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 24)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 25)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 26)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 27)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 28)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 29)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–5 DIESEL GENERATOR RELATED (SHEET 30)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–6 NOT USED

FIGURE 8.3–7 DIESEL GENERATOR FUEL OIL SYSTEM/AUXILIARY STEAM CONDENSATE AND HEATER INDEX (SHEETS 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–7 DIESEL GENERATOR FUEL OIL SYSTEM/AUXILIARY STEAM CONDENSATE AND HEATER INDEX (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–7 DIESEL GENERATOR FUEL OIL SYSTEM/AUXILIARY STEAM CONDENSATE AND HEATER INDEX (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–8 DIESEL GENERATORS' ANCILLARY SYSTEMS (SHEET 1A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–8 DIESEL GENERATORS' ANCILLARY SYSTEMS (SHEET 1B)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–8 DIESEL GENERATORS' ANCILLARY SYSTEMS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.3–8 DIESEL GENERATORS' ANCILLARY SYSTEMS (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

8.4 480 VOLT SYSTEM

8.4.1 DESIGN BASES

8.4.1.1 Functional Requirements

The 480 volt system provides power for unit auxiliary loads below 250 horsepower. Those auxiliaries required for a safe shutdown of the unit or for maintaining it in hot standby condition are served by emergency load centers and emergency motor control centers. These emergency 480 volt sources feed battery chargers for the DC system and regulating transformers for 120 volt AC instrumentation.

8.4.1.2 Design Criteria

The 480 volt system, including emergency load centers and emergency motor control centers, is designed, built and tested in accordance with Sections 4 and 5.2 of the IEEE Standard 308-1971, Criteria 1, 2, 3, 4, 17, and 18 of Appendix A of 10 CFR Part 50, and Safety Guide 6.

All of the 480 volt equipment in safety-related services is designed for the seismic conditions described in Sections 5.8.1 and 5.8.1.1.

8.4.2 SYSTEM DESCRIPTION

8.4.2.1 System

The 480 volt system shown in Figure 8.2–3 consists of six buses; two double-ended and two single-ended load centers. The double-ended load centers supply normal station loads. Each consists of two transformers and two bus sections with a tie breaker. Each transformer is connected to a different 4160 volt bus and is sized to support the loads of both 480 volt bus sections for maintenance purposes. The double-ended load center feeder and tie breakers are interlocked to prevent simultaneous connection of both 4160 volt sources to a single bus. The interlock can be defeated through the use of sync selector switches which permit momentary paralleling of load center 22A with 22B or load center 22C with 22D across the respective tie breakers 22A-1T-2 or 22C-1T-2. The sequence allows: (1) paralleling across and closing of the tie breaker immediately followed by the automatic opening of one of the selected feeder breakers, or (2) paralleling across and closing of a selected feeder breaker immediately followed by the automatic opening of the associated tie breaker. The two single-ended load centers (B5 and B6) each consist of a single transformer and bus section and supply loads necessary for safe shutdown or post-incident conditions.

In addition, a portable generator connection point has been provided to back feed bus 22F. This connection is a defense-in-depth feature that is available for coping with an extended loss of AC power (ELAP) event. The connection is shown on Figure 8.2–3. “Main Single Line Diagram”.

Motor control centers, located throughout the unit, are fed from these load centers. Four of these are considered emergency motor control centers (B51, B52, B61, B62), and are fed from the

emergency load centers. The loads carried by the emergency motor control centers include auxiliaries such as shown in Tables 8.3-2 and 8.3-3.

Normal station lighting is provided by 480-208/120 volt and 480-277 volt transformers located throughout the plant. Power for lighting of critical operating areas, such as the control room, diesel generator rooms, DC switchgear rooms, emergency shutdown panels and refueling areas is supplied as described above but has additional backup lighting panels fed from the emergency motor control centers.

On a loss of all AC sources, the life safety lighting requirements for egress are satisfied by the 1.5-hour rated, self-contained battery units installed throughout the station. In addition, the MP2 Appendix R Compliance Report, Section 6.3 credits 8-hour rated battery backed emergency lighting units (ELU's) and security lighting, which illuminate outdoor access and egress routes, to cope with a design basis fire. The security lighting power supply is backed by a security diesel generator. These Appendix R lighting fixtures (8-hour rated, self-contained battery units, and the security lighting) are credited by the Station Blackout Safe Shutdown Scenario. The 1.5-hour rated, life safety battery units provide additional portable emergency lighting for ingress / egress into plant areas during a SBO Event.

8.4.2.2 Components

All 480 volt power supply and utilization equipment in safety-related services is housed in Class I structures. These spaces are equipped with smoke and fire detection systems and portable fire extinguishers.

The 480 volt load centers are free standing, indoor type with dripproof enclosures containing drawout-type air circuit breakers, motor starters, and associated relays, instruments and fuses. Their rating is as follows:

Transformers		
	Transformers	1500 kVA (AA), 2000 kVA (FA)
	Temperature rise	80°C (40°C ambient)
	Impedance	7.59 to 8.7%
Bus		2000 amp continuous
Circuit Breakers		
	Main incoming feeders	3000 amp continuous
		65,000 amp interrupting (without instantaneous trip)
	MCC feeders and ties	42,000 amp interrupting (without instantaneous trip)
	Load feeders	600 amp continuous
		30,000 amp interrupting (with instantaneous trip)

Overcurrent and motor overload protection are provided by overcurrent trip devices on each load center feeder breakers as follows:

Emergency MCC - short and long time elements on all phases

Mag-jack M-G - instantaneous on all phases; long-time element on two phases

Other feeders - instantaneous and long time elements on all phases

Vital 480V buses 22E & 22F are equipped with class 1E voltage monitoring devices. These devices initiate an alarm in the control room if the devices respective bus voltage exceeds 516 VAC. This warns operators to correct the situation using plant procedures to minimize the overvoltage conditions which increase the potential for long-term equipment degradation.

A test report dated July 21, 1970, from General Electric, describes the vibration and shock tests given to a prototype of this switchgear. The seismic qualification of the transformers was verified by a mathematical analysis. A summary of the results of the 480 volt switchgear tests is given below:

a. GE-type AKD-5 switchgear has been vibration tested in each of the three directions at 0.5 g over the frequency spectrum of 5 to 500 hertz.

b. AKD-5 switchgear has been shock tested to 40 g on a Navy medium-weight shock machine.

c. GE-types AK-50 and AK-25 breakers have been vibration tested in each of the three directions at 0.5 g over the frequency spectrum of 5 to 500 hertz.

d. AK-50 and AK-25 breakers remained operative during shock at accelerations up to 15 g on a Navy medium-weight shock machine.

e. The AK-50 breaker has been vibration tested without loss of function during vibration at its lowest resonant frequency of 29 hertz at 5.0 g input.

f. The AK-25 breaker has been vibration tested without loss of function during vibration at its lowest resonant frequency of 44 hertz at 3.0 g input.

The motor control centers are General Electric free standing types with dripproof indoor enclosures containing combination starters, molded case circuit breakers, individual starter control transformers, and associated relays and fuses. Two Emergency Control Centers, B51 and B61, are encapsulated in environmentally controlled enclosures due to their location in areas which could be subject to a steam environment resulting from a high-energy line break. The ratings for the motor control centers are as follows:

Bus (main)	600 or 800 amp continuous
(vertical)	300 amp continuous

Circuit breakers 22,000 amp interrupting (min.)

Overcurrent and motor overload protection are provided on each feeder as follows:

Thermal-magnetic circuit breaker

Three thermal overload relays in each motor starter (except for MOV)

A certification from General Electric, dated December 20, 1971, testifies that tests they made on August 21, 1970, were on a prototype of the emergency motor control centers furnished for this project. A summary of these tests is given.

a. The test article was swept in frequency from 5 to 500 hertz at a one-half octave per minute sweep rate at a constant 0.5 g input acceleration.

b. The equipment was vibrated in each of its three orthogonal axes; vertical, horizontal in-breadth, and horizontal fore and aft.

c. All vibration sweeps were made with a 480 volt AC, 60-hertz source connected to the control center main bus.

d. Vibration sweeps were made in two modes of functional status:

    1. With each starter unit disconnect in the ON position but starter not energized.

    2. With each starter energized.

e. The tests conducted indicate the equipment, as tested, is suitable for applications up to at least 0.5 g base input accelerations through a frequency band width from 5 to 500 hertz.

f. The resonant level of the equipment structure appears to be at 5 to 6 hertz, while component resonant levels are at higher frequencies.

g. Interchanging of starter units vertically within the equipment will not significantly alter the response of the individual components inasmuch as the structure is nonresonant at the frequencies at which components are resonant.

h. These tests indicate that the 7700 line motor control center is suitable to at least 0.5 g base input accelerations through a frequency band width from 5 to 500 hertz.

Each of the two control element assembly drive power sustainers is connected to a 480 volt emergency load center. The control element assembly drive power sustainer system consists of two ride-through flywheel motor generator sets. The motor generator sets have the ability to hold

all the control elements and to sustain the motion of any element already being stepped during a one-second interruption in the station service voltage.

The only motors inside the containment required during a LOCA are those driving the containment air recirculation and cooling fans and the post incident recirculation fans noted in Section 8.4.1.2. The suitability of these motors to meet the environmental and seismic shock requirements is presented in Sections 6.5.4.2 and 6.6.4.2, respectively.

8.4.3 SYSTEM OPERATION

8.4.3.1 Normal Operation

Normally, each emergency load center transformer is fed from a corresponding emergency 4160 volt bus, which in turn receives power from a station service transformer as shown in Figure 8.2–3. There is no way in which these two load centers can be tied together. Each of these load centers feeds two emergency motor control centers.

8.4.3.2 Emergency Operation

Under emergency conditions, there is no change in the mode of operation of these load centers and motor control centers. The only changes are those associated with the means of supplying power to the 4160 volt emergency buses, as described in Sections 8.2 and 8.3.

8.4.4 AVAILABILITY AND RELIABILITY

8.4.4.1 Special Features

The load groups on each emergency load center and its associated motor control centers are redundant to each other and separate. Hence, a complete failure of power, or failure of components on one channel does not affect the availability of services to care for a LOCA.

One charging pump is fed from power source Z1 and one from Z2. The third charging pump can be aligned with either a channel Z1 or channel Z2 source. Two power selector switches with electrical and Kirk key interlocks prevent tying the two motor control center buses together, as shown in Figure 8.4–1. If one pump is out of service, the selector switches are positioned (under administrative control) to feed the third pump from the power source redundant to that which is feeding the operating pump. Thus, in case of an accident, two charging pumps are available, each fed from a separate and redundant power source.

As noted in Section 8.2.3.3 and in Figure 8.2–3, the third service water pump is fed from 4160-volt bus A5, which can be energized from either a channel Z1 or Z2 source. The service water strainer associated with this pump has a feeder from each of two 480 volt emergency motor control centers fed from these two vital sources. Two power selector switches (under administrative control) with electrical and Kirk key interlocks prevent tying the two motor control center buses together, as shown in Figure 8.4–2. If the selector switches are not positioned to complete the supply circuit, an alarm is given on the main control board. Interlocking also assures

that the power supply for the third strainer is aligned to the same source as the power for the associated service water pump. Thus, two service water pumps and their related strainers are always available, each with a separate and redundant power source.

Shutdown Cooling Suction Header Containment Isolation valve 2-SI-651 also has the capability of being supplied power from either Facility Z1 or Facility Z2. This valve is not an installed spare unit of emergency equipment as discussed in Section 8.2.3.3. The alternate power provides a means to open 2-SI-651 post-LOCA for boron precipitation control and will not be utilized during normal plant operation. Therefore, this valve is not considered a Facility Z5 component. To prevent placing the plant in an unanalyzed condition for separation concerns, the cross connection of 2-SI-651 to Facility Z2 is limited to boron precipitation control post-LOCA combined with a loss of power to MCC B51 or for testing purposes. Emergency operating procedures provide guidance for aligning 2-SI-651 to its alternate power.

8.4.4.2 Tests and Inspections

The 480 volt circuit breakers, motor starters, and associated equipment are tested while individual equipment is shut down or not in service. Circuit breakers and starters are withdrawn individually and their functions tested. Accidental grounds on the load center buses or feeders are monitored continuously by ground detectors and are alarmed in the main control room.

Load center transformers are periodically given an insulation resistance test either on line or during a refueling outage in accordance with a preventive maintenance program.

FIGURE 8.4–1 CHARGING PUMP - POWER SUPPLY CROSSOVER P18B

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.4–2 SERVICE WATER STRAINER ML1B POWER SUPPLY CROSSOVER SCHEME

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

8.5 BATTERY SYSTEM

8.5.1 DESIGN BASES

8.5.1.1 Functional Requirements

Battery power is needed for a variety of uses, such as “trip” and “close” of electrically operated circuit breakers, certain solenoid operated valves, certain control systems, and devices used during turbine coastdown. Through DC/AC inverters, the batteries also provide power to the reactor protective system, engineered safety features actuation, and to vital instrumentation.

8.5.1.2 Design Criteria

The components of the battery system are designed, built, and tested in accordance with the requirements of:

- Safety Guide 6
- Section 5.3 of IEEE Standard 308, 1971, with the exception of the requirements for the frequencies of battery discharge tests (battery service tests) and battery charger performance tests. (These tests are required to be conducted in accordance with the Surveillance Frequency Control Program.)
- Section 6.3.4 of IEEE 308, 1980
- Criteria 1, 2, 3, 17, and 18 of Appendix A of 10 CFR Part 50
- Section 4 of IEEE 279, 1971
- IEEE 650, 1979

Seismic criteria are defined in Sections 5.8.1 and 5.8.1.1 of this report.

8.5.2 SYSTEM DESCRIPTION

8.5.2.1 System

A 125 volt DC supply system consisting of two isolated switchgear bus sections (with key interlocked tie breakers) and associated distribution panelboards is provided. Each of the two DC switchgear bus sections is supplied by a separate 60 cell battery and two 400 amp battery chargers, normally connected in parallel to form a single 800 amp charger. Figure 8.5–1, Sheet 1, illustrates this system. An additional battery/charger/switchgear combination is provided for nonvital loads associated with the turbine and its auxiliaries, and for backup of selected vital instrument AC panels through DC/AC inverters. Figure 8.5–1 Sheet 2 illustrates this system.

8.5.2.2 Components

The ratings of the principal components of the main (vital) battery systems are as follows:

Batteries—2 (C&D Type LCR-33)

Rating 8 hr (amp-hr)	2320
1 min (amp)	2240
Voltage, nominal (volts)	125
min	105
max	140

Chargers—2 pairs

Capacity, individually (amp)	400
Capacity, total each pair (amp)	800
Voltage (volts, DC)	100-147
Regulation (%)	±1.0
Ripple, max (%)	2.0
Current limit (%)	125

Switchgear—2 groups

Circuit breaker s	
Continuous (amp)	1600
Interrupting (amp)	50,000
Bus	
Continuous (amp)	2000

A separate 125 volt battery system has been provided to supply power to the DC emergency oil pumps associated with the main turbine and the turbine-driven feedwater pumps and to two separate nonvital DC/AC inverters for backup of the 120 VAC vital 1 and 2 instrument panels. The system consists of a 1500 amp-hr battery, two 200 ampere chargers, and a DC switchgear assembly that includes the required motor starters and 600 amp continuous, 25,000 amps interrupting air circuit breakers. A DC panelboard, fed from the turbine battery, supplies turbine generator control power.

Station battery rooms are in Class I structures and have continuous induced ventilation as described in Section 9.9.6.

A 400 Amp Charger System was subjected to a Seismic Simulation Test Program as required by the Ametex Purchase Order C288264 and Clark Dynamic Test Laboratory's Seismic Test Procedure (Test Plan Number JID 21-00581-TP-Seismic Rev. 0, Issue Date 4/29/21). This test program was performed on 9/15/2021 - 9/17/2021, as documented in Test Report Number JID 21-00581, Rev. 0, Issue Date 10/04/2021.

The test program consisted of single axis Resonance Search testing and triaxial Random Multifrequency testing. The specimen was instrumented with accelerometers, electrically powered, loaded and monitored as directed by Ametek during the test program.

It was demonstrated that the battery charger specimen possessed sufficient structural integrity to withstand, without compromise of structure or monitored functions, the prescribed simulated seismic environment.

Each station battery is contained in two 2-step free standing racks bolted to the floor. In the seismic calculations, the lower step was omitted in calculations of natural frequency and resisting frictional forces of top step cells were neglected. The system is symmetric about the mid-frame. Calculations included the determination of stresses due to horizontal motion in short direction, horizontal motion in longitudinal direction, vertical motion, and side rail stresses due to short motion. All checked out to indicate a conservative design.

A 2000A fused disconnect switch is used with each station battery. The fused disconnect switches are seismically qualified using the GIP-2 methodology for resolution of USI-46 as discussed in Report Number 03-0240-1367, Revision 0, dated December 14, 1995.

Two LCU-33 [(1)] battery cells were installed in a battery rack which was attached to the table of a vibration machine. The specimens were vibrated between 3.5 and 19.5 Hz along each of the two mutually perpendicular horizontal axes and the vertical axis simultaneously. Acceleration was applied at an angle of 36° from the horizontal. The two cells were connected in series and the output voltage monitored during the test. There was no evidence of electrical discontinuities or mechanical damage during or as a result of these dynamic exposures.

8.5.3 SYSTEM OPERATION

8.5.3.1 Normal Operation

The 125 volt DC vital system shown in Figure 8.5–1, sheet 1, provides a reliable power and control source for systems required for a safe shutdown and for post-incident operation. A typical protective scheme is shown in Figure 8.5–2, Reactor Trip Switchgear schematics.

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(1) The LCR-33 replacement cells are similar to the tested LCU-33 cells. See C&D Charter Power Systems, Inc., Environmental and Seismic qualification Report Number QR 163409-01, dated May 24, 1995.

The system includes two 60 cell lead calcium standby batteries and four 400 amp battery chargers. One pair of chargers is associated with each battery. The tie breakers are interlocked to prevent the paralleling of the two batteries but allow either battery to feed both buses, if necessary during cold shutdown or refueling operations. While in this alignment (one battery connected to both buses, each battery is sized to supply the total connected vital loads for one hour without charger support. In all other plant modes of operation, each battery is sized to supply the most severe postulated load conditions on its associated bus for an eight-hour duty cycle. During normal operation, the 125V DC load is supplied from the battery chargers with the batteries floating on the 125V DC bus. On loss of AC power to the battery chargers, the DC load is supplied from the battery. Power is available to these DC loads for a period of 8 hours. After 8 hours, it is assumed that AC power is restored to the chargers either from the system or the emergency generators.

Each 400 amp of charger is able to recharge a fully discharged battery in less than 12 hours while supporting the maximum continuous load of that battery. (A fully discharged battery is a battery where the cells are discharged to 1.75 volts per cell). Each battery/charger combination feeds its own independent and separate distribution channels via a main DC switchgear assembly. Each switchgear bus supplies two independent and physically separated 125 volt DC distribution panelboards for vital loads, two 125 volt DC panelboards for nonvital loads, and two separate 125 volt DC/120 volt AC static inverters for vital instrumentation. Each of the two inverter outputs is connected to an independent and physically separated 120 volt AC panelboard. Thus, four separate DC feeders and two separate AC feeders (via the inverters) are available from each battery for distribution within the plant.

In the event of the loss of all sources of AC, minimal lighting requirements for egress per Life Safety Code (NFPA) are satisfied by 1.5-hour rated, self-contained automatically charged battery pack, lighting units.

8.5.3.2 Emergency Conditions

There is no change in the mode of operation of the battery system during emergency conditions except for the loads associated with loss of offsite power. On loss of AC power, vital battery loads include breaker tripping and closing circuits, continuous control circuits, inverters, EDG air compressors and EDG field flash current. The battery service test profile shown in Table 8.5-1 envelopes the battery discharge current versus time profile for both vital batteries on loss of AC power.

8.5.4 AVAILABILITY AND RELIABILITY

8.5.4.1 Special Features

Each of the two 125 volt DC emergency power sources is equipped with the following instrumentation in the control room to enable continual operator assessment of emergency power source condition:

a. Direct current bus undervoltage alarm.

b. Battery current indication.

c. Bus current indication.

d. Charger malfunction alarm (including input AC undervoltage, output DC undervoltage and overvoltage, high temperature, low airflow , high and low output current).

e. Direct current bus voltage indication.

f. Direct current ground alarm.

The undervoltage relay is designed specifically to monitor the charging supply for a station battery and to sound an alarm if this supply fails. The setting of the relay is above 125 volts, and the equipment is capable of operating down to 95 volts.

The 125 volt switchgear circuit breakers have magnetic series trip elements for overcurrent protection as follows:

Battery charger - instantaneous, short-time and long-time elements

Other feeders - short-time and long-time elements

Auxiliary spray charging header supply valve 2-CH-517 and Loop 1A charging header supply valve 2-CH-519 have the capability of being supplied power from either Facility Z1 or Z2. These valves are not an installed spare unit of emergency equipment as discussed in Section 8.2.3.3. The alternate power provides a means to open 2-CH-517 and close 2-CH-519 post-LOCA for boron precipitation control and will not be utilized during normal plant operations. Therefore, these valves are not considered Facility Z5 components. To prevent placing the plant in an unanalyzed condition for separation concerns, the cross connection of 2-CH-517 and 2-CH-519 to Facility Z1 is limited to boron precipitation control post-LOCA, coincident with a loss of normal power, combined with a loss of Facility Z2 DC power, or for testing purposes. Emergency operating procedures provide guidance for aligning 2-CH-517 and 2-CH-519 to their alternate power.

8.5.4.2 Tests and Inspections

To ensure battery functional capability and to assure detection of battery degradation, the following tests and inspections are performed periodically:

a. Float voltage is measured.

b. Cells are checked for cracks or leakage.

c. The plates of cells are checked for buckling, discoloring, grid cracks and plate growth.

d. Specific gravity of each cell is measured.

e. Voltage of each cell is measured.

f. Electrolyte level of each cell is checked, and all water additions are recorded.

g. Temperature and specific gravity of the electrolyte of a pilot cell are measured.

h. Battery charger alarms and battery charger voltages are checked.

i. Bottoms of cells are visually inspected for flaking buildup and for abnormal cell plate deterioration.

j. Periodically a battery charger AC supply breaker will be opened to verify the load-carrying ability of the battery. During this test an undervoltage annunciator will indicate that the battery chargers are out of service.

Batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of cell degradation long before it fails. Battery replacement will be made when a capacity test indicates the battery capacity is at or below 80 percent of the manufacturer’s rating.

TABLE 8.5-1 BATTERY SERVICE PROFILE FOR BATTERIES 201A AND 201B

Discharge Step	Duration	Nominal Test Current (DC Amperes)
1	0 to < 1 Minute	311
2	1 to < 29 Minutes	224
3	29 to < 30 Minutes	242
4	30 to < 120 Minutes	221
5	120 to < 479 Minutes	197
6	479 to < 480 Minutes	243

FIGURE 8.5–1 SINGLE LINE DIAGRAM 125 VDC / VAC SYSTEMS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–1 SINGLE LINE DIAGRAM 125 VDC / VAC SYSTEMS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 5)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 6)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 7)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 8)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 9)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 10A)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 10B)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 10C)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 10D)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 11)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.5–2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER CONTROL CIRCUITRY (SHEET 12)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

8.6 ALTERNATING CURRENT INSTRUMENTATION AND CONTROL

8.6.1 DESIGN BASES

8.6.1.1 Functional Requirements

The 120 volt AC instrument power for reactor protection, engineered safety features and vital instrumentation is supplied by four physically isolated and electrically independent vital instrument panels. Each vital instrument panel is powered by one of four physically isolated inverters. Two inverters are powered by each of the two redundant batteries.

8.6.1.2 Design Criteria

This power source is normally derived from the station battery system described in Section 8.5, hence the same criteria are met.

8.6.2 SYSTEM DESCRIPTION

8.6.2.1 System

This special 120 volt AC power supply and distribution system is shown in Figure 8.5–1, Sheets 1 and 2. It consists of two separate and redundant systems composed of four essential buses for vital instrumentation and control, and two regulated buses for nonvital instrumentation and control.

The four vital AC buses are normally supplied from DC/AC static inverters. The 120-volt regulated AC instrument panelboards supply the nonvital instrumentation requirements.

Each regulated panel is powered through a transfer switch from a normal or alternate source which includes one Uninterruptible Power Supply (UPS) fed by a nonemergency motor control center and one 480/208-120 volt regulating transformers fed by an emergency motor control centers. During normal operation, the transfer switch is in the normal position and power is fed from the normal MCC and UPS. Should this voltage drop to a predetermined low voltage setpoint, the transfer switch transfers to the alternate supply and power will be fed from the alternate MCC and regulating transformer.

In addition, portable generator connection points have been provided on 120VAC vital panels VA20 and VA40. These connections are defense-in-depth features that are available for coping with an extended loss of AC power (ELAP) event. The connections are shown on Figure 8.5–1, “Single Line Diagram”.

In addition, for increased flexibility during planned maintenance periods, the normal source UPS can be supplied with an administratively key controlled source, an emergency MCC.

To provide increased reliability, each of the four vital instrumentation panels has an alternate power supply via a “zero break” static transfer switch. Vital channels 1 and 2 are fed from

separate DC/AC inverters whose source of DC is the turbine battery. Vital channels 3 and 4 are fed from one of the two regulated AC instrument power panels.

8.6.2.2 Components

In addition to the battery system components described in Section 8.5.2.2, four DC/AC static inverters with static transfer switches are required.

The Inverters are Ametek Ferroresonant style inverters.

The Inverter characteristics are as follows:

Rating	125 volts DC/122 volts AC 10 kVA, any pf, 2 wire, undergrounded
Voltage regulation (%)	±2
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:

- Inverter Output Overload(1)
- Out of Sync(1)
- Low DC Voltage(1)
- Low DC Disconnect(1)
- Battery Supplying Load (Local Alarm Only)
- Bypass Source Failure(1)
- Low AC Output Voltage(1)
- Bypass Source Supplying Load (Independent Alarm to Main Control Board)
- Over Temperature(1)
- L1 To Ground (Local Alarm Only)
- Fan Failure(1)

- Common Alarm

(1)Denotes input sources to the Common Alarm at Main Control Board

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 inverters and static switches was demonstrated by National Technical Systems Laboratories and documented in their Test Report Number PR036428-TP-15, Rev. 1 dated 8/14/2017.

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 representative of the eight being used. One was tested in five directions, and the other two in two biaxial positions. The seismic simulation was accomplished by utilizing a pulse amplitude modulated sine wave. The test specimens were subjected to five such pulses at each resonant frequency and the frequencies of the building resonances. The acceleration amplitudes were as required by the criteria of this project. All specimens were found satisfactory at even higher levels. Circuit breakers and an undervoltage relay were monitored and all performed satisfactorily.

8.6.3 SYSTEM OPERATION

8.6.3.1 Normal Operation

The power supplied by the above sources is used for essential control systems as illustrated in Figure 8.5–2.

8.6.3.2 Emergency Operation

No changes are required in this system to cope with any unit emergency situation.

8.6.4 AVAILABILITY AND RELIABILITY

8.6.4.1 Special Features

By means of a synchronizing signal, the inverter output is always in synchronism with its associated static switch's alternate source. Under this condition, transfer to the alternate source can be made at any time with no “bump” due to voltage difference or phase displacement. The static switch will transfer the AC load to the alternate source under any of the following conditions:

- Inverter overload
- Inverter failure

8.6.4.2 Tests and Inspections

Each of the four vital inverters can be removed from service for inspection of its components, and for an operational test. These tests can be done individually by manually transferring the inverter load (zero break) to its associated alternate source. The static switch can be removed from service via a manual bypass switch. However, during this period of time, the associated alternate source is no longer available.

8.6.4.3 Additional Features

The backup sources for inverters 1 and 2 are inverters 5 and 6, respectively. Inverters 1 and 2 are synchronized to inverters 5 and 6 via static switches VS1 and VS2, respectively.

Inverters 3 and 4 are backed up by and synchronized to the normal source UPS or the backup 480/208-120 regulating transformers via VR11 and VR21 and static switches VS3 and VS4, respectively.

As internal frequency reference is used to keep the output of the inverter at 60 Hz ±0.5% if the alternate source is lost or out of tolerance. It is also used to prevent an out-of-synchronization transfer.

8.7 WIRE, CABLE AND RACEWAY FACILITIES

8.7.1 DESIGN BASES

8.7.1.1 Functional Requirements

In addition to transmitting electric power from the proper source to the designed load device, these facilities must be of a type and be properly installed and segregated to function during all postulated accident conditions.

8.7.1.2 Design Criteria

The electrical loading of conductors does not exceed, and is generally less than, the ampacities recommended by American Institute of Electrical Engineers - Insulated Power Cable Engineers Association (AIEE-IPCEA) “Power Cable Ampacities,” (joint publications S-135-2 and P-46-426, 1962), and in open-top cable trays without maintained spacing in between cables the ampacities recommended by IPCEA-Institute of Electrical and Electronics Engineers (IEEE), joint publication, IPCEA Publication Number P-54-440, and National Electrical Manufacturers Association (NEMA) Publication Number. WC 51-1972.

The percent cross-section fill of wireways is governed by the allowable cable ampacities.

The physical support of wireways meet the recommendations of Chapters 2 and 3 of the National Electrical Code, 1971.

Separation of conductors and of their wireways meets the requirements of Section 4 of IEEE 279, 1971, Sections 4 and 5 of IEEE Standard 308, 1971, Criteria 1, 2, 3, 4, 17 and 18 of Appendix A of 10 CFR Part 50. Electrical penetration assemblies conform to IEEE 317, 1971 or 1976.

8.7.2 SYSTEM DESCRIPTION

8.7.2.1 System

Cable types required to operate inside the containment after an accident are tested in an environment more severe than that expected in service. All cables have a sufficient degree of flame resistance to obviate the need for flame retardant coating or special fire extinguishing systems.

Fire detection is provided by a system of fire and smoke detector heads in the areas listed below. The power supply for this detector system comes from a 125 volt DC panelboard, as described in Section 8.6.2.1. In the event of smoke or flame in these areas, an annunciator in the control room displays the alarm. Additional fire protection is provided as described in Section 9.10. The areas covered by ionization type detector heads are as follows:

Computer room

Control room ventilating ducts

Main exhaust equipment room

Fuel handling area

Auxiliary building and radwaste ventilating room

Cable spreading room

Electrical penetration rooms

Cable vault

Medium-voltage switchgear rooms

Low-voltage switchgear rooms

Cable chases

8.7.2.2 Components

The raceway system is made up of cable trays, conduits and underground ducts, with the electrical cables contained therein.

Cable trays are of galvanized steel, ladder type or solid bottom, with solid covers where required. Hangers for trays carrying vital circuits are designed to withstand seismic disturbances as described in Sections 5.8.1 and 5.8.1.1.

Conduits are galvanized rigid steel where embedded in reinforced concrete in building slabs. The duct banks going to the intake structure are heavily reinforced and will withstand a seismic disturbance, as noted in Section 5.8.2.3.

All in-line splices of conductors are made only in metal enclosures such as terminal boxes and junction boxes or in designated splicing areas of the cable raceways.

Table 8.7-1 lists the physical and electrical characteristics of the cables that are used, and indicates the qualification tests. The certified results of such tests are available for inspection. Cable from vendors and of materials other than those listed in Table 8.7-1 are used when qualified in accordance with the applicable characteristics, standards, tests and the required service conditions.

The electrical penetration assemblies through the wall of the containment structure form part of the containment pressure boundary, as described in Section 5.2.6.1.1. 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 power penetrations are designed to meet or exceed all requirements of IEEE Standard 317, 1971. Replacement medium voltage power penetration modules are designed to meet or exceed all requirements of IEEE Standard 317, 1976. A complete prototype and production test program demonstrated the suitability of the assemblies for operation under the prescribed service conditions. These tests

include leak integrity, current carrying capacity (continuous, short time overload, and fault current) dielectric strength, insulation resistance, and resistance to seismic and thermal transients. The conductors contained therein meet all criteria applying to each class of service. The high-voltage conductors terminate in a stress cone and lug. Low voltage power cables larger than 4/0 AWG terminate in lugs rigidly fixed in a terminal box at each end of the penetration assembly. All other low-voltage power, control, and instrument cables except for Class 1E instrumentation and EEQ required circuits are terminated on terminal blocks enclosed in the penetration boxes at both ends of the penetrations. The Class 1E instrumentation and required EEQ circuits are terminated with EEQ qualified terminations at both ends of the penetrations. Incore detection cables, coaxial cable, and certain special multiconductor cables are terminated in connectors mounted in terminal boxes at both ends of the penetration assembly. All terminal boxes are designed for NEMA IV service.

A leak rate test is performed on each penetration assembly following its installation. This test is capable of detecting a leak rate of 1×10^{-2} cc/sec of dry nitrogen at ambient temperature when maximum design pressure is applied across the penetration assembly barrier. To effect this test, each assembly is fitted with a gage to monitor the pressure, and is then charged with 30 psig of nitrogen. The assembly is so designed that all seals, including conductor seals, are monitored by the gage.

8.7.2.3 Cable Ampacities

The maximum ampacities for cable installations at Millstone Unit 2 are based upon the following:

- Conductor temperature does not exceed 90°C.
- General Plant Areas are normally considered to be 50°C for ampacity analyses except specific locations (e.g., intake structure).
- Standard tray fill is 35 percent. When tray fill is above 35 percent the higher tray fill number is used.

Ampacities for cable installations in Free Air, Conduit, Maintained Space Cable Trays, and Underground Ductbanks are based upon IEEE S-135, IPCEA P-46-426, 1962. Ampacities for cable installations in Random Spaced Cable Trays are based upon ICEA P-54-440, NEMA WC-51, 1972. Cables with sizes or installations not identified in the standards above, have ampacities based on published industry papers or published vendor data.

8.7.3 AVAILABILITY AND RELIABILITY

8.7.3.1 Separation

The raceway systems are so designed that any one raceway channel may be physically sacrificed under accident conditions. The layout drawings in Figure 8.7–1 show typical examples of the separation of raceways serving different channels.

The separation of redundant cables is accomplished by spatial separation of their cable trays. This spatial separation is normally not less than four feet vertically and 18 inches horizontally to guard against damage from external fire, missile, or other accidents.

Where these spacings between trays of redundant systems cannot be maintained (physical obstructions, points of necessary convergence, crossovers, etc.), barriers are provided to preserve the physical and electrical integrity of the cables.

Vertical stacking of separate redundant trays is avoided where possible.

Where separate redundant trays must be stacked with less than 4 feet vertical separation and/or their horizontal separation is less than 18 inches, rated fire barriers must be used. In the cable vault, where an existing automatic detection and suppression system is located, separation barriers are not required.

In the case of crossover of trays carrying redundant cables, there shall be a minimum separation of 6 in. clear space between them with a barrier (equivalent to 0.5 inch of Marinite 36).

Typically, rated fire barrier material employed to enhance raceway separation is one-half inch Marinite 36, or equivalent. Installation will be as follows:

a. Horizontal Separation. A vertical barrier, one foot above and one foot below the trays, or to the ceiling or floor.

b. Vertical Separation. A horizontal barrier between trays extending one foot each side of the tray system.

c. Cross-overs. A horizontal barrier extending out one foot from each side of each tray, and five feet along each tray from the crossover.

In lieu of the above, conduit or a totally enclosed tray may be used and the two channels do not touch each other. For certain configurations, trays with ventilated covers, or cables in Sil-temp wrap, are considered enclosed raceways.

Generally, no more than one channel of separate redundant systems is run through a compartment containing machines with flywheels. Where this cannot be avoided, each case is evaluated for additional protection. Similarly, no more than one channel is generally routed through an area containing high pressure (275 psi and over) piping. Where necessary, the redundant raceway will be run at least ten feet from such piping. Where this spacing cannot be achieved, pipe restraints are provided and each case is evaluated for additional protection.

Where routing is unavoidable through an area subject to a possible open accumulation of quantities (gallons) of oil or other combustible liquids as a result of rupture or leakage of a fluid system, a single separation channel only is routed through this area. Furthermore, the cables are protected from dripping liquids by conduit or covered tray.

Raceways (exposed conduits, trays, penetrations, etc.) are generally stacked vertically in the following relative order:

a. 6900 volt power

b. 4160 volt power

c. 480 volt load center subfeeders

d. *480 volt power and general control

e. *Shielded control and instrumentation

(*Shielded control and instrumentation cables may be run with unshielded control and instrumentation for short distances such as risers into equipment).

Within each of these classifications, nonvital cable may be run with vital cables. However, a nonvital cable is never routed in raceways of more than one separation channel.

Vital circuits, components, and equipment are those that are safety related. That is, the safe operation and shutdown of the nuclear system is dependent on them. Vital systems meet the single failure criterion and therefore are redundant and separate.

Where indicators and other devices are not essential for the safe functioning of a vital system, current and potential transformer secondaries or other low-energy circuits feeding such devices are considered nonvital circuits.

Equipment, devices, cables and raceways have an assigned number that indicates if they are in vital service or not, and also indicates the channel. These designations are shown on one-line and three-line diagrams, schematics, circuit and raceway schedules, and the instrument index.

A “Z” prefix on a cable, conduit or tray number indicates a vital system. The absence of the Z prefix indicates nonvital service. The first figure of a cable, conduit or tray number designates the channel. Such an alpha-numeric prefix is called the Facility Code, and its use is further explained in Table 8.7-4.

Vital power and control cables fall mainly into two redundancy classifications; Channel Z1 and Channel Z2. In a few cases there is also a Channel Z5, which is a system that can be transferred from one source to another, and is run as described below. Cables such as those in reactor protection service are assigned to Channels Z1, Z2, Z3 and Z4. As shown in Table 8.7-4, nonvital Channel 1 may be routed with vital Channel Z1, and Channel 2 with Channel Z2. Low level buffered signal outputs from Z3 and Z4 channels of a four-channel instrument system may be run with nonvital channels 1 and 2 respectively. Where the system lacks a current limiting feature, Z3 and Z4 are run separately.

Channel Z5 is associated with the spare units fed from 4160-volt emergency bus A5; namely, service water pump P5B, Reactor Building Closed Cooling Water (RBCCW) pump P11B, and High Pressure Safety Injection (HPSI) pump P41B. The power circuits and the control circuits for this equipment are all transferred simultaneously to Channel Z1 or Z2 sources. Thus, their circuits are routed together as Channel Z5. The Z5 control circuit and power circuit for the spare 480 volt charging pump P18B, are transferred to Z1 or Z2 sources independent of the above circuits. Hence, the Z5 charging pump circuits are routed separately from those associated with bus A-5. Nonvital Channel 5 circuits are those associated with instrument loops or metering circuits. Channels 5 and Z5 circuits are routed together only where it can be demonstrated that their transfer to Channel 1 (Z1) or 2 (Z2) sources does not impair the separation of redundant safety related circuits.

The turbine driven AFP, the steam inlet valve, and speed adjuster motor have the capability of being transferred from their normal power supply Facility Z2 125VDC (Panel DV-20) to Facility Z1 125VDC (Panel DV-10). The transfer is accomplished by switching the position of two key-locked isolation switches on panel C-05 in the event of a loss of Facility Z2 125 or a loss of DC panel DV-20. The associated wiring from panel C-05 is routed in dedicated Z5 conduit to panel C-21, panel C-10 and ultimately to the steam inlet valve and the speed adjuster motor.

Shutdown Cooling Suction Header Containment Isolation valve 2-SI-651, auxiliary spray charging header supply valve 2-CH-517 and Loop 1A charging header supply valve 2-CH-519 have the capability of being supplied power from either Facility Z1 or Z2. However, these valves are not Facility Z5 components (Sections 8.4 and 8.5). The load side of the disconnect switches for 2-SI-651 are routed Z1 and the load side of the transfer switches for 2-CH-517 and 2-CH-519 are routed Z2. The transfer from the normal 480 volt for 2-SI-651, is accomplished through manual operation of local Kirk-keyed transfer switches which prevent tying the two motor control center buses together. Upon transfer to Facility Z2, the control for valve 2-SI-651 will be transferred from the control room to the local control panel. The transfer from the normal 125VDC for 2-CH-517 and 2-CH-519 is accomplished through manual operation of key-locked selector switches, located on panel C02, which prevent tying the two 125 VDC power sources together. These manual transfer schemes are consistent with the requirements of Safety Guide 6.

Computer and annunciator circuits are considered nonvital. Their inputs are from nonvital Channels 1 and 2 that may be routed with vital circuits as shown in Table 8.7-4. The Channel 1 and 2 segregation for the nonvital circuits is lost when they enter the final raceways at the computer or the annunciator terminal cabinets. The 480 volt power supply to the computer is reduced to 120 volts by an uninterruptible power supply (preferred) or a regulating transformer (alternate). The internal power supply provides 36 volts (fused one-half amp) to the digital inputs, and the analog inputs are 10-50 mA. The power supply to the annunciator is from two separate redundant AC to DC power supply systems which isolate the annunciator DC voltage from the AC power sources and isolate the two AC power sources from each other.

The control element drive system (CEDS), including the CEDS logic cabinets, are also considered nonvital. Two separate feeders, one from each of the two nonvital 120 VAC instrument buses, supply control power to the logic cabinets. The feeder cables are routed in separate raceway from the distribution panels to the cabinets, but are ultimately bundled together within a

common logic cabinet. Separation of the nonvital 120 VAC instrument buses is maintained, however, because separate double pole circuit breakers installed in each of the nonvital distribution panels provide isolation between the two buses. No redundancy is intended, or required, for the CEDS logic cabinet power supplies.

All power supply equipment is identified with respect to its source. Odd first digits are assigned to Channel 1; i.e., B1, B12, etc. Even first digits are assigned to Channel 2; i.e., B2, B21, etc.

To assist in verifying proper separation, the jackets of all cables are color coded. Table 8.7-4 indicates the physical separation applied to cables and raceways, and the cable jacket color for each case.

Apertures for entrance of redundant vital cables into control boards, panels and relay racks are separated by at least twelve inches of air space. Where this cannot be accomplished, the entrance is made with conduit or enclosed tray.

Redundant vital cables terminate on terminal blocks at least six inches apart. Internal wiring of redundant vital circuits, and any associated devices, is separated by a minimum of six inches. Where the minimum spatial separation of six inches is not feasible, nonflammable heat shrinkable tubing, noncombustible barriers or conduit are used to provide separation. Exceptions to these criteria may be permitted on an individual basis with analysis and documentation of acceptability in the Electrical Separation specification. Acceptability of lesser separation should be based on degree of hazard and mitigative measures which demonstrate that the effects of lesser separation do not degrade Class 1E safe shutdown circuits and equipment below an acceptable level. Exceptions cannot be taken between redundant vital wires/devices inside control panels. Nonvital channels may be wired to the same device, but their conductors are bundled separately.

Whenever practicable, shipping splits and structural stiffeners are utilized as natural barriers. The barriers are comprised of two sheets of steel plates with a minimum of one inch air space or insulating fire-resistant material in between, if devices and/or wiring are mounted on both faces of the barrier. If devices and/or wiring are mounted on the barrier on only one face, a single sheet of steel plate for isolation is satisfactory provided devices and/or wiring on the other side are installed at least one inch away from the barrier. The barriers are properly supported for structural strength, and extend from top to bottom and front to back to a depth which provides a minimum of six inches separation between channels.

Typical layouts illustrating the separation of redundant wireways are shown in Figure 8.7–1.

8.7.3.2 Tests and Inspections

The various documents indicating the separate routing of redundant cables are carefully cross-checked during the design of the system. The color coded jackets of the cables permit a visual inspection to verify that the separation criteria are observed.

Insulation resistance of all power cables is measured initially and spot checks are made at refueling periods. Such tests indicate significant trends in the unlikely event there has been deterioration of the insulation.

The pressure gages on the electrical penetration assemblies are located in the auxiliary building penetration rooms and are readily accessible. These assemblies remain charged with nitrogen throughout their life, and a pressure reading will be taken and recorded periodically.

TABLE 8.7-1 CABLE CHARACTERISTICS

TYPE	VOLTAGE RATING	SERVICE	CONDUCTORS			INSULATION	JACKET	PROTOTYPE TESTS (CERTIFIED)					FACTORY TESTS (CERTIFIED)			APPLICABLE STANDARDS	VENDOR	COMMENTS
			TYPE	STRAND	TEMP			RADIATION RESISTANCE	PHYSICAL	ELECTRICAL	FLAME	POST-ACCIDENT ENVIRONMENT	FLAME	ELECTRICAL	OTHER			
MEDIUM VOLTAGE POWER	8 KV	6.9 KV	CU AL	Class B	90C	EPR/CSPE	-	A.O. Aged 7 days 121 C $5x10^7$ Rad Phys. Test Elec Test Flame Test	Per IPCEA S-68-516 Sect 6.0 Tensile, Elongation, Aging, Ozone, Moisture Absorption	Per IPCEA S-19-81 Sect 6.0 AC, DC, IR, PF, SIC, Corona, Level per AEIC No. 5	Passes Horiz. & Vert. Tests per SPEC, Passes IPCEA S-19-81 Sect 6.19	N/A	Passes IPCEA S-19-81 Sect 6.0	Per IPCEA S-19-81	Corona Level & Corona Factor	IPCEA S-68-516 S-19-81	General Cable	Each Conductor Shielded
	5 KV	4.16 KV	CU AL	Class B	90C	EPR/CSPE	-					N/A	Passes IPCEA S-19-81 Sect 6.0	Per IPCEA S-19-81	Corona Level & Corona Factor	IPCEA S-68-516 S-19-81	General Cable	Each Conductor Shielded
LOW VOLTAGE POWER	1 KV Note (1)	480 VAC 125 VC	CU	Class B	90C	EPR/CSPE	CSPE	Preaged 7 days, 121 C $5x10^7$RAD	Per IPCEA S-68-516 Sect 6.0	Per IPCEA S-19-81 Sect 6.0	Passes Horiz. & Vert. Tests per SPEC, passes IPCEA Sect 6	$4.9x\ 10^8$ R 52 psig 292°F 100% RH 11/2 1/2% boric acid by wt	Passes IPCEA S-19-81 Sect 6.0	Per IPCEA S-19-81	-	IPCEA S-68-516 S-19-81	Anaconda	
CONTROL	1 KV Note 1	125 VDC 120 VAC	CU	Class B	70C	Kerite FR	Kerite FR	Results Available	Stated minimum tensile & elongation	HI POT & IR Tests on previous cable	Passes Horiz. & Vertical Tests	Results available	Passes IPCEA S-19-81 Sect 6.0	HI POT & IR Test on each Length	-	-	Kerite	Test Results Are Proprietary
INSTRUMENT	600 V	-	CU	Class B	-	Flame Retardant XLPE	Neoprene	$2x10^8$ Rad	See Note (2)	Passed AC hi-pot and IR tests as specified	2/C number 16 passed per Spec para 8.4.3.2	See Note (3)	Passes IPCEA S-19-81 Sect 6.0	HI POT, IR & Cdr Res on each reel	-	IPCEA S-66-524 S-19-81	Cerro	
SPECIAL INCORE CABLES	600 V	-	CU and TC Type K	Class B and Solid	-	Flame Retardant XLPE	Neoprene								-	IPCEA S-66-524 S-19-81	Cerro	
COAX	-	-	-	-	-	XLPE	CSPE	Results of previous tests on similar cable; $1x10^8$ rad min.; physical, elect. & flame test data			Available 9/73	N/A	Passed IPCEA S-19-81, Sect 6.0	Per MIL-C-17D HI POT Imped, IR Cap	Physicals per IPCEA S-66-524, S-19-81	MIL-C-17D IPCEA S-66-524 S-19-81	BIW	
COMMUNC	600	-	CU	16	-	Flame Resistant 105C PVC	Flame Resistant 105C PVC	-	-	-	-	-	-	-	-	-	Gai-Tronics	
THERMO-COUPLE	300	-	Type E	Solid	-	Flame Retardant XLPE	CSPE	N/A	N/A	N/A	2/C number16 passed per Spec para 8.3.2.2	N/A	Passed IPCEA S-19-81, Sect 6.0	Per IPCEA S-66-524	Conductor DC resistant	IPCEA S-66-524 ANSI C96.1-1964	BIW	

(1) In addition to the 1 kV voltage to be used for Low Voltage Power and Control Cables, it has been justified, September/October 1981 memos GEE-81-776/GEE-81-845, that 600V Insulation can also be used for new and replacement cables.

(2) Tested for tensile, elongation, aging, heat distortion, ozone and moisture absorption.

(3) Pre-aged 7 days at 121C, $2x10^8$ Rad, 104 psig, 340°F, 2% BA, Hipot, IR, Flame Tests, also Note 2.

TABLE 8.7-2 OMITTED

TABLE 8.7-3 OMITTED

TABLE 8.7-4 FACILITY IDENTIFICATION

Function		Facility Code				
		Channel 1	**Channel 3**	**Channel 2**	**Channel 4**	**Channel 5**
Power	Vital	Z1	Z3	Z2	Z4	Z5
	Nonvital	1	--	2	--	5
Control	Vital	Z1	Z3	Z2	Z4	Z5
	Nonvital	1	--	2	--	5
2-Channel Instrument	Vital	Z1	--	Z2	--	--
	Nonvital	1	--	2	--	--
4-Channel Instrument	Vital	Z1	Z3	Z2	Z4	--
	Nonvital	--	--	--	--	--
C-E Designation	Vital	A	C	B	D	--
	Nonvital	X	--	Y	--	--
Color Code	Vital	Red	Green	Yellow	Blue	Orange
	Nonvital	Black	--	White	--	Gray

FIGURE 8.7–1 RACEWAY PLANS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.7–1 RACEWAY PLAN CABLE VAULT AREA -6, ELEVATION 25 FEET 6 INCHES (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.7–1 RACEWAY PLAN CABLE VAULT AREA -7, ELEVATION 25 FEET 6 INCHES (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 8.7–1 RACEWAY PLAN CONTAINMENT AREA ELEVATION 14 FEET 6 INCHES (SHEETS 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 9: Auxiliary Systems

CHAPTER 9—AUXILIARY SYSTEMS

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9.13–1	P&ID Auxiliary Steam and Condensate (Sheet 1)

CHAPTER 9 - AUXILIARY SYSTEMS

9.1 GENERAL

The auxiliary systems discussed in this section are those supporting systems which are required to ensure the safe operation, protection or servicing of the major unit systems and, principally, the reactor coolant system. In some cases the dependable operation of several systems is required to fulfill the above requirements and, additionally, certain systems are required to operate under emergency conditions. The extent of the information provided for each system is commensurate with the relative contribute of, or reliance placed upon the system in relation to the overall plant safety. The systems considered are:

a. Chemical and Volume Control System (CVCS);

b. Shutdown Cooling System;

c. Water Treatment System;

d. New Fuel Handling System;

e. Spent Fuel Pool Cooling System;

f. Cooling Water System;

g. Plant Protection System;

h. Fire Protection System;

i. Compressed Air System;

j. Sampling System;

k. Auxiliary Steam System

The majority of the active components within these systems are located outside of the containment. Those systems with connecting piping or ductwork between the containment and the auxiliary building are equipped with containment isolation valves as described in Chapter 5.

Drawing symbols and abbreviations used throughout the FSAR are shown in Figures 9.0–1, 9.0–2, and 9.0–3. Power supplies for these auxiliary systems are discussed in Chapter 8.

FIGURE 9.0–1 P&ID LEGEND

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.0–2 P&ID LEGEND

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.0–3 GENERAL LEGEND NOTES

GENERAL P&ID LEGEND NOTES

1.0 PIPING CLASS CODE DEFINITION

1.1 Piping Classes are designated by a three-letter code. The first letter indicates the Primary Valve and Flange rating; the second letter the type of material; and the third letter the code to which the piping is designed.

1.2 The designations are as follows:

First Letter	**Rating (pounds)**
B	2500
C	1500
D	900
E	600
F	400
G	300
H	150
J	125
K	Class 250 AWWA
L	XH AWWA
M	150 (Gravity)

Second Letter	**Material**
B	Carbon Steel
C	Stainless Steel
D	Copper
E	Ductile Cast Iron, Mechanical Joint
F	Cast Iron, Soil
G	Carbon Steel, Epoxy Lined
R	Carbon Steel Radwaste
S	Stainless Steel Radwaste

FIGURE 9.0–3 GENERAL LEGEND NOTES (CONTINUED)

Third Letter	Design Code	Shop Fabrication	Field Fabrication & Installation
A	Nuclear Power Piping, Class I	ANSI B31.7-69	ASME Section III, 1971
B	Nuclear Power Piping, Class II	ANSI B31.7-69	ASME Section III, 1971
C	Nuclear Power Piping, Class III	ANSI B31.7-69	ASME Section III, 1971
D	Code for Pressure Piping, ANSI B 31.1.0		

1.3 Certain piping systems or portions of piping systems designed and fabricated to ANSI B 31.1.0 shall have additional testing and examination requirements over and above those required by that code. These additional requirements will be identified by a fourth code letter. The code letters and corresponding requirements are described as follows:

Fourth Letter — Additional Specific Requirements

A. ANSI B 31.1.0 Requirements Plus:

1. 100 percent volumetric examination of Butt Welds
2. Seismic Analysis
3. Base Material Traceability (as per Paragraph 1–723.1.3, ANSI B 31.7)

B. ANSI B 31.1.0 Requirements Plus:

1. Random volumetric examination of Butt Welds (1)
2. Base Material Identification (as per Paragraph 3-723, ANSI B 31.7)
3. Seismic Analysis

C. ANSI B 31.1.0 Requirements Plus:

1. Base Material Identification (as per ASTM requirements)
2. Seismic Analysis

(1) On Butt Welds over four inch nominal pipe size, a minimum of 10 percent of the Butt Welds in a specified class shall be examined volumetrically. Those welds examined must include welds by each welder or welding operator performing welding in the specified class.

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

9.2.1 DESIGN BASES

9.2.1.1 Functional Requirements

The chemical and volume control system (CVCS) is designed to perform the following functions:

a. Maintain the chemistry and purity of the reactor coolant system;

b. Maintain the required volume of water in the reactor coolant system by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature and for other coolant losses or additions;

c. Provide a controlled path for transferring fluids to the radioactive waste processing system;

d. Inject concentrated boric acid into the reactor coolant system upon a pressurizer low pressure and/or a containment high pressure signal.

e. Control the boron concentration to obtain optimum control rod positioning, to compensate for reactivity changes associated with major changes in coolant temperature, core burnup, and xenon concentration variations, and to provide shutdown margin for maintenance and refueling operations or emergencies;

f. Provide auxiliary pressurizer spray for operator control of reactor coolant system pressure during shutdown;

g. Provide a means for functionally testing the check valves which isolate the safety injection system for the reactor coolant system;

h. Provide periodic sampling analysis of reactor coolant boron concentration and fission product activity;

i. Collect the controlled bleed off from the reactor coolant pump seals;

j. Pressure test the reactor coolant system.

9.2.1.2 Design Criteria

The CVCS is designed in accordance with the following criteria.

1. The system is designed to accept the discharge when the reactor coolant is heated at the design rate of 100°F/hr and to provide the required makeup when the reactor coolant is cooled at the design rate of 100°F/hr.

2. The system is designed to automatically divert the letdown flow to the radioactive waste processing system (RWS) when the volume control tank is at its highest permissible level and to automatically line up charging pump suction to the RWST on low VCT level.

3. The system is designed to supply makeup or accept discharge due to power decreases or increases.

4. The system is designed with the capability to allow design transients of ±10 percent of full power step changes and ramp changes of ±5 percent of full power per minute between 15 and 100 percent power.

5. The system is designed so that the volume control tank has sufficient capacity to accommodate the reactor coolant system water inventory change for a full to zero power decrease with no makeup system operation, with the volume control tank initially at the normal operating level.

6. The system is designed to assure that the activity in the reactor coolant system does not exceed 411 μCi/cc (at 77°F) for an assumed 1 percent failed fuel condition.

7. The system is designed to maintain the reactor coolant chemistry within the limits specified in Table 9.2-2.

9.2.2 SYSTEM DESCRIPTION

9.2.2.1 System

The CVCS is shown in the simplified block diagram, Figure 9.2–1, and in the detailed piping and instrumentation drawing, Figure 9.2–2. The normal flow path is shown by the heavy lines in Figure 9.2–2. Coolant letdown flow from one reactor coolant loop cold leg first passes through the tube side of the regenerative heat exchanger, where the temperature is reduced to approximately 260°F at normal conditions, and then through the letdown control valves. The letdown control valves are modulated by the pressurizer level control program and control the letdown flow to maintain proper pressurizer level. The letdown coolant temperature is reduced to 120°F in the letdown heat exchanger downstream of the letdown control valves. This final temperature reduction is made to prevent possible damage to the ion exchange resins. Flashing of the hot liquid between the letdown control valves and the letdown heat exchanger is prevented by controlling back pressure with a pressure control valve downstream of the letdown heat exchanger.

The cooled letdown flow is passed through the letdown prefilter to remove suspended solids. The letdown flow is then directed through one of the two purification ion exchangers. The ion exchangers contain resins for removal of corrosion and fission products. The third ion exchanger, the deborating ion exchanger, is used for boron removal near the end of core cycle life when it is

no longer practical to use the feed and bleed method for boron dilution. The third ion exchanger is also used for RCS cleanup during plant refueling outages.

The letdown strainer located downstream of the ion exchanger retains particulate matter that may pass through the ion exchanger retention element. It also serves as a backup to the ion exchanger retention element, to prevent resin from entering the volume control tank, in the event of gross failure of this element.

The coolant then flows through the letdown post filter and is sprayed into the volume control tank where gaseous fission products are released and hydrogen gas is dissolved in the coolant. The charging pumps take suction from the volume control tank and pump the fluid through the shell side of the regenerative heat exchanger for recovery of heat from the letdown flow before being returned to the reactor coolant system.

Chemical shim and shutdown control for the reactor is maintained by control of the boron concentration. Boron concentration is monitored by periodic sample analysis. Changing the boron concentration in the reactor coolant system is accomplished by dilution with primary makeup water, addition of concentrated boric acid, or any desired blend of both to the reactor coolant system by the feed and bleed operation.

If the level in the volume control tank reaches the high level setpoint, the letdown flow is automatically diverted to the radioactive waste processing system. If the level in the volume control tank reaches the low-level setpoint, primary makeup water, borated to the existing concentration of the reactor coolant system, can be automatically supplied to the volume control tank by the makeup system. The system is designed to automatically line up charging pump suction to the RWST on low-low VCT level.

With the level in the normal control band, the volume control tank has sufficient capacity to accommodate the reactor coolant system water inventory change for a full to zero power decrease with no makeup system operation.

Concentrated boric acid is prepared in the batching tank and then transferred to the boric acid storage tanks for storage. The boric acid tank contents may be recirculated by the boric acid pumps. These pumps are also used to transfer boric acid to the volume control tank or to the suction of the charging pumps. They also transfer boric acid to the charging pump suction on a safety injection actuation signal (SIAS). On SIAS, both boric acid pumps transfer boric acid directly to the charging pump suction header. Should the boric acid pumps fail to start on SIAS, an additional line is provided for gravity-feeding concentrated boric acid from the storage tanks to the charging pump suction header.

The reactor coolant system may be leak tested when the plant is shut down in accordance with plant procedures. The system is also provided with connections for installing a hydrostatic test pump.

The general CVCS parameters are given in Table 9.2-1. The location numbers on the CVCS Piping and Instrumentation Diagram serve to indicate process flow reference points in the system.

Table 9.2-1 also includes a tabulation of the CVCS process flow data for the three modes of purification loop operation and six modes of makeup system operation using these numbers as reference points. Basically, a letdown flow of 40 gpm is normal purification operation, a letdown flow of 84 gpm is intermediate purification operation and a letdown flow of 128 gpm is maximum purification operation. Typical operating conditions are given for the various makeup system operating modes.

Volume Control

The CVCS automatically controls the volume of water in the reactor coolant system using a signal from the level instrumentation located on the pressurizer. The system reduces the amount of fluid that must be transferred between the coolant system and the CVCS during power changes by employing a programmed pressurizer level setpoint which varies with reactor power level. The programmed level setpoint varies linearly with the average reactor coolant temperature. This linear relationship is shown in Figure 4.3–9. The control system compares the programmed level setpoint with the measured pressurizer water level. The resulting error signal is used to control the operation of the charging pumps and the letdown valves, as described below.

The pressurizer level control program regulates the letdown flow by adjusting the letdown control valve so that the reactor coolant pump controlled bleed off plus the letdown flow matches the input from the operating charging pump. When the equilibrium is disturbed by a power change or for any other reason, a decrease in level will start the available standby charging pumps to restore level, and an increase in level will increase the letdown flow rate and initiate a backup signal to stop the operating standby charging pumps. This relationship is shown in Figure 4.3–10.

The volume control tank coolant level is controlled manually or automatically. When the level in the tank reaches the high level setpoint, the letdown flow is automatically diverted to the radioactive waste processing system. When the level in the tank reaches the low level setpoint, makeup water borated to the existing concentration of the reactor coolant system can be automatically supplied. When the level in the tank reaches the low-low level setpoint the charging pump suction is transferred to the refueling water storage tank.

The volume control tank can be vented to the radioactive waste processing system. The volume control tank is designed to handle all gases that come out of solution when the letdown flow is sprayed into the volume control tank.

Chemical Control

Chemistry control of the reactor coolant consists of operational control of oxygen concentration by maintaining excess hydrogen concentration in the reactor coolant. A chemical addition tank and pump are used to inject hydrazine into the suction side of the charging pumps for subsequent injection into the reactor coolant system. The hydrogen concentration in the reactor coolant system is controlled by the hydrogen overpressure in the volume control tank. The lithium that is produced in the reactor coolant by the reaction B^{10} (n, α) Li^{7} is used as the pH control agent. The production rate of lithium from this reaction is approximately 100 ppb per day at the beginning of core life, and decreases with core lifetime in proportion to boric acid concentration. One

purification ion exchanger is used intermittently to control the lithium concentration. Lithium is also added to the RCS late in core life to maintain pH.

Purification is accomplished by a combination of ion exchange and filtration. Using one purification ion exchanger intermittently, the reactor coolant lithium concentration is controlled within specification limits. The control of other impurities such as chlorides and fission products is accomplished by the continuous operation of the second CVCS purification ion exchanger. This second ion exchanger has been converted to the lithium form and does not remove lithium. These ion exchangers remove soluble nuclides by the ion exchange mechanism and insoluble particles by impingement of these particles on the surface of the resin beds.

Cartridge type filters are located upstream and downstream of the ion exchangers to remove insoluble particles. These filters are removed from service when the pressure drop across the filters, becomes excessive, or when the radiation dose exceeds a predetermined level. A strainer is located downstream of the ion exchangers to insure against the gross release of resin in the unlikely event of an ion exchanger retention element failure.

The volume control tank is provided with a vent to the waste processing system to allow the venting of hydrogen, nitrogen and fission gases.

Instead of controlling the buildup of hydrogen in the waste gases, oxygen is minimized in the gases vented to the waste processing system. All systems and components containing hydrogen are maintained under slightly positive pressure, with nitrogen if required, to prevent the leakage of air into these systems and to preclude the formation of a potentially dangerous mixture of H_2-O_2.

In the event that oxygen concentration exceeds safe limits, nitrogen will be introduced into the waste gas system to lower the concentration of oxygen.

The chemical and volume control system is designed to prevent the reactor coolant fission and corrosion product activities from exceeding the design basis values given in Table 11.A-1 when operating with 1 percent failed fuel.

Reactivity Control

The boron concentration of the reactor coolant is controlled by the makeup portion of the CVCS to:

a. Optimize the position of the control rods;

b. Compensate for reactivity changes caused by variations in the temperature of the coolant, core burnup, and xenon concentration variations;

c. Provide a margin of shutdown for maintenance, refueling or emergencies.

The major components of the makeup system include a batching tank for preparing boric acid solution, two tanks for storing the solution, and two pumps for transferring boric acid solution.

Normally, the CVCS adjusts the boron concentration of the coolant by the feed and bleed operation. To change concentration, the makeup system supplies either demineralized water, concentrated boric acid or any intermediate blend to the volume control tank or charging pump suction. During this time, the letdown stream is diverted to the radioactive waste processing system. Toward the end of a core cycle, the deborating ion exchanger is used to reduce the boron concentration. This avoids the generation of an excessive quantity of waste from feed and bleed operation.

The CVCS can add boron to the reactor coolant at a sufficient rate to override the maximum increase in reactivity due to a cooldown and the decay of xenon in the reactor.

9.2.2.2 Components

Components are designed, manufactured, tested and inspected according to the applicable codes. The following code classifications apply to the CVCS. Components not listed are classified as noncode items.

Component	Design Code	Code Effective Date
Regenerative Heat Exchanger Tube Side (Primary)	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Regenerative Heat Exchanger Shell Side (Secondary)	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Letdown Heat Exchanger Tube Side (Primary)	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Letdown Heat Exchanger Shell Side (Secondary)	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Ion Exchangers	ASME III, Class C	1968 Edition through Winter of 1969 Addenda
Charging Pumps	ASME III, Class 2	1974 Edition, Summer 1975 Addenda
Boric Acid Makeup Pumps	P&V, Class II	November 1968 Draft
Volume Control Tank	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Boric Acid Storage Tanks	ASME III, Class C	1968 Edition through Summer of 1969 Addenda
Strainers	B31.7, Class II & III	1969 Edition
Pulsation Dampeners (Discharge)	ASME III, Class 2	1971 Edition with Addendum through Winter 1973

The regenerative heat exchanger has been classified as an ASME Section III, Class C vessel for the following reasons:

1. It is possible to isolate the regenerative heat exchanger on both the shell and tube sides with isolation valves which are remotely operable from the control room;

2. Should it ever become necessary to completely isolate the regenerative heat exchanger an alternate charging path exists through the high-pressure safety injection header;

3. Additional Quality Control and Fatigue Analysis requirements have been placed on the regenerative heat exchanger beyond those normally required of an ASME Section III, Class C vessel.

Regenerative Heat Exchanger

The regenerative heat exchanger transfers heat from the letdown stream to the charging stream. Materials of construction are primarily austenitic stainless steel. The characteristics of the regenerative heat exchanger are given in Table 9.2-3.

Letdown Control Valves

Normally, one of the letdown control valves (CH110P, CH110Q) regulates letdown flow, as required by the pressurizer level control system. The valves reduce the pressure of the letdown fluid to about 500 psig. The letdown flow is nominally 40 gpm, for coolant purification, but will vary as the pressurizer water level changes. The valves are pneumatically operated and fail closed. All parts in contact with reactor coolant are of austenitic stainless steel. The valve characteristics are given in Table 9.2-4.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown flow to a temperature compatible with the purification ion exchanger resins. Reactor building closed cooling water (RBCCW) is the cooling medium on the shell side of the letdown heat exchanger. Tube side materials of construction are austenitic stainless steel; shell side materials of construction are carbon steel. The characteristics of the letdown heat exchanger are given in Table 9.2-5.

Ion Exchangers

Two purification ion exchangers purify the reactor coolant by removing corrosion and fission products. Each unit is designed to handle the maximum letdown flow of 128 gpm. Maximum flow during shutdown purification operation is 300 gpm. The vessels and resin retention element are of austenitic stainless steel construction.

The normal method of adjusting boron concentration is by feed and bleed. Toward the end of a core cycle, the quantities of waste produced due to feed and bleed operations become excessive,

and the deborating ion exchanger is used to reduce the boron concentration. The anion resin that is initially in the hydroxyl form, is converted to a borate form as boron is removed.

The characteristics of the ion exchangers are given in Table 9.2-6.

Letdown Filters

The letdown filters remove insoluble materials from the reactor coolant. One filter, the letdown pre-filter, is located in the letdown line upstream of the ion exchangers. The other filter, the letdown post-filter, is located downstream of the ion exchangers. Each filter can accommodate maximum letdown flow of 128 gpm. Maximum flow during shutdown purification operation is 300 gpm. The filter housings are austenitic stainless steel. The characteristics of filters are given in Table 9.2-7.

Volume Control Tank

The volume control tank is used to accumulate letdown flow from the reactor coolant system, to maintain the desired hydrogen concentration in the reactor coolant, and to provide a reservoir of reactor coolant for the charging pumps. A vent to the radioactive waste processing system permits removal of hydrogen, nitrogen and gaseous fission products released from solution in the volume control tank. The tank is of austenitic stainless steel construction and is provided with overpressure protection. Level controls divert coolant to the radioactive waste processing system on high level or operate the makeup system on low level. The characteristics of the tank are given in Table 9.2-8.

Charging Pumps

The charging pumps return the purification flow to the reactor coolant system during plant steady state operations. The pumps are horizontal, positive displacement type with a leakage collection system. Each charging pump has a capacity of 44 gpm. On a safety injection actuation signal (SIAS), the available pumps are started (normally, one of the pumps will already be running) and pump concentrated boric acid into the reactor coolant system. The pressure containing portions of the pump are austenitic stainless steel with internal materials selected for compatibility with boric acid. The characteristics of the pumps are given in Table 9.2-9. Pulsation dampeners are located on the suction and discharge sides of each pump to minimize vibration induced failures.

Boric Acid Storage Tanks

The operable boric acid storage tank(s), in conjunction with the RWST, store enough concentrated boric acid solution to bring the reactor to a cold shutdown condition at any time during the core lifetime. The solution is prepared in the boric acid batching tank and flows through the boric acid batching strainer before entering the storage tanks. With boric acid concentrations in the BASTs less than or equal to 3.5 weight percent, tank heaters are not required to prevent boron precipitation at Auxiliary Building ambient temperatures. Sampling connections are used to verify that the proper concentration is maintained. The tanks are constructed of stainless steel.

The characteristics of the batching tank, boric acid batching strainer and the boric acid tanks are given in Table 9.2-10.

Boric Acid Pumps

The boric acid pumps take suction from the concentrated boric acid tanks and provide boric acid for blended flow to the volume control tank or direct transfer to the charging pump suction header. On initiation of SIAS, these pumps line up with the charging pumps to permit direct transfer of concentrated boric acid into the reactor coolant system. Each boric acid pump is capable of supplying boric acid to three operating charging pumps. All wetted parts, except seals, are stainless steel. The pump and strainer characteristics are given in Table 9.2-11.

9.2.3 SYSTEM OPERATION

9.2.3.1 Startup

During RCS fill, the CVCS back pressure control valve is normally used to control RCS pressure. RCS heatup commences after a steam bubble is established in the pressurizer. During RCS heatup, CVCS (charging pumps and letdown) are used to maintain pressurizer level and RCS inventory; pressurizer heaters and spray are used to maintain RCS pressure. The level controls in the volume control tank automatically divert the letdown flow to the radioactive waste processing system (RWS).

When the reactor is shut down, the volume control tank is vented to the radioactive waste processing system and then pressurized with nitrogen. Prior to reactor startup, a hydrogen blanket is established in the tank. Any oxygen in the reactor coolant is normally removed by radiolytic recombination with excess hydrogen in the coolant.

Throughout startup, the letdown filters are in service to reduce the activity of wastes entering the RWS. The purification ion exchanger is also normally in service. Within limitations placed on the shutdown margin, the boron concentration in the reactor coolant system may be reduced during heatup. The operator may inject a predetermined amount of demineralized makeup water by operating the makeup system in the “Dilute” mode. The concentration of boron in the reactor coolant is verified by chemical analysis.

9.2.3.2 Normal Operation

Normal operation includes both the hot standby condition as well as normal plant operation.

During normal operation:

a. Level instrumentation in the pressurizer automatically controls the volume of water in the reactor system by adjusting the letdown flow.

b. The water level in the volume control tank is maintained manually or automatically, in accordance with plant procedures.

c. The hydrogen concentration and pH of the coolant are controlled in accordance with plant procedures.

d. Changes in reactivity may be compensated for by adjusting the concentration of boron in the reactor coolant system by the feed and bleed operation. Late in core life, the boron in the reactor coolant is maintained at a very low concentration. At this time the feed and bleed operation would generate excessive quantities of wastes so further reduction is accomplished by use of the deborating ion exchanger. During steady state operation, precise reactivity control is accomplished by utilizing the feed and bleed method. Over core life, an increasing quantity of demineralized water is metered to the suction of the charging pumps in order to minimize power level fluctuations.

e. The makeup system may be operated in four modes:

1. In the “Dilute” mode, a quantity of demineralized makeup water is selected and introduced into the charging pump suction or the volume control tank. When the integrating flowmeter indicates that the selected quantity of makeup water has been added, the makeup flow is automatically stopped.

2. In the “Borate” mode, a quantity of concentrated boric acid is selected and introduced into the charging pump suction or the volume control tank, as described above.

3. In the “Manual Blend” mode, the flows of the demineralized water and concentrated boric acid can be set for any blend concentration between demineralized makeup water and concentrated boric acid. This mode is primarily used to supply makeup to the volume control tank and refueling water storage tank.

4. In the “Automatic” mode, the flow rates of the demineralized water and concentrated boric acid are set to match the concentration present in the reactor coolant system at that time in core life. The solution is automatically blended and introduced into the volume control tank according to signals received from the volume control tank level program.

The makeup system is normally set for the dilute mode of operation. This allows the operator to closely monitor all additions to the RCS.

f. The letdown flow is normally routed through one of the purification ion exchangers to reduce coolant activity.

9.2.3.3 Shutdown

Plant shutdown is accomplished by a series of operations which bring the reactor plant from a hot standby condition at normal operating pressure and zero power temperature to a cold shutdown condition.

Before the plant is cooled down, the RCS is degassed to reduce the concentration of fission gases and hydrogen in the reactor coolant system. The operator may also increase the letdown flow rate to accelerate the degasification, ion exchange, and filtration processes. All CEAs are inserted and CEDMs deenergized to begin the mode change to the shutdown condition to prevent inadvertent CEA withdrawal from subcritical conditions. During cooldown, boron is added, as needed, to the reactor coolant system in accordance with plant procedures to maintain adequate shutdown margin.

During the cooldown, the charging pumps, letdown control valves and letdown backpressure control valves are used to adjust and maintain the pressurizer water level. The charging system is aligned to the Boric Acid Storage Tank(s) or RWST for RCS makeup during cooldown until the required shutdown margin is verified. The charging suction is then switched to the Refueling Water Storage Tank. These steps ensure the proper shutdown margin is maintained. A portion of the charging flow is used as an auxiliary spray to cool the pressurizer when the pressure of the reactor coolant system is below that required to operate the reactor coolant pumps.

9.2.3.4 Safety Injection Operation (Emergency Operation)

Under emergency conditions, the charging pumps function to inject concentrated boric acid into the reactor coolant system. Either the pressurizer level control or the SIAS will automatically start the available charging pumps. Normally one of the pumps will already be running. The SIAS signal will also function to transfer the charging pump suction from the volume control tank to the discharge of the boric acid pumps. If the boric acid pumps are not operable, boric acid flows by gravity from the concentrated boric acid tanks to the charging pump suction header. If the charging line inside the reactor containment building is inoperative, this line may be isolated outside the reactor containment building, and the concentrated boric acid solution may be injected by the charging pumps through the high pressure safety injection header.

Portions of the charging system may be employed to provide long term cooling and boron precipitation control in the event of a LOCA by simultaneous hot and cold leg injection. In the event that the preferred LPSI hot leg injection method is unavailable, HPSI pump P-41A is aligned to inject to the pressurizer auxiliary spray line, and thus the hot leg, through piping in the charging system.

9.2.4 AVAILABILITY AND RELIABILITY

9.2.4.1 Special Features

To assure reliability, the design of the CVCS incorporates component redundancy as well as operational redundancy. This is provided as follows:

Component	Redundancy
Purification Ion Exchangers	Parallel Standby Unit
Charging Pumps	Two Parallel Standby Units
Letdown Flow Control Valves	Parallel Standby Valve
Boric Acid Pumps and Tanks	Parallel Standby Unit
Backpressure Control Valves	Parallel Standby Valve

In addition to the component redundancy it is possible to operate the CVCS in a manner such that some components are bypassed. While the normal charging path is through the regenerative heat exchanger, it is also possible to charge through the high pressure safety injection header. It is possible to transfer boric acid to the charging pump suction header by bypassing the volume control tank, or by bypassing the makeup flow controls and the volume control tank. On SIAS two separate paths to the charging pump suction header are lined up for boric acid transfer (through boric acid pumps and through the gravity feed line). If the letdown temperature exceeds 140°F, the flow will automatically bypass the ion exchangers and flow to the process radiation monitor and boronometer is stopped.

The charging pumps and boric acid pumps are powered by an emergency bus if normal power is lost. The boric acid pumps and the motor-operated gravity feed boric acid valves are powered from different buses. Separation is provided between the power and control circuits for the redundant boration paths.

Standby features are provided so that at least one charging pump is running after SIAS. Separate power supplies and control circuits are provided among the charging pumps, boric acid pumps and the gravity feed boric acid motor-operated valves. Since both boric acid pumps are powered from the same power supply, the redundancy for boration is achieved by the design of the boric acid gravity feed piping arrangement and gravity feed motor operated valve as the redundant path for boration.

Heating of the boric acid solution in the Boric Acid Storage Tanks and the piping between the BASTs and the charging pumps is not required since the boric acid concentration is limited to 3.5 wt. percent and will remain in solution under normal ambient area temperatures. Heat tracing will be maintained on the Boric Acid Batching Tank, the batching strainers, and the piping upstream of the BASTs because of the possibility of filling these sections with cold water.

NOTE: The boric acid heat tracing circuits have been isolated awaiting retirement.

If the boric acid pumps are not available, boric acid from the concentrated boric acid tanks may be gravity fed into the charging pump suction. If the charging line inside the reactor containment building is inoperative, the charging pump discharge may be routed via the safety injection system to inject concentrated boric acid into the reactor coolant system.

The failure position of the backpressure control valves is the closed position. If the valve fails to open, relief valve CH-354 will protect the low pressure portion of the letdown line under all circumstances. However, if the CVCS is in the maximum purification mode there is a possibility that flashing downstream of the drag valves could occur. However, the maximum purification mode of operation requires maximum component cooling water flow to the letdown heat exchanger; thus, the letdown heat exchanger would prevent any two-phase flow condition downstream of the heat exchanger. In addition, if the temperature out of the letdown heat exchanger exceeds 140°F, the letdown flow would bypass the ion exchangers. Also, should the valves remain open, the low pressure alarm on the backpressure controller would alert the operator that the valve is no longer functional. Note that in none of these cases would the VCT be subject to over-pressure. In any case, the VCT has a relief valve with a relieving capacity of 250 gpm which is set to protect the VCT from exceeding its design pressure. Also, a VCT high pressure alarm provides notification that the VCT is above its normal operating pressure but below the VCT relief valve setting. The volume control tank also has a high temperature alarm above 130°F. Therefore, there is adequate warning and protection to prevent any pressure buildup in the low-pressure portion of the system.

Should the backpressure control valves fail closed, the intermediate pressure relief valve (CH-345) would preclude any overpressure condition from existing. The operator would be alerted to this condition by a high pressure alarm on the pressure controller (PIC-201). There would not be any flashing flow because of the higher set pressure of this relief valve. Since this backpressure control valve fails closed, there is no overpressurization problem downstream of this valve and the relief valve relieves the pressure upstream of this valve.

TABLE 9.2-1 CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS

Normal Letdown and Purification Flow, gpm	40
Normal Charging Flow, gpm	44
Reactor Coolant Pump Controlled Bleedoff (4 pumps), gpm	4
Normal Letdown Temperature at Loop, °F	550
Normal Charging Temperature at Loop, °F	395
Ion Exchanger Operating Temperature, °F	120

Chemical and Volume Control System Process Flow Data

CVCS Makeup System Operation - Manual Mode

CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	175	175	10	10	0	10	50	50	60	0
Press., psig	10	90	90	85	15	22	155	20	20	10
Temp., °F	140	140	140	140	140	140	60	60	140	70
CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	0	0	165	155	10	10	0	0	0	
Press., psig	90	15	90	90	90	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	

CVCS Makeup System Operation - Emergency Boration (SIAS)

CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	142	142	132	0	0	0	0	0	0	0
Press., psig	10	105	105	105	105	15	165	15	15	10
Temp., °F	140	140	140	140	140	140	60	60	140	140

TABLE 9.2-1 CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS (CONTINUED)

CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	132	132	10	0	10	10	0	0	0	
Press., psig	103	100	105	105	105	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	
CVCS Makeup System Operation - Shutdown Boration										
CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	180	180	15	0	0	0	0	0	0	0
Press., psig	10	90	90	90	90	15	165	15	15	10
Temp., °F	140	140	140	140	140	140	60	60	140	70
CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	15	15	165	155	10	10	0	0	0	
Press., psig	90	90	90	90	90	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	
CVCS Makeup System Operation - Emergency Boration (SIAS) Via Gravity Feed										
CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	0	0	0	0	0	0	0	0	0	0
Press., psig	10	10	10	10	5	15	165	15	15	10
Temp., °F	140	140	140	140	140	140	60	60	140	70
CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	0	0	0	0	0	0	66	66	132	
Press., psig	10	5	10	10	10	10	10	10	5	
Temp., °F	140	140	140	140	140	140	140	140	140	

TABLE 9.2-1 CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS (CONTINUED)

NOTE:
(1) See Figure 9.2–2, Bechtel CVCS P&ID No. 25203-26017, for location numbers.
(2) The data shown for the various modes of operation is typical.
(3) The pressure in the isolated piping of the CVCS makeup system will normally be 0 psig, but may range as high as 140 psig before the thermal relief valves lift.

Chemical and Volume Control System Process Flow Data

CVCS Normal Purification Operation (One Charging Pump in Operation)

CVCS Location:	1	2	3	4	5	6	7	8	9	10
Flow, gpm	40	40	40	40	40	39	1	39	39	40
Press., psig	2205	2195	465	460	27	26	27	25	24	23
Temp., °F	550	262	262	120	120	120	120	120	120	120

CVCS Location:	10a	10b	10c	11	12	13	14 (a,b,c,d)	14e	14f	14g
Flow, gpm	40	40	44	44	44	44	1	0	4	4
Press., psig	22	20	20	20	2310	2302	100	100	100	21
Temp., °F	120	120	120	120	120	395	120	120	120	120

CVCS Maximum Purification Operation (Two Charging Pumps in Operation)

CVCS Location:	1	2	3	4	5	6	7	8	9	10
Flow, gpm	84	84	84	84	84	83	1	83	83	84
Press., psig	2205	2162	481	460	50	46	50	45	41	37
Temp., °F	550	320	320	120	120	120	120	120	120	120

CVCS Location:	10a	10b	10c	11	12	13	14 (a,b,c,d)	14e	14f	14g

TABLE 9.2-1 CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS (CONTINUED)

Flow, gpm	84	84	88	88	88	88	1	0	4	4
Press., psig	33	20	20	20	2310	2278	100	100	100	21
Temp., °F	120	120	120	120	120	350	120	120	120	120
CVCS Maximum Purification Operation (Three Charging Pumps in Operation)										
CVCS Location:	1	2	3	4	5	6	7	8	9	10
Flow, gpm	128	128	128	128	128	127	1	127	127	128
Press., psig	2205	2105	510	460	92	82	92	81	71	61
Temp., °F	550	675	375	120	120	120	120	120	120	120
CVCS Location:	10a	10b	10c	11	12	13	14 (a,b,c,d)	14e	14f	14g
Flow, gpm	128	128	132	132	132	132	1	0	4	4
Press., psig	51	20	20	20	2310	2240	100	100	100	21
Temp., °F	120	120	120	120	120	120	120	120	120	120

NOTE:
(1) See Figure 9.2–2 Bechtel CVCS P&ID No. 25203-26017, for location numbers.
(2) The pressure drop across the purification filter and ion exchanger will vary with loading. The pressure drops shown are typical.
(3) The pressure in the volume control tank will vary and this will affect the pressures at locations 5 through 11 proportionally.

CVCS Makeup System Operation - Automatic Mode										
CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	175	175	10	10	0	10	140	140	150	0
Press., psig	10	90	90	85	15	22	140	20	22	10
Temp., °F	140	140	140	140	140	140	60	60	70	70

TABLE 9.2-1 CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS (CONTINUED)

CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	0	0	165	155	10	10	0	0	0	
Press., psig	90	20	90	90	90	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	
CVCS Makeup System Operation - Borate Mode										
CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	180	180	15	15	0	15	0	0	15	0
Press., psig	10	90	90	75	15	22	165	20	22	10
Temp., °F	140	140	140	140	140	140	60	60	140	70
CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	0	0	165	155	10	10	0	0	0	
Press., psig	90	20	90	90	90	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	
CVCS Makeup System Operation - Dilute Mode										
CVCS Location:	15	16	17	18	19	20	21	22	23	24
Flow, gpm	160	160	0	0	0	0	130	130	130	0
Press., psig	10	90	90	15	15	22	140	22	22	10
Temp., °F	140	140	140	140	140	140	60	60	140	70
CVCS Location:	25	26	27	28	29	30	31	32	33	
Flow, gpm	0	0	160	150	10	10	0	0	0	
Press., psig	15	20	90	90	90	20	10	10	10	
Temp., °F	140	140	140	140	140	140	140	140	140	

NOTE:

(1) See Figure 9.2–2, Bechtel CVCS P&ID No. 25203-26017, for location numbers.

(2) The data shown for the various modes of operation is typical

(3) The pressure in the isolated piping of the CVCS makeup system will normally be 0 psig but may range as high as 140 psig before the thermal relief valves lift.

TABLE 9.2-2 REACTOR COOLANT AND DEMINERALIZED WATER CHEMISTRY

PARAMETER	REACTOR COOLANT	DEMINERALIZED WATER
Suspended Solids, ppm maximum	0.35 prior to reactor startup	-
pH at 25°C	Determined by the concentration of boric acid and lithium present. Consistent with the Primary Chemistry Control Program.(4)	-
Chloride, ppm Cl^-, maximum	0.15	0.05
Fluoride, ppm F^-, maximum	0.10	-
Sodium, ppm Na^+, maximum	-	0.01
Hydrogen as H_2, cc(STP)/Kg H_20	25-50	-
Dissolved O_2, ppm maximum	0.1 (1) (2) (3)	0.1
Lithium as Li^7, ppm	Consistent with the Primary Chemistry Control Program. (4)	-
Boron, ppm	0-2620 (5)	-
Conductivity, μS/Cm at 25°C	Relative to lithium and boron concentration	2.0

TABLE 9.2-2 REACTOR COOLANT AND DEMINERALIZED WATER CHEMISTRY (CONTINUED)

PARAMETER	REACTOR COOLANT	DEMINERALIZED WATER

NOTES:

(1) The temperature at which the Oxygen limit applies is > 250°F.

(2) The at power operation residual Oxygen concentration control value is ≤ 0.005 ppm.

(3) During plant startup, Hydrazine may be used to control dissolved Oxygen concentration at ≤0.1 ppm.

(4) During power operation lithium is coordinated with boron to maintain a $pH_{(t)}$ of ≥ 7.0, but ≤ 7.4, consistent with the Primary Chemistry Control Program. Lithium is added to the RCS during plant startup, but prior to reactor criticality, and is in specification per the Primary Chemistry Control Program within 72 hours after criticality. Lithium may be removed from the reactor coolant immediately before, or during, shutdown periods to aid in the cleanup of corrosion products. By evaluation, a maximum lithium concentration of 4.5 ppm is permissible with a target lithium concentration of 4.3 ppm for 100% power operations.

(5) RCS boron concentration is maintained as necessary to ensure core reactivity or shutdown margin requirements are met. Although the RCS and related auxiliary systems containing reactor coolant are designed for a maximum concentration of 2620 ppm boron, it should be noted the design basis for the TSP baskets in the containment sump assumes the RCS, SITs, and RWST are at a maximum boron concentration of 2400 ppm.

TABLE 9.2-3 REGENERATIVE HEAT EXCHANGER

Design Parameters

Quantity	1
Type	Shell and Tube, Vertical
Code	ASME III, Class C (1968 Edition through Summer of 1969 Addenda)
Tube Side (Letdown)	
Fluid	Reactor Coolant, 1 wt % Boric Acid, Maximum
Design Pressure, psig	2485
Design Temperature °F	650
Materials	Stainless Steel, Type 304
Pressure Loss, psi	99 (at 128 gpm)
Normal Flow, gpm	40
Design Flow, gpm	128
Shell Side (Charging)	
Fluid	Reactor Coolant, 12 wt % Boric Acid, Maximum
Design Pressure, psig	3025
Design Temperature °F	650
Materials	Stainless Steel, Type 304
Pressure Loss at 44 gpm, psi	5
Normal Flow, gpm	44
Design Flow, gpm	132

TABLE 9.2-3 REGENERATIVE HEAT EXCHANGER (CONTINUED)

Operating Parameters

Tube Side (Letdown)	Normal	Maximum Letdown / Maximum Charging	Maximum Letdown / Maximum Charging	Maximum Letdown / Minimum Charging
Flow - gpm at 120°F	40	30	128	128
Inlet Temperature °F	550	550	550	550
Outlet Temperature °F	262	165	375	450

Shell Side (Charging)	Normal	Maximum Letdown / Maximum Charging	Maximum Letdown / Maximum Charging	Maximum Letdown / Minimum Charging
Flow - gpm at 120°F	44	132	132	44
Inlet Temperature °F	120	120	120	120
Outlet Temperature °F	395	212	310	452

TABLE 9.2-4 LETDOWN CONTROL VALVES

Quantity	2
Manufacturer	Control Component Incorp.
Design Pressure, psig	2485
Design Temperature, °F	550
Maximum Flow (each), gpm	128
Minimum Flow (each), gpm	30

TABLE 9.2-5 LETDOWN HEAT EXCHANGER

Design Parameters

Quantity	1
Type	Shell and Tube, Horizontal
Code	ASME III, Class C (1968 Edition through Summer of 1969 Addenda)
Manufacturer	Whitlock Manufacturing Co.
Tube Side (Letdown)	
Fluid	Reactor Coolant, 1 weight percent Boric Acid, Maximum
Design Pressure, psig	650
Design Temperature °F	550
Pressure Loss at 128 gpm	21
Materials, psi	Stainless Steel, Type 304
Normal Flow, gpm	40
Maximum Flow, gpm	128 [1]
	300 [2]
Shell Side (Cooling Water)	
Fluid	RBCCW
Design Pressure, psig	150
Design Temperature °F	250
Materials	Carbon Steel
Normal Flow, gpm	190
Design Flow, gpm	1200

1. During letdown system operation.
2. During shutdown purification operation.

TABLE 9.2-5 LETDOWN HEAT EXCHANGER (CONTINUED)

Operating Parameters

Tube Side (Letdown)	Normal	Maximum Letdown / Maximum Charging	Maximum Letdown / Maximum Charging	Maximum Letdown Minimum Charging
Flow - gpm at 120°F	40	30	128	128
Inlet Temperature °F	263	165	375	450
Outlet Temperature °F	120	120	120	127

Shell Side (Cooling Water)	Normal	Maximum Letdown / Maximum Charging	Maximum Letdown / Maximum Charging	Maximum Letdown Minimum Charging
Flow - gpm at 120°F	190	21	1200	1200
Inlet Temperature °F	100	65	100	100
Outlet Temperature °F	130	130	127	135

TABLE 9.2-6 ION EXCHANGERS

Quantity	3
Type	Flushable
Manufacturer	Air Preheater Co.
Design Pressure, psig	200
Design Temperature,°F	250
Normal Operating Pressure, psig	60
Normal Operating Temperature, °F	120
Resin Volume, each (total), ft^3	36.2
Resin Volume, each (useful), ft^3	32.0
Normal Flow, gpm	40
Maximum Flow, gpm	128 [1]
	300 [2]
Code for Vessel	ASME III, Class C (1968 Edition thru Winter of 1969 Addendum)
Retention Screen, U. S. Mesh	80
Material	Stainless Steel, Type 304
Fluid, Boric Acid, wt %	1 Maximum

1. During letdown system operation.
2. During shutdown purification operation.

TABLE 9.2-7 LETDOWN FILTERS

Quantity 2

Type Elements Disposable Cartridge

Data

Manufacturer	Weight % Efficiency	Normal Operating Temperature, °F	Design Pressure, psig	Maximum Allowable Pressure Loss, Clean, psi	Maximum Allowable Pressure Loss, Loaded, psi	Filter Cartridge Rating (Micron)	Design Temperature, °F	Maximum Flow, gpm	Normal Flow, gpm
Filterite	80 (minimum)	120	200	5 at 132 gpm	40 at 132 gpm	3 Nominal	250	132	40
Pall Cartridge	99 (minimum)			3.44 at 132 gpm	75 at 132 gpm	6, 2, 0.45 Absolute	250	132 [a] 300 [b]	40

a. During letdown system operation.
b. During shutdown purification operation.

Code for Vessel ASME III, Class C (1968 Edition through Summer of 1969 Addendum).

Material Austenitic Stainless Steel.

Fluid, Boric Acid, wt % 1 Maximum.

TABLE 9.2-8 VOLUME CONTROL TANK

Quantity	1
Type	Vertical, Cylindrical
Manufacturer	Air Preheater Co.
Design Pressure, Internal, psig	75
Design Pressure, External, psig	15
Design Temperature, °F	250
Normal Operating Pressure, psig	15 - 30
Normal Operating Temperature, °F	120
Normal Spray Flow, gpm	40
Blanket Gas (during plant operation)	Hydrogen
Code	ASME III, Class C (1968 Edition through Summer of 1969 Addenda)
Fluid, Boric Acid, wt %	6.25 Maximum
Material	Austenitic Stainless Steel

TABLE 9.2-9 CHARGING PUMPS

Quantity	3
Type	Positive Displacement
Manufacturer	Gaulin Corp. n
Design Pressure, psig	3010
Design Temperature, °F	250
Capacity, gpm	44
Normal Discharge Pressure, psig	2735
Normal Suction Pressure, psig	20
Normal Temperature, °F	120
Maximum Discharge Pressure (Short Term), psig	3010
Minimum Available Net Positive Inlet Pressure, psia	9.0
Net Positive Inlet Pressure Required, psia	8.25 at 44 gpm
Driver Rating, hp	100
Materials in Contact with Pumped Fluid	Austenitic Stainless Steel
Fluid	12 wt % Boric Acid, Maximum

TABLE 9.2-10 CONCENTRATED BORIC ACID PREPARATION AND STORAGE

Boric Acid Storage Tanks

Quantity	2
Manufacturer	Air Preheater Co.
Internal Volume, each, ft^3	870
Design Pressure, psig	15
Design Temperature, °F	250
Normal Operating Temperature, °F	60-100
Fluid, wt % Boric Acid	
Minimum	2.5
Maximum	3.5
Material	Austenitic Stainless Steel
Code	ASME III, Class C (1968 Edition through Summer of 1969 Addenda)

Boric Acid Batching Strainer

Quantity	1
Type	“Y” type, in line
Manufacturer	Mueller Steam Specialty Co.
Design Pressure, psig	150
Normal Operating Pressure, psig	5
Design Temperature, °F	200
Screen Size, U.S. Mesh	10
Design Flow, gpm	130
Materials	Austenitic Stainless Steel
Fluid, Boric Acid, wt %	3.5
Maximum Pressure Loss, psi	USAS B31.7, Class II & III
Code (1969 Edition)	10

TABLE 9.2-10 CONCENTRATED BORIC ACID PREPARATION AND STORAGE (CONTINUED)

Boric Acid Batching Tank

Quantity	1
Manufacturer	Air Preheater Company
Useful Volume, gallons	635
Design Pressure	Atmospheric
Design Temperature, °F	250
Normal Operating Temperature, °F	85-100
Type Heater	Electrical Immersion
Heater Capacity, Min., kW	45
Fluid, Boric Acid, wt %	12
Material	Austenitic Stainless Steel
Code	Non code

TABLE 9.2-11 BORIC ACID PUMPS AND STRAINERS

PUMPS

Quantity	2
Manufacturer	Goulds Pump Co.
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	250
Design Head, feet	231
Design Flow, gpm	143
Normal Operating Temperature, °F	60-100
NPSH Available (Minimum), feet	20
NPSH Required, feet	8
Horsepower, hp	2.5
Fluid, Boric Acid, wt % maximum	3.5
Material in Contact with Liquid Code (November 1968 Draft)	Austenitic Stainless Steel Pump and Valve Code, Class II

STRAINER

Quantity	1
Manufacturer	Mueller Steam Specialty Co.
Type	“Y” type, in line
Screen Size U.S. Mesh	200
Normal Operating Pressure, psig	80
Design Pressure, psig	150
Pressure Loss, Clean, psi	1 at 30 gpm
Design Temperature, °F	200
Normal Operating Temperature, °F	85-100
Design Flow, gpm	150
Materials	Austenitic Stainless Steel
Fluid, Boric, Acid, wt % Code (1969 Edition)	3.5 USAS B31.7, Class II & III

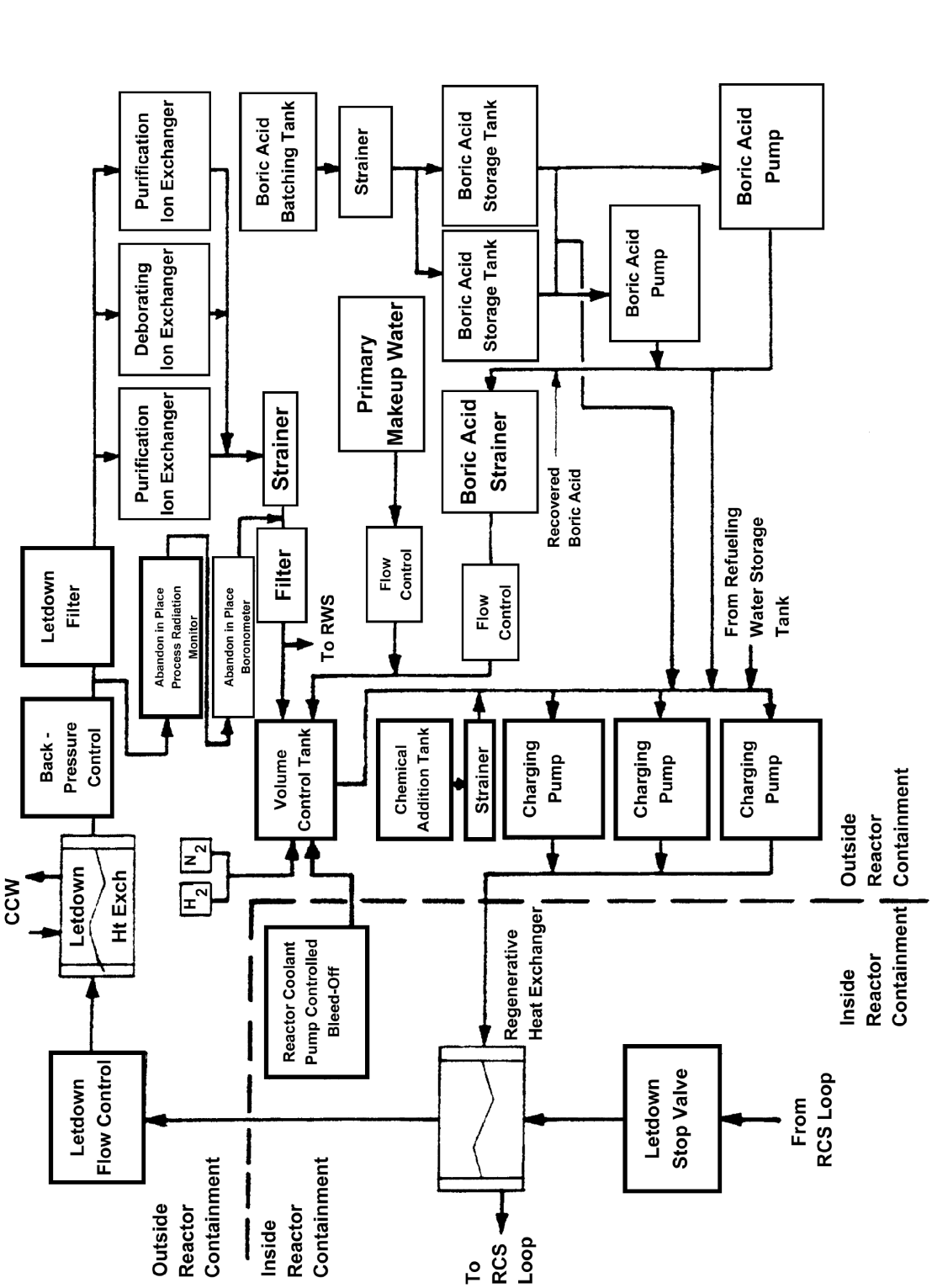


FIGURE 9.2–1 CHEMICAL AND VOLUME CONTROL SYSTEM FLOW SCHEMATIC (NORMAL OPERATION)

FIGURE 9.2–2 P&IDS: CHARGING; DEBORATING; PURIFICATION; BORIC ACID SYSTEMS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.2–2 P&IDS: CHARGING; DEBORATING; PURIFICATION; BORIC ACID SYSTEMS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.2–2 P&IDS: CHARGING; DEBORATING; PURIFICATION; BORIC ACID SYSTEMS (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.3 SHUTDOWN COOLING SYSTEM

9.3.1 DESIGN BASES

9.3.1.1 Functional Requirements

The shutdown cooling system, in conjunction with the main steam and feedwater systems, is designed to reduce the temperature of the reactor coolant in post shutdown periods from normal operating temperature to the refueling temperature. The main steam and feedwater system is utilized in the initial phase of the cooldown. The shutdown cooling system functions to reduce the coolant temperature to the refueling temperature and also maintains this temperature during refueling.

Refueling water transfer from the refueling water storage tank (RWST) to the refueling cavity in the containment building is normally accomplished after the reactor vessel head has been removed by using the high pressure safety injection pumps, low pressure safety injection pumps or the refueling water purification pumps. The high pressure safety injection pumps take suction from the RWST and discharge into the reactor coolant system and then to the refueling cavity via the open reactor vessel. Pumping continues until the refueling cavity is filled. The low pressure pumps can be used to return the refueling water from the refueling cavity to the RWST.

During shutdown or refueling, the shutdown cooling system can be aligned to cool the spent fuel pool system using the low pressure safety injection pumps (see Section 9.5)

The shutdown cooling system heat exchangers are normally used during the recirculation phase by either the containment spray system or the shutdown cooling system, when RCS conditions permit following a LOCA. If permissible RCS conditions exist post-LOCA, the shutdown cooling heat exchangers may be aligned to the shutdown cooling system and shutdown cooling may be initiated for longterm cooling. Otherwise, the shutdown cooling heat exchangers remain aligned to the containment spray system which is used to cool the recirculated water.

The functional performance requirements for the shutdown cooling heat exchangers during normal alignment or during LOCA recirculation mode are provided in Table 9.3-1.

Following a LOCA, when the containment spray system is no longer required, the RCS is filled, and the shutdown cooling system entry conditions have been met, the preferred method by which to provide long term cooling and boron precipitation control is with the shutdown cooling system in its normal plant operation alignment. In the event that the RCS is not filled 8 to 10 hours after the start of the LOCA, then portions of the shutdown cooling system may be aligned for simultaneous hot and cold leg injection to provide long-term cooling and boron precipitation control.

9.3.1.2 Design Criteria

Code selection and materials requirements are primarily related to the emergency operation capability of the system and are presented in Chapter 6. The shutdown cooling system components are designed to meet the design parameters listed in Table 9.3-2.

System components whose design pressure and temperature are less than the reactor coolant system design limits are provided with overpressure protection devices and redundant isolation means. System discharge from over-pressure protection devices is collected in closed systems.

Materials are selected to meet the applicable code material requirements of the codes given in Table 6.3-3. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel.

9.3.2 SYSTEM DESCRIPTION

9.3.2.1 System

The shutdown cooling system is shown schematically in Figure 9.3–1. The system uses portions of other systems, i.e., the reactor coolant system (Chapter 4), safety injection system (Section 6.3) and containment spray system (Section 6.4).

In the shutdown cooling system, reactor coolant is circulated using the low pressure safety injection pumps. The flow path from the pump discharge runs through normally locked closed valves SI-452 and SI-453, through the shutdown cooling heat exchangers, through normally closed valves SI-456 and SI-457, through normally locked closed valve SI-657 to the low pressure safety injection header, and enters the reactor coolant system through the four safety injection header legs. (As required for temperature control, a portion of the pump flow may bypass the heat exchangers, going directly to the low pressure safety injection header by flowing through the shutdown cooling heat exchanger bypass valve SI-306.) The circulating fluid flows through the core and is returned from the reactor coolant system through the shutdown cooling nozzle in the loop number 2 reactor vessel outlet (hot leg) pipe. The coolant is returned to the suction of the low pressure safety injection pumps through normally closed valves SI-651 and SI-652. These valves are interlocked to prevent opening when the reactor coolant system pressure exceeds the design pressure of the shutdown cooling system. Further, if either of these valves is open when reactor coolant system pressure exceeds a set point of 280 psia, an annunciator will alarm. Each valve is independently controlled by separate instrumentation channels. The interlock on these valves is also described in Sections 4.3.8.2.3 and 9.3.4.1.

A pressure equalizing line is installed between the valve body of SI-651 and upstream piping 12"-CCA-10. The line will ensure that over-pressurization of the valve bonnet for SI-651 and pressure locking of the valve does not occur.

Shutdown cooling and total low pressure injection flow are measured by an orifice meter installed in the low pressure safety injection header. Flow is indicated in the control room. The flow element also transmits a signal to a controller.

The cooldown rate is controlled by adjusting flow through the heat exchangers with the throttle valve SI-657 on the discharge of the heat exchangers. Relatively constant total shutdown cooling flow to the core is maintained by adjusting the heat exchanger bypass valve controller FIC-306 and/or the injection valves to compensate for changes in flow through the heat exchangers.

9.3.2.2 Interface With Other Systems

Safety Injection System

The low pressure safety injection pumps recirculate the reactor coolant during the shutdown cooling mode of heat removal. The flow path is described in Section 9.3.2.1.

Reactor Coolant System

Temperature control during refueling and the initial cooldown to refueling temperature of the reactor coolant system is accomplished by recirculating reactor coolant through the shutdown cooling heat exchangers. Reactor coolant is circulated from the hot legs to the cold leg side of the reactor vessel, thus maintaining the normal flow direction in the reactor coolant system during the refueling cycle. During normal operation two closed valves provide isolation from the reactor coolant system. The two isolation valves are also provided with pressure interlocks which preclude them from being opened at a reactor coolant system pressure above the design pressure of the shutdown cooling loop.

Reactor Building Closed Cooling Water System

The RBCCW system provides the heat sink to which the reactor coolant system residual heat is rejected. RBCCW water flows through the shell side of the shutdown cooling heat exchangers and also functions to cool the shaft seals on the low pressure safety injection pumps as they circulate the heated reactor coolant.

Containment Spray System

During normal plant operation the containment spray pumps are aligned to flow through the shutdown cooling heat exchangers. In the shutdown cooling mode of operation isolation valves act to separate the heat exchangers from the containment spray system.

Chemical and Volume Control System

Piping and valves are provided in the CVCS such that during shutdown cooling a portion of the flow can be bypassed from the outlet of the shutdown heat exchangers through the letdown portion of the CVCS and returned to the suction line of the low pressure safety injection pumps. Flow through this bypass stream provides filtration and ion exchange of the reactor coolant via the purification filter and ion exchanger.

Spent Fuel Pool Cooling System

During shutdown or refueling, the shutdown cooling system can be aligned to cool the spent fuel pool system using the low pressure safety injection pumps (see Section 9.5).

9.3.2.3 Components

Shutdown Cooling Heat Exchanger and Pumps

The Shutdown Cooling System is made up of components of the Safety Injection System and the Reactor Coolant System. The principal characteristics of the major components in those systems are given in Sections 6.3 and 4.0, respectively.

The shutdown Cooling Heat Exchangers and low pressure safety injection pumps are described in Section 6.3. Additional information covering the characteristics of heat exchangers in the shutdown cooling mode is given in Table 9.3-1.

Shutdown Cooling Pumps

The low pressure safety injection pumps are part of the Shutdown Cooling System but are shared with the safety injection system. During all periods of plant operation when safety injection system operability is required, they are aligned for the emergency core cooling operation.

The low pressure safety injection pumps are vertical centrifugal pumps with mechanical seals to minimize reactor coolant leakage to the atmosphere. The low pressure safety injection pumps are described in Chapter 6.

Shutdown Cooling Valves

Manual isolation valves are provided to isolate equipment for maintenance. Manual valves have backseats to facilitate repacking. Control valves are provided for remote and manual control of heat exchanger tube side flow. Control valve 2-SI-306 has a lantern ring at the bottom of each of two packing boxes followed by a set of packing. It is not equipped with and does not require packing leak-off lines. 2-SI-657 has two sets of packing and intermediate leak-off connections that discharge to the waste processing system. Check valves prevent shutdown cooling reverse flow through the low pressure safety injection pumps.

Shutdown Cooling System Piping

All shutdown cooling system piping is austenitic stainless steel. All piping joints and connections are welded except for a minimum number of injection pumps, for example high pressure safety, and orifice plates.

9.3.3 SYSTEM OPERATION

9.3.3.1 Plant Heatup

Prior to plant heatup, RCS temperature is held stable with the shutdown cooling system removing decay heat. When heatup is started, the flow through the shutdown cooling heat exchangers is reduced to allow decay heat to start warming up the RCS. Since shutdown cooling return temperature is monitored as the indicator of reactor vessel temperature until RCPs are started, this flow reduction must be gradual. After taking into account the possible temperature rise if the steam generators are hotter than the reactor vessel, two RCPs are started. Shutdown cooling flow and temperature may then be adjusted as required to maintain the desired heatup rate within the limits of the Technical Specifications. When heat removal via the steam generators is available, the shutdown cooling system is secured and lined up for emergency operation.

9.3.3.2 Normal Plant Operation

During normal plant operation there are no components of the system in operation. The system is normally aligned for possible emergency operation.

9.3.3.3 Plant Cooldown

Plant cooldown is the series of operations which bring the reactor from a hot standby condition to cold shutdown.

Cooldown to less than 300°F is accomplished by dumping steam to the condenser or to atmosphere. During this time, after safety injection is no longer required, the shutdown cooling system is prepared for operation. This includes aligning the shutdown cooling heat exchangers to the LPSI pumps and the LPSI injection valves. This may also include boron equalization with the RWST (through the SIT test line) or warm-up (through valve 2-SI-400). When dumping steam is no longer effective in cooling down the reactor, and pressure has been reduced to within the design pressure of the shutdown cooling system, shutdown cooling is initiated. This is done preferably with two reactor coolant pumps still operating, but may be done while cooling down on natural circulation. Use of the reactor coolant pumps is preferred due to better control of the cooldown rate and the ability to cool the entire RCS simultaneously.

Initiation of shutdown cooling is done by opening the suction isolation valves and gradually opening the injection valves with the heat exchanger flow control valve shut. When flow through the system is established, the heat exchanger flow control valve is opened gradually to begin removing heat. If reactor coolant pumps are running, vessel temperature is taken as the cold leg temperature. If initiating shutdown cooling from natural circulation, vessel temperature is taken as cold leg temperature until shutdown cooling return temperature becomes lower than cold leg temperature. Then vessel temperature is taken as shutdown cooling return temperature. The RCS is then cooled down at a rate governed by Technical Specifications. Heat removal is controlled by controlling flow through the heat exchangers with the flow control valve. As cooldown progresses, this flow is increased to compensate for reduced temperature difference. Relatively constant total flow is maintained through the use of the heat exchanger bypass valve and/or the

injection valves. Component cooling water flow through the heat exchangers may need to be adjusted at some point in the cooldown dependent on whether the reactor coolant pumps are being used and the level of decay heat.

During cold shutdown and refueling, as long as there is fuel in the reactor, shutdown cooling operation is continuous (with brief exception to support refueling).

9.3.3.4 Refueling

The transfer of refueling water from the refueling storage tank to the reactor cavity may be accomplished using the safety injection system at the start of refueling. The reactor vessel head is removed and the high pressure safety injection pumps are started. These pumps take water from the RWST and inject it into the reactor coolant loops through the normal flow paths. The low pressure safety injection pumps or the containment spray pumps may also be used for this operation.

During shutdown or refueling, the shutdown cooling system can be aligned to cool the spent fuel pool system using the low pressure safety injection pumps (see Section 9.5).

At the end of refueling operations, refueling water is returned from the reactor cavity through the reactor coolant system and safety injection system to the RWST. A connection is provided from the shutdown cooling heat exchanger discharge to the refueling water storage tank for this purpose. The low pressure safety injection pumps are used for the transfer operation.

9.3.3.5 Emergency Conditions

Components of the shutdown cooling system are also used for emergency core cooling and their operation in this mode of operation is discussed in Chapter 6.

9.3.4 RELIABILITY AND AVAILABILITY

9.3.4.1 Special Features

The unit can be cooled to refueling temperature within the 27.5 hour time period using one LPSI pump and two heat exchangers. The unit can still be cooled to refueling temperature if only one heat exchanger is available, but it would take considerably longer. The unit can be maintained at refueling temperature with one pump and one heat exchanger after decay heat has decreased sufficiently.

In the event of a tube to shell leak in the heat exchanger, a high level alarm on the component cooling water surge tank will be generated. The tank overflows to the waste disposal system.

Two motor-operated valves in series isolate the shutdown cooling system suction piping from the reactor coolant system. An interlock with pressurizer pressure prevents them from opening when reactor coolant system pressure exceeds the design pressure of the shutdown cooling system. Refer to Section 4.3.8.2.3 for further details. Additionally, one of the valves (2-SI-651) has a

disconnect switch in its power circuit locked open while it is closed and the Unit is in Modes 1, 2 or 3, due to Appendix R fire concerns. 2-SI-651 has the disconnect switch in its power circuit locked open while it is open and in Modes 5 or 6, to preclude inadvertent operation during shutdown cooling.

Pressure relief valves are provided to protect isolated sections of piping from overpressure. The source of overpressure for which they are sized is thermal expansion for all but 2-SI-468, which is sized to protect the system from the simultaneous injection of all three charging pumps into a solid system. Insulation is another means for protecting isolated sections of piping from thermal over pressurization due to an ambient temperature rise.

Relief valve 2-SI-469, which is located between two suction isolation valves, has a setpoint of 300 psig and relieves to the primary drain tank. If reactor coolant system pressure is increased with 2-SI-652 inadvertently left not fully closed, or if it is opened prior to sufficiently reducing reactor coolant system pressure, 2-SI-469 will open and relieve to the drain tank which is equipped with pressure, temperature, and level alarms. This arrangement results in an extremely low interfacing systems LOCA frequency associated with the SDC suction path.

The SDC System is susceptible to an overpressure transient due to an inadvertent start of a HPSI pump since the injection capability of a HPSI pump exceeds the installed SDC System relief capacity. To eliminate an over pressurization of the SDC System caused by an inadvertent start of a HPSI pump, the HPSI pumps will be prevented from automatically injecting into the RCS when the SDC System is connected to the RCS, unless the RCS is sufficiently vented. This can be accomplished by the use of administrative controls.

The pressure interlock associated with valve 2-SI-651 in the shutdown cooling system has an installed local control switch which can override the pressure interlock to align the shutdown cooling system for boron precipitation control post-LOCA. Use of the override would only be required if vital bus power to motor control center B51 were lost in conjunction with the LOCA. The operators would verify RCS pressure below the shutdown cooling relief valve setting prior to overriding the pressure interlock. The pressure interlock for 2-SI-652 does not have override capability and cannot be opened if reactor coolant system pressure is greater than 280 psia.

9.3.4.2 Test and Inspections

Each component is inspected and cleaned prior to installation. Demineralized water is used to flush each system. Initially, the system is operated and tested to verify that the flow path, flow, thermal capacity, and mechanical operability meet the design requirements. Instruments are calibrated during testing. The automatic flow control is tested.

Periodic testing of the low pressure safety injection pumps as described in Chapter 6, assures the availability of this equipment for shutdown cooling. Data can be taken during refueling operations to confirm heat transfer capacity.

TABLE 9.3-1 SHUTDOWN COOLING HEAT EXCHANGERS DESIGN BASIS PARAMETERS

Tube Side	Shutdown Cooling Mode (27.5 hours after shutdown)	Recirculation Mode (after LOCA)
Flow, gpm	3000	1350
Heat Load, Million Btu/hr	31	32

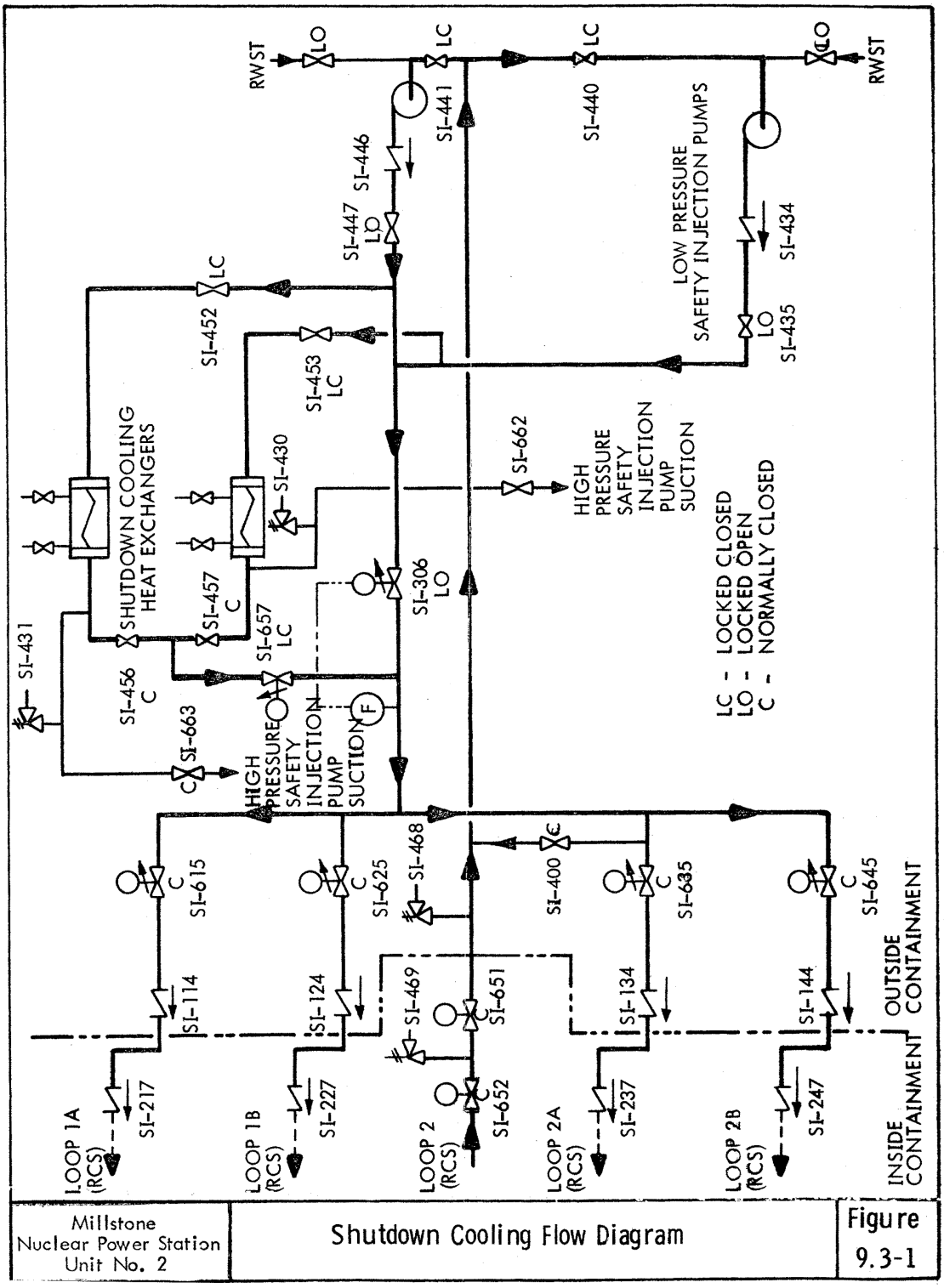
Shell Side	Shutdown Cooling Mode (27.5 hours after shutdown)	Recirculation Mode (after LOCA)
Flow, gpm	3500	2000
Heat Load, Million Btu/hr	31	32

Note: The above flow and heat loads are the minimum required to ensure adequate heat removal during shutdown cooling and post-LOCA recirculation modes.

TABLE 9.3-2 SHUTDOWN COOLING SYSTEM DESIGN PARAMETERS

Shutdown Cooling System Startup	Approximately 3.5 hours after reactor shutdown or trip
Suction Piping Design Pressure, psig	300
Discharge Piping Design Pressure, psig	500
Design Temperature, °F	350
Reactor Coolant System Cooldown Rate, °F/hr	75
Refueling Temperature, °F	130
Material - Piping and Valves	Austenitic Stainless Steel

FIGURE 9.3–1 SHUTDOWN COOLING FLOW DIAGRAM



9.4 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

9.4.1 DESIGN BASES

9.4.1.1 Functional Requirements

The function of the reactor building closed cooling water (RBCCW) system is to transfer heat from safety related structures, systems, and components to an ultimate heat sink. The RBCCW system safety function is to transfer the combined heat load of these structures, system and components under normal operating and loss-of-coolant accident (LOCA) conditions.

9.4.1.2 Design Criteria

The following criteria have been used in the design of the RBCCW system:

a. The system shall have two independent redundant subsystems having 100 percent heat removal capacity following a LOCA.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with associated components.

c. Capabilities shall be provided to assure the system operation with either on site power (assuming off site power is not available) or with off site power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

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

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

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The RBCCW system shall be an intermediate barrier between the service water (seawater) system and the radioactive or potentially radioactive fluids contained in the systems and components being cooled by the RBCCW system.

i. The RBCCW system components shall be designed to operate within the environment to which they are exposed.

j. The RBCCW system shall be evaluated to assure that it maintains structural integrity and pressure boundary for the postulated water hammer loading transient resulting from concurrent occurrences of a Design Basis LOCA event (i.e., LOCA or MSLB) with Loss of Normal Power event (i.e., LNP) as described in NRC Generic Letter GL 96-06. The structural design of piping and supporting components shall accommodate the aforementioned water hammer loading transient, with other applicable concurrent loads, so that continued function of the system is assured to mitigate the consequences of the aforementioned accident.

k. The CAR and CEDM Cooler units shall be evaluated to assure the integrity of the pressure boundary and structural adequacy for the GL 96-06 based water hammer loading condition.

l. The RBCCW supply temperature shall not exceed 85°F in Modes 1, 2 and 3.

9.4.2 SYSTEM DESCRIPTION

9.4.2.1 System

The RBCCW system is shown schematically in Figures 9.4–1 through 9.4–6. The logic diagram is shown in Figure 7.3–1.

The RBCCW system consists of two independent headers, each including one motor-driven RBCCW pump, one RBCCW heat exchanger, and associated piping, valves, instrumentation, controls and a downcomer from the RBCCW surge tank. A third RBCCW pump and heat exchanger is provided as a spare for the system.

Redundant safety feature components, cooled by the RBCCW system, are split between the two independent RBCCW headers. The other systems and components cooled by the RBCCW system are divided between the RBCCW headers to equalize header heat loads.

The items cooled by the RBCCW system are listed in the following. Further information about each may be found in the section referenced in the parentheses.

Containment air recirculation and cooling units (Section 6.5)

Reactor vessel support cooling coils (Section 6.5.4.1)

Containment spray pump seal coolers (Section 6.4)

High and low pressure safety injection pump seal coolers (Section 6.3)

Shutdown cooling heat exchangers (Section 6.4, 6.3)

Engineered safety features room air recirculation and cooling units (Section 9.9.7)

Reactor coolant pump thermal barrier and oil coolers (Section 4.3.3)

Primary drain and quench tank heat exchanger (Section 11.1)

CEDM coolers (Section 9.9.1)

Letdown heat exchanger (Section 9.2)

Degasifier effluent cooler (Section 11.1)

Degasifier vent condenser

Sample coolers (Section 9.6.2.1)

Spent fuel pool heat exchangers (Section 9.5)

Waste gas compressor aftercoolers

Quench tank heat exchanger (Section 10.4.6)

Each RBCCW pump is designed to circulate 7000 gpm of water through the RBCCW system for removing heat from systems and components that handle the reactor coolant, containment atmosphere and engineered safety features. The RBCCW is cooled in the RBCCW heat exchangers by the service water (seawater) (Section 9.7.2.2.1).

The RBCCW System is designed to provide cooling for the following system operating conditions:

I. Normal Operation

II. 3.5 Hours after Shutdown

III. 27.5 Hours after Shutdown

IV. Loss of Coolant Accident (LOCA) Injection Operation

V. LOCA Recirculation Operation

System hydraulic analysis and flow balancing were performed to ensure that the minimum required flows can be provided to essential safety related loads for the above operating conditions. The flow balance was field tested to demonstrate that when the safety injection actuation signal (SIAS) and the sump recirculation actuation signal (SRAS) automatically realigns the system, the minimum required flows will be provided to the essential safety related loads during LOCA injection operation (condition IV) and LOCA recirculation operation (condition V). Table 9.4-3 identifies the minimum required flows for these two post-accident operating conditions.

The RBCCW pump and heat exchanger design is based on requirements during normal operation, normal shutdown cooling and a LOCA. This results in a design which provides an adequate cooling capacity for normal and emergency conditions, including reactor shutdown.

A RBCCW surge tank provides sufficient NPSH for the RBCCW pumps and absorbs the volumetric change due to temperature changes imposed on the water within the RBCCW system. The lower half of the RBCCW surge tank is divided into two equal sections by a vertical weir. If one of the two independent RBCCW headers ruptures, the vertical weir in the RBCCW surge tank assures that 1000 gallons of makeup water will be available to the functional header. Demineralized water is used for makeup and comes from the primary water storage tank (Section 9.12). A flow orifice is located in each surge tank discharge header downstream of the outlet isolation valves to dampen pressure surges that result from cold starts of the RBCCW Pumps that could occur following an LNP scenario.

Makeup water supplied to the RBCCW surge tank enters from the top of the surge tank at two locations on either side of the weir, therefore, makeup water can be supplied to both sections of the tank. To stop the flow of water to the failed header, manual valves can be closed.

The following instrumentation and alarms located on the main control board give sufficient information to assure the operator that makeup water is being supplied to the RBCCW surge tank section supplying the functional header:

1. Position lights ZS-6000 on makeup valve LV-6000 indicate whether the valve is open or closed. A timed alarm ZAH-6000 indicates that the valve has been open longer than required for normal makeup.

2. Level controller LC-6000 controls 6 inches above the weir, so regardless of which side fails, the controller will keep the makeup water valve LV-6000 open until the failed header is repaired or until the valve is closed by the hand switch HS-6000 or by using manual valves.

3. Level indicators LI-6001 and LI-6730 indicate the water level in each of the two sections of the surge tank. Alarm LAHL-6892 indicates high and low water level in the surge tank.

The valves, piping, instrumentation and alarms described above are shown on Figure 9.4–1.

Corrosion inhibitor can be injected into the RBCCW system to increase the optimal concentration during normal operation, as required.

External leaks, relief valve discharges and drains are collected by the drains system and processed through the radioactive waste processing system. Leakage into the RBCCW system can be determined by a radiation monitor installed on the discharge of the RBCCW pumps.

The RBCCW system components located in the containment building are subject to a maximum temperature of 289°F and a maximum relative humidity of 100 percent during post-incident operation. The RBCCW system components located in the auxiliary building are subject to a temperature of 110°F and a maximum relative humidity of 100 percent. The RBCCW pumps and motors are subjected to a maximum temperature of 155°F after a LOCA.

The RBCCW system is designed to permit venting of components, obtaining system water samples, and draining of components to the radioactive waste system.

Components and heat exchangers served by the RBCCW system, which can be isolated are equipped with self-actuated, spring-loaded relief valves for overpressure protection.

9.4.2.2 Components

A description of the major RBCCW system components is given in Table 9.4-1.

9.4.3 SYSTEM OPERATION

9.4.3.1 Normal Operation

During normal operation, RBCCW is supplied to the components described in Section 9.4.2.1 with the exception of the shutdown heat exchangers and the engineered safety features room air recirculation and cooling units.

Two RBCCW pumps and two RBCCW heat exchangers are required for cooling service during normal operation. The heat load per RBCCW heat exchanger during normal operation is 16.8 x 10^6 Btu/hr (header 'A') and 23.4 x 10^6 Btu/hr (header 'B') and the RBCCW discharge temperature is 85°F when the RBCCW heat exchangers are cooled by service water up to maximum temperature of 80°F.

The operation of the RBCCW system is monitored with the following instrumentation:

a. A temperature detector in the inlet and outlet line of the RBCCWS heat exchangers and temperature alarms in the outlet lines.

b. Pressure indicators in the lines between the pumps and the RBCCWS heat exchangers.

c. Level alarms and a controller on the surge tank.

d. Temperature detectors located on the outlet lines of the components being cooled.

e. Flow indicators and high flow alarms in both headers on the discharge side of the RBCCW heat exchangers.

f. Low pressure alarm in both headers on the discharge side of the RBCCW heat exchanger.

g. Hand switches and indicating lights for the pumps and remotely operated control valves.

9.4.3.2 Emergency Conditions

Following a LOCA, the RBCCW system is automatically aligned for post-incident cooling by the safety injection actuation signal (Section 7.3), the containment isolation actuation signal (Section 7.3) and the sump recirculation actuation signal (Section 7.3). The SIAS and CIAS actuates valves to stop the RBCCW flow to components not required during a LOCA and to initiate RBCCW flow to the engineered safety features room air recirculation and cooling units. The SRAS signal will open the RBCCW shutdown heat exchanger outlet valves. Prior to the SRAS signal, the RBCCW shutdown heat exchanger outlet valves have a safety function to remain closed (during the injection phase of a LOCA) to ensure adequate RBCCW flow is directed to the CAR coolers. The RBCCW is also supplied to the HPSI, LPSI and containment spray pump seal coolers.

The RBCCW system operation during a LOCA uses two RBCCW pumps, two RBCCW heat exchangers and two headers for LOCA cooling. If one RBCCW header is lost, one RBCCW pump, heat exchanger and one header can provide adequate cooling.

Calculations provide the basis for the FSAR Section 14.8.2 Containment Pressurization Analysis. To maximize the containment pressurization consequences following a MSLB or LOCA accident, these analyses minimize the energy removed from containment by the CAR cooling units and the SDC heat exchangers. This is accomplished by assuming design fouling of the CAR cooling units, the SDC heat exchanger, and the RBCCW to Service Water heat exchangers, and minimum RBCCW flow distributions assuming degraded RBCCW pumps. Prior to the sump recirculation (SRAS), two CAR cooling units transfer a maximum of 160 million BTU/hour of energy to the RBCCW system. Following SRAS, two CAR cooling units transfer a maximum of 70 million BTU/hr and the SDC heat exchanger transfers a maximum of 47 million BTU/hour of energy to the RBCCW System.

The RBCCW peak temperature analysis, which maximizes the RBCCW temperatures by assuming clean CAR cooling units and SDC heat exchangers, with maximum RBCCW flow distributions was performed to evaluate the system design capability.

Based on the RBCCW peak temperature analysis, the maximum RBCCW heat load from the two containment air recirculation and cooling units plus the balance of plant components is 204.4 x 10^6 Btu/hr during the injection mode following a LOCA. During the recirculation mode following a LOCA the maximum calculated heat load of 130.7 x 10^6 Btu/hr is transferred to the RBCCW system through the containment air recirculation and cooling units, the shutdown heat exchanger plus the balance of plant components. The design heat removal capacity of the RBCCW heat exchanger is 160 x 10^6 Btu/hr during the injection mode, but calculations demonstrate that the heat exchangers can adequately handle the 204.4 x 10^6 Btu/hr heat load.

The RBCCW peak temperature analysis calculated the maximum system supply temperatures based on maximum RBCCW flow distribution values, clean CAR coolers and SDC heat exchangers, and fouled (design) RBCCW heat exchangers. The component maximum outlet temperatures were then determined based on design heat loads. The system components and

piping, as well as the affect on the equipment and rooms being cooled post-accident were evaluated at the maximum RBCCW temperatures. The calculated peak RBCCW temperature supplied to the components during a LOCA/MSLB with LNP (GL 96-06 scenario), and considering no fouling in the CAR Cooler units is less than 149°F.

The following is a list of essential components cooled by the RBCCW System during a LOCA:

Containment Air Recirculation And Cooling Units

Shutdown Cooling Heat Exchangers

Engineered Safety Features Room Air Cooling Coils

High Pressure Safety Injection Pump Seal Coolers

Low Pressure Safety Injection Pump Seal Coolers

Containment Spray Pump Seal Coolers

The nonessential components which are on line after a LOCA are the waste gas compressors, Primary drain and quench tank cooler, CEDM coolers, Reactor vessel support concrete cooling coils, reactor coolant pump thermal barriers and lube oil coolers, and degasifier vent condenser. After SRAS all the nonessential components except for the waste gas compressors and degasifier vent condenser shall be off the line. However, the spent fuel pool heat exchanger will be returned to service within 4 to 5 hours after the start of the LOCA. The Primary Sample Coolers may also be returned to service as necessary to support PASS cooling.

A failure mode analysis is given in FSAR Table 9.4-2 which describes how leaks and malfunctions are detected and what corrective actions must be taken to insure that the safety functions performed by the RBCCW System are not impaired. Once the operator has detected a leak or malfunction, the operator will isolate the equipment for repair maintenance if environmental conditions permit operation of manual valves. If environmental conditions do not permit isolation of the equipment, the entire header may be isolated remotely from the main control room. The safety functions of the RBCCW system can be performed with only one header in operation. If the leak or malfunction involves the RBCCW pump and heat exchanger, the pump and heat exchanger can be isolated remotely from the main control room. Also, equipment in the containment can be isolated remotely in the main control room by closing the containment isolation valves.

Relief valves, on the RBCCW supply and return headers serving the reactor Coolant Pumps integral heat exchangers, ensure that the RBCCW containment isolation valves, on these headers, can be closed during an intersystem LOCA, by ensuring that the valves will not be overpressurized to the point that the valve motors cannot close the valves. The relief valves, which have a set pressure of 165 psig may open upon a rupture of an integral heat exchanger tube (a beyond design basis event). This is required in response to NRC IN 89-54 to preclude an unisolable release of radioactive fluid outside of the containment that may exceed 10 CFR 100 limits.

The operation of the RBCCW system during a LOCA is monitored as described in Section 9.4.3.1.

The RBCCW system, consisting of two independent headers, is at a higher pressure than the service water system during normal operations to prevent service water leaking into the RBCCW system. The only time that the RBCCW system pressure will be lower than the service water system is (1) during shutdown for routine maintenance, (2) if the RBCCW pumps malfunction or (3) if there is a major RBCCW leak greater than 400 gpm. Loss of RBCCW system pressure is indicated by low pressure alarms (PAL-6036 & PAL-6037) located in the RBCCW heat exchanger discharge headers. Once the operator has observed the low pressure alarm, it will be decided whether this branch can be easily isolated; if this is not possible, the RBCCW header will be taken out of service and at the same time the shell side of the RBCCW heat exchanger will be isolated. Therefore, the possibility of service water leaking into the RBCCW heat exchanger is remote. If service water does leak into the RBCCW system during periods of low pressure, it will be detected by manual sampling on a routine basis.

Since the RBCCW system is potentially capable of contaminating the service water system with radioactivity due to leakage, a detection system is located within the RBCCW system.

The RBCCW System is continuously monitored for radioactivity by using a self-contained closed-loop radiation monitoring system consisting of (1) a gamma NaI(T1) scintillation detector assembly, (2) a 4 inch schedule 40 stainless steel sample chamber shielded with lead to reduce the ambient background radiation level, (3) solid-state, control room readout module, (4) local meter indication with visual and audible alarms, and (5) auxiliary support equipment such as piping, valves and recorder.

The RBCCW sample is continuously taken from the RBCCW pump discharge and is circulated through the sample chamber and detector assembly. The sample returns to the RBCCW pump suction.

The radioactivity present in the system is monitored by the gamma detector assembly, displayed and recorded in the control room.

Alarm set points based on the limits established by 10 CFR Part 20, will result in alarm annunciations in the control room and at the monitoring site.

9.4.3.3 Shutdown

The RBCCW system is required for continuous operation during normal unit shutdown cooling (Section 6.3) beginning 3.5 hours after the start of unit shutdown. The RBCCW system supplies cooling water to the shutdown heat exchangers to cool the reactor coolant from 300 to 130°F in 24 hours.

During normal shutdown, such as for refueling, two RBCCW pumps and two RBCCW heat exchangers are used for cooling. The heat load per RBCCW heat exchanger is 84.1×10^6 Btu/hr

(header ‘A’) and 84.5 x 10^6 Btu/hr (header ‘B’) at 3.5 hours after start of Unit shutdown. The RBCCW heat exchanger discharge is less than 130°F when the RBCCW heat exchangers are cooled by service water at a maximum temperature of 80°F. At 27.5 hours after the start of Unit shutdown, the heat load per RBCCW heat exchanger is reduced to 40.8 x 10^6 Btu/hr.

If one RBCCW heat exchanger or pump should malfunction, the spare can be immediately placed in service, or the unit can be cooled down at a slower rate with one heat exchanger and one pump. During shutdown the operation of the RBCCW system is monitored as described in Section 9.4.3.1.

9.4.4 AVAILABILITY AND RELIABILITY

9.4.4.1 Special Features

The components of the RBCCW system are designed to general design requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip from high pressure piping as described in Section 6.1.

To ensure pump availability, each RBCCW pump motor is supplied from a specific 4160 volt bus. The buses may be powered from alternate sources (i.e., NSST, RSST, D/G, or via Unit 3 bus 34A or 34B). In the event that reserve station service power is lost, the RBCCW pumps, required for engineered safety features systems cooling or normal shutdown cooling, are automatically supplied by the emergency buses.

The RBCCW pump motor is designed to withstand a maximum temperature of 155°F during a LOCA and maximum relative humidity of 100 percent.

The RBCCW system is provided with a radiation monitor to alarm if radioactive fluids leak into the RBCCW.

A failure mode analysis is given in Table 9.4-2. A rupture in the system is considered an initiating event only; it is not postulated concurrent with a LOCA (or any other Chapter 14 event). System redundancy and header separation have been provided to maintain continuous cooling in the event of a single passive failure during post-accident long term cooling.

9.4.4.2 Tests and Inspection

One RBCCW pump was shop-tested for hydraulic performance. Seven test points were used to generate the performance curve which is shown in Figure 9.4–7.

Each pump was hydrostatically tested at 375 psi, 1.5 times the design pressure of the pump.

Nondestructive testing for the RBCCW heat exchangers were performed in accordance with the ASME Code Section VIII.

The RBCCW heat exchangers were leak tested as follows:

a. Shell side gas test (inert gas) at 188 psig which is 1.25 times the design pressure.

b. Shell side hydrostatic test at 225 psig which is 1.5 times the design pressure.

c. Channels hydrostatic test at 250 psig which is twice the tube side design pressure.

d. Tubes, hydrostatic test at 150 psig, which is 1.2 times the tube side design pressure.

Each component is inspected and cleaned prior to installing it in the system. Following installation, the RBCCW system undergoes a preoperational test before startup. The detail test procedure is described in Section 13.

Instruments are calibrated during testing and the automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during preoperational testing.

Initially the system was operated and tested with regard to flow paths, flow capacity and mechanical operability. At least one pump was tested online to demonstrate head and capacity (Section 13).

Data are taken periodically during normal operation to confirm the heat transfer capabilities.

Major components of the system such as pumps, tanks, and heat exchangers are accessible for periodic inspection during operation.

The RBCCW system and components are periodically tested to demonstrate the operability, performance, structural and leaktight integrity of the system and components. The testing is accomplished as follows:

a. During normal operation the RBCCW system components and piping are constantly pressurized by the RBCCW pump, therefore, they are constantly being tested for structural and leaktight integrity.

   The operability and performance of all components except the engineered safety features room air recirculation and cooling units, the shutdown cooling heat exchanger, the spare RBCCW pump and the spare RBCCW heat exchanger is determined by observing the temperature, pressure and flow instrumentation associated with each component.

   Online testing of the spare RBCCW component(s) (pump and/or heat exchanger) is performed by bringing the spare component(s) to be tested into operation while removing the operating component(s) from service. The operation and

performance of the spare RBCCW pump and heat exchanger can be determined by observing the pressure and flow instrumentation.

Online testing of the remotely operated valves actuated by SIAS signals and located on the discharge side of the containment air recirculation and cooling units, the spent fuel pool heat exchangers, the safeguards room air cooler units and the degasifier effluent cooler is performed by operating each valve's manual control switch. The position of the valves is verified by position indication lights in the control room. Testing these valves during power operation by initiating SIAS signals is not performed as it results in unnecessary starting of the emergency diesel generators and causes RBCCW system flow transients which operators must respond to.

Online testing of the other remotely operated valves actuated by SRAS is performed by closing the appropriate manual valves and by manually initiating the SRAS signals for the valves on the discharge side of the shutdown heat exchangers. The position of the valves is verified by position indicated lights in the control room.

b. During shutdown the operability and performance of the RBCCW components and the RBCCW system as a whole is determined by observing the flow, pressure and temperature instrumentation.

c. After normal unit shutdown for refueling, the applicable portion of the RBCCW system used for engineered safety feature systems cooling is tested by actuating SIAS and SRAS signals to demonstrate operability and position of valves on a per facility basis. Structural and leaktight integrity of the system is tested separately as part of the Unit 2 Inservice Inspection Program.

The RBCCW containment isolation valves for the supply and return lines for the Containment Air Recirculation and Cooling heat exchangers have "T-Ring" seats. In lieu of test results under 10 CFR 50, Appendix J, Type C testing, the "T-Ring" seats will be replaced based on observed degradation. These valves are not included in the Type C testing program because the valves are open during accident conditions. This eliminates a commitment made under letter A06107, dated 1/16/87 under Docket Number 50-336.

9.4.5 CODE STRUCTURAL QUALIFICATION

The code of record for the RBCCW piping system is USAS B31.1-1967. The portion of piping in the containment penetration area is designated as ANSI B31.7, Class 2.

The piping and support components are designed for the design basis loads for the appropriate load combinations and criteria. The original code qualification was performed for the following load conditions and effects: (a) internal pressure, (b) dead weight, (c) thermal expansion at maximum temperatures, (d) OBE inertia loads, (e) DBE inertia loads, and (f) seismic anchor movements for OBE and DBE, as applicable. The system and components are evaluated for the

newly identified LOCA/water hammer loads, including the original design basis loads, considering temperature changes based on zero fouling in the CAR Cooler units.

The postulated LOCA induced dynamic water hammer loads based on the scenario described in GL 96-06 is classified as “Faulted Plant Occasional Mechanical Load” because LOCA is a “Faulted Plant Condition.” Although not part of the original design basis, this load is recognized as a design basis load.

The acceptance criteria for piping is based on maintaining the combined primary stress due to concurrent dead weight, internal pressure and dynamic LOCA/water hammer transient within the material yield stress of piping at operating temperature. The acceptance criteria for design verification of support components corresponds to conservative Emergency stress limits (i.e., ASME Level C Limits) as follows, with a minor exception as noted in item b: (a) structural steel elements-within 90% of yield per AISC Code (b) vendor supplied components Load Capacity per Load Capacity Data Sheets (LCD) Level C, except those supports which are not protecting the safety related function of the RBCCW system in the post-accident scenario may be evaluated using LCD Level D (c) anchor bolts safety factors shall be 4 for Wedge type (e.g., HILTI KWIK BOLT) and 5 for shell type (e.g., Phillips Red Head) consistent with NRC I.E. Bulletin 79-02.

The LOCA/water hammer is analyzed using dynamic time history analysis method. The time history forcing functions applicable to various segments of RBCCW piping are determined by detailed thermal hydraulic analysis of the system using RELAP 5/Mod 3.2 computer code.

TABLE 9.4-1 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS

RBCCW PUMP	
Manufacturer	Ingersoll-Rand
Model	12 x 14 SD
Type	Centrifugal, horizontal split case, dual volute with mechanical seals
Quantity	3
Capacity each (gpm)	7000
Head (feet)	150
Pump (rpm)	1750
BHP	312
NPSH required/available (feet)	23/80
Material	
Case	ASTM A48 Gr 30
Impeller	ASTM B143 Alloy 2A
Shaft	AISI-C1045
Motor	350 hp, 400 volt, 60 Hz, 3 phase, 1750 rpm
Codes	
Motor: NEMA	
Pump: Standards of the Hydraulic, Institute ANSI B 16.1, ANSI B 31.1	
Seismic	Class I
RBCCW HEAT EXCHANGER	
Manufacturer	Struthers Nuclear
Model	
Horizontal, single-pass, counter-flow straight tubes rolled into tube sheets	
Type	TEMA Type AEL
Quantity	3

TABLE 9.4-1 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS (CONTINUED)

	Design duty each (Btu/hr)
26.33×10^6 - normal	36.0×10^6 - 27.5 hours after shutdown
90.25×10^6 - 3.5 hours after	160.0×10^6 - loss-of-coolant shutdown
Design Overall fouling factor	0.0008
Design pressure (psig)	Shell side 150; tube side 125
Design temperature (°F)	Shell side 250; tube side 120
Material	
Shell ASTM-A-285-Gr C	
Tubes Aluminum brass ASME SB-111 Alloy 687	
Tube sheets	Aluminum bronze ASME SB-171- CDA614
Channel	Nickel Copper, UNS N04400, unlined
Seismic	Class 1
Code	TEMA Class R, ASME Section VIII
RBCCW SURGE TANK	
Manufacturer	PX Engineering
Type	Vertical with vertical weir divider in lower half of the tank
Quantity	1
Design pressure (psig)	15
Design temperature (°F)	250
Volume (gal)	4480
Material	
Shell	ASTM A 285-Gr C
Dished head	ASTM A 285-Gr C
Seismic	Class I
Code	ASME Section VIII

TABLE 9.4-1 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS (CONTINUED)

Pipe:

Size – 12 inch and larger	Wall Thickness – 0.375
Size – 2.5 inch through 10 inch	Wall Thickness – Schedule 40
Size – 2.5 and smaller	Wall Thickness – Schedule 80
Material	Seamless ASTM-A53-B

Code ANSI B31.1.0 plus Random Radiography of Butt Weld Base Material Identification Seismic Analysis ANSI B31.7, Class II Containment Penetration.

Joints Welded except at flanged equipment connections and butterfly valves.

Fittings

2.5 inches and larger

Butt welding, ASTM A-234 WPB or WPB-W, wall thickness to match pipe.

2.5 inches and smaller

3000 lb. socket welding, ASTM A-181, Gr. I Mod

Flanges

150 lb ANSI Std. F&D ASTM A-181 Gr. II (original construction) or ASTM A-105.

ASTM A-181

12 inches and larger	Slip-on, RF
2.5 inches through 10 inches	Weld neck, RF
2 inches and smaller	Socket Weld, RF

Bolting: Bolt-studs, ASTM A-193 B-7 Hex nuts, heavy steel, ASTM A-194 2H.

Valves

2 inches and smaller	800 lb. ANSI, socket weld, carbon steel*
2.5 inches and larger	150 lb. ANSI, butt weld, carbon steel (except butterfly valves)
Non-nuclear Butterfly valves	150 lb., wafer, cast iron
Nuclear Butterfly	150 lb, wafer, carbon steel

TABLE 9.4-1 REACTOR BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS (CONTINUED)

Valves Code ASME Section III, Nuclear Power Plant Components, Class II

Seismic Class

Intersystem LOCA relief valves
6 inch x 10 inch, ANSI Class 300, RF, carbon steel, ASME Section VIII

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* 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 800 lb, based on the design conditions of the specific application.

TABLE 9.4-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION AND QUANTITY	*FAILURE MODE*	*METHOD OF DETECTION AND MONITOR*	*DETRIMENTAL EFFECT ON SYSTEM*	*CORRECTIVE ACTION*	*RESULTANT SYSTEM STATUS*	*REMARKS*
RBCCW Return Header (2)	a. Break equivalent to 0.5 inch 0.5 inch diameter line	Makeup valve opens alarm for a and b. Surge tank low level alarm for a and b.	a. Loss of water equivalent to makeup capacity	Isolate failed header	One header and its components out of service; Alternate header in operation	Header out of service for repairs; one header sufficient for safe shutdowns
	b. Large break	Low pressure alarm for b. All CRI	b. Loss of one header			
Piston-Operated Valve (6) HV6002 through HV6007	Fails, as is	Position indication CRI	None	Valve operator repaired	Normal operation	Valve can be manually operated
RBCCW pump Suction Header (2)	a. Break equivalent to 1.5 inch diameter line	Makeup valve opens alarm and sump high level alarm for a and b. Surge tank low level alarm for a and b.	a. Loss of water equivalent to makeup capacity	Failed header isolated; standby pump and header put	Normal operation	Suction and discharge header and associated pump out of service for repairs
RBCCW pump Discharge Header (2)	b. Large break	Low pressure alarm for b. All CRI	b. Loss of one header			
RBCCW Pump (3)	Pump stops	Header low pressure alarm; standby pump starts alarm; both CRI	Loss of flow in one header	Isolate pump; put standby pump in service; valve lined up	Normal operation	Pump out of service for repairs
RBCCW Heat Exchanger Supply Header (3)	a. Break equivalent to 1.5 inch diameter line	Makeup valve opens alarm and sump high level alarm for a and b. Surge tank low level alarm for a and b	Loss of water equivalent to makeup capacity	Failed header isolated; standby pump and headers put in service	Normal operation	Header out of service for repair
RBCCW Heat Exchanger Discharge Header (3)	b. Large break	Low pressure alarm for b. All CRI	Loss of one header			
RBCCW Heat Exchanger (3)	a. Shellside rupture; b. Tube rupture	Sump level alarm for a. Header low pressure alarm; Surge tank low level alarm; Makeup valve opens alarm for a and b. All CRI	Loss of water in one header	Failed heat exchanger isolated; standby put into service	Normal operation	Heat exchanger out of service for repairs
Piston-Operated Valve (8) HV6011 through NV6018	Fails, as-is	Position indication	None	Valve operator repaired	Normal operation	Valve can be operated manually
RBCCWS Supply Header (2)	a. Break equivalent to 1.5 inch diameter line	Makeup valve opens alarm for a and b Surge tank low level for a and b	a. Loss of water equivalent to makeup capacity	Isolate failed header	One header and equipment out of service; alternate header in normal operation	Header our of service for repairs; one sufficient for safe shutdown
	b. Large break	High pressure alarm for b.	b. Loss of one header			
	All CRI					
Spent Fuel Pool Heat Exchanger (2)	a. Shell side rupture	a. Surge tank low level alarm and makeup valve opens alarm	a. Loss of water from half of system	a. Isolate failed heat exchanger	a. Redundant heat exchanger available	a, b. Heat exchanger out of service for repairs
	b. Tube rupture	b. High radiation alarm, CRI	b. Radioactivity in header	b. Isolate heat exchanger and	b. Header out of service	b. Drain header

TABLE 9.4-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	*FAILURE MODE*	*METHOD OF DETECTION AND MONITOR*	*DETRIMENTAL EFFECT ON SYSTEM*	*CORRECTIVE ACTION*	*RESULTANT SYSTEM STATUS*	*REMARKS*
Piston-Operated Valve (2) HV6315, HV6316	Fail closed	Position indication	None	Valve operator repaired	Normal operation	Valve can be operated manually
Shutdown Heat Exchanger (2)	During shutdown:					
	a. Shell side rupture	a. Surge tank low level alarm and makeup valve opens alarm	a. Loss of water from half of system	a. Isolated failed heat exchanger	Normal shutdown operation at reduced rate	a, b. Redundant heat exchanger out of service for repairs; shutdown heat exchangers not in use during normal operation
	b. Tube rupture	b. Radiation monitor. Both CRI	b. Radioactivity in header	b. Isolate heat exchanger and header		
Piston-Operated Valve (2) HV6050, HV6055	Fails open	Position indication	None	Valve operator repaired	Normal operation	Valve can be operated manually
HPSI Pump Seal (3)			Loss of water less than makeup system water	HPSI pump isolated	Normal operation	Pump out of service for repairs. Pumps not in use during normal operation
	a. Out leakage	Seal rupture:	a. Makeup valve open alarm; sump pump alarm in engineered safety features room			
	b. In leakage	b. High RBCCW outlet temperature alarm and surge tank high level alarm. All CRI				
LPSI Pump Seal (2)	Seal rupture:		Loss of water less than makeup system water	LPSI pump isolated	Normal operation	Pump out of service for repairs; pumps not in operation during normal operation
	a. Out leakage	a. Makeup valve opens alarm and sump pump alarm in engineered safety features room				
	b. In leakage	b. High RBCCW outlet temperature alarm and surge tank high level alarm. All CRI				
Containment Spray Pump Seal (2)	Seal rupture:		Loss of water less than makeup system water	Pump isolated	Normal operation	Pump out of service for repair; pump not in service during normal operation
	a. Out leakage	a. Makeup valve opens alarm and sump pump alarm in engineered safety features room				
	b. In leakage	b. High RBCCW outlet temperature alarm and surge tank high level alarm. All CRI				
Sample Cooler (7)	Shell side rupture	Surge tank low level alarm and makeup valve opens alarm. Both CRI	Loss of water from header B.	Cooler isolated	No sample	Cooler not available until repaired.

TABLE 9.4-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	*FAILURE MODE*	*METHOD OF DETECTION AND MONITOR*	*DETRIMENTAL EFFECT ON SYSTEM*	*CORRECTIVE ACTION*	*RESULTANT SYSTEM STATUS*	*REMARKS*
Letdown Heat Exchanger (1)	a. Shell side	a. Surge tank low level alarm makeup valve opens alarm; possible high temperature in letdown line	a. Loss of water from header B	a. Heat exchanger isolated	a, b. No letdown	a, b. Letdown heat exchanger out of service until repaired
	b. Tube side	b. Surge tank high level alarm and high radiation monitor alarm. All CRI	b. Radioactivity in header B	b. Isolate heat exchanger and header		b. Drain header
Control Valve (1) 2-RB-402	Fails close	Position indication; CRI; Abnormally low letdown temperature	None	Valve operator repaired	Normal operation	
Piston-Operated Valve (4) HV6072, HV6073, HV6075, HV6077	Fails open	Position and flow indications, both CRI	None	Valve operator repaired	Normal operation	Valves can be operated manually
Containment Air Cooler (4)	Tube side	Surge tank low level alarm; makeup valve opens alarm; low flow indication, all CRI	Loss of water from one half of system	Failed cooler isolated	Normal operation	Failed cooler out of service for repairs; only 3 units needed for normal operation
Piston-Operated Valve (8) HV6079, HV6080, HV6083, HV6084, HV6087, HV6088, HV6091, HV6092	Fails open	Position and flow indications, both CRI	None	Valve operator repaired	Normal operation	Valves may be operated manually
Motor-Operated Valve (2) HV6095, HV6096	Fails as is	Position indication, CRI	None	Valve operator repaired	Normal operation	Valves may be operated manually
Reactor Coolant Pump Thermal Barrier and Oil Cooler (4)	a. Rupture of thermal barrier	a. High radiation alarm and high temperature cooling water alarm	a. Partial loss of reactor coolant	a. Isolated failed cooler and header	a. One header out of service	Reactor coolant pump out of service until cooler repaired
	b. Rupture of oil cooler tube side	b. Low oil level alarm and low oil pressure indication	b. Oil in system	b, c. Failed cooler isolated	b, c. Reactor coolant pump out of service	
	c. Rupture of oil cooler shell side	c. Makeup valve opens alarm; low flow alarm; all CRI	c. Loss of water equivalent to makeup capacity			
Reactor Vessel Support Cooling Coils (3)	a. Coil plugs up	a. Low flow alarm	a. Partial loss of cooling capacity redundant independent coil on other header	Failed cooler isolated open valves to	Normal operation	Coil out of service for repair
	b. Coil connection ruptures	b. Low flow alarm and makeup valve open alarms; all CRI	b. Loss of water from header			
CEA Drive Air Cooler (2)	Tube side	Air temperature indicators; makeup valve opens alarm; both CRI	Loss of water in header	Failed cooler isolated	Normal operation	One redundant cooler out of service for repair
Primary Drain Tank and Quench Tank Heat Exchanger (1)	a. Shell side rupture	Surge tank level alarm and makeup valve open alarm for a and b	Loss of water from header A	Isolate heat exchanger	Heat exchanger out of service until repaired	
	b. Tube rupture					

TABLE 9.4-2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	*FAILURE MODE*	*METHOD OF DETECTION AND MONITOR*	*DETRIMENTAL EFFECT ON SYSTEM*	*CORRECTIVE ACTION*	*RESULTANT SYSTEM STATUS*	*REMARKS*
Diaphragm operated Valve (1) HV6118	Fails open	Position indication; CRI	None	Repair operator	Normal operation	Valve can be manually operated
Motor Operated Valve (2) HV6108, HV6106	Fails as is	Position indication; CRI	None	None; operator repaired during shutdown	Normal operation	Valve can be manually operated
RBCCW Surge Tank (1)	Rupture	Surge tank low level alarm and makeup valve open alarm	Loss of water	Isolate tank	Normal operation	Surge tank out of service until repaired
Diaphragm-Operated Valve (1), LV6000	Fails open	Position indication; CRI	Loss of water	Isolate valve	Normal operation	Manual makeup until repaired
Degasifier Efficient Cooler (1)	a. Shell side rupture	Surge tank low level alarm; makeup valve open alarms for a and b. All CRI	Loss of water from header B	Isolate cooler	Cooler out of service until repaired	
	b. Tube side rupture					
Piston-Operated Valve (1) HV6739	Fail closed	Position indication; CRI	Loss of degasifier efficient cooler and letdown heat exchanger	Valve operator repaired	Normal operation	Valve can be manually operated
Degasifier Vent (1) Condenser	a. Tube side rupture	Surge tank low level alarm and makeup valve open alarm for a and b. All CRI	Loss of water from header B	Isolate vent condenser	Vent condenser out of service until repaired	
	b. Shell side rupture					
Diaphragm Operated Valve (1) HV9309	Fail open	Visual	None	Valve operator repaired	Normal operation	
Waste Gas Compressors (2) (after cooler)	a. Shell side	Surge tank low level alarm and makeup valve open alarm; all CRI	Loss of water from header A	Isolate waste gas compressors	One redundant compressor available	Compressor out of service for repairs
	b. Tube side rupture					
Engineered Safety Features Room Air Recirculation and Cooling Units (2)	Tube side rupture	Surge tank low level alarm, makeup valve open alarm and low flow indication; all CRI	Loss of water from header	Isolate coils	Coil out of service for repairs	Coil not required during normal operation
Quench Tank Heat Exchanger	Tube side rupture	Surge tank low level alarm, makeup valve opens alarm; all CRI	Loss of water from header A	Isolate heat exchanger	Heat exchanger out of service for repairs	
Inter system LOCA Relief Valves	Inadvertent opening	Reduction in RBCCW inventory; surge tank low level alarm	Reduction in RBCCW inventory	Isolate non-essential loop in containment; repair valve	One header out of service	Relief valves are installed to mitigate a beyond design basis event and address NRC IN 89-54

TABLE 9.4-3 MINIMUM REQUIRED RBCCW FLOW TO ESSENTIAL SAFETY-RELATED LOADS LOCA INJECTION AND RECIRCULATION OPERATIONS

Essential Load (Train A Shown)	Condition IV LOCA Injection Flow (gpm)	Condition V	
		LOCARecirculation (Spent Fuel Pool Cooling Isolated) (Flow gpm)	LOCA Recirculation (Spent Fuel Pool Cooling Restored) Flow (gpm)
CAR Cooler A	2000	1600	1400
CAR Cooler C	2000	1600	1400
SDC Hx A	Isolated	2000	1800
HPSI Pump Seal Cooler A	15	15	15
HPSI Pump Seal Cooler B	15	15	15
LPSI Pump Seal Cooler A	3	3	3
CS Pump Seal Cooler A	11	11	11
ESF Room Cooler A	59	59	59
SFP Cooler A	Isolated	Isolated	1100

FIGURE 9.4–1 P&ID RBCCW SYSTEM RBCCW PUMPS AND HEAT EXCHANGERS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.4–2 RBCCW SYSTEM SPENT FUEL POOL AND SHUTDOWN HEAT EXCHANGERS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.4–3 P&ID RBCCW. SYSTEM CONTAINMENT SPRAY PUMP AND SAFETY INJECTION PUMP SEAL COOLERS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.4–4 P&ID RBCCW. SYSTEM REACTOR COOLANT PUMP, THERMAL BARRIERS AND LUBE OIL COOLERS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

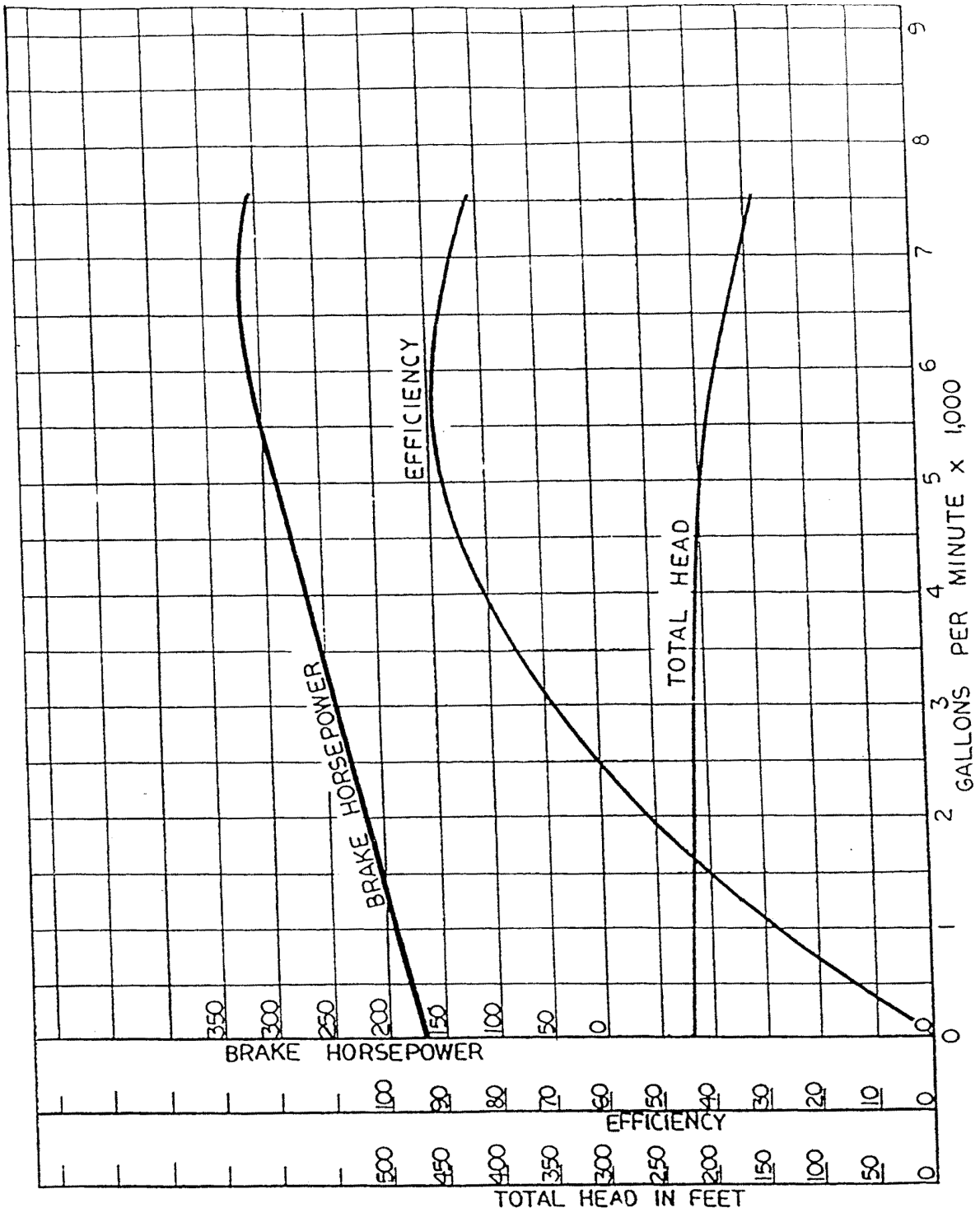
FIGURE 9.4–5 P&ID RBCCW. SYSTEM CONTAINMENT AIR RECIRCULATION AND COOLANT UNIT

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.4–6 P&ID DIAGRAM RBCCW. SYSTEM REACTOR VESSEL SUPPORT CONCRETE COOLING COILS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.4–7 REACTOR BUILDING CLOSED COOLING WATER PUMP PERFORMANCE CURVE



9.5 SPENT FUEL POOL COOLING SYSTEM

9.5.1 DESIGN BASES

9.5.1.1 Functional Requirements

The function of the spent fuel pool cooling system is to remove decay heat generated by spent fuel assemblies stored in the pool by limiting the temperature of the borated pool water to an acceptable level, thereby ensuring the cladding integrity of stored spent fuel assemblies. In order to ensure that spent fuel pool water temperature limits and fuel integrity are maintained two different refueling operations are analyzed: (1) Normal refueling involves movement of a maximum of 80 fuel assemblies or the equivalent heat load from the reactor vessel to the spent fuel pool, and (2) A full core offload of all 217 fuel assemblies from the reactor vessel to the spent fuel pool. A full core offload comprises of an end-of-cycle full core offload or a mid-cycle emergency full core offload. In the event of a full core offload, the spent fuel pool water temperature will be limited to the Technical Requirements Manual (TRM) limit of 150°F. This would utilize one train of the shutdown cooling system for the limiting emergency full core offload. Under less limiting full core offload conditions, the Spent Fuel Pool Cooling system or Spent Fuel Pool Cooling supplemented by the Shutdown Cooling system may be used, provided that a Spent Fuel Pool temperature of less than 150°F is maintained

The spent fuel pool cooling system is provided with a cleanup system for maintaining the purity and clarity of water in the spent fuel pool, refueling pool and the refueling water storage tank after completion of refueling operations. The cleanup systems limit operating personnel radiation exposures from these sources by reducing the concentrations of radioactive constituents introduced into these waters.

9.5.1.2 Design Criteria

The spent fuel pool cooling system is designed in accordance with the following criteria:

a. The system shall be designed to ensure adequate decay heat removal capability under normal and postulated accident conditions.

b. The system shall be designed with the ability to permit appropriate periodic inspection and testing of components important to decay heat removal.

c. The system shall be provided with suitable shielding for radiation protection.

d. Reliable and frequently tested monitoring equipment to detect conditions that may result in loss of decay heat removal shall be provided.

e. The system shall be designed with an adequate spent fuel pool makeup system with appropriate provisions for a backup system for filling the pool.

f. Design shall prevent significant reduction in fuel storage cooling water inventory due to equipment failure, maloperation or accident conditions.

g. Appropriate containment and cleanup system for spent fuel pool cooling water shall be provided.

9.5.2 SYSTEM DESCRIPTION

9.5.2.1 System

The spent fuel pool cooling system, shown in Figure 9.5–1 is designed to remove decay heat generated by stored spent fuel assemblies by circulating the borated pool water through a heat exchanger unit consisting of two heat exchangers in parallel. Cooling water for the heat exchangers is supplied by the RBCCWS as described in Section 9.4. The two motor-driven pumps take suction from the spent fuel pool at elevation 35 feet 6 inches, one foot below the normal operating level of elevation 36 feet 6 inches and at elevation 13 feet 0 inches through the lower suction line. The normal depth of the spent fuel pool is approximately 38.5 feet of water. The spent fuel pool cooling water is returned to the bottom of the pool through three headers which penetrate the pool walls approximately one foot below the normal operating water level, on the opposite side of the pool from the supply headers to provide convective circulation of the pool water through the stored fuel assemblies.

There is a make-up connection attached to the emergency make-up water line for a portable diesel driven pump to deliver make-up water to the Spent Fuel Pool. This connection is a defense-in-depth design feature that is available for coping with an extended loss of AC power (ELAP) event. The location of the BDB SFP FLEX make-up connection is shown on Figure 9.5–1, Sheet 2.

Normal Refueling (fuel shuffle)

The first design basis for the spent fuel pool cooling (SFPC) system is for normal refueling (fuel shuffle). Fuel movement to the SFP is assumed to start between 110 and 160 hours of decay, and proceeds at an average rate of 6 fuel assemblies per hour. The case analyzed is refueling at the end of plant life to maximize fuel inventory and therefore decay heat of spent fuel in the SFP. Decay heat calculations are performed with ORIGEN-ARP. This normal refueling, started at 160 hours, produces a maximum conservative heat loading of 14.52×10^6 BTU/hr. With both trains of SFPC in service and 85°F RBCCW spent fuel pool water temperature will be maintained to less than 131°F.

A normal refueling with fuel movement starting after 110 hours of decay, and proceeding at an average rate of 6 assemblies per hour over the refueling is acceptable as long as RBBCW temperature is maintained at or below 75°F. The resulting maximum conservative heat loading in the spent fuel pool is 16.18×10^6 BTU/hr. In the event that more than 81 fuel assemblies need to be moved to the spent fuel pool the evolution may still be defined as a Fuel Shuffle as long as the

entire core is not offloaded and the maximum heat load in the spent fuel pool remains less than or equal to this value.

With RBCCW temperature maintained at or below 75°F the spent fuel pool cooling system can maintain the spent fuel pool temperature at approximately 125°F with both pumps and heat exchangers in service.

A single active associated failure analysis of the spent fuel pool cooling system shows the limiting failure to be failure of RBCCW valve 2-RB-8.1A or 8.1B. Failure of either valve closed results in loss of cooling flow to the associated SFPC heat exchanger. This failure is more limiting than the loss of a SFPC pump. Failure of RBCCW valve 2-RB-8.1A or 8.1B in the closed position would result in a SFP bulk water temperature greater than 150°F but less than 200°F, if the failure occurred at maximum heat load. Should this limiting single failure occur, 1 train of the shutdown cooling (SDC) system may be used to cool the SFP, or the SDC system may be used to supplement the SFPC system. Used in this fashion, depending on heat load, between 0 to 1000 gpm of SDC flow would be able to maintain SFP water temperature ≤ 150°F. See Section 9.3 for a description of the SDC. Since the SDC system will eventually be re-aligned for its ECCS requirements for Mode 4, the SFPC system must be capable of sustaining a single failure without assistance from SDC upon entry into Mode 4. Analysis shows that once the SFP heat load drops to 10.16×10^6BTU/HR, even if the limiting SFPC single failure occurs, the SFPC system will be able to maintain SFP bulk water temperature ≤ 150°F. At least 616 hours of being subcritical or a heat load of 10.16×10^6 BTU/HR is required prior to re-entry into Mode 4 from a refueling outage.

Key SFP Cooling and SDC parameters used in this analysis are:

- Credited SFP cooling flow is 850 gpm per SFP cooling pump.
- Credited RBCCW flow to each SFP cooling heat exchanger is 1100 gpm.
- Credited RBCCW flow to SDC heat exchangers is 3500 gpm.
- Credited RBCCW temperature to SFP and SDC heat exchangers is 75°F for fuel movement starting after 110 hours of decay and 85°F for fuel movement starting after 160 hours of decay.
- SFP and SDC heat exchangers have a 1% tube plugging penalty.

Periodic maintenance of the Spent Fuel Pool Cooling system and associated supporting systems is required. During the performance of maintenance activities it may be necessary to remove system components and portions of various support systems from service. While these components and systems are out of service, the ability of the Spent Fuel Pool Cooling system to meet single failure criteria will be limited. These maintenance activities do not conflict with system design or licensing basis. Suitable redundancy is ensured by the shutdown risk program.

Therefore the requirement to maintain the SFP ≤ 150°F can be met during modes 1through 4 with a single train of SFP Cooling. During refueling, for a fuel shuffle, under limiting postulated conditions, 2 trains of SFP cooling are sufficient to maintain SFP bulk water temperature ≤150°F. During refueling, for a fuel shuffle, under limiting postulated conditions, 1 train of SFP cooling will need to be supplemented by Shutdown Cooling to maintain SFP bulk water temperature ≤ 150°F. A single train of Shutdown Cooling is sufficient to remove all decay heat from both the reactor vessel and the SFP. Eventually during refueling, 1 train of SFP cooling is sufficient to maintain SFP bulk water temperature ≤ 150°F. For the final refuel outage, this point is reached 616 hours after shutdown or when the SFP heat load is at or below 10.16×10^6 BTU/HR, when 1 train of SFP cooling is sufficient to maintain SFP bulk water temperature ≤ 150°F.

Full Core Offload as Normal Refueling

The second design basis for the SFP cooling system is for a full core offload as normal refueling. Fuel movement to the SFP is assumed to start after 110 hours with a RBCCW temperature of less than or equal to 75°F or after 160 hours with RBCCW temperature of less than or equal to 85°F. The fuel movement is assumed to proceed at an average rate of 6 assemblies per hour until all 217 fuel assemblies are in the SFP and the reactor vessel is empty of fuel. The case analyzed is the final core offload at the end of plant life to maximize the fuel inventory, and therefore decay heat, of spent fuel in the pool. At end of plant life, decay heat removal is needed for a total of 1343 fuel assemblies and 3 consolidated fuel storage boxes. The number of fuel assemblies analyzed in the heat load analysis bounds the number of fuel assemblies which can be physically stored in the SFP. Decay heat calculations are performed with ORIGEN-ARP.

The last full core offload at end of plant life, calculated in the above manner, produces a maximum conservative heat loading of 34.59 MBTU/hr with a 110 hour core hold time and produces 30.90 MBTU/hr with a 160 hour core hold time. With 1 train of SDC available during the core offload to cool both the reactor vessel and the core, spent fuel pool water temperature will be maintained to less than 150°F.

During the refueling evolution, cooling is transitioned from the spent fuel pool cooling system to the shutdown cooling system when the temperature in the spent fuel pool is observed to rise. This transition begins before challenging the conservatively established normal operating pool temperature limit of 120°F.

One train of SDC is assumed to be capable of delivering 3000 gpm, however the split of SDC flow going to the reactor vessel, or cooling the SFP will change during the core offload as the heat load shifts from the core to the SFP. With this limiting heat load, at the start of the offload most of the SDC flow will go to the reactor vessel (approximately 2000 gpm), with lesser SDC flow going to the SFP (approximately 1000 gpm). By the end of the offload most of the SDC flow will go to the SFP (up to 1900 gpm), with lesser SDC flow going to the reactor vessel (>1000 gpm). See Section 9.3 for a description of the SDC system.

Key SDC parameters used in this analysis are:

- Credited SDC flow is 3000 gpm per LPSI pump, with a maximum of 1900 gpm diverted to cooling the SFP.

- Credited RBCCW flow to SDC heat exchangers is 3500 gpm.

- Credited RBCCW temperature to SDC heat exchangers is 75°F for fuel movement starting after 110 hours of decay and 85°F for fuel movement after 160 hours of decay.

- SFP and SDC heat exchangers have a 1% tube plugging penalty.

While the above analysis is for the limiting heat load case at end of plant life, it is acceptable to use SFP cooling, or SFP cooling supplemented by lesser amounts of SDC, during any portion of the core offload, provided that spent fuel pool bulk water temperature can be maintained below 150°F.

Full core offload as a normal activity is acceptable because there are 2 redundant trains of Shutdown Cooling (SDC). One train of SDC is sufficient to ensure that the Spent Fuel Pool (SFP) can be maintained at a bulk water temperature ≤ 150°F even with the entire core offloaded to the SFP. The other train of SDC is available as backup, should there be a failure in the operating SDC train. In the event that maintenance activities during an outage cause the redundant train of SDC to be unavailable, the Containment Spray (CS) pump has sufficient capacity (Design Temperature of 300°F and Design Head of 390 feet at 1600 GPM) in this circumstance to cool the SFP and to ensure that SFP temperature is maintained ≤150°F. The CS pump has no required function with the core defueled. The CS pump supplemented by a SFP cooling train (if required) will be the means of backup to SDC during SDC maintenance activities. Suitable redundancy is ensured per the Shutdown Risk program.

Therefore, for a full core offload as normal refueling, the system is conservatively designed to maintain the water in the spent fuel pool water below 150°F during all normal conditions or single active failure conditions.

Emergency Full Core Offload

The third design basis for the spent fuel pool cooling system is for an emergency full core offload. Fuel movement to the SFP is assumed to start after 110 hours with a RBCCW temperature of less than or equal to 75°F or after 160 hours with RBCCW temperature of less than or equal to 85°F. The fuel movement is assumed to proceed at an average rate of 6 assemblies per hour until all 217 fuel assemblies are in the SFP and the reactor vessel is empty of fuel. The case analyzed is during the final fuel cycle at end of plant life to maximize the fuel inventory, therefore decay heat, of spent fuel in the pool. At end of plant life, decay heat removal is needed for a total of 1343 fuel assemblies and 3 consolidated fuel storage boxes. The number of fuel assemblies analyzed in the heat load analysis bounds the number of fuel assemblies which can be physically stored in the spent fuel pool. Decay heat calculations are performed with ORIGEN-ARP. Significant conservatism is used in the calculation of the decay heat values. A conservatively short decay time of 36 days is chosen for the previous batch of discharge fuel, and 400 days decay for the previous discharge batch, to maximize decay heat of fuel stored in the pool.

The emergency full core offload during the last fuel cycle of operation, calculated in the above manner, produces a maximum conservative heat loading of 39.40 MBTU/hr with a 110 hour core hold time and produces 35.72 MBTU/hr with a 160 hour core hold time. There is no single failure requirement for emergency core offload calculations, per our current licensing basis. With two trains of SDC available during the emergency core offload, approximately 1900 gpm of SDC flow is available to cool the SFP during the entire core offload. With 1900 gpm of SDC flow available to cool the SFP, the SFP water temperature will be maintained to less than 150°F. See Section 9.3 for a description of the SDC system.

Key SDC parameters used in this analysis are:

- Credited SDC flow is 3000 gpm per LPSI pump, with a maximum of 1900 gpm diverted to cooling the SFP.
- Two trains of SDC are assumed to be available.
- Credited RBCCW flow to SDC heat exchangers is 3500 gpm.
- Credited RBCCW temperature to SDC heat exchangers is 75°F for fuel movement starting after 110 hours of decay and 85°F for fuel movement after 160 hours of decay.
- SFP and SDC heat exchangers have a 1% tube plugging penalty.

While the above analysis is for the limiting heat load case at end of plant life, it is acceptable to use SFP cooling, or SFP cooling supplemented by lesser amounts of SDC, during any portion of the emergency core offload, provided that SFP water can be maintained below 150°F.

Therefore, for an emergency full core offload, the system is conservatively designed to maintain the water in the SFP below 150°F.

Also since the emergency core offload represents the limiting heat load in the SFP, an additional analysis is performed to verify that the local water temperatures and local fuel cladding temperatures were less than boiling. Results of that analysis show that at the maximum emergency core offload heat load, the maximum local SFP water temperature along the fuel assembly is less than 178°F, and the maximum local fuel clad temperature is less than 234°F. These temperatures are below the local boiling temperature, at the top of the fuel assemblies, of 240°F.

Normal makeup to the spent fuel pool water inventory is supplied from the primary water storage tank at a rate of 50 gpm. The 50 gpm makeup capability is adequate to provide water at a rate greater than normal water loss due to evaporation and any system leakage. The maximum makeup capability of this permanently installed system is approximately 200 gpm. Backup makeup for the pool is from the RWST by using one Refueling Pool Purification Pump to transfer the water at a rate of approximately 125 gpm. The RWST contents can also be transferred by the use of one LPSI pump and the interconnecting piping between the spent fuel pool cooling system and shutdown cooling system. A makeup capability of 3000 gpm for 15 minutes is possible from this

source. Makeup at approximately 100 gpm can be supplied by an Auxiliary Feedwater Pump taking suction from the Condensate Storage Tank. An alternate backup source for makeup is obtained from the fire protection system by using temporary hose connections. Flow is at a rate of 200 gpm.

All leakage and drains from the spent fuel pool cooling system components are collected in the open drains system and processed as aerated liquid waste as described in Section 11.1.3. The maximum system leakage for an extended period of time for which adequate processing by the RWS is possible is 50 gpm.

Sufficient monitoring equipment is provided to detect and alert operating personnel to conditions that may result in loss of decay heat removal capability. Table 9.5-1 lists the monitoring equipment provided.

A low-flow alarm will alert operating personnel that there is either a low pool water level or one or both cooling water pumps has failed to operate. Excessive radiation levels are detected by local area radiation monitors as described in Section 7.5.6.

To prevent significant reduction in the fuel storage cooling water inventory, all connections to the pool penetrate the pool walls near the normal operating water level so that the pool cannot be gravity drained by leaking pumps, valves, etc. The cooling water return piping which extends to the pool bottom is provided with anti-siphon devices.

Section 5.4.3 describes the way the spent fuel pool has been designed to prevent cooling water inventory loss. The spent fuel pool liner leak monitoring and detection system consists of a series of channels welded behind each seam of the pool liner. Any leakage through the pool liner seams is collected in the channels which are piped to the open drain system. The drain header contains a level switch which annunciates in the main control room to alert operating personnel to any leakage. The drain header is provided with valves to isolate the leak and is of a small pipe size (0.5 inch) to prevent significant water loss prior to isolation. The location of the leaks in the pool liner seams is accomplished by pressurizing the channels with air and observing air bubbles rising to the pool water surface. Sections 5.4.3.1.9 and 9.8.4 describe the provisions incorporated in the station design to preclude and/or limit the consequences of a dropped fuel shipping cask.

Boron concentration in the spent fuel pool water is maintained consistent with Technical Specification requirements. The design of the spent fuel storage racks to preclude criticality and the bases for the safe geometry are described in Section 9.8.2.

A cleanup system consisting of pumps, filters, and a demineralizer is provided to maintain the purity and clarity of the water in the spent fuel pool, refueling pool, and the RWST after completion of reactor refueling operations. The cleanup system is designed to remove corrosion and fission products introduced into these waters by leaking fuel assemblies and mixing with reactor coolant during refueling operations. The purity and clarity of these waters are maintained to limit operating personnel doses. The radiation levels are closely monitored during refueling operations to establish the allowable exposures times for personnel in accordance with 10 CFR Part 20.

To facilitate the removal of accumulated dust from the surfaces of both the spent fuel pool and refueling pool, skimmer assemblies with a pump and filters are provided for each pool.

9.5.2.2 Components

A description of the components of the spent fuel pool cooling system is given in Table 9.5-2.

9.5.3 SYSTEM OPERATION

9.5.3.1 Normal Operation

During normal operation of the pool, both pumps and heat exchangers are in continuous service. As the decay heat generated by the spent fuel decreases, one pump is stopped. As the decay heat further decreases, operation of the system is intermittent as required to limit the pool water temperature to less than or equal to 150°F.

As described in Section 9.5.2.1, during refuel operations, the spent fuel cooling system, with assistance from the shutdown cooling system when needed, is capable of maintaining spent fuel pool water temperature below the Technical Requirements Manual (TRM) limit of 150°F for a normal refueling through (and including) the end of plant life refueling. This includes allowance for a single active failure of the spent fuel pool cooling system, or shutdown cooling system, as appropriate.

As described in Section 9.5.2.1, the spent fuel pool heat load must decay to a value of 10.16×10^6 BTU/HR, for the SFP cooling system to be capable of withstanding the worst single active failure, and maintain spent fuel pool water temperature ≤ 150°F. For the time period before the SFP heat load drops to 10.16 MBTU/hr, should the limiting single failure occur, 1 train of the shutdown cooling (SDC) system may be used to cool the SFP, or the SDC system may be used to supplement the SFP cooling system, to maintain spent fuel pool water temperature ≤ 150°F.

As a result of this need to depend on the SDC for potential SFP cooling single failure per Technical Requirements Manual (TRM), entry into Mode 4 following a refueling is not allowed until either:

(a) 616 hours has passed since subcriticality for the fuel bundles remaining in the spent fuel pool which were discharged from the previous refueling, of ≤ 81 fuel bundles, or

(b) the heat load to the SFP is less than 10.16×10^6 BTU/hr. At a SFP heat load of 10.16×10^6 BTU/hr, adequate cooling is available to maintain the SFP bulk water temperatures less than or equal to 150°F should a single failure occur in the SFP cooling system.

These TRM limits assure that the shutdown cooling system would be available for supplemental cooling and not committed to its emergency core cooling system functions.

While water temperatures up to 150°F are allowed in the spent fuel pool, the demineralizers in the spent fuel pool cleanup system should not be exposed to spent fuel pool water temperatures in excess of 140°F, to prevent demineralizer resin degradation. Alarms and procedural controls are used to prevent spent fuel pool water from reaching the demineralizer resin prior to the temperature reaching 140°F.

A portion of the spent fuel pool cooling water flow, approximately 125 gpm, is passed through the cleanup system. Operation of this system as well as the skimmer system is intermittent as required to maintain the clarity and purity of the spent fuel pool water. During refueling operations, the refueling water purification pumps and spent fuel pool filters may be used for purification of the water in the refueling pool. The demineralizer is used as required for purification service on the spent fuel pool or refueling pool. At the completion of the refueling operations, the pumps are used for transferring the remaining refueling water from the pool to the RWST after the water level is lowered to the elevation of the reactor vessel flange.

9.5.3.2 Abnormal Operation

If a mid-cycle emergency full core offload is placed in the spent fuel pool, the cooling capacity for this additional spent fuel is provided by connecting the spent fuel pool cooling system with the shutdown cooling system, if additional cooling is required to limit the pool water temperature. The shutdown cooling system is placed into service by manual initiation.

9.5.3.3 Emergency Conditions

In the event that a serious leak develops in the spent fuel pool liner, makeup water is supplied to the pool from the primary makeup water system by manual initiation from the 14 foot 6 inches level of the auxiliary building. Should the leakage exceed the 50 gpm normal makeup capability, additional makeup is available from the RWST via the refueling water purification system, and the fire protection system by temporary hose connections.

Following a postulated LOCA, RBCCW flow to the spent fuel pool cooling system is stopped by the automatic closing of valves in the cooling water discharge piping from the heat exchangers on SIAS. The availability of cooling water, normally supplied to the spent fuel pool system, allows for greater heat removal capability in the engineered safety features components. As discussed later in Section 9.5.4.1, spent fuel pool cooling loss for a period of 9.76 hours will raise the pool water temperature to 212°F. RBCCW cooling to the spent fuel pool cooling heat exchangers is restored 4 to 5 hours after the start of the postulated LOCA when the heat load on the RBCCW system is substantially reduced.

For the normal case where spent fuel is stored in the spent fuel pool, it is unlikely that a seismic event will cause loss of cooling water flow. The lines from the spent fuel pool to the suction of the spent fuel pool cooling pumps and from the spent fuel pool heat exchangers to the spent fuel pool have been designed and analyzed to Seismic Category I requirements and the remainder of the system, including the spent fuel pool cooling pumps, heat exchangers and their connecting piping, has been analyzed and found to be in accordance with Seismic Category I requirements. This provides assurance that cooling water would be available from the spent fuel pool cooling system.

In the event that the cooling water flow is lost, makeup water is available from one of the two seismic Category I sources. The seismic category 1 makeup sources for the spent fuel pool are:

The auxiliary feedwater system (AFW) interconnection to the spent fuel pool cooling system. This allows the use of any AFW pump and the inventory of the condensate storage tank as makeup to the SFP. This path is nominally designed to deliver 100 gpm by a flow orifice (FO-8954).

The low-pressure safety injection (LPSI) system interconnection to the spent fuel pool cooling system. This allows the use of a LPSI pump and the inventory of the refueling water storage tank (RWST). The LPSI interconnection is also capable of taking water from the refueling cavity during outages when the inventory from the RWST has been utilized to fill the refueling cavity.

The events or malfunctions that may require makeup to the spent fuel include the long-term loss of spent fuel pool cooling heat removal capability, the failure of a steam generator nozzle dam with the fuel transfer tube isolation valve (2-RW-280) open or the refueling pool cavity seal with the fuel transfer tube isolation valve (2-RW-280) open. The catastrophic failure of the cavity seal is not considered to be a credible event, but has been analyzed and is dealt with the unit procedures. Failures of the spent fuel pool structure are not considered to be credible. The aforementioned events or malfunctions could allow the spent fuel to boil thus requiring makeup. The maximum required makeup would be approximately 85 gallons per minute which would be for core offload conditions.

In the case where a full core offload is stored in the spent fuel pool, it is improbable that cooling water flow will be lost entirely due to the redundancy of the cooling equipment. In the event that the cooling water flow is lost while a full core offload is being stored in the spent fuel pool, makeup water may be available from the refueling water storage tank, refueling cavity, or the condensate storage tank.

9.5.4 SYSTEM AVAILABILITY AND RELIABILITY

9.5.4.1 Special Features

The spent fuel pool purification system can be used as a backup to maintain the RWST contents above 50°F. However, this flow path has been administratively prohibited in certain plant operating modes and when the RWST is credited as a borated water source due to seismic/non-seismic piping interconnection concerns.

The spent fuel pool cooling pumps and heat exchanger units can be supplemented by the shutdown cooling system.

The most serious failure of the system would be the complete loss of the spent fuel pool water inventory. To protect against this possibility, all connections to the pool enter near to or above the pool operating water level, so that the pool cannot be gravity drained through leaking valves, piping and equipment maloperation. Piping which extends to the bottom of the pool is provided with antisiphoning devices.

Reliable monitoring equipment, as described in Section 9.5.2.1 is provided to detect and alert plant operating personnel to any conditions which could result in loss of decay heat removal capability.

During refueling, following the loss of all cooling, the minimum times to reach boiling conditions in the SFP occur when the core offload begins at 110 hours after shutdown. During Normal Refueling, Full Core Offload as Normal and Emergency Full Core Offload the minimum times to reach boiling are conservatively determined to be in excess of 6.4 hours, 3.1 hours and 2.8 hours, respectively.

To conservatively estimate the time for the SFP bulk water temperature to reach the SFPC system temperature of 200°F in Modes 1 through 4, the SFP bulk water temperature is assumed to be 150°F at 616 hours after subcriticality and SFP heat load is at or below 10.16×10^6 BTU/HR. If loss of SFP cooling is assumed to occur at that time, the SFP bulk water temperature will reach the SFPC system temperature of 200°F in 7.87 hours or 212°F in 9.76 hours. Alternately, for SFP starting bulk water temperatures of 130°F and 117.7°F, it takes 11 hours and 12.96 hours respectively to heat the SFP to 200°F.

Sufficient time is available for operating personnel to locate and correct the malfunction.

That portion of the spent fuel pool cooling system that is used in seismic Category I makeup consists of the section of the spent fuel pool heat exchangers outlet piping downstream of check value 2-RW-8 as shown on Figure 9.5–1. HCC-11 has been designed and analyzed to seismic Category I requirements and additionally all connections have been analyzed up to and including the first seismic restraint beyond the isolation value in each connecting line. Following is a list of the line and valve numbers that constitute the Category I makeup:

Piping

10" - HCC-11, 6" - HCC-11

Valves

2-RW-8, 2-RW-15, 2-RW-65, 2-RW-66, 2-RW-67, 2-RW-71, 2-RW-76, 2-RW-119, 2-RW-213, 2-RW-222

9.5.4.2 Tests and Inspections

Components of the spent fuel pool cooling system are nondestructive tested in accordance with the applicable codes as listed in Table 9.5-2.

All pumps in the spent fuel pool cooling system were manufacturer shop performance tested to demonstrate compliance with design head and capacity requirements. A performance curve for the spent fuel pool cooling pumps is shown in Figure 9.5–2.

Prototype filter cartridge testing was conducted by the filter manufacturer confirming a filter efficiency of 92 weight percent when tested with fine Arizona air dust.

All system components are visually inspected and manually adjusted if required to ensure proper installation and arrangement.

The completely installed spent fuel pool cooling system was pre operation tested prior to startup. The detailed test procedure is described in Section 13.

The components of the spent fuel pool cooling system are located in a low-radiation area which permits access for periodic testing and maintenance.

9.5.5 REFERENCES

9.5-1 Letter from W. G. Counsil to Director of Nuclear Reactor Regulation, “Millstone Nuclear Power Station, Unit Number 2 Control of Heavy Loads Near Spent Fuel,” dated July 17, 1978.

TABLE 9.5-1 SPENT FUEL POOL MONITORING EQUIPMENT

Equipment	Function
Spent fuel pool temperature instrumentation	Provides continuous monitoring and recording of pool water temperatures by main control room personnel. Annunciates high temperature alarm in control room.
Spent fuel pool level instrumentation	Annunciates low pool water level alarm in main control room. * Continuous wide range level indication is provided remotely in the Cable Vault and East 480V Switchgear Rooms of the Auxiliary Building.
Spent fuel pool cooling water flow instrumentation	Annunciates low spent fuel pool cooling water flow alarm in main control room.
Spent fuel pool heat exchanger outlet instrumentation	Annunciates high temperature alarm in main control room.

* It is not necessary for this alarm to sound locally since personnel in the area would either visually notice an abnormal pool level or be notified of the abnormal pool level by control room personnel. (Reference 9.5-1).

TABLE 9.5-2 COMPONENTS OF SPENT FUEL POOL COOLING SYSTEM

Cooling Pumps

Type	Horizontal centrifugal
Quantity	2
Design temperature (°F)	200
Design head tdh (feet)	50
Design capacity, each (gpm)	850
NPSH available (feet)	60
Minimum NPSH required (feet)	10
Horsepower	15
Materials	
Case	ASTM A351, Gr CF8M
Impeller	ASTM A351, Gr CF8M
Shaft	ASTM A351, Gr CF8M
Codes and Standards	
Draft ASME Code for Pumps and Valves for Nuclear Service, Class III (1968 Edition)	
Seismic design class	I
Design integrated radiation dosage (rad)	10^6
Supplier	Gould
Heat Exchangers Unit Original Design	
Type	U-tube with single shell pass
Number	2 in parallel
Design heat transfer rate, each (Btu/hr)	5.5×10^6
Heat transfer area, each (ft^2)	589
Design RBCCW temperature, in/out (°F)	85/95
Design spent fuel pool water temperature, in /out (°F)	120/107
Tube design, spent fuel pool water	
Design pressure (psig)	125
Design temperature (°F)	200
Maximum Flow (gpm)	1050
Design Flow (gpm)	850
Pressure drop at 850 gpm flow (psi)	7
Materials	ASTM A249 Type 304
Relief valve setting (psig)	125

TABLE 9.5-2 COMPONENTS OF SPENT FUEL POOL COOLING SYSTEM (CONTINUED)

Shell design (RBCCW)	
Design pressure (psig)	150
Design temperature (°F)	130
Maximum Flow (gpm)	1500
Design Flow (gpm)	1100
Pressure drop at 1100 gpm (psi)	4
Materials	ASTM A106
Relief valve setting (psig)	150
Codes and Standards	ASME Section VIII TEMA "R"
Seismic design class	I
Supplier	Struthers Nuclear

Spent Fuel Pool Demineralizer

Type	Mixed bed, nonregenerable
Number	1
Design pressure (psig)	100
Design temperature (°F)	150
Normal operating pressure (psig)	75
Normal operating temperature (°F)	107-120
Resin volume (ft^3)	42
Design flow (gpm)	132
Normal operating flow (gpm)	125
Material	ASTM A-240 Type 304
Code Edition)	ASME Section III for Class 3 Components (1971
Seismic design class	II
Supplier	Illinois Water Treatment

Spent Fuel Pool Filters

Type	Disposable cartridge
Number	2
Design pressure (psig)	200
Design temperature (°F)	250
Normal operating pressure (psig)	75
Normal operating temperature (°F)	107-120

TABLE 9.5-2 COMPONENTS OF SPENT FUEL POOL COOLING SYSTEM (CONTINUED)

Filter rating (micron)	3
Design Filter efficiency (%)	80 minutes
Design flow (gpm)	132
Normal operating flow (gpm)	125
Material	
Vessel	ASTM A 312 Type 304
Internals	Type 304 SS, Micarta
Cartridges	Ethylene propylene
Seismic design class	II
Code	ASME Section III Class C (1968 Edition)
Supplier	Filterite Corporation

Refueling Water Purification Pumps

Type	Inline centrifugal
Number	2
Design temperature (°F)	150
Design head, tdh (feet)	165
Design capacity (gpm)	125
NPSH available (feet)	26
Minimum NPSH required (feet)	5
Horsepower	15
Materials	
Case	ASTM A351 Gr CF8M
Impeller	ASTM A351 Gr CF8M
Shaft	ASTM A276 Type 316
Codes and Standards	ASME Section III for Class 3 Components (1971 Edition)
Seismic design class	II
Design integrated radiation dosage over lifetime (rad)	10^6
Supplier	Ingersoll Rand

TABLE 9.5-2 COMPONENTS OF SPENT FUEL POOL COOLING SYSTEM (CONTINUED)

Spent Fuel Pool and Refueling Pool Skimmer Pumps

Type	Inline centrifugal
Number	2
Design temperature (°F)	175
Design head, tdh (feet)	75
Design capacity (gpm)	60
NPSH available (feet)	46
Minimum NPSH required (feet)	6
Horsepower (hp)	3
Materials	
Case	ASTM A351 Gr CF8M
Impeller	ASTM A351 Gr CF8M
Shaft	ASTM A276 Type 316
Code	ASME Section III for Class 3 Components (1971)
Seismic design class	II
Design integrated radiation dosage over lifetime (rad)	10^6
Supplier	Ingersoll Rand

Spent Fuel Pool and Refueling Pool Skimmer Filters

Type	Disposable cartridge
Number	4
Design pressure (psig)	200
Design temperature (°F)	250
Normal operating pressure (psig)	50
Normal operating temperature (°F)	120
Filter rating (micron)	10
Design filter efficiency (%)	80
Design flow (gpm)	132
Normal operating flow (gpm)	60
Material	
Vessel	ASTM A312 Type 304
Internals	Type 304SS, Micarta
Cartridges	Ethylene propylene

TABLE 9.5-2 COMPONENTS OF SPENT FUEL POOL COOLING SYSTEM (CONTINUED)

Code	ASME Section III Class C (1968 Edition)
Seismic design class	II
Supplier	Filterite Corporation

Piping and Valves

Piping

Material	ASTM A-312 Type 304
Design pressure (psig)	100
Design temperature (°F)	150

Joints:

2.5 inches and larger -Butt welded except at flanged equipment

Codes:

Fabrication, ANSI B31.7 Class III

Testing and installation, ASME Section III for Class 3 Components (1971 Edition)

Valves

Materials 2.5 inches and larger - ASTM A351 Grade CF-8

2 inches and smaller - ASTM A182 Type F-304

Ratings 2.5 inches and larger - 150 lb. butt welded ends.

2 inches and smaller - 600 lb. socket welded ends.*

Code Draft ASME Code for Pumps and Valves for Nuclear Service (1968 Edition).

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* 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 600 lb, based on the design conditions of the specific application.

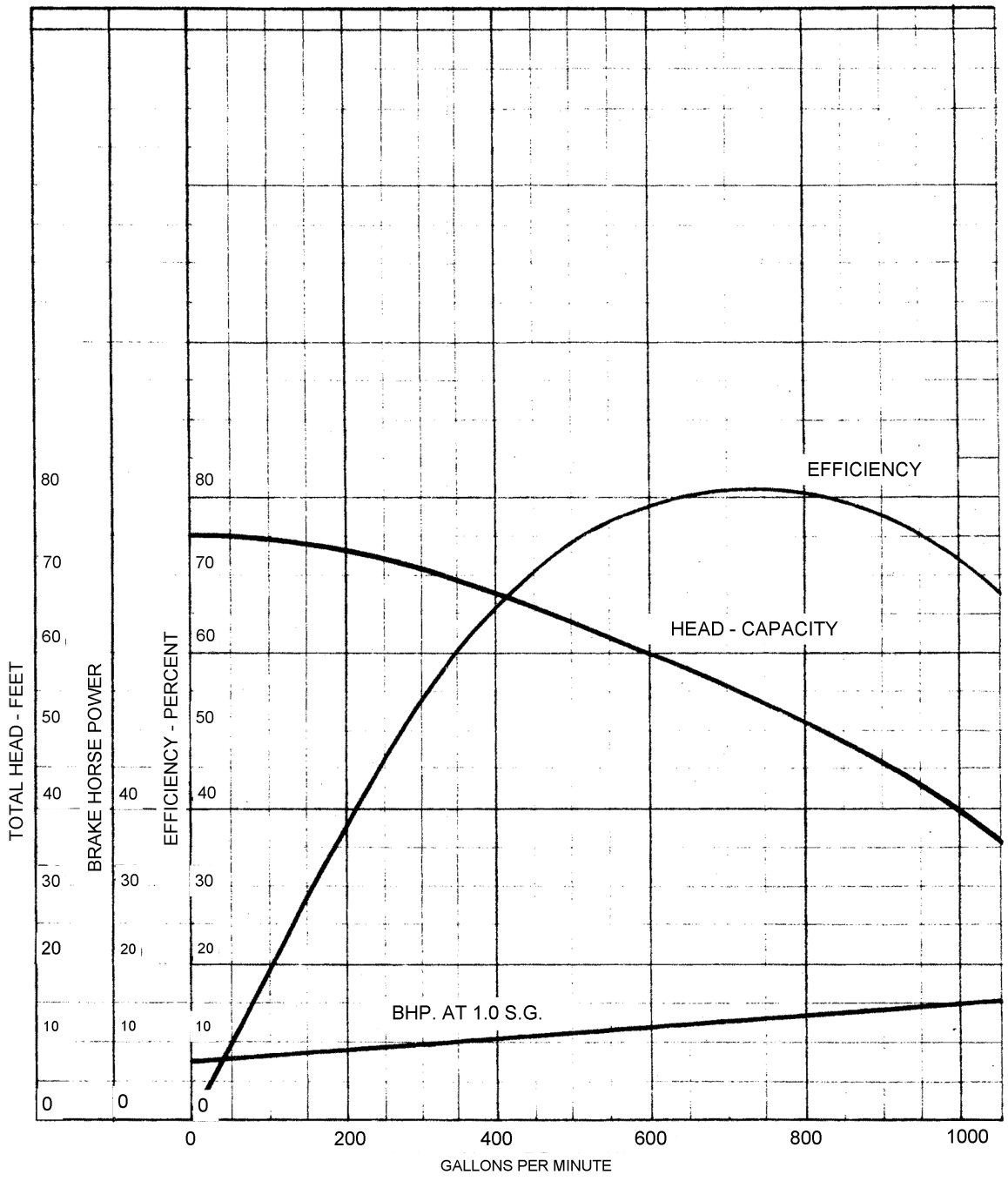
FIGURE 9.5–1 P&ID SPENT FUEL POOL CLEANING AND CLEANUP SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.5–1 P&ID SPENT FUEL POOL CLEANING AND CLEANUP SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.5–2 SPENT FUEL POOL PUMP PERFORMANCE CURVE



9.6 SAMPLING SYSTEM

9.6.1 INTRODUCTION

The sampling system consists of Sampling Stations 1 and 2, the Corrosion Monitoring Sample Station, the Waste Gas Sample Sink, and the Post Accident Sampling System (PASS). These provide the means for determining physical, chemical and radioactive conditions of process fluids, waste gas, and containment air.

Station 1 is in the sampling room located in the auxiliary building. It handles radioactive sampling of the reactor coolant system, the chemical and volume control system (CVCS), and the radioactive waste processing systems. The sample room contains the piping, valves, and cooling equipment necessary to reduce the pressure and temperature of the sample fluid or gas to acceptable levels for grab sampling or collection in sample bottles. The sample streams are radioactive or potentially radioactive and may contain boric acid. Grab samples are taken to the radiochemistry laboratory for analyses. The system also contains online hydrogen and oxygen monitors. (See Figure 9.6–1)

Station 2 is located in the turbine building. Continuous sampling by semi-automatic and manual methods can be performed. This station contains pressure reducing valves, cooling equipment, pressure, temperature and flow control regulators, valves, piping, grab sample sink, continuous O_2 and conductivity monitors, recorders, indicators, ion chromatograph, and on-line sodium monitors. (See Figure 9.6–2)

The Corrosion Monitoring Sample Station provides a means for sampling condenser hotwells for evidence of sea water in-leakage. (See Figure 9.6–3)

The Waste Gas Sample sink provides the means for sampling gasses in the volume control tank, waste gas surge tank, and the waste gas decay tanks. See Section 11.1 for additional discussion of the Gaseous Waste Processing System.

The PASS has the capability to obtain samples of reactor coolant and containment atmosphere under accident conditions. The PASS is comprised of two independent units, designated reactor coolant PASS and containment air PASS. The reactor coolant PASS is designed to obtain representative samples of reactor coolant or liquid from the containment. The containment air PASS is designed to obtain a containment air sample. Once these samples are obtained, radiological and chemical analyses can be performed.

9.6.2 REACTOR COOLANT PASS

9.6.2.1 Equipment Purpose and Description

The reactor coolant PASS is a dual module unit consisting of one sample module and one remote operating panel. Samples are trapped within the sample module. The equipment within the sample module is operated remotely via the remote operating panel. The motive force for obtaining reactor coolant samples is the differential pressure between the primary plant and the volume

control tank. Two sampling modes may be chosen, depending upon whether the sample is to be shipped off site for analysis or analyzed onsite. Samples to be analyzed off site are collected in a 2 ml shielded, removable sample chamber within the sample module. Samples to be analyzed on site are collected in shielded containers within the sample module. (See Figure 9.6–4)

9.6.2.1.1 Sample Module

This module contains the valves and components required to physically collect the sample. All components are located within a stainless steel cabinet measuring approximately 22 inches wide, by 24 inches deep, by 36 inches high, which sits on a 2 foot high stand. An exhaust blower is built into the top of the cabinet and discharges into the plant radioactive exhaust ventilation system. Doors are provided on the cabinet for access to remove samples and to perform maintenance. The module is located in the primary sample room. At this location, levels of radiation created at the module during the purge of reactor coolant through the sample lines will not result in significant exposure to the operator at the remote panel or to other individuals.

Sample influent and effluent lines are connected to the sample module. Influent samples are taken from several points at Millstone Unit Number 2 via two flow paths. Reactor coolant samples can be taken from the hot leg of the reactor coolant system. Samples of containment sump liquid can be obtained from the high pressure safety injection pumps, low pressure safety injection pumps, or the containment spray pumps. Both influent samples pass through sample coolers prior to being delivered to the sample module. Effluent lines connected to the module are directed to the volume control tank.

9.6.2.1.2 Remote Operating Panel

This module contains the sample system mimic board, electrical controls, and instrumentation readout necessary to remotely operate the sample module. The remote operating panel is located in the turbine building in an area which will have low radiation levels during an accident. The remote operating panel is connected to the sample module through electrical cables which carry power and instrumentation indication lines. Nitrogen gas supply lines, used to operate valves and purge the radioactive gas sample after sampling is completed, are hard-piped to the sample and remote modules. The face of the module is normally protected by a lockable closure to prevent damage and unauthorized operation.

9.6.2.1.3 Reactor Coolant Auxiliary Valve Operating Panel (RCAVOP)

This panel contains switches and electrical controls to operate the sample system isolation valves and the system flush valves. The RCAVOP is located above the remote operating panel.

9.6.2.1.4 Deionized Water Flushing Module

Deionized water flushing module is a modular unit designed to provide deionized water flushing capability at approximately one gallon per minute and up to 375 psig. The module is located adjacent to the remote operating panel.

The flushing system is a self contained system consisting of a water storage tank, positive displacement pump, and controls to operate the equipment. The equipment is mounted on bedplate to form a modular unit.

9.6.2.2 Design Features

Both operating and equipment failure modes are analyzed to maintain exposure ALARA. Personnel radiation exposure is minimized through the use of remote control operation, flushing techniques, and minimal sample volume and shielding.

Radiation exposure to the operator taking the sample is estimated to be well below the exposure limits defined by 10 CFR 50, Appendix A, GDC-19.

The PASS system piping downstream of the sample coolers is designed for 2500 psig and 165°F. The inherent design of pH probes limits functional usage to pressure of 250 psig. A pressure regulating valve, installed in the influent line, allows fluid to flow through the pH probe. The pH probe outer housing is designed to withstand system design pressure of 2500 psig in the event that the pressure regulating valve fails and the pH probe internals are inadvertently overpressurized.

All fluid boundary materials are of either 300 series stainless steel or Inconel.

The normal fail position of each solenoid valve was selected such that failures will still allow flushing the system to minimize radiation levels.

Solenoid valves are equipped with positive position indication at the remote operating panel.

Radioactive airborne contamination control is provided by a blower which maintains capture velocity into the sample module. This blower exhausts into existing plant ventilation.

Commercially available components are utilized to the maximum extent possible and have been selected based upon a reputation for high quality. Swagelok fittings are utilized wherever possible to be consistent with existing utility sample system components.

PASS piping is sized to maintain turbulent flow thereby minimizing crud buildup and plate-out of radioactive products.

Two sample paths and one effluent return path are available on the reactor coolant PASS providing operational redundancy.

Reactor coolant PASS solenoid containment isolation valves are operated from two control boards, located outside the control room.

The valves required to be operated during the sampling operation are operated by chemistry personnel at the RCAVOP and remote operating panel.

The reactor coolant PASS sample module is powered from vital instrument panel VA20 to allow for operation with a loss of off site power.

Solenoid valves isolating the reactor coolant PASS from the safety class systems are built to the ASME Code, Section III requirements, and are fully qualified for postulated accident conditions.

9.6.3 CONTAINMENT AIR PASS

9.6.3.1 Equipment Purpose and Description

The containment air PASS has the capability of collecting a sample of containment air. The PASS is a dual module unit consisting of one sample module and one remote operating panel. The motive force for obtaining a sample is supplied by the hydrogen analyzer sample pumps which draw the sample from the hydrogen analyzer system through the PASS sample module, returning it back to the hydrogen analyzer system. (See Figure 9.6–5)

9.6.3.1.1 Sample Module

This module contains the valves and components required to physically collect a 10 ml sample of containment air. The equipment is housed within a wall mounted cabinet, located in the spent fuel pool skimmer pump area. An exhaust blower is built into the top of the cabinet and discharges into the plant radioactive ventilation exhaust system. A door is provided on the cabinet for access to remove samples and to perform maintenance. Influent and effluent connections are provided for connection to the sample system piping.

9.6.3.1.2 Remote Operating Panel

This module contains the sample system mimic panel and slave valves required to remotely operate the sample module. The module is located above the Millstone Unit Number 2 railway access area where radiation levels during an accident are low. The remote operating module is connected to the sample module through electrical cables and piping. A control switch is provided to operate a remote valve to purge the sample lines with nitrogen following sample collection and isolation.

9.6.3.2 Containment Air Samples

A 10 ml sample of containment air is isolated in a shielded sample chamber. Samples are withdrawn from the chamber via septum using a syringe, injected into a sealed container, and transported to the laboratory for subsequent analysis.

9.6.3.3 Design Features

Both operating and failure modes have been analyzed to maintain exposure ALARA. Personnel radiation exposure is minimized through the use of remote control operation, flushing techniques, minimal sample volume, and shielding.

Radiation exposure to the operator taking the sample is estimated to be well below the exposure limit defined by 10 CFR 50, Appendix A, GDC-19.

Materials in contact with the containment air sample are of 300 series stainless steel and have been selected to ensure system integrity up to a pressure of 100 psig at 300°F.

Heat generated by flow of hot fluid through the sample module piping is dissipated by a blower which provides capture velocity air flow into the sample module and exhausts to the plant radioactive ventilation system.

Commercially available components have been utilized to the maximum extent possible and are selected based upon a reputation for high quality. Swagelok fittings have been utilized to be consistent with existing plant equipment.

Isolation valves and breakdown connections are provided for solenoid valve and other equipment for maintenance considerations.

The containment air PASS piping and sample pump are sized to maintain turbulent flow, thereby minimizing crud buildup and plate-out of radioactive products.

Two sample paths and two return paths are available on the containment air PASS providing operational redundancy.

All valves are operated from the remote operating panel. Isolation from containment is provided by valves in the hydrogen analyzer system.

The containment air PASS sample module is powered from vital instrument panel VA20 to allow for operation with a loss of off site power.

9.6.4 TEST

The reactor coolant and containment air PASS can be operated during normal plant operations for testing purposes to obtain actual reactor coolant or containment air samples.

FIGURE 9.6–1 P&ID DIAGRAM SAMPLING SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.6–2 P&ID DIAGRAM SAMPLING SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.6–3 P&ID DIAGRAM SAMPLING SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.6–4 P&ID POST ACCIDENT SAMPLING SYSTEM REACTOR COOLANT SAMPLE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.6–5 P&ID POST ACCIDENT SAMPLING SYSTEM CONTAINMENT AIR SAMPLE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.7 COOLING WATER SYSTEMS

9.7.1 CIRCULATING WATER SYSTEM

9.7.1.1 Design Bases

9.7.1.1.1 Functional Requirements

The function of the circulating water system is to provide a continuous supply of cooling water to the main condenser.

9.7.1.1.2 Design Criteria

a. The circulating water system shall be designed to remove the heat load from the condenser.

b. The circulating water system shall be designed to use sea water for cooling.

9.7.1.2 System Description

9.7.1.2.1 System

The circulating water system is shown in Figure 9.7–1. Four one-fourth capacity (137,000 gpm each) motor driven vertical wet pit pumps circulate the water from the intake structure through the main condenser and discharge into the quarry. The water is taken from Long Island Sound through an intake trash rack and four traveling screens located inside the intake structure. The screens are designed to prevent debris larger than three-eighths inch from passing into circulating water pump, condenser and the service water pumps.

A sodium hypochlorite system capable of chlorinating circulating water on a shock treatment basis is provided. The method of circulating water chlorination is gravity feed injection of sodium hypochlorite solution directly to the running pump suction. The chlorination of the circulating water system, including the condensers, controls biological fouling while maintaining chlorine concentrations within NPDES permit limits.

The total inventory of the circulating water system is as follows:

1. Circulating Water Pump up to Turbine Building Condenser Pit Floor elevation equals 232,000 gallons.

2. Circulating Water above Condenser Pit Floor elevation equals 160,000 gallons.

3. Discharge piping below Condenser Pit Floor elevation to quarry equals 990,000 gallons.

The maximum flow rate through the main condensers is 548,800 gpm.

The minimum time required to stop the circulating water flow would be the coastdown time of the circulating water pumps, 10 seconds. Factors affecting the actual time elapsed would be the operators evaluation of the alarms received in the control room and the severity of the failure. Drop-out time for the control circuitry is in the order of a fraction of a second and is insignificant.

In the event of a gross failure in the circulating water transport system resulting in the flooding of the condenser pit by the discharge from all four circulating water pumps, the total time required to stop all water flow is approximately 16 seconds and occurs automatically.

The circulating water transport system is designed to withstand the maximum operating discharge pressure of the circulating water pumps, 10 psig. Piping is designed for 100 psig, with the expansion joints, butterfly valves, condenser water boxes and tube bundles designed for 25 psig. Normal operating pressure for these components has a range of 0-7 psig, with the outlet waterboxes operating under a vacuum of about 15 to 20 inches Mercury Absolute.

Small leaks around valves and fittings would go undetected except for visual inspection and would drain into the condenser pit sump. The condenser pit sump is provided with 200 gpm capacity sump pumps and provided with high level alarms to indicate water level in the pit 6 inches below the top of the sump. Large leaks due to pipe or expansion joint ruptures would be indicated in the control room by both a gradual loss of vacuum in the condenser shell and high-high condenser pit level switches, set at 10 inches and initiating a trip of the circulating water pumps.

The potential for ruptures is minimized by interlocking the circulating water pumps with the water box butterfly valves, requiring the valves to be open between 20 to 25% prior to starting the pumps. The water box cross connect valves are also interlocked with the water box discharge valves and initiate a pump trip in the event of an incomplete flow circuit. Therefore, the system is adequately protected against over pressuring by the circulating water pumps due to improper valve operation. However, all components in the system are hydrostatically tested at 1.5 times the design pressure of 25 psig to ensure system integrity before normal operation.

Failures in the circulating water transport system would be inconsequential except those occurring above ground in the Turbine Building. The condenser pit is designed to contain these failures and has the capacity to store 284,450 gallons at a maximum depth of 9 feet before spilling over to the Turbine Building ground floor.

The worst case of a major failure would be the complete rupture of all the circulating water piping to both condensers. This would result in the draining into the condenser pit of all circulating water in use in the above ground components, approximately 160,000 gallons, and flooding the pit to a level of 47 inches. When the water level in the pit reaches 10 inches, approximately 4 seconds after the rupture, the high-high level switches in the condenser pit are actuated and cause the circulating water pumps to trip 2 seconds later. During this 6 seconds, the circulating water pumps will have continued to pump approximately 55,500 gallons, which will also drain to the condenser pit raising the water level to 6 feet 2 inches. During the 10 second coastdown time of circulating

water pumps, they continue to add about 45,600 gallons to the condenser pit raising the final water level to 7 feet 11 inches, leaving an additional capacity available in the pit of approximately 23,750 gallons before overflowing. During the postulated failure and subsequent flooding of the condenser pit, the water level is rising an average of 5.9 inches per second.

The condenser pit will contain the water from any postulated failures in the circulating water system without affecting any safety related systems.

9.7.1.2.2 Components

The ratings and materials of construction of the circulating water system are given in Table 9.7-1.

9.7.1.3 System Operation

All four circulating water pumps are in operation during normal operation, with the operating speed of the pumps controlled by the variable frequency drives (VFD). In the event of VFD unavailability, the VFD can be bypassed, allowing operation of the pumps at rated speed; however, no speed control is available in this condition. A cross connection is provided between the two water boxes at each end of each condenser shell. Valves are provided so either pump on a condenser shell can provide water to both sides of the condenser if one pump is out of service. The valving also provides capability to back flush each condenser section.

The inlet cross-connect valve between the condenser inlet water boxes will be opened manually (remote) by the operator under the following conditions:

1. With one circulating water pump inoperable.

2. Following the trip of an operating circulating water pump.

3. In anticipation of a circulating water pump trip.

   Condition 1:

   If a circulating water pump is to be out for maintenance or otherwise inoperable, load will be limited as necessary to maintain vacuum. The inlet cross connect valve will be opened to supply water to both halves of the condenser from the operating pump.

   Condition 2:

   If during normal operation a circulating water pump(s) should automatically trip, the plant load will be reduced as necessary to maintain vacuum. The discharge valve of the tripped pump(s) will be closed, and the inlet cross connect valve will be opened. The circulating water pumps will trip on the following:

   a. Loss of lubricating water.

This trip is preceded by a low lube water pressure alarm.

b. Pump breaker trip.

c. VFD trip.

d. Traveling screens High-High differential pressure.

This trip is preceded by a high differential pressure alarm.

Condition 3:

If during normal operation, it is apparent that a circulating water pump trip is imminent, the plant load will be reduced as necessary to maintain vacuum. The affected pump will be shut down and the pump discharge valve closed. The inlet cross connect valve will be opened. If VFD malfunctions occur, the VFD may be isolated and the affected circulating water pump operated with the VFD bypassed. This operation requires that the affected circulating water pump be shut down during the transfer to the bypass mode of operation.

9.7.2 SERVICE WATER SYSTEM

9.7.2.1 Design Bases

9.7.2.1.1 Functional Requirements

The function of the service water system is to supply a dependable continuous flow of cooling water to the reactor building closed cooling water (RBCCW) heat exchangers, turbine building closed cooling water (TBCCW) heat exchangers, diesel engine heat exchangers, vital AC switchgear room cooling coils, chilled water heat exchangers and the circulating water pump bearings.

9.7.2.1.2 Design Criteria

The service water system is designed to the following design criteria:

a. The service water system shall be designed with suitable redundancy that in the event of a LOCA, concurrent with a loss of off site power and a single active failure, the service water system can perform its safety function. (Note: The postulation of a LOCA concurrent with a seismic event is outside the design basis.)

b. The service water system shall be sized and shall have capacity to provide sufficient water for all modes of operation at a maximum ultimate heat sink temperature of 80°F.

c. The system shall be designed in accordance with the general criteria as described in Section 6.1.

d. The system shall be designed as a moderate energy system.

e. The system shall be designed to withstand a pipe rupture (as an initiating event). Single failures in the redundant train shall not be postulated to occur. Multiple failures of the service water piping will not occur.

The ultimate heat sink consists of the service water cooling system serving the components mentioned in Section 9.7.2.1.1. The water source for the heat sink is the Long Island Sound sea water. It provides sufficient cooling for more than 30 days:

a. to permit safe shutdown and cooldown of the reactor and can maintain a safe shutdown condition for unit 2 and other units at the site;

b. to allow control of an accident in the event one occurs.

The UHS temperature determination method shall use SWS header/branch temperature measurements. Precision instruments installed at the inlet to the reactor building closed cooling water (RBCCW) heat exchangers will normally be used. All in-service precision instruments must be within the limit. If all of the precision instruments are out of service, alternative instruments that measure SW supply side temperature will be used. In this case, an appropriate instrument uncertainty will be subtracted from the acceptance criteria.

SWS supply temperature measurements shall be obtained for each operable SWS loop and the highest valid SWS supply loop temperature shall be used to determine the UHS temperature.

The safety related equipment in the intake structure (pump motors and controls) are protected against the maximum hypothetical tide and storm surges including wave action by being located above elevations 22 feet 0 inches. The pumps are also designed for drought condition for low-low water level of minus 7 feet.

The ultimate heat sink is served only by one source of water, that of the Long Island Sound sea water. The Long Island Sound sea water capacity is sufficient to provide a total maximum quantity of 24,000 gpm cooling water for more than 30 days. A second source of water is not required.

Canals or conduits between the water source and the intake structure are not provided. The intake structure is located on the water source.

9.7.2.2 System Description

9.7.2.2.1 System

The service water system is described by Figure 9.7–1. Three half capacity (12,000 gpm each) vertical wet pit motor driven pumps take suction downstream from the traveling screens in the intake structure. Two independent cross-connected supply headers with isolation valves are provided to all heat exchangers. Two independent discharge headers are provided for the RBCCW heat exchangers that run above ground from the discharge of the heat exchangers to the discharge canal. Two discharge headers are provided, one for each diesel, for the diesel generator cooling water heat exchangers. One discharge header is provided for the TBCCW heat exchangers as the three discharge pipes are combined above ground prior to entering the discharge canal. The discharge header for the chilled water system and for the vital switchgear room cooling heat exchangers combine with the TBCCW heat exchanger discharge header before entering the discharge canal. Normally, one pump and service water header are required to provide cooling of the RBCCW and diesel generator cooling water heat exchangers following a loss-of-coolant accident (LOCA) or for Unit shutdown. In the event of a loss of station power, the pump motors are supplied power from the emergency buses. The TBCCW headers are isolated for either a SIAS or LNP signal.

9.7.2.2.2 Components

A description of the service water system components is given in Table 9.7-2.

9.7.2.3 System Operation

During normal operation two pumps are operating and provide water to the RBCCW and the TBCCW heat exchangers and vital switchgear ventilation system cooling coils. Each pump is capable of supplying the required service water to one train for accident mitigation and safe shutdown. A third standby pump will be started remotely by manual means upon loss of an operating pump. During winter time operation (ultimate heat sink temperature less than 60°F), the standby RBCCW and TBCCW heat exchangers may be placed on-line (service water side only) in order to maintain service water pump minimum flow requirements. This may be necessary since much less service water flow is required to remove RBCCW and TBCCW heat loads as service water temperature decreases.

9.7.2.4 Availability and Reliability

Redundant headers and one spare pump are provided. Sea water is chlorinated to prevent slime, algae, and mussel growth. Thermal backwashing (mussel cooking) is also performed to prevent fouling of the intake structure pump bays. A sodium hypochlorite system capable of chlorinating service water is provided. The service water chlorination system is capable of injecting sodium hypochlorite solution into the service water pump suction bells via an injection pump system. The solution is taken up by the service water pumps and enters the service water system to provide biological fouling control. Pump motors and controls are located at Elevation 22 feet 0 inches for flood protection. The intake structure is designed to withstand any probable tornado missile.

Various types of internal protective coatings and liners are utilized in piping and components throughout the system. Periodic preventive maintenance, surveillances, and inspections ensure that significant coating or lining degradation will be promptly detected. This and proceduralized investigative and corrective actions ensure that coating or lining degradation will not impair service water system safety functionality.

The service water pump has been modified to an open-line shaft design. Thordon bearings are lubricated by sea water flowing through the pump. No external lube water is required.

The service water pumps and motors are not susceptible to a failure from a common cause. Failure of a circulating water line cannot flood the intake structure since the discharge line is low energy and located below grade. There are no sources of potentially damaging internally generated missiles within the intake structure. Rotating components are limited to the relatively slow speed 300 rpm and 900 rpm motors on the circulating water pumps and the service water pumps respectively. The rotating masses therein are essentially encapsulated within heavy motor windings. There is no high-energy fluid system to cause any missile within the intake structure.

The intake structure louvers have been redesigned to resist penetration by any of the design basis tornado driven missiles which were discussed in Section 5.2.5.1.2 of the FSAR.

The exterior concrete walls are of adequate thickness to prevent perforation by any of the design basis missiles. An investigation of generation of a secondary missile due to concrete spalling indicates that only minor spalling may occur due to the impactive energy of the utility pole missile. This spalling is limited to the thickness of the protective cover on the steel reinforcing on the interior of the structure. Secondary missiles thus assumed to form would not reach the closest service water pump. The intake structure is constructed of materials which will not support combustion. Any potential fire, such as a service water pump motor, would be localized in a small area and would normally be mitigated by a dry type of fire protection. The spatial separation (approximately 20 feet.) would prevent damage to the redundant pumps.

The only safety related system in the intake structure is the service water system.

The service water pump motors and controls are protected against the maximum hypothetical tide and storm surges including wave action.The pumps are also designed for low-low water of minus 7 feet. Flooding of the intake structure due to a circulating water pipe rupture is not credible because the circulating water pipes are below the floor on the intake structure. In case of a service water pipe rupture no water accumulation can be sustained in the intake structure due to the floor grating. Catastrophic failure resulting in missile generation have not been experienced with circulating water or screen wash pumps. Therefore, the service water pumps have not been protected against missiles generated by the failure of circulating water or screen wash pumps.

Because of the redundancy of the service water system even if a hypothetical missile is generated and causes the inoperability of one service water pump it will not effect the service water system operation because only two service water pumps are required for normal operation while only one is needed for emergency operation.

The service water system also provides cooling water to the Emergency Diesel Generators when these are started such as in the case of a loss of off site power. If the intake structure becomes flooded and the service water to the diesels must be isolated, cooling to one diesel is provided by a cross-connection to the Fire Water System.

A failure mode analysis is given in Table 9.7-4. A rupture in the system is considered an initiating event only; it is not postulated concurrent with a LOCA (or any other Chapter 14 event). System redundancy and header separation have been provided to maintain continuous cooling in the event of a single passive failure during post-accident long term cooling.

9.7.2.5 Test and Inspections

Each service water pump and strainer was hydrostatically tested at 1.5 times the design pressure.

The service water pump was tested per ASME Power Test Code Section 8.2, 1965 for hydraulic performance. The service water pump design curve is given in Figure 9.7–3.

Each component is inspected and cleaned prior to installation into the system. The service water system undergoes a preoperational test prior to startup; the detailed test procedure is described in Section 13.

Instruments will be calibrated during testing. Automatic controls will be tested for actuation at the proper set points. Alarm functions will be checked for operability and limits during preoperational testing.

The system will be operated and tested initially with regard to flow paths, flow capacity and mechanical operability. At least one pump will be tested on line to demonstrate head and capacity (Section 13).

Major components of the system such as pumps and strainers are accessible for periodic inspection during normal operation.

Heat exchanger fouling and other component degradation is common in open cycle Service Water systems due to both macro and micro fouling. This fouling can lead to an inability to provide the safety related cooling that is the function of the Service Water system. NRC Generic Letter (GL) 89-13 required that actions be taken to confirm and maintain the capability of the Service Water system to perform its design basis functions. Actions performed to ensure the capability of the system to provide the required safety related cooling include:

- Injecting sodium hypochlorite to minimize biofouling. This injection prevents the attachment and subsequent growth of large quantities of mussels.

- Inspecting and cleaning the intake bays to minimize fouling. This cleaning removes fouling that might be drawn into the Service Water system and clog downstream components.

- Flowing Service Water through normally stagnant portions of piping such as supply to the diesel generators and vital chillers to minimize fouling. Stagnant and near stagnant water can be conducive to the biological growth.

- Monitoring of available heat exchanger parameters to detect gross debris loading. This periodic monitoring detects fouling build up in a heat exchanger between visual inspections.

- Inspecting heat exchanger, pipe, and other component internals on the Service Water Side to remove fouling and repair as needed. Periodic inspections identify component degradation or the slow buildup of fouling prior to it affecting component operability.

- Cleaning heat exchangers on the Service Water side to minimize fouling buildup. Cleaning removes deposits of fouling that tend to occur in heat exchanger tubes.

- Tested heat exchangers to confirm design heat transfer capability. This testing and subsequent analysis verified heat exchanger performance was capable of meeting design requirements.

- Filling heat exchangers with fresh water during layup to minimize buildup of fouling and component degradation caused by stagnant seawater.

9.7.3 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

9.7.3.1 Design Bases

9.7.3.1.1 Functional Requirements

The turbine building closed cooling water (TBCCW) system provides cooling water to the auxiliary equipment (associated with the power control system) which is located in the turbine building and for the condensate polishing facility located in warehouse number 5.

9.7.3.1.2 Design Criteria

The turbine building closed cooling water system shall be designed to cool auxiliary equipment over the full range of operation.

9.7.3.2 System Description

9.7.3.2.1 System

The TBCCW system is described by Figure 9.7–2.

The turbine building closed cooling water system uses treated water to remove heat from the turbine components and sample coolers. The system transfers the heat to the service water system

in the TBCCW heat exchangers. The system has three half capacity heat exchangers and three half capacity pumps to circulate the cooling water through the system. The system has a head tank and chemical addition provision to take off thermal expansion of water and maintain the water chemistry.

The components cooled by this system include:

1. Stator liquid coolers
2. Electro hydraulic coolers
3. Station air compressor and after cooler
4. Steam generator feed pump lube oil coolers
5. Exciter air cooler
6. Heater drains pumps motors and bearings
7. Condensate pumps motor cooling
8. Sampling Station Number 2 sample cooling
9. Generator isolated phase bus cooler
10. Generator hydrogen coolers
11. Turbine lube oil coolers
12. Local sample coolers
13. Warehouse number 5 air conditioning unit
14. Deleted by MP-PACKAGE-FSC-00-MP2-011
15. Multi-circuit cooler
16. Fuel handling area air conditioning units
17. Auxiliary chilled water system mechanical refrigeration units

9.7.3.2.2 Components

A description of the TBCCW system components is given in Table 9.7-3.

9.7.3.3 System Operation

During normal operation two pumps and two heat exchangers are operating. The third heat exchanger will be operated upon failure of an operating pump or heat exchanger.

TABLE 9.7-1 CIRCULATING WATER SYSTEM COMPONENTS

Traveling Water Screen

Type	Vertical through flow
Quantity	4
Speed (feet per minute)	5 and 20
Material	
Screen	3/8 inch square openings of No. 14 Stainless steel wire
Frame	Stainless steel
Motor Manufacturer	Leeson Electric
Motor	1800/450 rpm, 10/2.5 hp, 3 phase, 60 Hz, 460 V
Codes	NEMA

Screen Wash Pumps

Pump Manufacturer	Fairbanks Morse
Model	14 inch M. C.
Type	Vertical turbine
Quantity	2
Capacity each (gpm)	1760
Head (feet)	210
Material	
Case	ASTM A-48 Class 30, 2-3% Ni
Impeller	ASTM A-296 CF 8M (316 SS)
Shaft	AISI 316 Stainless Steel
Motor Manufacturer	General Electric
Motor	125 hp, 460 V, 3 phase, 60 Hz, 1800 rpm
Codes	NEMA, Standards of the Hydraulic Institute, ASME Boiler and Pressure Vessel Codes, Section VIII ANSI B16.5

TABLE 9.7-1 CIRCULATING WATER SYSTEM COMPONENTS (CONTINUED)

Circulating Water Pumps

Pump Manufacturer	Babcock & Wilcox Canada Ltd.
Model	78BN
Type	Vertical, wet pit
Quantity	4
Capacity each (gpm)	137,200
Head (feet)	24
Material	

Case Materials such as: ASTM A-240, Type 316-L; ASTM A-48 CL-30; ASTM A-436, Type 1B, Ni-Resistant; ASTM A-296, CF 8M (316 SS).

Impeller Shaft - AISI 304 Stainless Steel

Motor Synchronous 1500 hp, 4000 V, 60 Hz, 3 phase, 300 rpm

Codes

NEMA, Standards of the Hydraulic Institute, ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, ANSI B16.5

Variable Frequency Drive

Manufacturer	Siemens Energy & Automation, Inc.
Model	Harmony Drive
Quantity	4
Type	Air Cooled
Blowers	Transformer Cabinet - 3, Cell Cabinet - 3
Horse Power	1500 hp
Input	4160V, +10%, -15%, 3 phase, 60 Hz, 247A RMS
Output	0-4000V, 3 phase, 0-60 Hz, 260A RMS
Weight	15,580 pounds
Enclosure	NEMA1

TABLE 9.7-2 SERVICE WATER SYSTEM COMPONENTS

Service Water Pumps

Type	Vertical, wet pit
Quantity	3
Capacity each (gpm)	12,000
Head (feet) TDH	100
Material	Stainless Steel
Motor Manufacturer	General Electric
Motor	450 hp, 4000 V, 3 phase, 60 Hz
Codes	NEMA, Standards of the Hydraulic Institute, ASTM, ANSI, G16.5
Seismic	Class 1

Service Water Pipe (Selection based on applicable service)

High Performance Stainless Steel, Unlined
Carbon Steel Pipe, Epoxy or PVC Lined
Stainless Steel Pipe, Epoxy Lined or Unlined
Cast Ductile Iron Pipe, Cement and/or Epoxy Lined
Copper Nickel Pipe, Unlined
Red Brass Pipe, Unlined
Nickel Copper, Unlined

Service Water Valves (Selection based on applicable service)

High Performance Stainless Steel, Unlined
Carbon Steel Valves
Stainless Steel Valves
Cast Ductile Iron Valves
Copper-Nickel Valves
Cast Bronze Valves

Service Water Strainer

Type	Automatic self cleaning
Quantity	3
Design flow rate (gpm)	12,000
Design pressure (psig)	100
Design temperature (°F)	100
Materials	Selected for Salt Water Service
Motor	3/4 hp, 460 V, 1725 rpm, 3 phase, 60 Hz
Codes	NEMA, ASME Section VIII
Seismic	Class 1

TABLE 9.7-3 TURBINE BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS

TBCCW Heat Exchanger

Manufacturer	Southwestern Engineering Co.
Model	TEMA type NEN
Type	One pass shell and tube
Quantity	3
Design duty each (BTU/hr)	28.5×10^6
Design pressure (psig)	
Shell side	125
Tube side	125
Design temperature (°F)	
Shell side	125
Tube side	120
Material	
Shell	ASME SA-516-70 (carbon steel)
Channels	SB-171-715 (70-30 CuNi)
Channel covers	SA-516-70 with SB-171-715 liner
Tube	ASME SB-111-715 D.S.R.
Tube sheet	ASME SB-171-715
Codes	
TEMA - Class C, ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels	
Seismic	None
Specification	SP-ME-542

TBCCW Pump

Manufacturer	Goulds Pump
Model	3405-10 x 12-12
Type	Horizontal centrifugal
Quantity	3
Capacity each (gpm)	3400
Head (feet) TDH	110

TABLE 9.7-3 TURBINE BUILDING CLOSED COOLING WATER SYSTEM COMPONENTS (CONTINUED)

Material	
Case	ASTM A-278-59T, Class 30
Impeller	1106 Bronze
Shaft	AISI Type 316
Motor	125 hp, 460 V, 3 phase, 60 Hz, 1800 rpm
Codes	
NEMA, Standards of Hydraulic Institute, ASME Boiler and Pressure Code Section VIII, Pressure Vessels.	
Seismic	Class 2

TBCCW Head Tank

Manufacturer	PX Engineering
Type	Vertical
Quantity	1
Design pressure (psig)	15
Design temperature (°F)	110
Volume (gal)	1100
Material	ASTM A-285 Gr C
Code	ASME Section VIII
Seismic	Class 2

TABLE 9.7-4 SERVICE WATER COOLING SYSTEM

	COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
1.	Service Water Cooling Pumps (3)	Pump stops.	Pump trip alarm.	Loss of flow in one header.	Isolate pump. Standby pump is put into service. Valve line-up.	Normal operation.	Pump out of service for repairs.
2.	Piston Operated Valves (2) HV6482, HV6489	a. Fails to open.	a. & b. Position indication C.R.I.	a. None	a. & b. Repair valve operator.	Normal operation.	
		b. Fails to operate (open).		b. Loss of flow from standby pump.	b. Operate valve manually, or direct flow from standby pump to alternate header.		
3.	Service Water Cooling Pump Discharge Header (3)	Pipe break.	Low flow indication on the effected components C.R.I.	Loss of flow in header.	Break isolated. Standby pump is put into service.	Normal operation.	Header out of service for repairs.
4.	RBCCW Heat Exchanger Supply Header and Channels	Pipe break.	Heat exchanger outlet low flow indication. Sump level alarm. Both C.R.I.	Loss of flow in header.	Break isolated. Standby heat exchanger put into service.*	Normal operation.	Header out of service for repairs.

TABLE 9.7-4 SERVICE WATER COOLING SYSTEM (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY		FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
5.	RBCCW Heat Exchanger (3)	a. Tube side rupture.	a. Makeup valve. None to RBCCW Surge Tank open alarm. b. C.R.I.	None	Failed heat exchanger isolated. Standby heat exchanger put into service. *	Normal operation.	Heat exchanger out of service for repairs.
6.	Piston Operated Valve (2) HV6399, HV6400	Fails as is.	Position indication. C.R.I.	None	Repair valve operator.	Normal operation.	Valve can be manually operated.
7.	RBCCW Heat Exchanger Discharge Header (2)	Pipe break.	High flow indication. Sump level alarm. Both C.R.I.	None	Break isolated. Standby heat exchanger put into service. *	Normal operation.	Header out of service for repairs.
8.	Piston Operated Valve (3) TV6306, TV6306A, TV6307, TV6307A, TV6308, TV6308A	Fails open.	Position indication. Local	Possible rapid cooldown of RBCCW by service water leading to RCP seal problems.	Repair valve operator.	Normal operation.	Place standby heat exchanger into service if modulating control required.

TABLE 9.7-4 SERVICE WATER COOLING SYSTEM (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY		FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
9.	TBCCW Heat Exchanger	Pipe break.	1. Heat exchanger flow alarm (C.R.I.).	Reduction/Loss	Break Isolated Standby Heat Exchanger put into service.*	Normal operation.	Repair header
	a. Supply Header		2. Possible Surge Tank high level alarm on TBCCW heat up.	Loss of Service Water Flow.			
			3. TBCCW High Temperature Alarm (C.R.I.).				
	b. Discharge Header	Pipe break.	Heat exchanger flow alarm (C.R.I.).	Reduction / Loss of Service Water Flow.	Isolate Service Water to TBCCW Repair Header *	Potential trip / damage to components due to TBCCW heat up.	Repair header.
10.	TBCCW Heat Exchanger (3)	Tube side rupture.	Flow indication; TBCCW Surge Tank low level alarm.	None	Heat exchanger isolated.*	Normal operation.	Heat exchanger out of service for repairs.
11.	Piston Operated Valve (2)HV6438, HV6439	Fail as is.	Position indication. C.R.I.	None	Repair valve operator.	Normal operation.	Valve can be manually operated.
12.	Diaphragm Operated Valves (3) TV6303, TV6304, TV6305	Fail open.	Local visual indication only.	Loss of one heat exchanger.*	Repair valve operator.	Normal operation.	Not required for shutdown.

TABLE 9.7-4 SERVICE WATER COOLING SYSTEM (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY		FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
13.	Diesel Engine Cooling Water Heat Exchanger (2)	Tube side rupture.	Diesel Generator High Temperature Alarms. C.R.I. Jacket cooling pressure low, jacket cooling level low, jacket cooling temperature high / low. All local.	Loss of one diesel generator.	isolate heat exchanger and repair.	Normal operation (one diesel generator in service).	One diesel generator required during an emergency.
14.	Diesel Engine Cooling Water Heat Exchanger Supply & Discharge Headers	Pipe break.	Heat exchanger outlet low-flow indication, lube oil temperature, local jacket cooling temperature. Local	Loss of flow in system.	Break isolated.	Normal operation (one diesel generator in service).	One diesel generator required during an emergency. SBO allows eight hours to restore power.
15.	Piston Operated Valve (2) FV6389, FV6397	Fail open.	Position indication. C.R.I.	None	Repair valve operator.	Normal operation.	Valve can be manually operated.

TABLE 9.7-4 SERVICE WATER COOLING SYSTEM (CONTINUED)

COMPONENT IDENTIFICATION & QUANTITY		FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
16.	Service Water Header to Circulating Water Pump Bearings & Seals	Pipe break.	Low pressure alarm. C.R.I. Domestic water backup to lube water open. C.R.I.	1. Loss of water.	Isolate break.	Domestic Water System supplying bearings and seals on circulating water pumps.	Normal circulating water pump operation while service water header is repaired.
				2. Loss of service water supply to circulating water pumps.			
17.	Diaphragm Valve (1) PV6522	Fail open.	Visual	None	Repair valve operator.	Normal operation.	

* During wintertime operation (ultimate heat sink temperature less than 60°F), the standby RBCCW and TBCCW heat exchangers may be placed on line (service water side only) in order to maintain service water pump minimum flow requirements. If a standby heat exchanger is required to be placed in-service due to a failure of an RBCCW or TBCCW heat exchanger, then service water pump minimum flow concerns must be met by manually adjusting service water flow in the affected header. This may result in lower temperatures and will require close monitoring of RBCCW or TBCCW component temperatures.

FIGURE 9.7–1 P&IDS - CIRCULATING WATER, SERVICE WATER, SCREEN WASH, SODIUM HYPOCHLORITE (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–1 P&IDS - CIRCULATING WATER, SERVICE WATER, SCREEN WASH, SODIUM HYPOCHLORITE (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–1 P&IDS - CIRCULATING WATER, SERVICE WATER, SCREEN WASH, SODIUM HYPOCHLORITE (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–1 P&IDS - CIRCULATING WATER, SERVICE WATER, SCREEN WASH, SODIUM HYPOCHLORITE (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–2 P&ID TBCCW SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–2 P&ID TBCCW SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.7–2 P&ID TBCCW SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

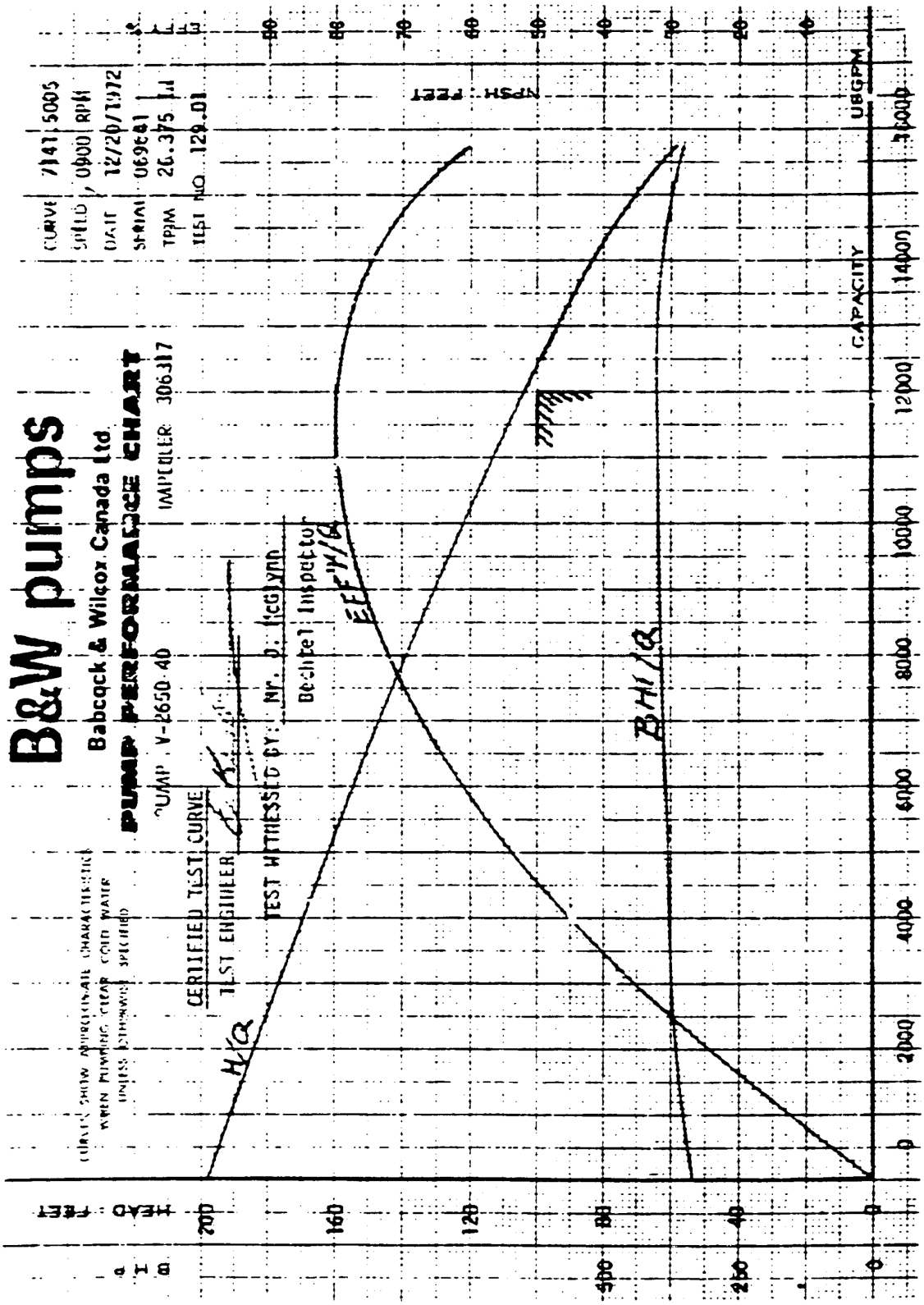


FIGURE 9.7–3 TYPICAL SERVICE WATER PUMP CERTIFIED TEST CURVE

FIGURE 9.7–4 DELETED BY FSARCR 03-MP2-011

9.8 FUEL AND REACTOR COMPONENT HANDLING EQUIPMENT

9.8.1 DESIGN BASES

The fuel and reactor component handling equipment provides for the safe handling, inspection and storage of fuel assemblies, control element assemblies (CEA) and reactor internals under normal conditions. It also provides for the safe disassembly, handling, and reassembly of the reactor vessel closure head, and in-core instrumentation (ICI).

9.8.1.1 Functional Requirements

a. The equipment shall be capable of operation in water with the following design chemistry:

pH (77°F)	4.5 to 10.6
Boric Acid, Maximum, weight percent	1.5
Ammonia, Maximum(1), ppm	50
Lithium, Maximum (1) , ppm	2.5
Dissolved Air, Maximum	Saturated
Chloride, Maximum, ppm	0.15
Fluoride, Maximum, ppm	0.1

b. The equipment will normally be used during refueling for a period of approximately three weeks during which time it must operate continuously without maintenance or service.

c. The equipment is capable of operating dry for the initial reactor core loading.

d. In the event of loss of power, the equipment and its load remains in a safe condition.

e. Equipment located within the reactor containment building is capable of withstanding, without damage, the internal building test pressure.

9.8.1.2 Design Criteria

a. Structural

(1) Concentrations do not occur simultaneously.

1. The stresses under the combined dead weight, live and seismic loads will not exceed the allowable stress derived from the applicable design code based on properties of the material per American Society for Testing and Materials (ASTM) requirements.

2. The equipment will withstand the loading induced by the hypothetical vertical and horizontal seismic loadings which are considered as acting simultaneously on this equipment in conjunction with normal loads without exceeding material yield stresses as specified by ASTM.

3. Where required, keepers are provided to preclude derailment of equipment under seismic loading.

The following refueling equipment has been designed to seismic Category I requirements.

a. reactor polar crane

b. spent fuel cask crane

c. spent fuel pool platform crane (1)

d. new fuel elevator (1)

e. spent fuel storage racks

f. new fuel storage racks

g. permanent reactor cavity seal

h. refueling machine (1)

i. fuel transfer machine (1)

j. fuel tilting mechanisms (1)

k. fuel transfer tube and isolation valve

The refueling machine and the fuel transfer system have been designed to accept the combined dead weight, live-load, and design seismic loads acting simultaneously without exceeding the allowable stress derived from the applicable

(1) Designated Seismic Class II components but designed for Class I earthquake basis.

design code based on properties of the material as specified by ASTM requirements.

The possibility of damage to a fuel assembly as a consequence of mishandling is minimized by extensive training, detailed procedures, and equipment design. Inadvertent disengagement of a fuel assembly from the spent fuel pool platform crane or refueling machine is precluded by positive interlocks. Consequently, the possibility of dropping or damaging a fuel assembly during handling, resulting in the failure of all rods in the fuel assembly with the highest radioactive inventory is minimized. In addition, the exclusion boundary doses resulting from a fuel handling accident have been shown to be within the guideline of 10 CFR Part 50.67 and Regulatory Guide 1.183. The safety aspects of a fuel handling accident are presented in detail in Section 14.7 of the FSAR.

b. Mechanical

1. In general, all hoisting and winching components conform to the requirements of Electric Overhead Crane (EOC)1, Specification number 61 and Crane Manufacturers Association of America (CMAA) Specification number 70.

2. Hoisting units are provided with two means of braking. Electric brakes are capable of holding 100 percent of the motor stall torque and shall permit convenient adjustments. Mechanical load brakes are capable of holding the rated load of the hoist.

3. Drives imparting equipment motions other than hoisting are equipped with automatically operated brakes sized to “soft stop” conditions.

4. Hoist drives are designed to regulate the speed during lowering so as to prevent undue acceleration.

5. Sheaves are supplied with keepers to prevent the wire rope from falling off.

6. If operation of gearing in oil is required, sealed enclosures are provided. Drive mechanisms are installed in pans or similar containment to prevent spillage of lubricants. No oil is used underwater.

7. Grapples and mechanical latches which carry fuel assemblies or CEA's are mechanically interlocked against accidental opening.

8. Equipment is provided with suitable locking devices or restraints to prevent parts, fasteners, or limit switch actuators from becoming loose. In those cases where a loosened part or fastener can drop into, or is not separated by a barrier from, or whose rotary motion could propel it into the

water of the refueling cavity or spent fuel pool, these parts and fasteners are lockwired or otherwise positively captured.

9. The refueling machine is capable of removing and installing a fuel assembly at each operating location at the most adverse combined tolerance condition for the equipment, core internals and fuel assemblies.

10. Electrical limit switches and positive mechanical stops prevent the fuel from being lifted above the minimum safe water cover depth.

c. Electrical

1. Electrical equipment conforms to American National Standards Institute (ANSI) Standards C6.1, C19.1 and C50 and National Electrical Manufacturers Association (NEMA) Standards.

2. Motor housing are grounded with ground straps.

3. Each motor and related conductors is protected against running overload by the use of a separate overload device responsive to motor current, or, in the case of NEMA Design B motors on high reversing duty, by overload devices specifically selected for the applications, such as embedded thermal sensing devices.

4. Where equipment heaters are required, they conform to the following requirements:

a. All control panels and motors are provided with heaters to maintain the temperature inside the panels above the dew point during periods of system inactivity.

b. All heaters are supplied by an independent power supply and suitably protected with overload devices specifically designed for the application.

5. Receptacles are suitably protected by overcurrent devices that will open the power supply leads.

6. The refueling machine hoist and the spent fuel pool platform crane hoist are provided with load measuring devices with a visual display of the load and interlocks to interrupt hoisting if the load reaches the overload setpoint and interrupt lowering if the load reaches the underload setpoint.

7. The refueling machine is provided with dual redundant electric brakes.

d. Materials

1. Where wire rope penetrates the borated water, it is unlubricated stainless steel wire rope with stainless steel core. Hoisting rope is stretched to twice the normal working load. Catch pans are provided where the hoisting rope might drip on the equipment or operator.

2. Equipment which will be located or operate submerged in dilute boric acid are of austenitic stainless steel or other corrosion-resistant material.

3. In general, where 17-4 precipitation-hardened stainless steel is used in the design of the equipment, a minimum heat treat temperature of 1100°F (condition H-1100) is required.

4. Wire rope sheave units carrying stainless steel rope are fabricated of austenitic stainless steel or other corrosion resistant material. (Except for the five ton monorail hoist mounted to the spent fuel cask crane, which has a carbon steel bottom block assembly.

e. Radiological Considerations

1. All equipment is designed in an attempt to permit unrestricted access during disassembly, handling and reassembly operations.

2. Crevices in equipment have been minimized to facilitate decontamination.

f. Criticality Accident Requirements

Millstone 2 has chosen to comply with 10 CFR 50.68(b).

9.8.2 SYSTEM DESCRIPTION

9.8.2.1 System

9.8.2.1.1 New Fuel Storage

New fuel assemblies are transported by truck to the plant and into the auxiliary building in regulating agency approved containers. The new fuel assemblies are removed from the shipping containers by the auxiliary hook of the spent fuel cask crane inspected and stored in the new fuel storage racks which are designed for dry storage of 76 fuel assemblies. The arrangement of the racks is shown in Figure 9.8–1.

The new fuel assemblies with or without the CEA’s nested are stored in an environment of air, temperature of 55°F to 110°F ambient and a relative humidity of 30 to 100 percent. The base of the new fuel storage racks is located at Elevation 38 feet 6 inches. The receiving and removal operation of the new fuel assemblies into or out of the storage cells are accomplished by lateral

movement of the fuel assemblies above the platform at Elevation 47 feet 6 inches and vertical movement along the vertical guide in the cell while the fuel assemblies are suspended vertically on their longitudinal axis from a special tool attached to the auxiliary hook of the spent fuel cask crane.

The new fuel storage racks are fabricated of stainless steel and consisting of vertical cells in parallel rows with a center-to-center distance of twenty and one-half inches. The new fuel storage racks are designed to the requirements defined below:

a. Structural

The structure consists of vertical elongated cells in rows and is of open construction to permit ventilation. Solid plate partitions are used as shear plates to resist horizontal loads or to transmit shear forces. The storage cells are accessible from the aisles between two rows of storage racks above a platform at Elevation 47 feet 6 inches. Guides, hold downs, and new fuel assembly bracings are provided at suitable horizontal intervals to assist receiving and removal operation as well as securing and maintaining the vertical alignment of the new fuel assembly in place for storage. Vertical guides are continuous for the full length of the fuel assemblies. The bottom plates are raised from the floor level and supported by the base of the new fuel rack. The storage rack and the base are designed to withstand all anticipated loadings, including dropping or side swinging of a fuel assembly on the rack in an accidental event. The structural deformations are limited to preclude any possibility of criticality.

The storage racks are laterally braced one to another and supported in a manner which will not permit a reduction in separation space in the event of a design earthquake. The racks themselves are designed not to collapse or bow under the force of a fuel assembly if dropped or swung against the rack.

b. Stability

The new fuel storage racks are stabilized by floor anchorage only. This system will prevent significant lateral movement of the racks, containing any number of fuel assemblies and subjected to all anticipated loads, including the design basis earthquake (DBE). An insignificant lateral movement is one which will not reduce the center-to-center distance of the cells below the safe geometry margins. The maximum displacement of any member, joint or component of the structure is within the elastic limit of the material.

c. Safe Geometry

K_{eff} for the new fuel storage racks containing adjacent fresh fuel assemblies of 5.0 percent enrichment is less than 0.95 when flooded with clean unborated water, or with low density moderation. The 5 percent enrichment used in the analysis conservatively exceeds the 4.85 percent Technical Specification limit.

All external corners of the new fuel storage racks in the path of the new fuel element when being transported or inserted into the cell for storage are rounded to prevent possible damage to the fuel element if swung against the rack.

The racks consist of eleven modules for a total of 76 cells as shown on Figure 9.8–1.

With the base of the new fuel storage racks 38.5 feet mean sea level the possibility of new fuel being flooded with sea water is precluded. The new fuel vault is situated in an area over which no large piping is routed. Therefore, the only possible source of water into the vault would be minor leakage of rainwater through the roof or wall of the auxiliary building. The walls of the new fuel vault have been provided with block-outs which would allow any water in-leakage to drain out of the vault. Also, the active part of the new fuel assembly is approximately seven inches above the floor of the vault. In addition, the 20.5 inches center-to-center spacing precludes criticality in the highly unlikely event of flooding. Figure 9.8–2 shows a cross section of the new fuel storage vault.

Figures 9.8–3 through 9.8–6 show details of the new fuel storage racks.

As shown in Figure 9.8–2, the racks are a hybrid between side and top loading racks. This design was based on preventing the placing of two new fuel assemblies adjacent to each other and to minimize the amount of overlap between assemblies during handling in the new fuel vault.

The new fuel rack closure bar is an administratively controlled device which is used to laterally support the top of fuel assembly in the rack (see Figure 9.8–6).

The rack has been designed to take an uplift force equal to the dead weight of the empty rack module.

9.8.2.1.2 Spent Fuel Storage

The spent fuel pool is located in the fuel handling area of the auxiliary building. The pool is designed for the underwater storage of spent fuel assemblies after removal from the reactor core. The heat load analysis of the spent fuel pool conservatively accounts for a total of 1349 fuel assemblies. Of these 1349 fuel assemblies, 1343 are intact fuel assemblies, and the other 6 fuel assemblies are assumed to have been consolidated into 3 Consolidated Fuel Storage Boxes. As described later in this section, there are currently 1346 physical storage locations in the spent fuel pool. The spent fuel pool is designed to maintain approximately 24 feet of borated water above the stored fuel assemblies. CEAs removed from the core are stored in the guide tubes of the fuel assemblies.

The spent fuel pool (SFP) also stores Non-standard Fuel Configurations (discussed later in this section). A Non-standard Fuel Configuration is an object containing fuel that does not conform to the standard fuel configuration. The standard fuel configuration is a 14 x 14 array of fuel rods (or

fuel rods replaced by un-enriched fuel rods or stainless steel rods) with five (5) guide tubes that occupy four lattice pitch locations each. Fuel in any other array is a "Non-standard Fuel Configuration." Reconstituted fuel in which one or more fuel rods have been replaced by either un-enriched fuel rods or stainless steel rods is considered to be a standard fuel configuration.

The stainless steel storage racks consisting of vertical cells grouped in parallel rows, are designed for a center-to-center distance of 9.8 inches in Regions 1 and 2 and 9.0 inches in Regions 3 and 4. Spent fuel decay heat is removed by the spent fuel pool cooling system described in Section 9.5. The fuel storage racks are designed to the requirements defined below while maintaining a physical arrangement that results in a K_{eff} of 0.95 or less during all normal usage of the racks and under abnormal conditions. The arrangement also provides for adequate convective cooling of stored fuel assemblies.

Figure 9.8–7 provides a schematic of the spent fuel pool arrangement described below.

a. Structural

Region 1 consists of two 8 by 10 modules of spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches. These racks contain Boraflex which is no longer credited in the spent fuel pool criticality analysis. These modules are used to store 80 fuel assemblies (up to a maximum initial planar average enrichment of 4.85 weight percent U-235) in a 2 out of 4 storage pattern as shown in Figure 9.8–7. In a 2 out of 4 storage pattern two locations in a 2 x 2 storage array (4 spent fuel storage locations) can store a fuel assembly, and the other two locations are designated as Restricted Locations as shown in Figure 9.8–7. Fuel storage rack locations designated as Restricted Locations in Figure 9.8–7 shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.(1) There are 80 Restricted Locations in this region.

Region 2 consists of two 8 by 9 modules and one 8 by 10 module of spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches. These racks contain Boraflex which is no longer credited in the spent fuel pool criticality analysis. These modules are used to store spent fuel bundles that have achieved a specified fuel burnup vs. initial planar average enrichment. Region 2 also has two types of storage locations, designated Type 2A and Type 2B. Type 2A locations can store more reactive fuel assemblies than Type 2B. These modules are used to store 164 fuel assemblies in a 3 out of 4 storage pattern in which the fourth

(1) Note that Region 1 and 2 SFP rack storage locations contain removable Boraflex panel boxes which house the Boraflex panels. The Boraflex panel boxes were manufactured as an integral part the original SFP racks and as such are NOT stored components in SFP rack storage locations. Criticality analysis has shown that the Restricted Locations are acceptable with or without the Boraflex panel boxes.

location in a 2 x 2 storage array is a Restricted Location as shown in Figure 9.8–7. There are 60 Restricted Locations in this region.

Region 3 consists of twelve modules of spent fuel racks with nominal center-to-center cell spacing of 9.0 inches, which do not contain Boraflex. These modules are used to store spent fuel assemblies that have achieved a specified fuel burnup vs. initial planar average enrichment. Region 3 requires that fuel assemblies contain either three Borated Stainless Steel Poison Rodlets (installed in the assembly's center guide tube and in two diagonally opposite guide tubes), or a full length, full strength CEA (Note that credit is NOT allowed for the full-length, reduced-strength CEAs used in Cycles 1 through 6 [CEAs with serial numbers 66 through 73, inclusive] and for the part-length CEAs used in Cycle 1 [CEAs with identifier letters A through H, inclusive]). The region consists of 822 storage locations.

Region 4 consists of one 7 by 9 module and one 7 by 11 module of spent fuel racks with a nominal center-to-center cell spacing of 9.0 inches, which do not contain Boraflex. These modules are used to store spent fuel assemblies that have achieved a specified fuel burnup vs. initial planar average enrichment. These modules are used to store 100 fuel assemblies in a 3 out of 4 storage pattern with 40 Restricted Locations.

No structural changes to the storage rack modules were required as they were initially designed to withstand the static and dynamic loads resulting from the storage of both intact fuel assemblies and consolidated fuel in Regions 1, 2, 3, and 4.

The physical arrangement of the storage locations within the racks and the total number of racks provide for a total of 1346 storage locations. Of this number:

- 80 locations are available within Region 1 for the storage of any fuel assembly, with 80 cells designated as Restricted Locations
- 164 locations are available within Region 2 for the storage of fuel assemblies that meet the burnup requirements of this region, with 60 cells designated as Restricted Locations
- all 822 locations are available within Region 3 for the storage of fuel assemblies that meet the burnup requirements of this region (and contain Borated Stainless Steel Poison Rodlets or a CEA)
- 100 locations are available within Region 4 for the storage of fuel assemblies that meet the burnup requirements of this region, with 40 cells designated as Restricted Locations.

With these storage restrictions, the maximum number of rack locations usable for fuel assembly storage become:

80 Region 1 - Storage Locations (including Fuel Assemblies, Non-standard Fuel Configurations, and Non-fuel containing components)

80 Region 1 - Restricted Locations

164 Region 2 - Storage Locations

60 Region 2 - Restricted Locations

822 Region 3 - Storage Locations

100 Region 4 - Storage Locations

40 Region 4 - Restricted Locations

1346 Total Region 1 + 2 + 3 + 4 rack storage locations (including Restricted Locations)

Note that each Non-standard Fuel Configuration must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require insertion of Borated Stainless Steel Poison Rodlets or a full length, full strength CEA if stored in Region 3. A list of Non-standard Fuel Configurations that have been qualified for storage in any non-restricted pool location is included later in this section.

Fuel assemblies and Non-standard Fuel Configurations shall NOT be stored in Region 1 and 2 storage locations in which the Boraflex panel box has been removed. It is permissible to store non-fuel components in non-restricted locations whether or not the location contains a Boraflex panel box.

It should be noted, however, that both the spent fuel racks and the pool/building structure have been analyzed and qualified for the conditions and maximum loadings associated with consolidated fuel stored in each of the 1346 storage locations.

The spent fuel racks in all four regions are fabricated from 304 stainless steel which is 0.135 inches thick. Each cell is formed by welding along the intersecting seams. This enables each spent fuel rack module to become a free-standing module that meets the seismic design requirements without mechanical dependence on neighboring modules or fuel pool walls for support. The rack modules are classified ANS Safety Class III and Seismic Category I.

The mechanical design of the individual storage modules provides a flow path for convective cooling of the spent fuel assemblies and consolidated fuel storage boxes through natural circulation. Safety analyses of the thermal conditions within the spent fuel pool have been performed to assure that even with the most severe expected heat load, the stored spent fuel can be adequately cooled during normal and abnormal conditions.

b. Stability

The spent fuel racks have been designed for direct bearing onto the spent fuel pool floor. An adjustable pad (Figure 9.8–8D) is provided under each corner of the fuel rack. Fuel rack module leveling is accomplished by adjusting each pad height to conform to the pool floor. Each foot can be raised or lowered one-quarter inch and can rotate two to three degrees to accommodate pool floor variations during installation.

The spent fuel storage racks are designed to withstand forces generated during normal operation, an operating basis earthquake (OBE), or a safe shutdown earthquake (SSE) with the loads associated with consolidated fuel. Lateral and vertical seismic loads along with fluid forces are considered to be acting simultaneously on the fuel racks. The racks are designed to assure rack structural integrity while at the same time keeping the fuel in a subcritical state.

The seismic analysis of the spent fuel rack includes an assessment of the maximum sliding and tipping that can be expected. The racks are installed with a nominal gap of two inches between modules and the pool walls. The analysis has shown that the maximum motion of the racks, including tipping, sliding, and thermal expansion is less than the gap between adjacent modules.

Materials for all components of the rack support system are made of stainless steel with high temperature and corrosion resistance. The design and fabrication of the system is in accordance with the American Institute of Steel Construction (AISC) and American Society of Mechanical Engineers (ASME) Codes where applicable.

c. Safe Geometry

Analyses have been performed to ensure that the multiplication factor, K_{eff} of the storage racks including all uncertainties, is less than 0.95, with a 95% probability at the 95% confidence limit, under all normal conditions with water borated to a minimum of 550 ppm, and all accident conditions with water borated to a minimum of 2100 ppm. Analyses have also been performed to ensure that the K_{eff} of the storage racks, including all biases and uncertainties, is less than 1.00, on a 95% probability at the 95% confidence limit, under all normal conditions with unborated water. These analyses cover all four (4) regions of the spent fuel pool, which are designated Regions 1, 2, 3, and 4. These analyses also address other equipment in the spent fuel pool which may hold fuel assemblies on a temporary

basis, such as the new fuel elevator and the fuel transfer machine. Non-standard Fuel Configurations are also addressed by these analyses. A normal spent fuel pool water temperature range from 32°F to 150°F, and an accident condition temperature range from 150°F to boiling are bounded by these analyses.

1. Region 1, Region 2, and Region 4

Region 1 fuel storage racks do not credit fuel burnup and require fuel assembly storage in a 2 out of 4 storage pattern. Region 2 and Region 4 fuel storage racks credit fuel burnup and require storage in a 3 out of 4 storage pattern. Figure 9.8–7 shows the spent fuel rack storage patterns, including the designated Restricted Locations, for each region.

The criticality analysis for Region 1 assumes an infinite array of storage locations exist, and that 2 out of 4 storage locations are filled with fuel assemblies that have an initial planar average enrichment of 4.85 weight percent U-235. The criticality analysis for Regions 2 and 4 assumes that an infinite array of storage locations exist, and 3 out of 4 storage locations are filled with the maximum reactivity fuel assemblies allowed by the Technical Specifications. The moderator for each region is assumed to be water borated to 0 ppm for the $K_{eff} < 1.0$ case, and to a minimum level of 550 ppm for the $K_{eff} \leq 0.95$ case.

The Region 1 and 2 fuel storage racks contain Boraflex which is no longer credited as a neutron absorber.

For Regions 1, 2, and 4 the reactivity effects associated with the following manufacturing material and dimensional tolerances are accounted for: storage rack dimensional tolerances, and fuel dimensional and enrichment tolerances. Also accounted for are eccentric loading of fuel in the storage locations and calculational bias/uncertainties in the methods used to calculate K_{eff}. In addition the Regions 2 and 4 analyses account for the reactivity effects associated with axial fuel burnup variations, uncertainty in fuel isotopics, and uncertainty in the declared burnup, clad creep, fuel rod growth, and migration of volatile fission products.

2. Region 3 - Storage of Fuel Assemblies

Region 3 fuel storage racks do not contain a neutron poison. Fuel may be stored in all locations of Region 3, subject to meeting the fuel burnup restrictions specified by the Technical Specifications, and containing Borated Stainless Steel Poison Rodlets or a full length, full strength CEA (note that the criticality analysis for a given Non-standard Fuel Configuration may qualify it for Region 3 storage without these inserts). Fuel assemblies which have achieved the burnup required by the Technical Specifications and contain Borated Stainless Steel Poison Rodlets or a full length, full strength CEA may be stored in Region 3. The insertion of Borated Stainless Steel Poison Rodlets and full length, full strength CEAs are

credited in Region 3. The use of Borated Stainless Steel Poison Rodlets or CEAs reduces the required fuel burnup needed for fuel storage in Region 3. When Borated Stainless Steel Poison Rodlets are installed and credited, three poison rodlets must be placed inside the fuel assembly guide tubes. The three Borated Stainless Steel Poison Rodlets must be located in the center fuel assembly guide tube and any two other guide tubes which are diagonally opposite each other. Qualified fuel assemblies containing Borated Stainless Steel Poison Rodlets or a full length, full strength CEA, as well as qualified Non-standard Fuel Configurations, may be mixed in any manner in the Region 3 storage racks.

The criticality analysis for Region 3 assumes that an infinite array of storage locations exist, all filled with the maximum reactivity fuel assemblies allowed by the Technical Specifications. The moderator is assumed to be 550 ppm borated water at the water temperature which causes the maximum K_{eff}. In Region 3, a positive moderator temperature coefficient exists, therefore, higher water temperatures give larger values of K_{eff}. A water temperature of 150°F is the largest normal spent fuel pool temperature. Spent fuel pool temperatures greater than 150°F are evaluated as accident conditions.

For Region 3 the reactivity effects associated with the following manufacturing material and dimensional tolerances are accounted for: storage rack dimensional tolerances, Borated Stainless Steel Poison Rodlet diameter and boron loading tolerances, and fuel dimensional and enrichment tolerances. Also accounted for are eccentric loading of fuel in the storage locations, calculation bias/uncertainties in the methods used to calculate K_{eff}, the reactivity effects associated with axial fuel burnup variations, uncertainty in fuel isotopics, uncertainty in the declared burnup, clad creep, fuel rod growth, and migration of volatile fission products.

3. All Regions - Non-standard Fuel Configurations

Each region of the SFP may store Non-standard Fuel Configurations, except for the Restricted Locations and locations in which the Boraflex panel box is removed. However, each Non-standard Fuel Configuration must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require Borated Stainless Steel Poison Rodlets or a full length, full strength CEA if stored in Region 3.

The following list contains Non-standard Fuel Configurations which have been analyzed for SFP storage. They may be stored in all regions (except Restricted Locations and locations in which the Boraflex panel box has been removed). The list also indicates whether or not the Non-standard Fuel Configuration requires insertion of Borated Stainless Steel Poison Rodlets or a CEA if stored in Region 3. The analysis for each Non-standard Fuel Configuration concluded that the multiplication factor, K_{eff} of the storage racks including all uncertainties, is ≤ 0.95 at a 95% probability and 95% confidence level.

Non-standard Fuel Configuration	Description	Borated Stainless Steel Poison Rodlets or full length, full strength CEA Required for Region 3 Storage?
Failed Fuel Storage Grid Cage G-56	Failed fuel storage cage. Normal MP2 fuel lattice spacing	No
B&W Failed Rod Storage Container	NUSCO 40 failed rod storage basket. Contains 16 failed fuel rods and a tube with 5 fuel pellet fragments.	No
Consolidated Fuel Storage Box	Consolidated fuel from two depleted fuel assemblies in each basket.	No
Individual Rod Storage Container	Hollow stainless steel tube stored in a fuel assembly guide thimble with one failed fuel rod inside.	No
Storage Cage 8FR	Donor cage has normal assembly pitch and lattice, 52 depleted fuel pins, 12 natural enrichment pins, 86 non-fuel filler rods, and 26 empty locations.	No
Fuel Assembly P-26 (has missing fuel rod)	Failed assembly in which a fuel rod was removed and not replaced.	Yes

4. Other Equipment Which May Store Fuel on a Temporary Basis

There are two pieces of equipment in the spent fuel pool, other than the storage racks, which could store fuel on a temporary basis. They are: the new fuel elevator and the fuel transfer machine.

The new fuel elevator has been analyzed to be capable of storing a fresh 4.85 weight percent U-235 fuel assembly, and maintaining the K_{eff} of the storage racks including all uncertainties, to less than 0.95, with a 95% probability at the 95% confidence limit.

The fuel transfer machine is capable of storing two fuel assemblies. The fuel transfer machine has been analyzed to be capable of storing two fresh 4.85 weight percent U-235 fuel assemblies, and maintaining K_{eff} of the transfer machine configuration including all uncertainties, to less than 0.95, with a 95% probability at the 95% confidence limit.

5. Accident Conditions

Analyses have been performed to ensure that the multiplication factor, K_{eff} of the storage racks including all uncertainties, is less than 0.95, with a 95% probability at the 95% confidence limit, under all normal conditions with credit for 550 ppm borated water, and all accident conditions with credit for a total of 2100 ppm boron in the water.

The following conditions were considered in these analyses (Note that the following scenarios assume fuel contains zero gadolinia loading):

- Multiple misloads of 4.85 weight percent U-235 fresh fuel assemblies into Region 1 and 2 storage locations.

- Multiple misloads of higher than typical reactivity depleted fuel assemblies (initial enrichment of 4.20 weight percent U-235 and 10 GWD/MTU burnup) into Region 3 and 4 storage locations.

- Mislead or dropping of a single 4.85 weight percent U-235 fresh fuel assembly into a Region 4 storage location.

- Dropping or misplacing a 4.85 weight percent U-235 fresh fuel assembly between Region 3 and the new fuel elevator, with a 4.85 weight percent U-235 fresh fuel assembly in the new fuel elevator.

- Three 4.85 weight percent U-235 fresh fuel assemblies are allowed to touch edge to edge. This bounds a scenario in which two fresh fuel assemblies are in the fuel transfer cart and another fresh fuel assembly, by some undefined means, is allowed to come edge to edge with the fuel assembly in the transfer cart.

- Loss of pool cooling resulting in SFP temperature increase and voiding.

- Dropping of a fuel assembly or Non-standard Fuel Configuration which comes to rest on top of the racks in a horizontal position.

- Lateral rack movement due to a seismic event affecting the spacing between racks of any of the Regions.

The bounding scenario is multiple misloads of higher than typical reactivity depleted fuel assemblies (initial enrichment of 4.20 weight percent U-235 and 10 GWD/MTU burnup) into Region 3 and 4 storage locations. The analysis shows that a 2100 ppm boron concentration ensures that K_{eff} remains ≤ 0.95, including all uncertainties and biases.

The spent fuel shipping cask is placed in the spent fuel pool cask laydown area for loading. The spent fuel cask is designed such that spent fuel assemblies can be placed into the cask while maintaining adequate water coverage above the fuel assembly. After loading with fuel assemblies, the cask cover is fastened to the cask and the cask is transferred to the cask wash area by the spent fuel cask crane. Interlocks are provided on the cask crane to prevent handling of the shipping cask above the storage racks and stored fuel. The cask cover is fastened in place before the cask is moved from the pool to prevent the fuel assemblies from falling out in the unlikely event of a dropped cask accident. Also it prevents the cask cover from becoming a missile in the event of a dropped cask accident. While the cask is en route to the decontamination pit at elevation (+) 38 feet six inches, plastic or other means are provided to catch any contaminated water dripping off the cask. The cask is washed down with demineralized water which is drained to the radioactive processing system. Checks are made to ensure that surface contamination meets transportation regulatory requirements.

Temperature monitoring and alarm instrumentation as described in Section 9.5 are provided in the fuel pool to assure that the decay heat from the spent fuel elements is being removed. A local monitor is used to assure that proper radiation levels are maintained. Means are provided to control entry of personnel and to account for the flow of tools in and out of the area.

Figure 9.8–9 shows the physical arrangement of the ten leak collection zones of the spent fuel pool liner. Figure 9.5–1 of the FSAR shows, schematically, the leak detection equipment.

Leakage from any portion of the spent fuel pool liner would collect in one of the ten zone collector channels and would drain out of the channel into a common header. Once in the header the leakage would accumulate upstream of closed valve 2-RW-175. This accumulation would trip level switch LS-7321 which would, in turn, annunciate the appropriate alarm on the main control board.

The liner plate was manufactured under quality assurance requirements and was tested for leak tightness prior to plant operation. Therefore, no leakage is expected under normal conditions.

The spent fuel cask crane, which includes a monorail hoist, and spent fuel pool platform crane are capable of passing over the spent fuel pool. Since both are designed and analyzed for seismic Category l requirements, their failure and

subsequent falling into the pool is thought to be an incredible accident and will not be considered in this discussion.

The spent fuel cask crane handles the following items in the area of the spent fuel pool:

a. The spent fuel cask.

b. The spent fuel pool bulkhead gates.

c. The new fuel assemblies.

The spent fuel cask crane (SFCC), which includes the main hoist (125 tons) and auxiliary hoist (15 tons), has been designed to meet the single failure proof requirements in accordance with NUREG-0612 and NUREG-0554. The design features will ensure that a single failure will not result in the loss of the capability of the system to safely retain the load. Load drop events are not credible for loads lifted by the SFCC when handled and rigged in accordance with the single failure criteria.

The spent fuel pool bulkhead gates are essentially 26 foot long by three foot six inches wide by one inch thick flat plates weighing approximately 5100 pounds each. The gates are normally stored on the west wall of the spent fuel storage area just south of the transfer slot and on the east wall just south of the cask laydown area transfer slot. If for some reason either transfer canal or the cask laydown area must be drained, the spent fuel cask crane moves the gate from its storage location to the transfer slot. Movement of the gates will be performed under administrative controls to ensure compliance with the heavy loads guidelines of NUREG-0612 through use of a “single failure proof” lifting system or through the use of controlled safe load paths and therefore fuel damage from a spent fuel pool bulkhead gate drop is not a credible event.

New fuel assemblies are transferred to the spent fuel pool by using a short fuel handling tool attached to the auxiliary hook of the spent fuel cask crane and transferring the new fuel assembly into the new fuel elevator located in the spent fuel pool. The new fuel assembly is handled and rigged in accordance with the single failure proof criteria of NUREG-0612. Therefore, a new fuel assembly load drop is not a credible event.

The spent fuel pool platform crane manipulates the fuel under water by means of long tools attached to the hoist. Since there is some operator handling in attaching or detaching the tools, it is conceivable that a tool could be dropped. The tools are approximately 32 feet long by three inches in diameter, weighing approximately 270 pounds. The pool water would offer practically no resistance to this shape, but the weight and drop height are small enough that no damage would be caused to rack or the fuel stored inside even if the dropped tool hit directly on a fuel

assembly. It is also conceivable that a fuel assembly could be dropped while being handled over the spent fuel racks. The racks have been designed to withstand the dropping of the fuel assembly with no appreciable damage.

The rack has been designed and analyzed to take an uplift force equal to 6000 pounds. The normal capacity of the spent fuel pool platform crane is 2000 pounds, which is less than the analyzed weight so that it is unlikely that the hoist would cause any damage if it were caught on the rack and the crane's overload device failed to operate.

6. Spent Fuel Pool Dilution

Criticality analysis has shown that 600 ppm of soluble boron is sufficient under normal conditions in the SFP, to assure not to exceed the 0.95 k_{eff} design basis (including biases and uncertainties). The actual analysis shows that 550 ppm of soluble boron is sufficient to meet this k_{eff} requirement. The criticality analysis has also shown that 0 ppm of soluble boron, under normal conditions in the SFP, assures k_{eff} is maintained less than 1.00 (including biases and uncertainties). An engineering analysis of potential scenarios which could dilute the boron concentration in the SFP demonstrates that sufficient time is available to detect and mitigate a boron dilution prior to reaching 600 ppm, thus not exceeding the 0.95 k_{eff} design basis (including biases and uncertainties). It should be noted that the criticality analysis credits 2100 ppm of soluble boron in the SFP for certain accident conditions. However, a boron dilution event does not need to be considered for these conditions. This is because the simultaneous occurrence of two unlikely and independent events such as a boron dilution event and another independent accident condition is not required to be considered.

The systems which could dilute the spent fuel pool, either by direct connection to the spent fuel pool, or by a potential pipe crack/break have been analyzed. There is no automatic spent fuel pool level control system in the spent fuel pool, so that any dilution to the spent fuel pool will add water to the spent fuel pool. Therefore the addition of unborated water to the SFP will lead to increased SFP water level, and if not controlled, an overflow of the SFP.

The ability to prevent the SFP soluble boron concentration from being diluted from less than the TS minimum value of 2100 ppm (analysis was performed at 1720 ppm) to a value of 600 ppm has been demonstrated by showing that each potential dilution source meets one of the following two criteria:

- The dilution source does not have a sufficient volume of unborated water to be capable of diluting the SFP soluble boron concentration from 1720 ppm (less than the 2100 ppm TS requirement) to 600 ppm.

- The dilution source flowrate of unborated water is ≤ 200 gpm. At a dilution flowrate of 200 gpm of unborated water, at least 19 hours will be needed

for the SFP soluble boron concentration to be reduced from 1720 ppm (less than the 2100 ppm TS requirement) to 600 ppm. Should a dilution occur, the water added to the SFP will eventually cause a SFP high water level alarm in the control room alarm, or be detected by the Plant Equipment Operator (PEO) detecting high SFP water levels or SFP overflow. A time of 19 hours is ample time to detect and terminate the dilution event.

7. Spent Fuel Storage in an Independent Spent Fuel Storage Installation (ISFSI)

With the installation of the Millstone ISFSI, “dry” storage of Unit 2 spent fuel on the Millstone ISFSI provides an available fuel storage option. Millstone has selected the NUHOMS Standardized and NUHOMS EOS spent fuel storage systems under Orano-TN’s general license as authorized per 10 CFR 72 and approved by the NRC in Certificates of Compliance Number1004 and 1042, respectively, for an Independent Spent Fuel Storage Installation.

The ISFSI is designed to accommodate on site spent fuel storage through the end of plant life including 20-year license renewal, or until transfer of the spent fuel to a DOE repository. The spent fuel storage canisters for each system are dual purpose, designed for storage at an ISFSI per 10 CFR 72 requirements and transport off site in a spent fuel shipping cask per 10 CFR 71 requirements.

Both NUHOMS ® systems consist of reinforced concrete horizontal storage modules (HSMs) and steel dry shielded canisters (DSCs) assembled on a concrete pad within the site protected area boundary. System operation is totally passive crediting natural air circulation for cooling.

Spent fuel is selected based on the Unit 2 spent fuel strategy and the NUHOMS Standardized or EOS Technical Specification requirements for fuel qualification. The DSC consists of a shell and basket assembly, which can accommodate 32 (Standardized) or 37 (EOS) fuel assemblies. The DSC is inserted into a transfer cask and placed in the cask laydown area of the Unit 2 spent fuel pool for fuel loading. Once loaded, the transfer cask/DSC is relocated to the cask wash pit for draining, drying, closure operations and decontamination. The transfer cask is utilized to transfer the loaded DSC to the ISFSI pad for loading into an HSM. The HSM array consists of precast concrete components forming a series of concrete storage modules for dry shielded canisters storing spent fuel.

9.8.2.1.3 Fuel Transfer Tube and Isolation Valve

A fuel transfer tube extending through the containment wall connects the refueling pool with the spent fuel pool as shown in Figure 9.8–10. During reactor operation, the transfer tube is closed by means of a manually operated isolation valve located on the spent fuel pool side of the transfer tube and a blind flange located inside the containment. Section 9.8.3.2, "Standard System Operations" describes preparation of the fuel transfer tube for refueling operations.

Section 9.8.3.2 also includes a description of actions to isolate the transfer tube at the completion of refueling operations.

The transfer tube isolation valve is bolted to the spent fuel pool end of transfer tube. The valve is supported in such a manner to allow for horizontal movement along the centerline of the transfer tube. The manual operator for the valve is designed to allow for movement of the valve due to thermal expansions and still permit operation. The valve stem extends above the operating pool water level at elevation (+) 36 feet 6 inches and designed for manipulation from the operating deck at elevation (+) 38 feet 6 inches. The valve is furnished with a spur gear operator and position indicator.

9.8.2.1.4 Spent Fuel Pool Platform Crane

The spent fuel pool platform crane consists of a bridge which spans the width of the spent fuel pool and a trolley mounted hoist that traverses the length of the bridge. The crane is located above the spent fuel pool and rides on rails set in the concrete on the north and south sides of the pool. Electric motors position the bridge and trolley over the specified rack location, new fuel elevator, transfer carriage upender, and spent fuel shipping cask.

The spent fuel pool platform crane transfers:

1. spent fuel assemblies from the upender to the spent fuel storage rack,

2. new fuel assemblies from the new fuel elevator to the upender normally via temporary positions in the spent fuel storage racks,

3. spent fuel assemblies from the spent fuel storage racks to the spent fuel shipping cask,

4. damaged or leaking fuel assemblies into a special container which, in turn, is placed into a special damaged fuel shipping cask,

5. spent fuel assemblies to and from the new fuel elevator for inspection

   In addition, the spent fuel pool platform crane may be used for retrieving dropped items from the spent fuel pool or to place in and retrieve from the pool various items that may be stored in the spent fuel pool for radioactive decay.

6. CEAs in the spent fuel storage racks.

9.8.2.2 Components

9.8.2.2.1 New Fuel Storage Racks

The new fuel storage racks are a welded construction of stainless steel to provide dry, vertical storage for 76 assemblies. The new fuel assemblies are placed into the rack through the top of the

new fuel storage rack. All welding of the racks conforms to the applicable requirements of ASME Section VIII.

9.8.2.2.2 Design and Fabrication of Spent Fuel Racks

The modules of Regions 1, 2, 3, and 4 are designed and licensed to store fuel with a maximum initial planar average enrichment of 4.85 weight percent U_{235} depending on the fuel history and the specific requirements of the respective Region of the spent fuel pool where the fuel is to be stored.

The spent fuel storage racks are fabricated with 304 stainless steel having a maximum carbon content of 0.065 percent, with a 304 stainless steel module wall thickness of 0.135 inches. The racks are monolithic honeycomb structures with square fuel storage locations as shown in Figure 9.8–8A. Each storage location is formed by welding stainless steel sections along the intersecting seams, permitting the assembled cavities to become the load bearing structure, as well as framing the storage cell enclosures. Each module is free standing and seismically qualified without mechanical dependence on neighboring modules or pool walls. This feature allows remote installation (or removal if required for pool maintenance). Reinforcing plates at the upper corners provide the required strength for handling. The racks are designed to withstand the dropping of a fuel assembly or a consolidated storage box onto the top of the racks with no loss of design function. Figure 9.8–7 shows the arrangement of Regions 1, 2, 3, and 4 modules in the Millstone Unit Number 2 pool, including the Restricted Locations.

Stainless steel bars, which are inserted horizontally through the rectangular slots in the lower region of the module, support the fuel assemblies. The support bars are welded in place and support an entire row of fuel assemblies. The module is supported by adjustable pads to facilitate leveling at installation.

Loading of the fuel racks is facilitated via movable lead-in funnels. The openings of the funnels are symmetrical. The funnels are placed on top of the rack module.

Regions 1 and 2 have a combined maximum total of 244 locations available for storage. Each storage location, including Restricted Locations, contains or contained a Boraflex poison insert which is no longer credited. Some storage locations may have their Boraflex panel boxes removed. Fuel assemblies and Non-standard Fuel Configurations shall NOT be stored in locations where these boxes have been removed (it is permissible to store non-fuel components in these locations except for the Restricted Locations). These inserts are made up of Boraflex sheets enclosed, but not sealed, between two stainless steel sheets. The neutron absorbing material, Boraflex, is designed to be compatible with the pool environment. Positive venting of the poison is provided by the three-eighth inch diameter hole in the inner walls of the inserts. The stainless steel sheets are configured such that a minimum water gap is maintained between the insert and the cell wall. These inserts lock into the storage cavity using a spring locking mechanism on the upper end which snaps into a hole in the surrounding cell wall. These poison inserts are neutron absorbers. A typical Region 1 and 2 fuel rack module and poison box is shown in Figures 9.8–8A and 9.8–8C.

The neutron absorbing material, Boraflex, existing in Regions 1 and 2 spent fuel racks is manufactured by Brand Industrial Services, Inc., and fabricated in accordance with the quality assurance criteria of 10 CFR 50, Appendix B. Boraflex is a silicone based polymer containing fine particles of boron carbide in a homogeneous, stable matrix.

Region 3 consists of a maximum of 822 storage locations with no Restricted Locations, and spent fuel assemblies and qualified Non-standard Fuel Configurations can be stored in these locations. Region 3 employs a combination of burnup credit and insertion of Borated Stainless Steel Poison Rodlets or a full length, full strength CEA for reactivity control (Non-standard Fuel Configurations may be qualified for Region 3 storage without insertion of Borated Stainless Steel Poison Rodlets or a CEA).

Region 4 has a maximum of 100 storage locations available for storage of fuel assemblies and Non-standard Fuel Configurations, and also employs burnup credit for fuel assemblies.

Regions 3 and 4 are reserved for fuel that has sustained a specified burn-up and for qualified Non-standard Fuel Configurations. The spent fuel rack design is based on the criticality acceptance criteria specified in Revision 2 of Regulatory Guide 1.13 (1981 draft version [(1)]) which allows credit for reactivity depletion in spent fuel. (Previously, the physics criteria for fuel stored in the spent fuel pool were defined by the maximum unirradiated initial enrichment of the fuel.) A typical Region 3 and 4 module is shown in Figure 9.8–8B.

The entire fuel assembly storage rack is constructed of type 304 stainless steel. All welded construction is used in the fabrication of the fuel assembly storage rack. Design of the individual cells provides assurance of smooth, snag free passage in the storage cavities so that it is highly improbable that a fuel assembly could become stuck in the rack.

The storage racks are structurally sized in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, design allowable stress limits. Welding is performed in accordance with ASME Boiler and Pressure Vessel Code, Section IX. The spent fuel racks are Seismic Category I, Quality Class 2 structure, ANS Safety Class III.

9.8.2.2.3 Reactor Refueling Machine

The reactor refueling machine is shown in Figure 9.8–11. The refueling machine is located above the refueling pool and rides on rails set in the concrete on each side of the pool. The refueling machine consists of a bridge which travels along the north-south axis, a trolley which travels along the east-west axis on the bridge, and a hoist which raises and lowers fuel assemblies with the grapple. The grapple is a pneumatically-operated, rotating hook which engages the fuel

(1) The 1981 draft of Revision 2 of Regulatory Guide 1.13 was referenced in the Millstone Unit 2 spent fuel pool storage expansion licensing amendment request submitted in 1985. The NRC approved this request in 1986 (Amendment Number 109). Subsequently, the final version of the Revision 2 of Regulatory Guide 1.13, approved in 2007, removed discussion of credit for reactivity depletion in spent fuel.

assembly. The hoist lifts the fuel assembly and the grapple mechanism into the refueling machine's hoist box, which is then lifted into the refueling machine's mast, which acts as a protective shroud around the hoist box and fuel assembly. The hoist, hoist box, mast and grapple mechanism are all supported from the trolley.

Bridge, trolley, and hoist drives are variable speed with a range of 0 to 50 ft/min on the bridge, 0 to 30 ft/min on the trolley and 0 to 40 ft/min on the fuel hoist. All other devices on the RFM operate at a single speed. Operating speeds of the air actuated devices (grapple and spreader) are adjustable at a pneumatic control panel.

The refueling machine has three (3) modes of operation; automatic, semi-automatic and manual. All three axes of motion can be controlled via handwheels in the event that the control system is inoperable.

9.8.2.2.3.1 RFM Control System

The RFM control system monitors all hoist movements using dual-redundant position encoders on all three axes of movement, a load weighing system and the following condition detection limit switches:

- Slack Cable
- Fuel Spreader Retracted
- Fuel Spreader Extended
- Grapple Closed
- Hoist-up Limit
- Hoist Box Latched
- Grapple Open

The RFM control console is installed on the trolley deck during refueling operations only and moved to a storage location at the conclusion of refueling operations. All external electrical connections to the Control Console are made by quick-disconnect connectors. The console has a horizontal shelf and a vertical panel. The horizontal shelf has a control panel which contains the operating controls and indicator lights for the bridge, trolley and hoist mechanism, an operator consolet with an analog joystick for each axis of movement and grapple and fuel spreader control switches. The vertical panel contains additional indicator lights, override and emergency stop push buttons, hoist load meter, computer monitor and position readouts. The control console also includes a Gai-Tronics jack, phone and speaker to support refueling communication needs.

9.8.2.2.3.2 Bridge and Trolley

The bridge is operated by moving the joystick towards the FORWARD or REVERSE positions, the trolley is operated by moving the trolley joystick towards the LEFT or RIGHT position. Speed is determined by how far the joystick is displaced from the neutral position. The refueling machine control system monitors all bridge and trolley movements using dual-redundant, rack and pinion encoders. In the event of a power loss or emergency stop, the bridge brake will set and stop the bridge with a deceleration less than 0.1g and the bridge encoders will not lose track of the bridge position. The trolley will brake to a complete stop and the trolley encoders will not lose track of the trolley position. Both electric brakes can be manually released for handwheel operation. The trolley also features two deck-mounted viewing windows and a pneumatic control panel which operates the fuel grapple, fuel spreader and hoist box latch and a camera system that provides operators with an overall view of the area under the trolley deck.

9.8.2.2.3.3 Hoist

The main fuel hoist lifts and lowers the hoist box assembly and consists of an AC servo motor, motor drive, gearbox, two (2) electric-brakes, hoist drum, hoist cable and a limit switch. The hoist is mounted on the trolley and has a rated capacity of 3590 pounds. The main fuel hoist is operated by moving the hoist joystick in the RAISE and LOWER position. Speed is determined by how far the joystick is displaced from the neutral position. The refueling machine control system monitors all hoist movements using dual-redundant, stainless steel tagline encoders. The load weighing system provides a six (6) position selector switch on the control console to choose the fuel type, and thus adjust the individual setpoints for each type.

The mast is equipped with a retractable stop called the Latch, located near the top of the mast. During operation over the upender, the hoist box automatically engages this stop to hold it in its upper, retracted, position. When the weight of the hoist box is transferred to the Latch, a “Hoist Box Latched” message is displayed on the touch screen, indicating the hoist box is supported by the Latch. The hoist is latched automatically when the refueling machine is over the upender area, and unlatched automatically when it is not over the upender area.

9.8.2.2.3.4 Mast/Hoist Box

The mast is a vertically-oriented, cylindrical structure supported by the trolley. Mounted within the mast is the hoist box. The hoist box descends from the mast when the hoist is lowered. The grapple is rotated to an Open position via a pneumatic mechanism. Lowering the grapple further allows it to pass over the top fitting of the fuel assembly. The grapple is rotated to a closed position, engaging the fuel assembly. The hoist raises the fuel assembly and the grapple mechanism into the hoist box, which is then lifted into the mast, which acts as a protective shroud around the hoist box and fuel assembly. The hoist box is equipped with a fuel spreader to ensure adequate spacing between fuel assemblies. The mast can also be manually rotated via a hand wheel.

9.8.2.2.3.5 Auxiliary Hoist

The RFM Auxiliary Hoist is rated to 1,000 pounds capacity and includes a motorized trolley, load cell and pendant. The pendant is used for manual control of the auxiliary hoist, its trolley and the RFM bridge. The pendant also contains local load and height indicators and can be used to initiate an Emergency Stop. The auxiliary hoist travels along the east-west axis on the monorail frame mounted to the RFM bridge. In the event of a power loss, the auxiliary hoist will brake to a complete stop. The electric brake can be released for manual handwheel operations.

9.8.2.2.3.6 Modes of Operation

In Manual mode, control of all axes of motion (bridge, trolley and hoist) is accomplished using joysticks that provide analog speed commands to the PLC. All RFM travel boundaries, operator displays, and safety interlocks remain functional. Arrows will indicate bridge and trolley positions as the RFM moves in response to joystick commands. The encoder positions associated with the desired location will also be displayed for numerical comparison with the instantaneous position indications on the control console. Once over the core, hoist speed icons will indicate regions where the hoist may operate at high speed or low speed.

In Semi-automatic mode, the Operator manually enters a destination coordinate via the touchscreen. The bridge and the trolley will move simultaneously toward the destination coordinate. Hoist motion will be controlled manually along with the grapple.

Automatic mode includes all features of the Semi-automatic Mode with the added capability to execute a sequence of movements based on fuel management software stored on the computer. The sequence of movements will be displayed on screen. Operators will have the opportunity to verify each move before it is executed.

Interlock override mode allows the operators full control of the RFM in the event of a PLC failure. Interlock override mode bypasses the PLC via a key switch on the control console. With the PLC bypassed, the RFM will operate as commanded by the operators at intermediate speeds (Trolley - 5 ft/min, Bridge and Hoist - 10 ft/min) with no regard to PLC programmed travel limits and interlocks. The Hoist Up Limit and Slack Cable Limit switches will still prevent hoist movements in the up and down direction.

All three axis of motion can also be controlled via handwheels in the event that the control system is inoperable.

9.8.2.2.4 Transfer Carriage

A transfer carriage conveys the fuel assemblies between the refueling pool and the spent fuel storage area. Large wheels support the carriage and allow it to roll on tracks within the transfer tube. Track sections at both ends of the transfer tube are supported from the pool floor and permit the carriage to be properly positioned to the upending mechanisms. The carriage is driven by steel cables connected to the carriage and through sheaves to its driving winch mounted on the

operating floor located on the spent fuel pool side. A two-pocket fuel carrier is mounted on the carriage and is pivoted for tilting by the upending machines.

The support for the replaceable rails on which the transfer carriage rides are welded to the 36 inch diameter transfer tube. The rail assemblies are fabricated to a length which will allow them to be lowered for installation in the transfer tube. There are no rails in the valve on the fuel storage side of the transfer tube to allow closing of the valve.

9.8.2.2.5 Upending Machine

Two upending machines are provided, one in the containment refueling pool and the other in the spent fuel pool. Each consists of a structural steel support base from which is pivoted an upending straddle frame which engages the two pocket fuel carrier. When the carriage with its fuel carrier is in position within the upending frame, the pivots for the fuel carrier and the upending frame are coincident. Two hydraulic cylinders, attached to both the vertical and horizontal positions required by the fuel transfer procedure. Each hydraulic cylinder can perform the upending operation and can be isolated in the event of its failure. A long tool is also provided to allow manual rotation of the fuel carrier in the event that both cylinders fail or hydraulic power is lost.

9.8.2.2.6 Reactor Vessel Head Lifting Rig

The reactor vessel head lifting rig is shown in Figure 9.8–13.

The lifting rig is composed of a removable three part lifting frame and a three part column assembly which is attached to the reactor vessel closure head. The column assembly supports the three hoists which are provided for handling the hydraulic stud tensioners, the studs, washers and nuts, and links the lifting frame with the reactor vessel head.

The weight of the reactor vessel head, with the lift rig, CEDMs and CEDM cooling shroud attached, is approximately 140 tons and its shape is approximated by a right circular cylinder 17 feet in diameter and 37 feet high. The maximum height above the reactor vessel flange to which the reactor vessel head may be lifted is 30 feet. In operation, the head with lift rig attached will be lifted above the vessel only as high as required to clear the control element drive mechanism (CEDM) extension shafts, approximately 28 feet.

The equipment used in lifting the reactor vessel head, namely the head lift rig and the reactor polar crane, is analyzed for Seismic Class I requirements and has been manufactured under the quality assurance supervision of a “Q-Listed” item.

9.8.2.2.7 Core Support Barrel Lifting Rig

The core support barrel lifting rig, shown in Figure 9.8–14, is provided to withdraw the core barrel from the vessel for inspection purposes. The upper clevis assembly is a tripod shaped structure connecting the lifting rig to the containment crane lifting hook. The lifting rig includes a spreader beam providing three attachment points which are to be threaded to the core support barrel flange. This is accomplished manually from the refueling machine bridge by means of the

lifting bolt torque tool. Correct positioning of the lifting rig is assured by attached guide bushing which mate to the reactor vessel guide pins.

9.8.2.2.8 Upper Guide Structure Lifting Rig

The upper guide structure (UGS) lifting rig is shown in Figure 9.8–15.

This lifting rig consists of a delta spreader beam which supports three columns providing attachment points to the UGS. Attachment to the upper guide structure is accomplished manually from the working platform by means of lifting bolt torque tools. The integral ICI hoist connects to an adaptor which is manually attached to the ICI structure by utilizing an adapter torque tool The ICI is then lifted by the crane hook. The upper clevis assembly, which is common to this and the core support barrel lifting rig, is installed prior to lifting of the structure by the crane hook. Correct positioning is assured by attached bushings which mate to the reactor vessel guide pins.

The weight of the UGS and lift rig is approximately 58 tons and its shape can be approximated by a right circular cylinder 14 feet in diameter and 44 feet in height. The maximum height to which the UGS may be lifted is approximately 18 feet. The normal lift height is approximately 13.5 feet.

The equipment used in lifting the UGS, namely the UGS lift rig and the reactor polar crane, is analyzed for seismic Category I requirements and has been manufactured under the quality assurance supervision as outlined in Appendix 1B (located in the original FSAR dated August 1972). In addition the maximum capacity of the reactor polar crane is 160 tons which is approximately 100 tons greater than the lift weight of the UGS.

9.8.2.2.9 Stud Tensioners

Hydraulically operated stud tensioners are used to apply and remove the preload on the reactor vessel head closure studs. These tensioners are suspended from electric hoists which are attached to the head lift rig. The tensioner assemblies, when placed over the studs, rest upon the reactor vessel head flange. An internal socket is attached to the stud by engagement with the stud upper threads and when hydraulic pressure is applied to the stud tensioner pistons, the studs are elongated a predetermined amount. After closure nuts are seated, the hydraulic pressure is released which result in the preload necessary to maintain the seal between the reactor vessel and the reactor vessel head.

A portable pumping unit mounting on a two-wheel truck is the source of hydraulic power. Two air-operated pumps connected in parallel produce the hydraulic pressure which is routed by hose to the tensioner pistons. The control panel contains an air gauge indicating the regulated air pressure and air valve for operating the pump. An hydraulic gauge showing the pump pressure is also provided as is the hydraulic valve.

9.8.2.2.10 Surveillance Sample Handling Tool

The tool is of simple tubular construction and is approximately 40 feet long to allow it to be operated from the refueling machine. It has a female acme thread on the end which mates with a

male thread on the sample holder. Rotation of the tool handle threads the tool to the holder and also unlocks the latching mechanism, which allows removal of the holder by use of the overhead crane. (Figure 9.8–16).

9.8.2.2.11 Mechanism Uncoupling Tool

This tool is approximately 17 feet long and consists basically of two concentric tubes with a funnel at the end to facilitate engagement with the extension shafts. When installed, pins attached to the outer tube are engaged with the extension shaft outside diameter and the pins carried by the inner tube are inserted in the inner operating rod of the extension shaft. The inner tube of the tool is then lifted and rotated relative to the outer tube which compresses a spring allowing the gripper to release, thus separating the extension shaft from the CEA. The extension shaft is then handled by the tool.

9.8.2.2.12 Underwater Television

The underwater television camera has been removed from the Fuel Hoist Assembly for the Reactor-Refuel Machine. An updated video camera system is utilized during fuel movement activities.

9.8.2.2.13 In Mast Sipping Equipment

An in mast sipping system is provided to test irradiated fuel assemblies for fuel cladding defects. This system consists of both permanently mounted components and a portable skid that is set onto the refueling machine trolley when a sipping test is to be performed. The in mast sipping system takes a water sample from around each fuel assembly as it is lifted from the core into the mast. The water soluble and/or gaseous fission products escaping from a failed fuel rod during this handling process are drawn from the mast region of the fuel assembly handling machine through tubing, degassed continuously, and the gas is directed to a detector. The detector measures the fission products and determines the isotopes present. A determination of fuel integrity can be made based upon the concentrations and types of isotopes present. The sample system is then directed back to the refuel cavity and released below the water surface.

The tubing and nozzles are permanently installed. The sampling detectors and other equipment are part of the portable skid installed onto the refueling machine for each refueling that sampling is required.

9.8.2.2.14 Hydraulic Power Package

The hydraulic power package provides the motive force for raising and lowering the upender with the fuel carrier. It consists of a stand containing a motor coupled to a hydraulic pump, a sump reservoir, valves and the necessary hoses to connect the power package to the hydraulic cylinders on the upender. The valves can be lined up to actuate either or both upender cylinders. The hydraulic fluid is deionized water.

9.8.2.2.15 Fuel Transfer Tube and Isolation Valve

The 36 inch diameter transfer tube is contained in a 42 inch diameter tube which is sealed to the containment. The two concentric tubes are sealed to each other by a welding ring on the refueling pool side and a bellows-type expansion joint on the spent fuel pool side. The transfer tube is fabricated of ASTM-A-358, Type 304 material. All welding on the tube is in accordance with the requirements of ASME Section VIII, where applicable.

Description of the transfer tube isolation valve is given in Table 9.8-1.

9.8.2.2.16 Spent Fuel Pool Platform Crane

The general arrangement of the spent fuel pool platform crane is shown in Figure 9.8–17. The crane travel limits are shown in Figure 9.8–18. Details of the crane are given in Table 9.8-2.

The crane is provided with two hold-down bars bolted to each truck frame with the flange extending under the head of the rail to preclude derailment under seismic loads.

Bumpers - Four heavy duty spring bumpers are provided at the ends of each bridge truck to prevent crane derailment and smooth stopping. The bridge and trolley are provided with encoders as discussed in Section 9.8.4.1.10.

The factor of safety for the crane is 5.0, except that a factor of 10 is used for cables.

Welded fabrication is in accordance with the latest American Welding Society (AWS) Standard Specification for Welded Highway and Railway Bridges (AWS D2.0). Structural components of the crane which are not to be submerged in the spent fuel pool are fabricated from ASTM A-36 material or equal.

Structural components of this equipment which are designed for submerged service or come in contact with components which have been, are fabricated of austenitic stainless steel.

All three motions of this equipment (bridge, trolley, hoist) may be accomplished manually.

9.8.2.2.17 Spent Fuel Cask Crane

The spent fuel cask crane, located in the auxiliary building as shown in Figure 9.8–19, is described in Table 9.8-3.

9.8.2.2.18 New Fuel Elevator

The new fuel elevator provides a means of lowering new fuel assemblies from the fuel handling area at elevation (+) 38 feet 6 inches down to a level where the spent fuel platform crane can pick up the assembly by means of the long fuel handling tool. The elevator consists of carriage, that is similar in design to a spent fuel storage rack cell, that rides on a track mounted to imbeds in the spent fuel pool wall. Motive force is provided by an electric motor through a winch with

redundant hoisting cables. Electric interlocks are provided to give a “soft stop” at both the upper and lower limits of travel. All wetted parts are of stainless steel.

The new fuel elevator is also used as a spent fuel inspection stand. A manually operated turntable on the bottom of the carriage rotates the spent fuel assembly in front of a portable underwriter TV camera to allow inspection. Raising and lowering of the elevator allows full length inspection. An electrical interlock prevents raising of a spent fuel assembly above the minimum shielding depth. A mechanical interlock provides backup should the electrical interlock fail.

9.8.2.2.19 Spent Fuel Inspection Machine
(Deleted - Equipment Permanently Removed)

9.8.2.2.20 Heated Junction Thermocouple Handling Canister Design

The heated junction thermocouple (HJTC) probe handling hardware configuration is shown on Figure 9.8–20. The Grayloc flange is disassembled, and a protective bullet nose assembly is screwed onto the HJTC probe seal plug. The hub adapter is lowered over the bullet nose onto the Grayloc seal ring and secured (hand tight) using the Grayloc clamp assembly. The handling canister is then positioned over the bullet nose assembly and a cable is attached to the bullet nose assembly. The handling canister is then lowered over the bullet nose assembly with the weight of the handling canister supported by the overhead crane. The cable assembly supporting the HJTC Probe is then tied off to a lanyard attached to the canister. The winch at the base of the canister is removed with quick disconnect pins. The handling canister assembly, with the HJTC probe inside, is then removed to the storage area using the overhead crane. The operation is then repeated for the other HJTC probe location with a second handling canister. After the reactor vessel head has been reinstalled, the above procedure is reversed to reinsert the HJTC probes. Some of the specific features for the design of the handling hardware are as follows:

a. The bottom zone of the canister contains sealed attenuating material such that exposure to personnel from the HJTC probe will be kept reasonably low. Estimated radiation levels are provided in Table 9.8-5.

b. The attenuating material is encapsulated in stainless steel.

c. A winch, attached to the base of the canister with quick release pins, is operated by one man. The man would be standing on a work platform above the control rod drive mechanism housings. (Work would be done in parallel with the other normal head area work.)

d. Modifications to the head lift rig or any head area components are not required for probe handling.

e. The reactor head area/control rod drive (CRD) mechanism area is left completely unencumbered and servicing of the drive mechanisms and underside of the head may be performed without increased personnel radiation exposure and without risk of damage to the probes.

f. The weight of a handling canister is maintained at less than 1500 pounds, including the attenuating material.

g. A removable drop catcher is provided on the bottom of the handling canister to prevent water dripping while the canister is being transported from the reactor vessel to the storage area.

h. A hub adapter is provided to protect the Grayloc seal ring surface and to pilot the lower end of the handling canister.

9.8.3 SYSTEM OPERATION

9.8.3.1 New Fuel Transfer

New fuel assemblies are removed from their regulatory agency approved shipping containers inspected and placed in the new fuel storage racks or in the new fuel elevator. This is accomplished by using a short fuel handling tool attached to the auxiliary hook of the spent fuel cask crane. New CEAs are inserted into the guide channels of the fuel assemblies while the new fuel is in the new fuel elevator.

Prior to the start of reactor refueling operations the new fuel is removed from the new fuel storage racks and transferred to the new fuel elevator. This operation is accomplished by the auxiliary hook of the spent fuel cask crane and the short fuel handling tool.

The new fuel elevator lowers the fuel assembly down into the spent fuel pool where the spent fuel pool platform crane transfers the fuel assembly normally to a spent fuel rack, or, directly to the spent fuel pool side upending mechanism. Interlocks are provided to prevent the spent fuel pool platform crane from lowering the fuel assembly unless the upender is in the vertical position. Interlocks are also provided to prevent raising or lowering the upender if the spent fuel platform crane is in the upender zone with the hoist below the hoist-clear elevation. The spent fuel platform crane attaches to the fuel assemblies by means of a long fuel handling tool. After the spent fuel platform crane has placed a new fuel assembly in the upending mechanism, it removes a spent fuel assembly from the other position of the upending mechanism and transfers the spent fuel to a designated position in the spent fuel storage racks.

9.8.3.2 Standard System Operations

Refueling operations are initiated with the removal of the missile shield from over the reactor. The CEDMs are disengaged from their drive shaft extensions by de-energizing the electromagnets, and both CEDM and ICI cabling are disconnected in preparation for head removal. The stud tensioners are employed to remove the vessel head studs which are then removed and stored. Plugs are installed to protect the empty stud holes and two alignment pins are placed to assist in subsequent operations. The CEDM coolant shroud is disconnected from its duct work and the vessel vent line removed. The hatches in the reactor vessel cavity seal are closed to prevent water from entering the lower portion of the vessel cavity. A lifting frame is then installed on the head

assembly and, by means of the containment building overhead crane, the head is removed to its storage location.

A fuel transfer tube extending through the containment wall connects the refueling pool with the spent fuel pool as shown in Figure 9.8–10. During reactor operations, this transfer tube is isolated by (1) a manually operated isolation valve that is outside containment and within the fuel transfer canal and (2) a blind flange located inside containment.

During preparations for filling the refueling pool for refueling operations, the blind flange on the fuel transfer tube is removed.

Following verification that the water inventories of the spent fuel pool, fuel transfer canal and refueling pool are at a common elevation, the isolation valve of the fuel transfer tube is opened.

Using the refueling machine walkway as a work platform, the CEDM drive shaft extensions are disconnected from their CEAs by means of tools hung from the polar crane and the refuel machine. The UGS lift rig is installed and the ICI plate is withdrawn into the UGS and locked in place.

Provision is made in the refueling pool for the temporary storage of the UGS of the reactor vessel internals.

9.8.3.3 Refueling Operations

The normal fuel management sequence is initiated by positioning the refueling machine hoist above the desired fuel assembly in the core. The operators lower the hoist until it stops at the UPPER GRAPPLE OPERATING ZONE (UGOZ), the indicator illuminates and the operator energizes the grapple mechanism which rotates to OPEN the grapple. When the SPREADER is fully extended, the hoist is lowered until it stops at the LOWER GRAPPLE OPERATING ZONE (LGOZ), the indicator illuminates and the operator energizes the grapple mechanism to close the grapple onto the fuel assembly. The hoist motor is started and the fuel assembly is withdrawn into the hoist box which protects the fuel assembly during the traverse to the upender. The hoist is raised until the HOIST UP LIMIT is reached, then the hoist will automatically stop. The grapple has been designed to preclude inadvertent disengagement as the fuel assembly is lifted vertically from the core. Grapple operation is interlocked both mechanically and electrically to permit grapple actuation only in two operating zones UGOZ and LGOZ. Due to the mechanical interlocks, the grapple must be closed while it is being raised or lowered within the hoist box.

After removal from the core, the spent fuel assembly is lowered into one of the two fuel carrier pockets located in the refueling pool upender. The refueling machine ungrapples the spent fuel assembly, raises the hoist and returns to the reactor core to move the next fuel assembly in accordance with the fuel management procedure. Under certain fuel management procedures, the refueling operator then repeats the process and removes another spent fuel assembly and traverses to the refueling pool upender and the second spent fuel assembly is lowered into the second of the two fuel carrier pockets.

The upender lowers the fuel carrier to the horizontal position and the cable drive transports the carriage through the fuel transfer tube to the spent fuel pool upender. Once received in the spent fuel pool upender, the upending mechanism raises the fuel carrier to the vertical position. The spent fuel pool platform crane removes the spent fuel assemblies and places them into predetermined spent fuel rack locations. The fuel carrier is lowered to the horizontal position and returned to the refueling pool for one or two more spent fuel assemblies or the spent fuel platform crane operator grapples a new fuel assembly and traverses to the spent fuel pool upender in accordance with the fuel management procedure.

When the fuel carrier returns to the refueling pool upender, the fuel carrier is raised to vertical and the refueling machine operator either removes the new fuel assembly from the fuel carrier and transfers it to the core or places another spent fuel assembly into the fuel carrier. This sequence is repeated until the refueling sequence is completed.

For transferring CEAs among fuel assemblies in the refueling pool, a CEA handling tool attached to the auxiliary hoist of the refueling machine is used. For transferring CEAs among fuel assemblies in the spent fuel storage racks within the spent fuel pool, a CEA handling tool attached to the spent fuel pool platform crane is used (refer to Section 9.8.2.1.4).

Neutron sources are transferred between fuel assemblies in the spent fuel pool via a tool manipulated from the spent fuel platform crane. Capsules containing surveillance samples are removed from the reactor vessel assembly utilizing a tool manipulated from the refueling machine which then transports them to the upender station for insertion into a container similar to a dummy fuel assembly. The transfer carrier transports the container with the surveillance capsules to spent fuel storage area for eventual disassembly and sample evaluation.

In mast sipping of fuel assemblies can be conducted, if required, during normal fuel handling operations. A removable control console is installed on the refueling machine with connections to the tubing within the refuel mast. The in mast sipping takes a water sample from around each fuel assembly as it is lifted from the core into the mast. The water soluble and/or gaseous fission products escaping from a failed fuel rod during this handling process are drawn from the mast region of the fuel assembly handling machine through tubing, degassed continuously, and the gas is directed to a detector. The detector measures the fission products and determines the isotopes present. A determination of fuel integrity can be made based upon the concentrations and types of isotopes present. The sample stream is then directed back to the refuel cavity and released below the water surface.

9.8.3.4 Refueling Restoration

At the completion of the refueling operation, the transfer valve is manually closed. The UGS is reinserted in the vessel and the ICI plate placed in position. The drive shaft extensions are reconnected to the CEAs. The water in the refueling pool is lowered, utilizing one of the low-pressure safety injection (LPSI) pumps or one of the purification pumps. The head is then lowered until the drive shaft extensions are engaged by the CEDMs. Lowering of the head is continued until it is seated. Then the head is bolted down, and the transfer tube blind flange installed. The hatches in the cavity seal between the reactor vessel flange and the pool are opened to permit

cooling air flow. The CEDM and ICI cabling is reconnected. The coolant ducting is reconnected to the shroud, the vessel vent piping installed, and the missile shield placed in position.

9.8.3.5 Emergency Conditions

The failure mode analysis is shown in Table 9.8-4.

9.8.4 AVAILABILITY AND RELIABILITY

9.8.4.1 Special Features

9.8.4.1.1

The new fuel storage racks are designed to maintain the minimum center-to-center distance necessary to preclude criticality of the new fuel even with unborated water. In addition the structural design precludes any deformation of the racks during earthquake loads, that would reduce the center-to-center spacing to a point where the new fuel would approach criticality.

9.8.4.1.2

The spent fuel storage racks are designed to maintain the minimum center-to-center distance necessary to preclude criticality of the spent fuel even with unborated water under normal operating conditions. In addition the structural design prevents deformation of the racks that reduces the center-to-center spacing to a point where the spent fuel would become critical due to a DBE or a dropped fuel assembly (horizontal) on top of the storage racks. The structure provides for adequate convective cooling of stored fuel assemblies.

9.8.4.1.3

Reliability of the fuel handling equipment including the bridge and trolley, the lifting mechanisms, the upending machines, the transfer carriage, and the associated instrumentation and controls has been assured through implementation of pre-operational tests and routines. Prior to the first actual fuel loading, the equipment was cycled through its operations using dummy fuel. In addition, the following special features of the equipment assure safe and reliable operation:

9.8.4.1.4

Provision for seismic loading is made by proper sizing and arrangement of the structure and components of the equipment. Where required, mechanical means (clips, guide tools, etc.) are utilized to prevent overturning or derailing of the equipment during an earthquake.

9.8.4.1.5

A bumper ring mounted on the refueling machine mast interrupts the bridge and trolley drives should the mast be driven into an obstruction. This feature prevents damage to either the refueling machine or to members or components into which it may be driven.

9.8.4.1.6

The three major subsystems of the overall fuel handling complex (Refueling Machine, Spent Fuel Pool Platform Crane and Fuel Transfer System) are electrically interlocked within each other and conjointly with each other to prevent incorrect and potentially damaging sequences of operation. These interlocks accomplish the following:

a. Bridge, trolley and hoist speeds are restricted in areas where fuel assemblies are handled near potential interferences.

b. Hoist drives cannot be operated if either the bridge or trolley drives are operating.

c. Up motion of the hoist is stopped if the hoist is at the up limit or if the hoist is overloaded.

d. Hoist down motion is stopped if the hoist load is below a preset limit or if the cable is slack.

e. The refueling machine bridge and trolley drives may not be actuated when the spreader device is extended.

f. The refueling machine bridge and trolley are inoperative if the fuel hoist is in or below the Upper Grapple Operate Zone (UGOZ), or during interlock override at creep speed only.

g. The refueling machine hoist cannot be raised if the grapple is above the Upper Grapple Operating Zone (UGOZ) and is not closed.

h. The refueling machine hoist is latched automatically when the machine is in the upender zone and unlatched automatically when it is outside the upender zone.

i. The refueling machine fuel spreader cannot be extended when the machine is in the upender zone.

j. The refueling machine hoist cannot lower if in the upender zone and the upender is not vertical.

k. The upender cannot raise or lower if the refueling machine is in the upender zone and the hoist is not in the full up position.

l. The spent fuel pool platform bridge cannot traverse into the upender zone if the upender is not vertical, unless the hoist is full up (loaded) or above the hoist clear elevation (unloaded).

m. The spent fuel pool platform hoist cannot lower if in the upender zone and the upender is not vertical.

n. The upender cannot raise or lower if the spent fuel pool platform hoist is in the upender zone, unless the hoist is full up (loaded) or above the hoist clear elevation (unloaded).

o. The fuel transfer carrier cannot begin a transfer unless both upenders are horizontal. Once started, the carrier travel will be stopped if the cable tension exceeds a preset limit.

9.8.4.1.7

The refueling machine is equipped with various mechanical interlocks to provide for safe and proper operation. They include, positive mechanical locking of the refueling machine grapple to the fuel bundle, and a positive mechanical stop to prevent raising a fuel bundle above the minimum depth of water.

9.8.4.1.8

Miscellaneous special design features include — backup hand operation of hoist and traverse drives in the event of power failure; an available 4 to 1 gear reduction at the winch motor to permit applying an increased pull on the transfer carrier in the event it becomes stuck; a viewing port in the refueling machine trolley deck to provide visual access to the reactor for the operator; electronic and visual indication of the refueling machine position over the core; a protective shroud into which the fuel bundle is drawn by the refueling machine; transfer system upenders manual operation by a special tool in the event that the hydraulic system becomes inoperative; removal of the transfer system components from the refueling pool for servicing without draining the water from the pool.

9.8.4.1.9

The manual operator for the fuel transfer tube isolation valve extends from the valve to the deck at elevation +38 feet 6 inches. Also, the operator has enough flexibility to allow for operation of the valve due to thermal expansion of the fuel transfer tube.

9.8.4.1.10

The spent fuel pool platform crane is provided with encoders that monitor the bridge, trolley and hoist positions and a programmable logic controller that establishes appropriate boundaries to assure that fuel assemblies do not contact any of the walls or other equipment in the spent fuel pool area. The hoist is also provided with a geared limit switch that assures the minimum water necessary for shielding and an anti-two block switch that protects the hoist from damage if the geared limit switch fails. The fuel is also protected from damage by a load cell that interrupts hoist movement in the event of overload or underload conditions while raising or lowering. Interlocks between the platform crane and fuel transfer system prevent lowering a fuel assembly into the fuel carrier unless the upender is vertical and prevent the platform crane from entering the upender zone unless the hoist is at an elevation that avoids interference with the upender.

The spent fuel pool platform crane bridge and trolley will not derail under design seismic conditions.

9.8.4.1.11

The single failure proof design of the spent fuel cask crane utilizes limit switches to actuate special features that sense over travel (control and power), overweight (loads), overspeed (trolley and hoists), mis-spooling (wire rope on drum) and unbalanced reeving (wire rope). The electrical design addresses the effects of phase reversal or phase loss in the hoist power supply as well as undervoltage, overvoltage and overcurrent protection. Detection of any of the above faults removes power from the hoists, placing them in a safe condition. Bridge and trolley motions are limited by travel limit switches, which de-energize the motor at end travels.

The spent fuel cask crane travels within the Auxiliary/Warehouse buildings but has features that can restrict its path in certain areas. The cask crane has a position switch for NORMAL / CASK HANDLING / NEW FUEL modes. When in the NORMAL mode, the crane cannot travel over the fuel pool. When the switch is positioned in the CASK HANDLING position, the crane can only travel into the cask pit area of the fuel pool. When the switch is positioned in the NEW FUEL mode, the crane can only travel into the new fuel area of the fuel pool. This switch is a key lock switch and will be controlled administratively.

Both the main and auxiliary hoists have redundant brakes to allow portions of the hoist drive train to be repaired while retaining the load. The hydraulic brakes on the main hoist can be manually modulated to lower a load in the event of a hoisting equipment failure. The auxiliary hoist holding brakes are provided with manual override levers to permit manual load lowering.

The main and auxiliary hoisting systems also have various modes of operation. The main hoist lift mode selector switch is a two position switch that can be selected to critical or non-critical main hoist operation. In the critical (CRIT) position, the main hoist ultra lift feature is disabled and the overspeed system is set to trip at 110% of critical speed. The lifting speed is limited to 5 feet per minute (fpm). In the non-critical (NON) position, the main hoist ultra lift feature is enabled and the overspeed system is set to trip at 110% of non-critical speed. The lifting speed is limited to 7.5 fpm. The auxiliary hoist lift mode selector switch is a two position switch that can be selected to critical or non-critical auxiliary hoist operation. In the critical (CRIT) position, the auxiliary hoist ultra lift feature is disabled and the overspeed system is set to trip at 110% of critical speed. The lifting speed is limited to 20 feet per minute. In the non-critical (NON) position, the auxiliary hoist ultra lift feature is enabled and the overspeed system is set to trip at 110% of noncritical speed. The lifting speed is limited to 31 feet per minute.

The cask crane is equipped with laser distance measuring devices for the bridge and trolley. These devices are retro-reflective and utilize a target for accuracy. The target for the trolley system is attached to the trolley structure, while the bridge target is attached to the building wall. Each system measures the time for the laser light to return to the unit and communicates the calculated distance from the southeast corner of the building to the throat of the main hoist hook, then sends this number to the attached scoreboard display mounted on the bridge. The display reads in feet and inches continuously while power is applied to the crane. This feature enhances the manual

positioning capabilities of the cask crane when handling heavy loads particularly around sensitive areas of the spent fuel pool.

The cask crane is designed to retain control of the main hoist 125 ton maximum critical load (MCL) and auxiliary hoist 15 ton MCL for all load combinations including a single broken rope, two blocking, load hang up and OBE & SSE seismic events.

The seismic and structural analysis of the cask crane determined that there is no trolley or bridge uplift for any of the applied loading combinations. The bridge and trolley are provided with seismic restraints to trap it between the bridge girders and runway beams in the event of a wheel flange or rail failure.

Analysis determined that the bridge will remain on the runway and the trolley will remain on the bridge with the brakes applied during an OBE and SSE event. The cask crane is designed to remain in place and hold the load during an OBE and SSE event.

9.8.4.2 Test and Inspections

Equipment listed under this section has had nondestructive testing in accordance with the procedures of Section VIII of the ASME Code where applicable.

In addition the entire fuel handling system was tested using a dummy fuel element before the plant was put into operation.

The following specific tests are performed on the individual equipment in addition to the above test.

The new and spent fuel storage racks are tested with a dummy fuel assembly for ease of insertion and alignment each time a cavity is fabricated, and after the rack is assembled.

The fuel transfer tube is given a vacuum base test to check for leaks.

The spent fuel pool platform crane and spent fuel cask crane cables are given rope tests.

The spent fuel pool platform crane and spent fuel cask crane, after fabrication and assembly at the vendor’s shop, are given no-load running tests.

Hooks which are designed to support fuel assemblies are volumetrically tested to detect internal defects prior to testing. The hooks are tested at 150 percent of rated load. After the 150 percent load test, they are subjected to liquid penetrant inspection in accordance with ASME Section VIII Pressure Vessels Code.

After installation, the fuel handling devices are thoroughly field tested using a dummy fuel assembly.

During manufacture at the vendor’s plant, various in-process inspections and checks are required including certification of materials and heat treating, and liquid-penetrant or magnetic-particle inspection of critical welds. Following completion of manufacture, compliance with design and specification requirements is determined by assembling and testing the equipment in the vendor's shop. Utilizing a dummy fuel assembly and a dummy CEA, each having the same weight, center of gravity, exterior size and end geometry as an actual assembly, all equipment is run through several complete operational cycles. In addition, the equipment is checked for its ability to perform under the maximum limits of load, fuel mislocation and misalignment. All traversing mechanisms are tested for speed and positioning accuracy. All hoisting equipment is tested for vertical functions and controls, rotation, and load misalignment. Hoisting equipment is also tested to 125 percent of maximum working load. Setpoints are determined and adjusted and the adjustment limits are verified. Interlock function, and backup systems operations are checked. Those functions having manual operation capability are exercised manually. During these tests the various operating parameters such as motor speed, voltage, and current, hydraulic system pressures, and load measuring accuracy and setpoints are recorded. At the completion of these tests the equipment is checked for cleanliness and the locking of fasteners by lockwire or other means is verified.

When finally installed in the field, the equipment is again tested in a manner which, in effect, is a repeat of the tests performed at the vendor’s plant. This allows determination of any changes in adjustment and condition which may have ensued from transit to the site. In addition, the tests in the field permit determination of characteristics which are unique to the actual site installation and, therefore, cannot be duplicated in the vendor’s shop test.

TABLE 9.8-1 TRANSFER TUBE ISOLATION VALVE

Type	Wedge gate
Size, inches	36
Design pressure, psia	83
Normal operating differential pressure, psi	14.3 psi
Design temperature, °F	150
Normal operating temperature, °F	120
Materials	Stainless Steel
Design Code	Draft ASME Code for Pumps and Valves for Nuclear Power, Class II
Seismic Design Classification	1
Design Integrate Dosage	3.2×10^8 R
Supplier	W. G. Rovang & Associates

TABLE 9.8-2 SPENT FUEL POOL PLATFORM CRANE

Bridge	
Capacity, tons	26
Drive	2 hp motor
Speed, fpm (variable)	0-40
Manufacturer	Dwight-Foote/PaR Nuclear
Trolley	
Capacity, tons	2
Drive	0.5 hp motor
Speed, feet per minute (variable)	0-45
Manufacturer	Shepard Niles/PaR Nuclear
Hoist	
Capacity	2000 lb
Drive	3 hp motor
Speed, fpm (variable)	0-33 feet per minute (variable)
Maximum lift, feet	23
Manufacturer	Shepard Niles/PaR Nuclear

TABLE 9.8-3 SPENT FUEL CASK CRANE

Bridge	
Speed, fpm	1.25 - 50 (infinitely variable)
Drive	Variable Frequency
Trolley	
Speed, fpm	1 - 40 (infinitely variable)
Drive	Variable Frequency
Hoists	
Main hoist capacity	125 tons
Auxiliary hoist capacity, tons	15
Drive for main hoist	Variable Frequency n
Drive for auxiliary hoist	Variable Frequency
Speed of main hoist, fpm	0.25 - 5 critical load speed (infinitely variable) 0.25 - 7.5 non-critical load speed (infinitely variable)
Speed of auxiliary hoist, fpm	0.25 - 20 critical load speed (infinitely variable) 0.25 - 31 non-critical load speed (infinitely variable)
Seismic Design Classification	1
Supplier	Harnischfeger Corp. (Bridge & Monorail) American Crane and Equipment Corp. (ACECO) (MH, AH & Trolley)

TABLE 9.8-4 FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION	FAILURE MODE	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	REMARKS
Refueling Machine. Fuel Hoist weight system	Electrical Overload Trip fails	None	Continue refueling, repair on non-interfering basis	Use visual presentation of load on meter.
	Complete system fails	None	As above	Control System will automatically disable the hoist.
Fuel Carrier	Wheels lock in transfer tube	Transfer change completed	Switch to 4:1 gear reduction	Torque sufficient to move fuel carrier with all wheels locked
Hydraulic Power supply for upender	Line to cylinder on upender ruptures	None	Valve off defective line	Upender has two cylinders, each of which is capable of raising upender
	Loss of hydraulic power	Process can continue on slower basis	Upend manually	Use tool provided
Brake on R. M. fuel hoist	Does not provide required brake load	None	Continue, repair on non-interfering	Redundant brake system provided
Fuel Carrier Cable	Cable parts	Delays refueling	Move fuel carrier to safe position with manual tool	Remove fuel prior to repair
Refueling Machine Hoist Motor	Power Failure	Operation can be completed	Repair	Hoist using manual hand-wheel
Bridge Drive Motor	Power Failure	Operation can be completed	Repair	Drive using manual hand-wheel
Electronic Position Indication	Power Failure	None	Repair non-interfering basis	Redundant position indicator is available on the console’s touchscreen display

TABLE 9.8-4 FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION	FAILURE MODE	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	REMARKS
Fuel Carrier Position Sensing System	Electrical Failure	None	Repair non-interfering basis	Winch motor stalls on overload
Refueling Machine	Loss of air pressure	None	Repair	Continue, using manual mode
Dual under deck camera system	Power or Video failure	None	Continue, repair on non-interfering	Refueling can continue using direct visual observation of machine operation

TABLE 9.8-5 RADIATION LEVELS FROM HEATED JUNCTION THERMOCOUPLE (R/HR)

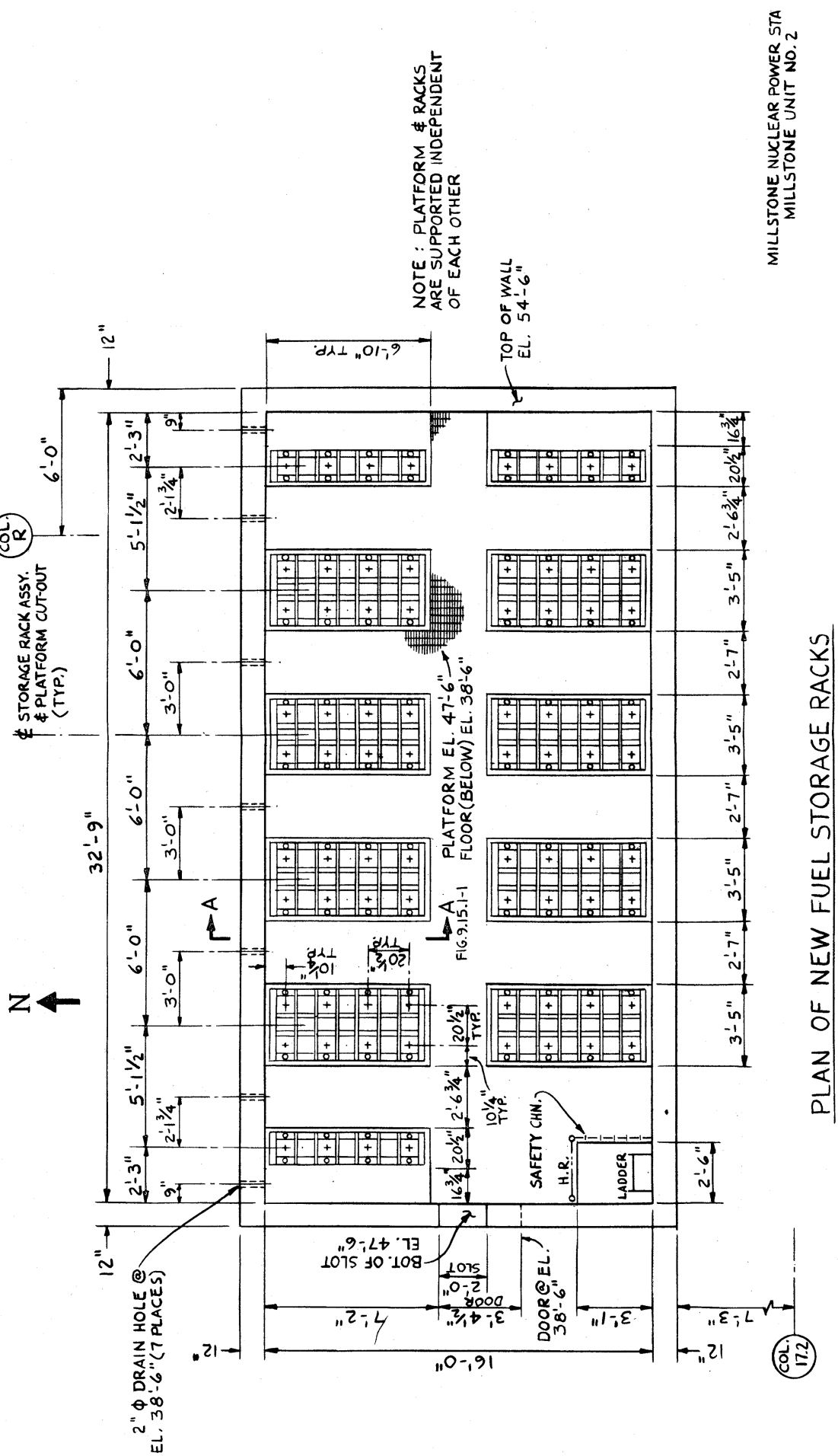
	Dose Rates				
	Crud Contribution		Activation Contribution		Total
Surface of Canister Zone A	0.007-.02	+	0.20	=	0.22
Surface of Canister Zone B	0.007-.25	+	0	=	0.25
Surface of Canister Zone C	0.07-2.5	+	0	=	2.5
1 foot from Surface Zone A	Insignificant	+	0.05	=	0.05
1 foot from Surface Zone B	0.125	+	0	=	0.125
1 foot from Surface Zone C	0.5-1.25	+	0	=	0.5-1.25

Notes: See Figure 9.8–20 for definition of zones.

Worker dose in Shielded Zone for ten minutes = (10/60) x (.125 R/hr) = 0.021 R

21mR

FIGURE 9.8–1 PLAN OF NEW FUEL STORAGE RACKS



PLAN OF NEW FUEL STORAGE RACKS

FIGURE 9.8–2 NEW FUEL STORAGE RACKS - SECTION "A-A"

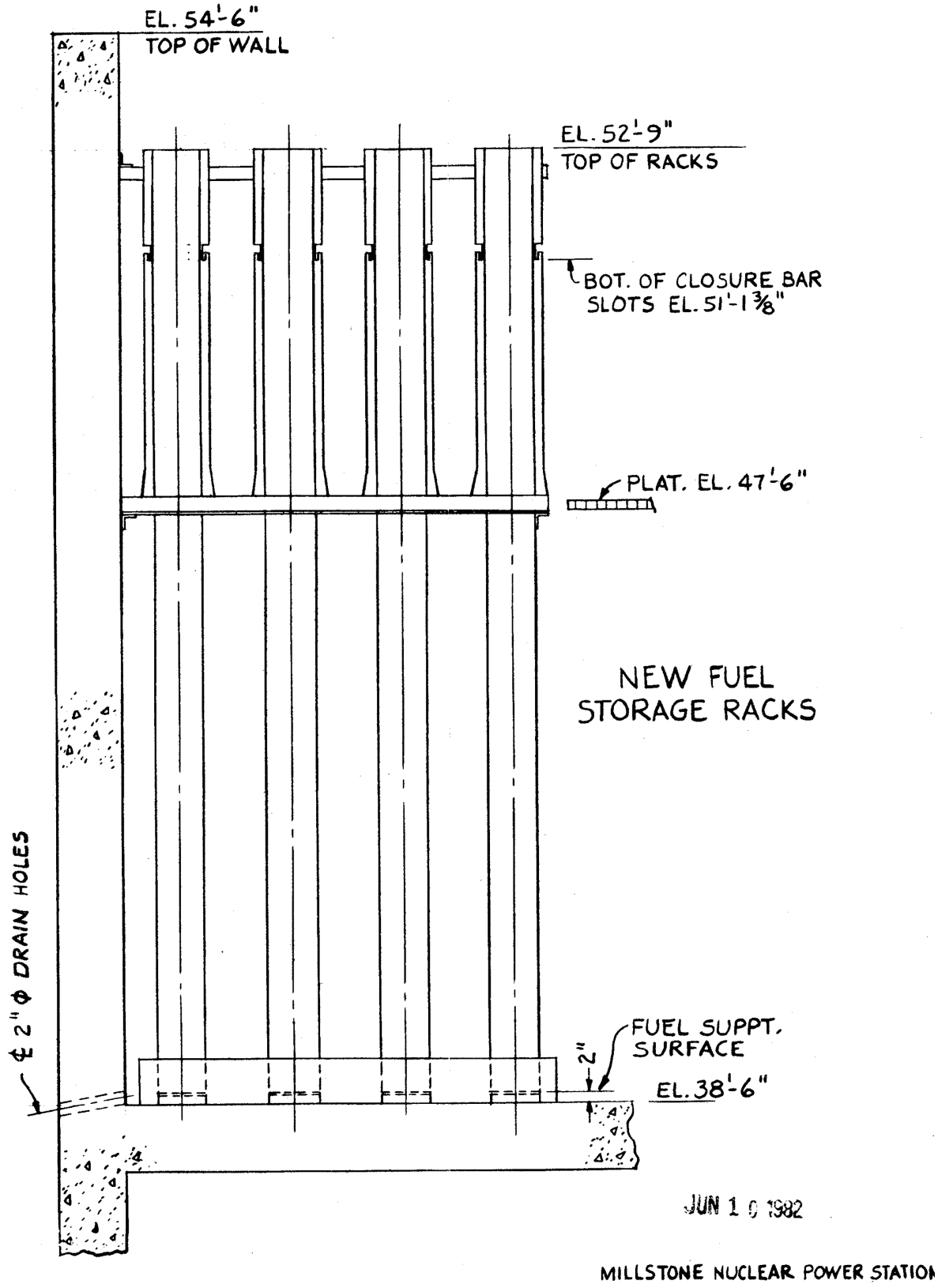


FIGURE 9.8–3 NEW FUEL STORAGE RACK ASSEMBLY (8 RACK MODULE)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.8–4 NEW FUEL STORAGE RACK ASSEMBLY (4 RACK MODULE)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.8-5 STORAGE RACK

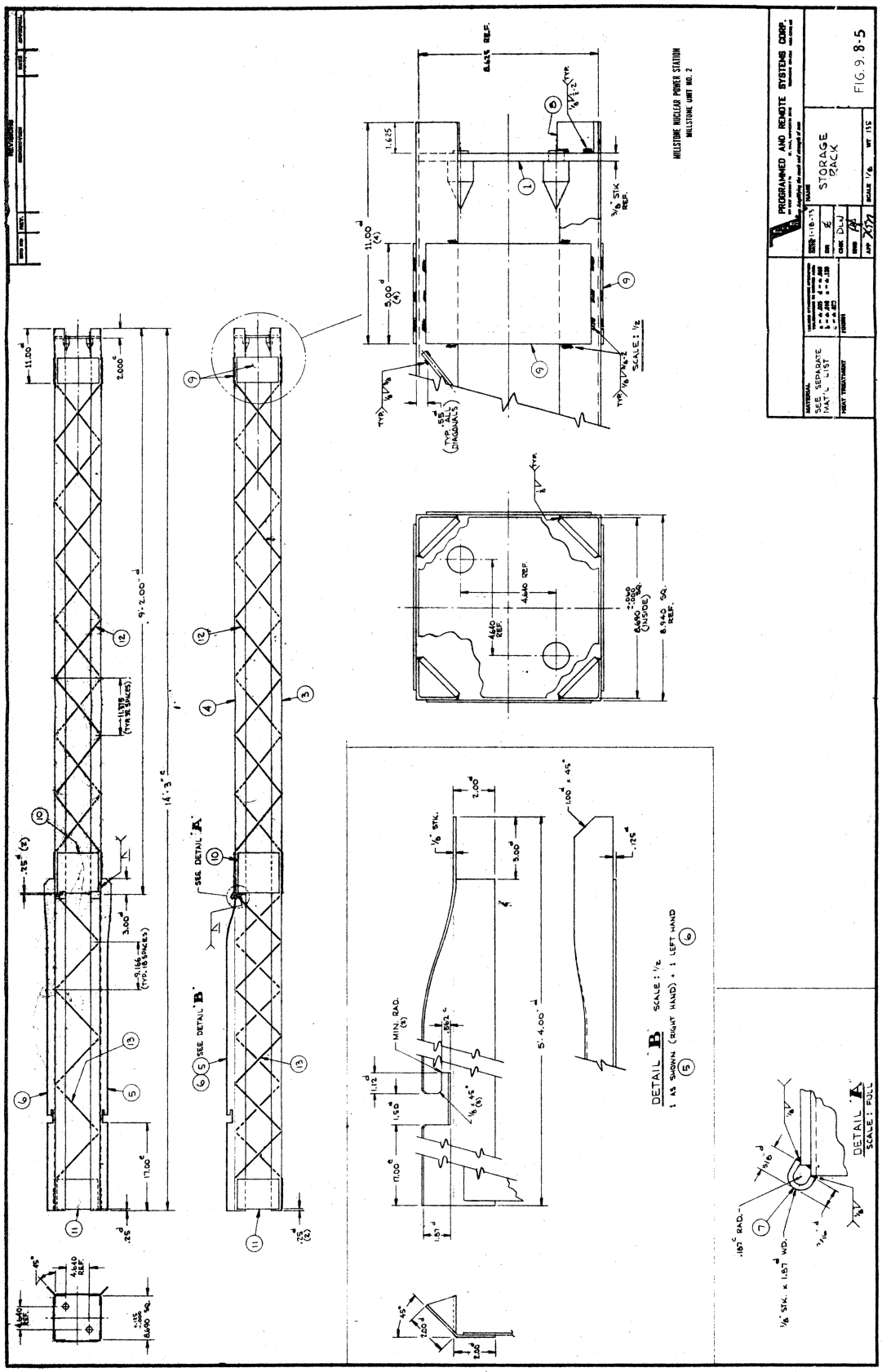
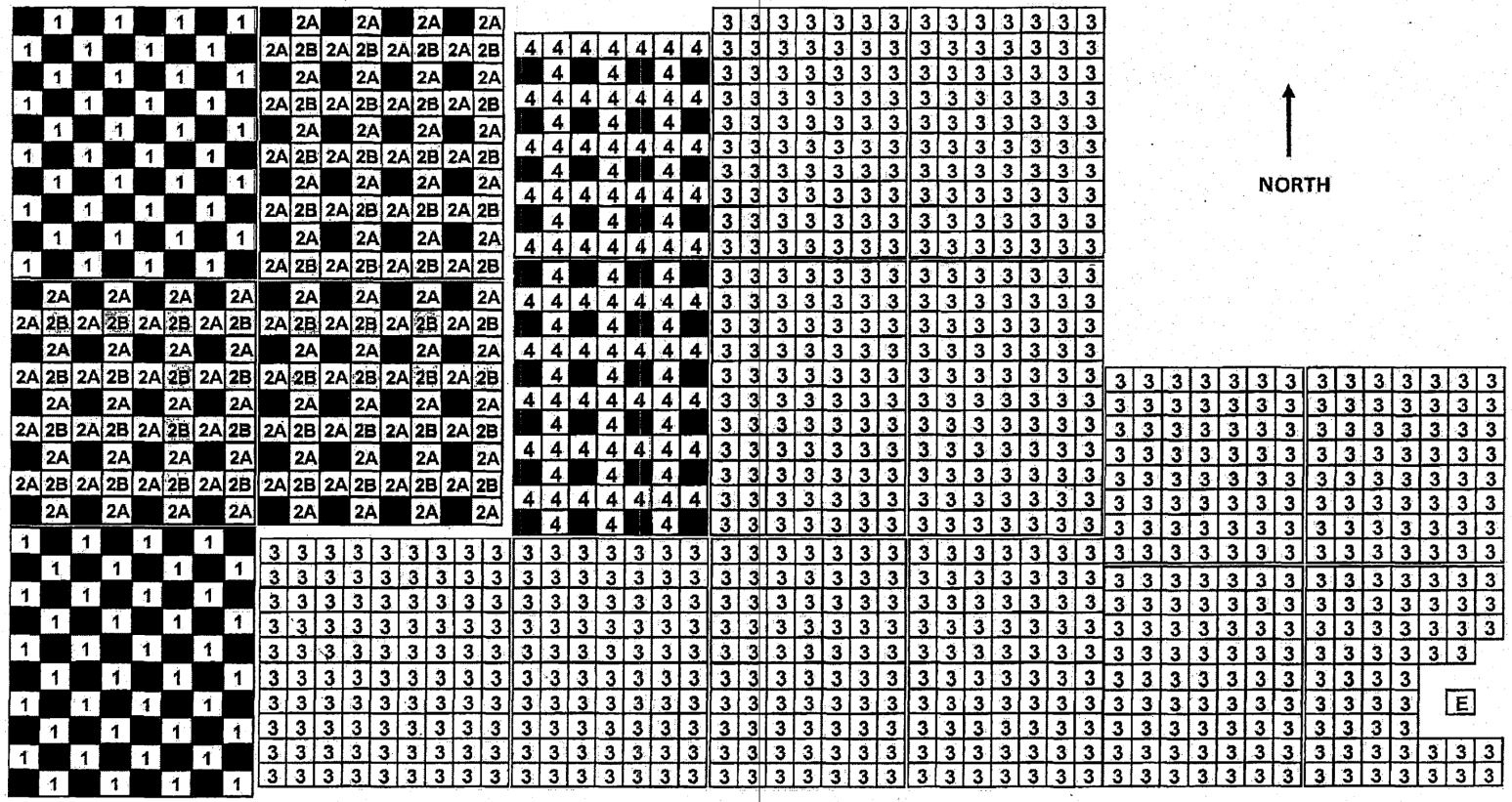


FIGURE 9.8–6 CLOSURE BAR NEW FUEL STORAGE RACKS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.8–7 FUEL POOL ARRANGEMENT - FOUR REGION



* A Restricted Location shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.

FIGURE 9.8–8A TYPICAL SPENT FUEL RACK MODULE/POISON BOX - REGIONS 1 AND 2

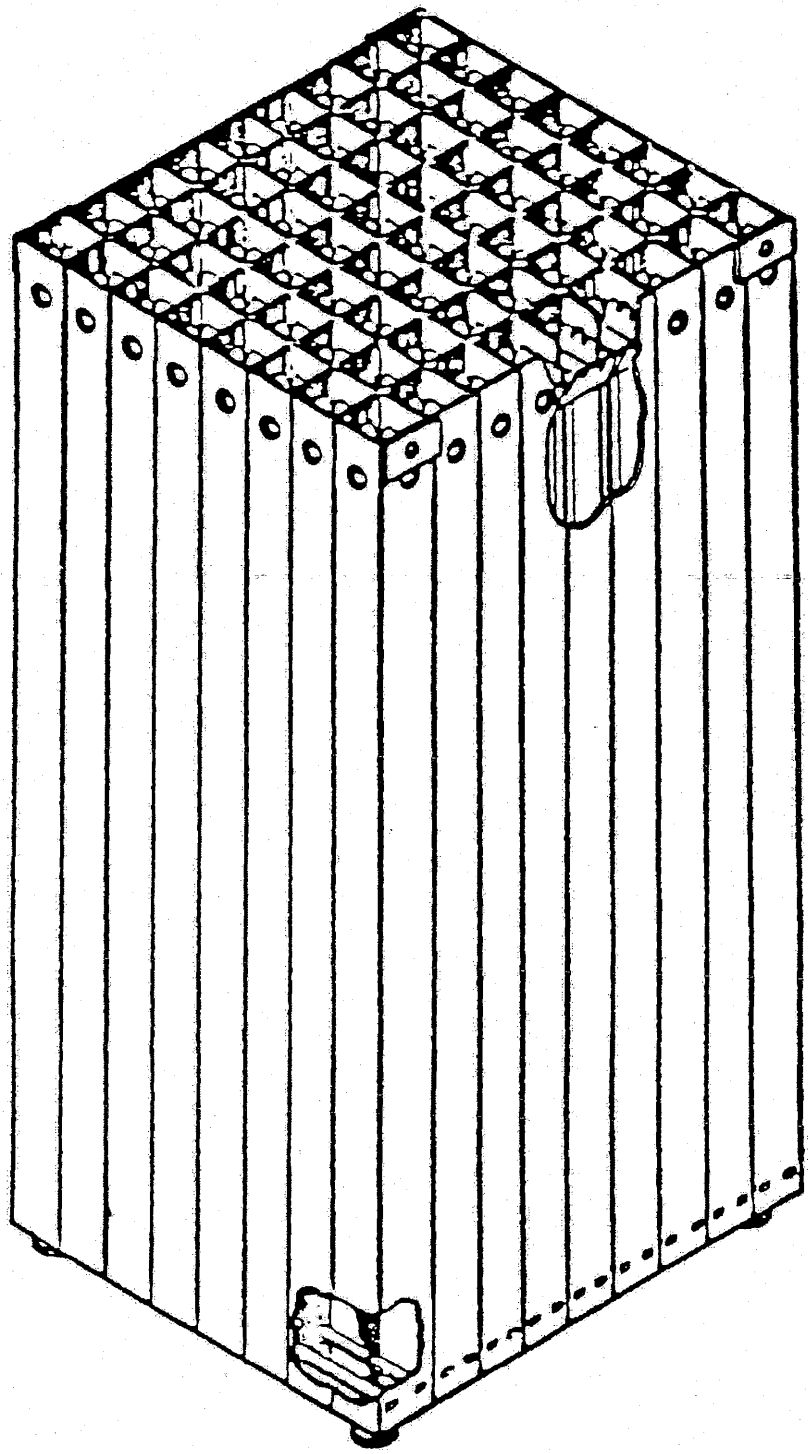


FIGURE 9.8–8B TYPICAL SPENT FUEL RACK MODULE/REGIONS 3 AND 4

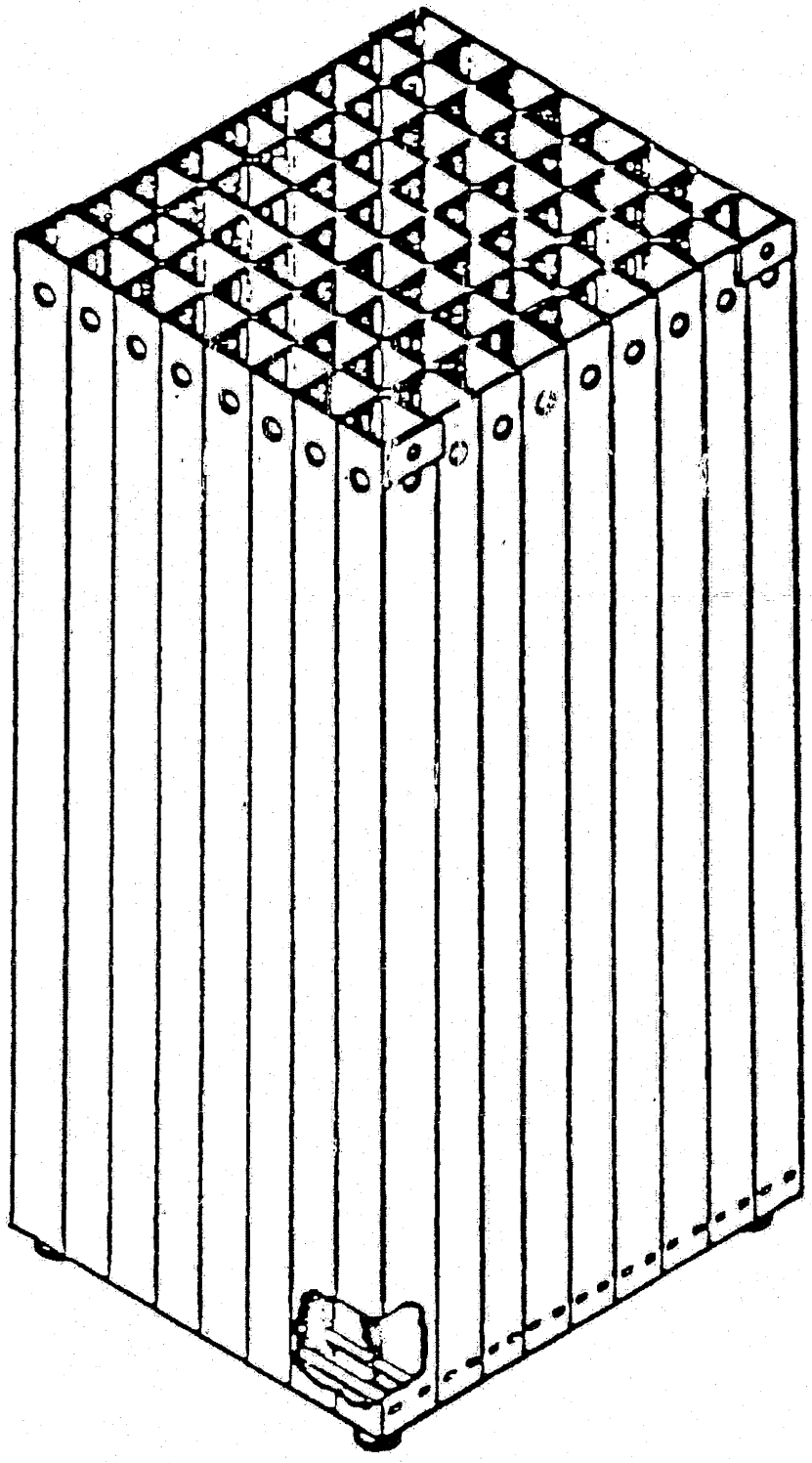


FIGURE 9.8–8C POISON BOX (BORAFLEX POISON NO LONGER CREDITED)

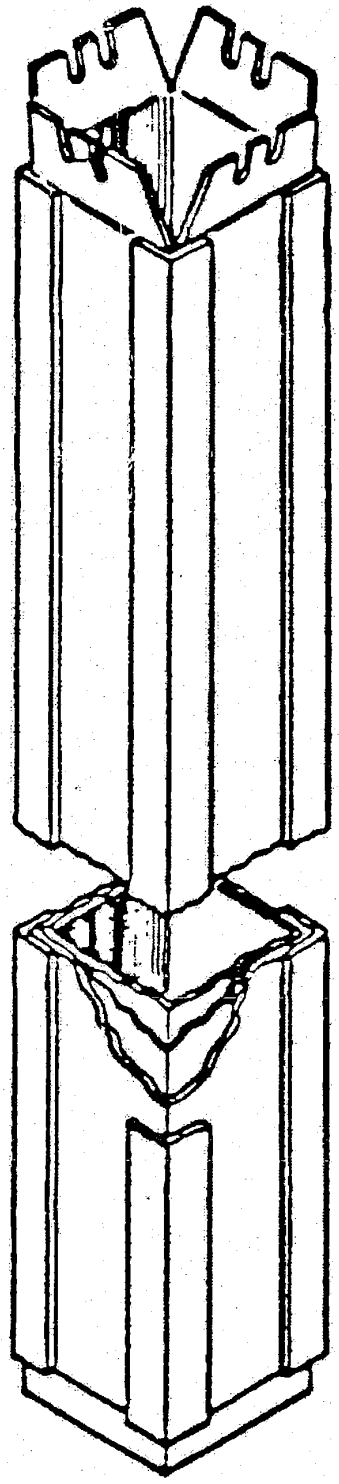


FIGURE 9.8–8D ADJUSTABLE FOOT

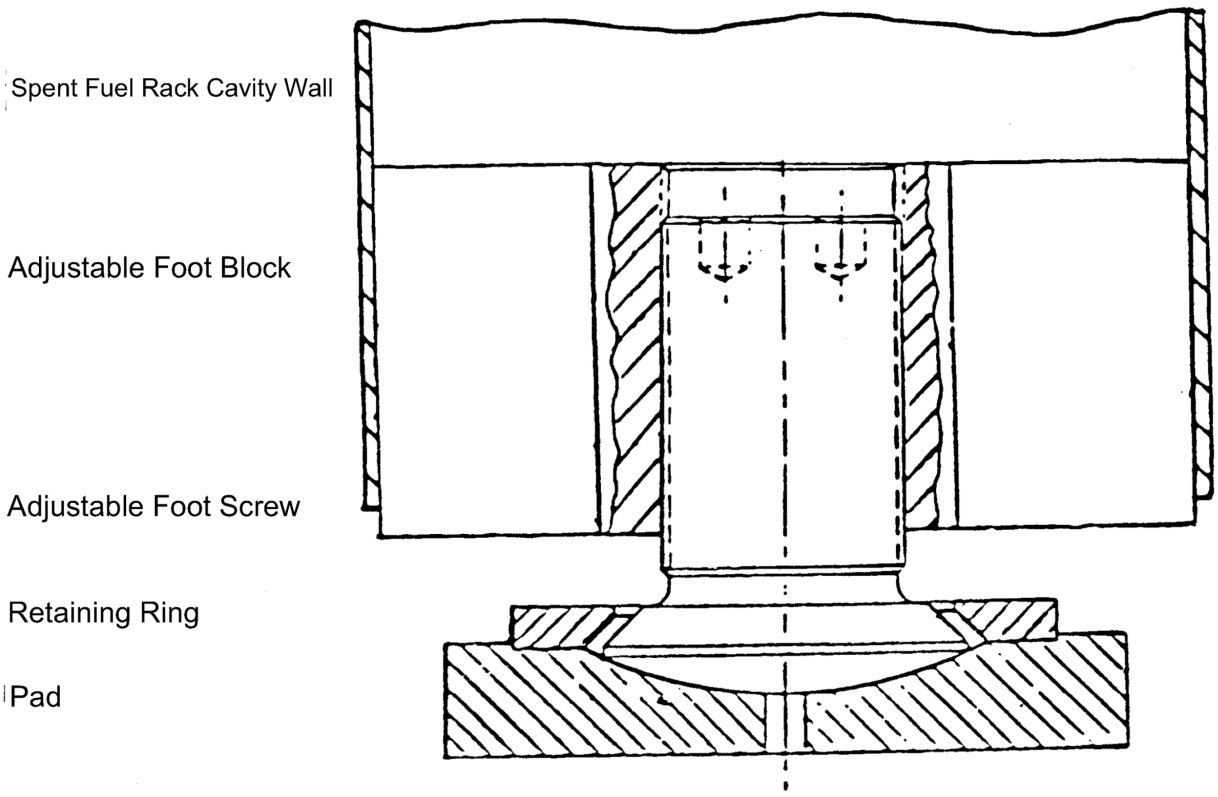
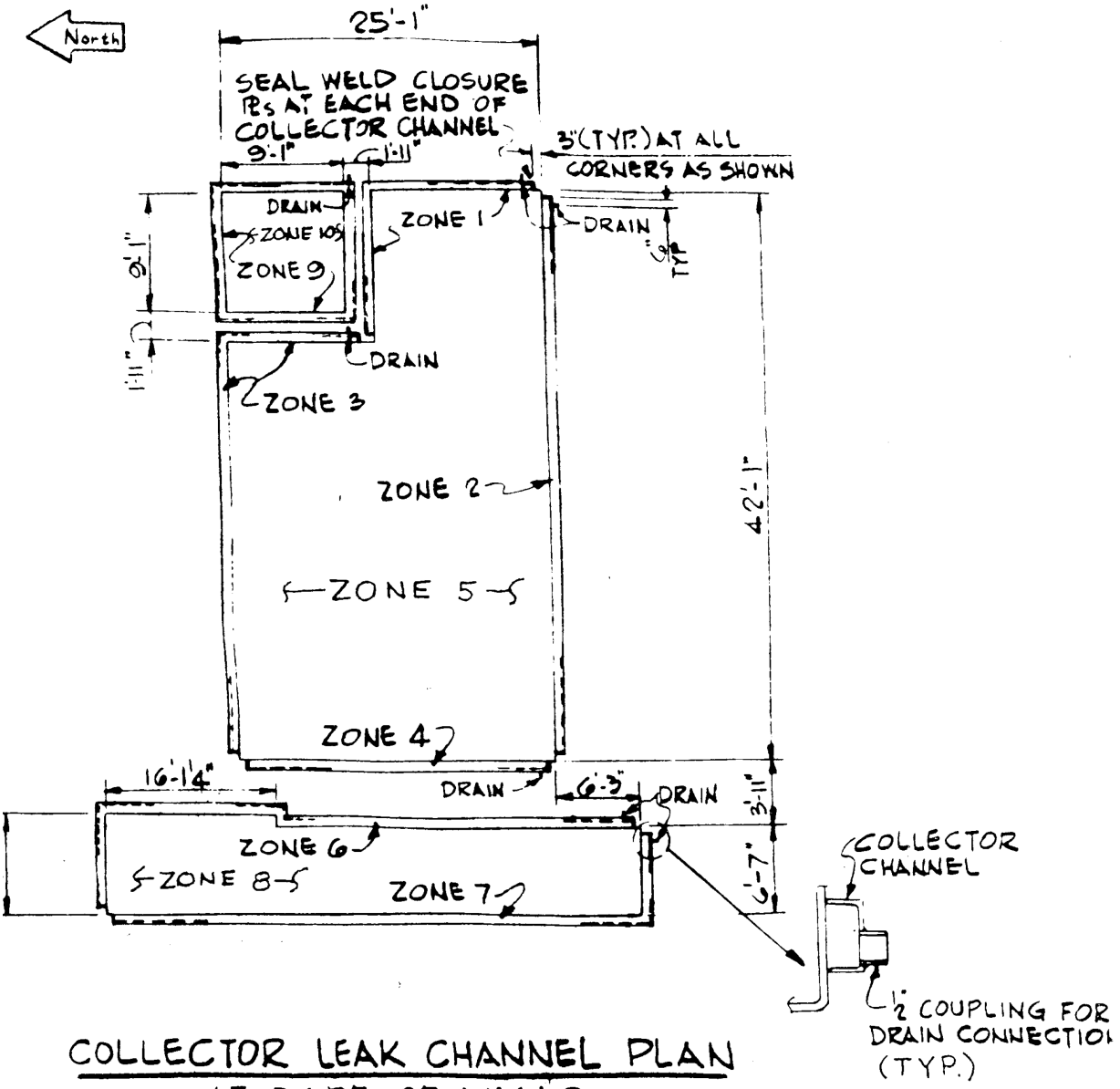


FIGURE 9.8–9 COLLECTOR LEAK CHANNEL PLAN AT BASE OF WALLS



COLLECTOR LEAK CHANNEL PLAN AT BASE OF WALLS

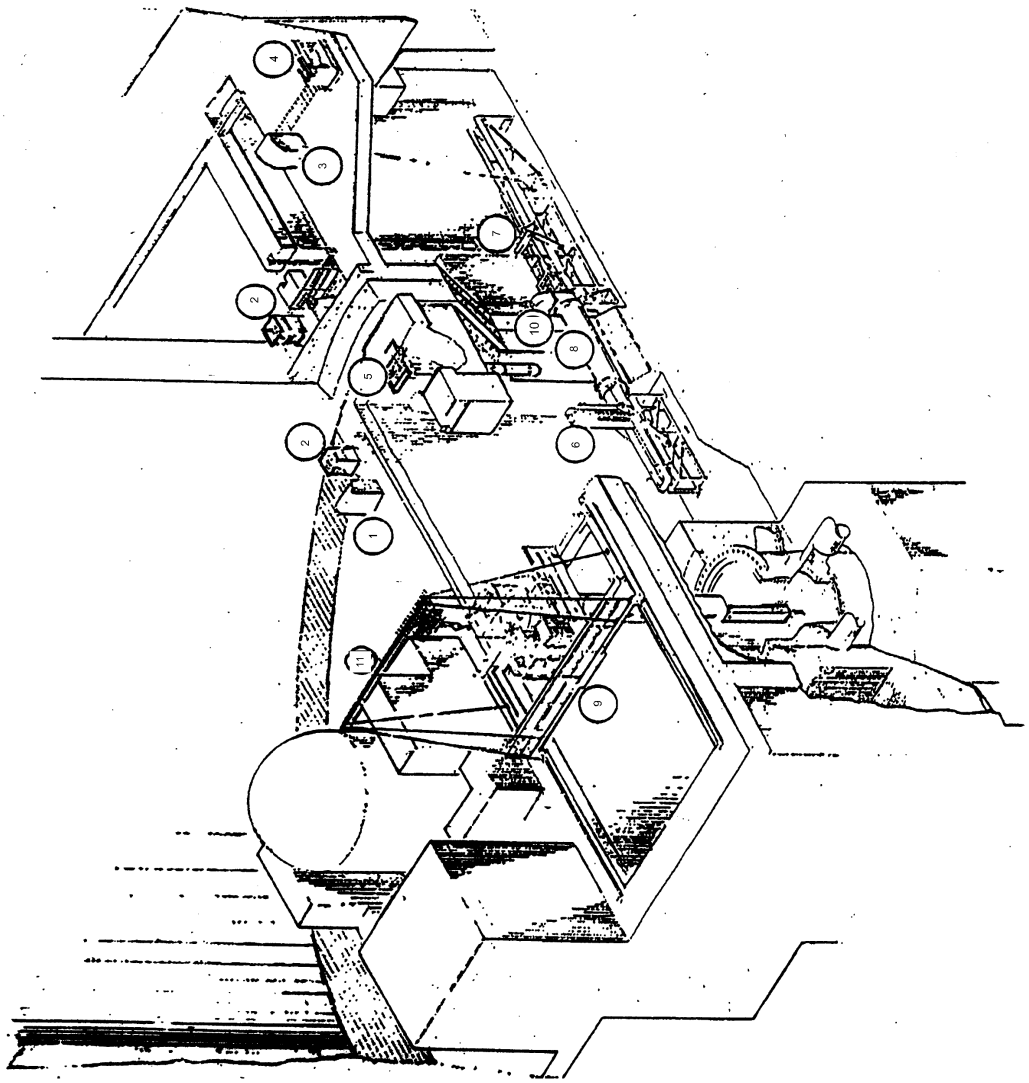
3/32" = 1'-0"

NOTE:

ZONES 5, 8 & 10 ARE USED TO DETECT LEAKAGE THROUGH THE FLOOR OF THE LINER IN THE SPENT FUEL POOL, TRANSFER CANAL & CASK LAYDOWN AREA, RESPECTIVELY.

FIGURE 9.8-9

FIGURE 9.8–10 REFUELING EQUIPMENT ARRANGEMENT



1. TRANSFER MACHINE AND CEA CONSOLE
2. HYDRAULIC POWER PACKAGE
3. TRANSFER MACHINE CONSOLE
4. TRANSFER MACHINE WINCH ASSEMBLY
5. CEA CHANGE MECHANISM REMOVED. REFER TO SAR CHANGE PACKAGE PKG FSC MP2-UCR-2009-006
6. FUEL CARRIAGE
7. UPENDER
8. TRANSFER TUBE
9. REFUELING MACHINE
10. ISOLATION VALVE
11. AUXILIARY HOIST

FIGURE 9.8–11 REFUELING MACHINE

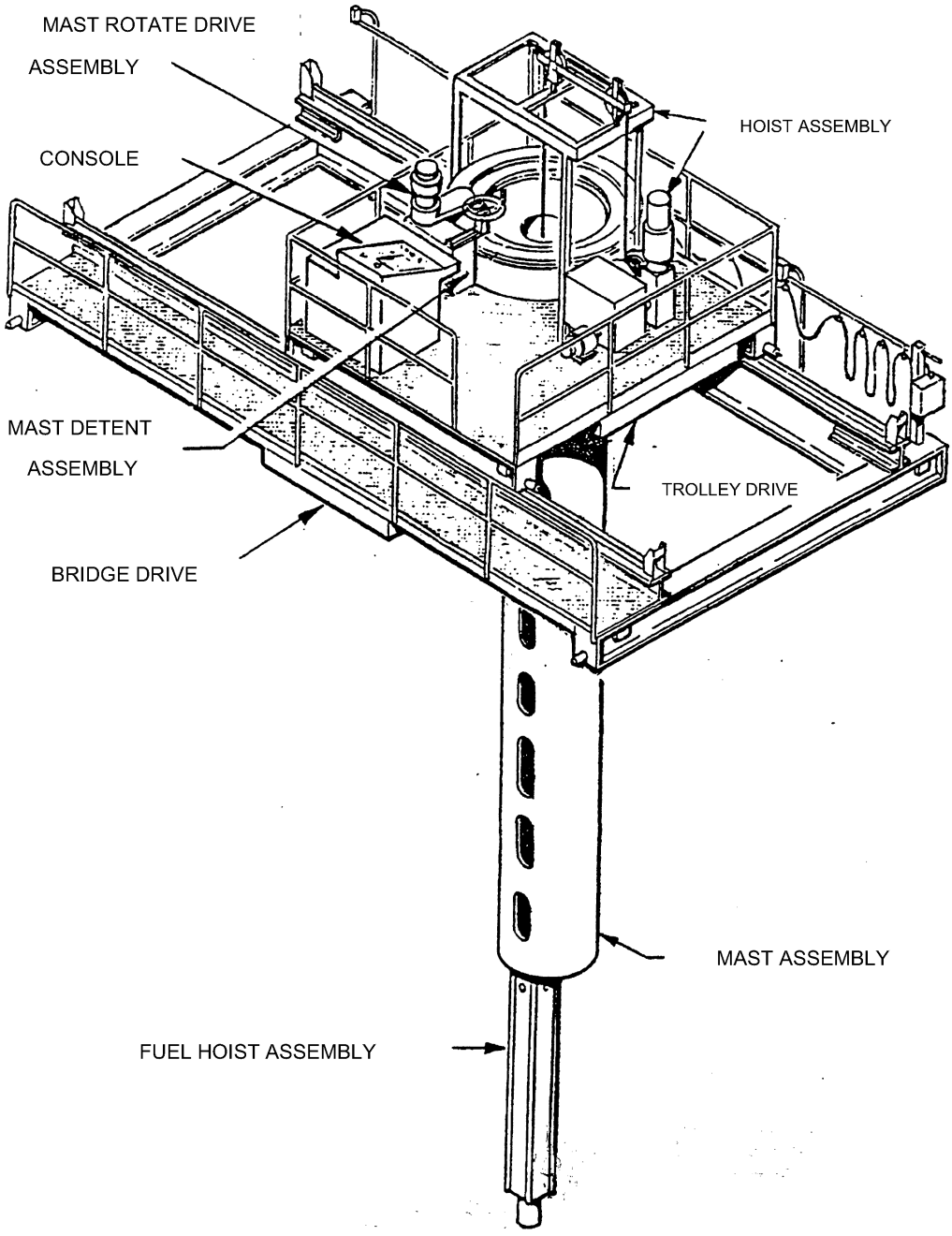


FIGURE 9.8–12 DELETED BY PKG FSC MP2-UCR-2009-006

FIGURE 9.8–13 REACTOR VESSEL HEAD LIFTING RIG

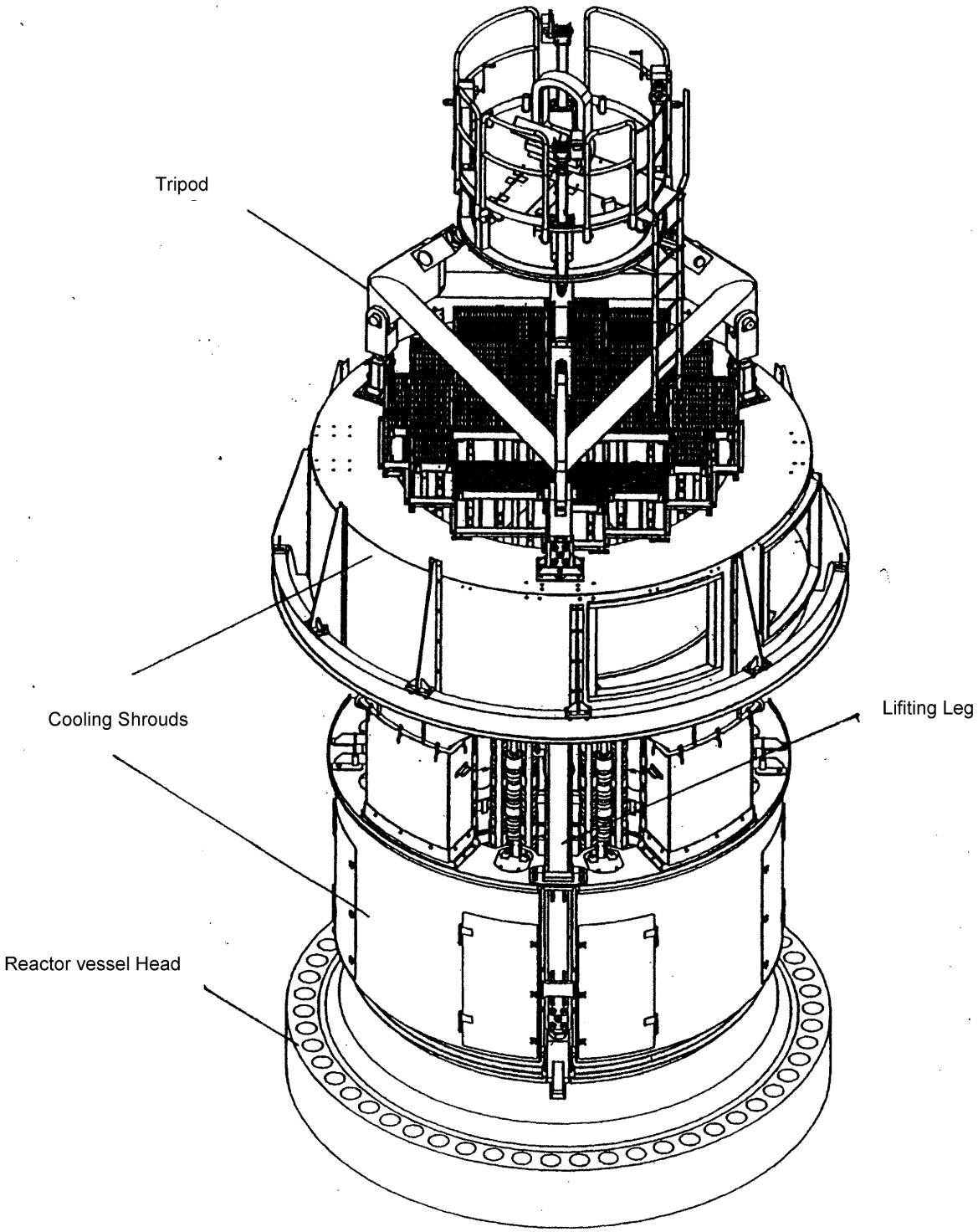


FIGURE 9.8–14 CORE SUPPORT BARREL LIFTING RIG

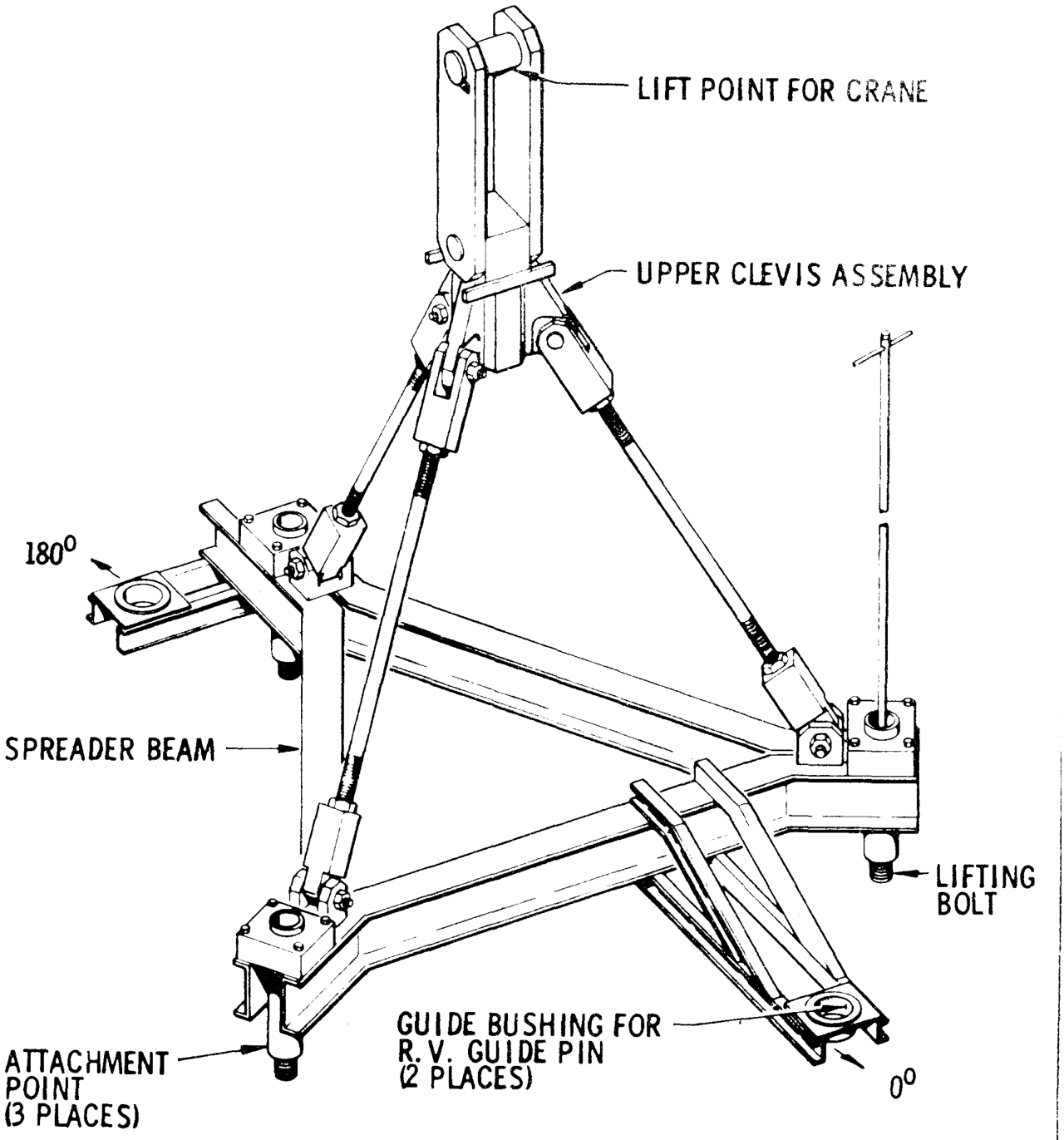


FIGURE 9.8–15 UPPER GUIDE STRUCTURE LIFTING RIG

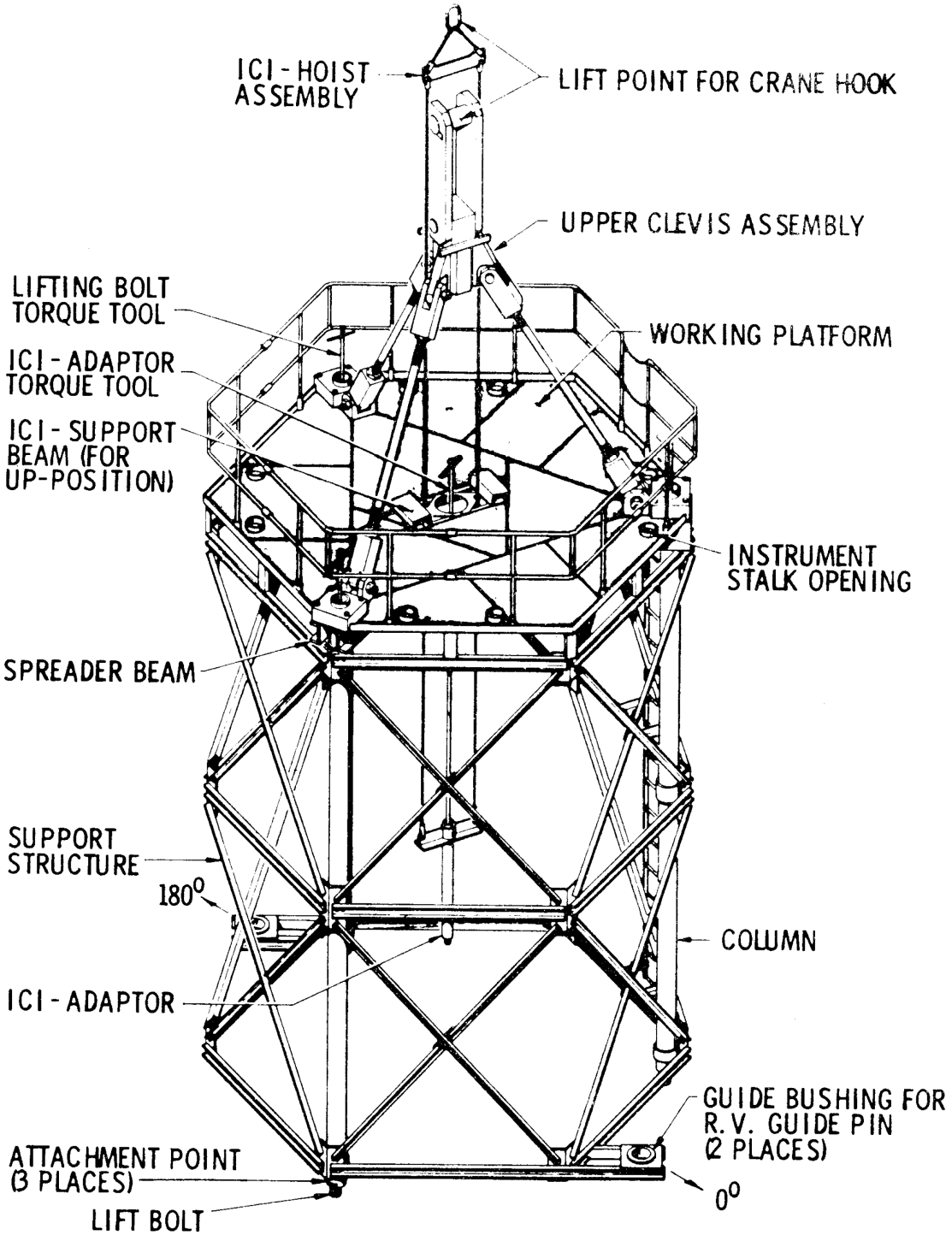
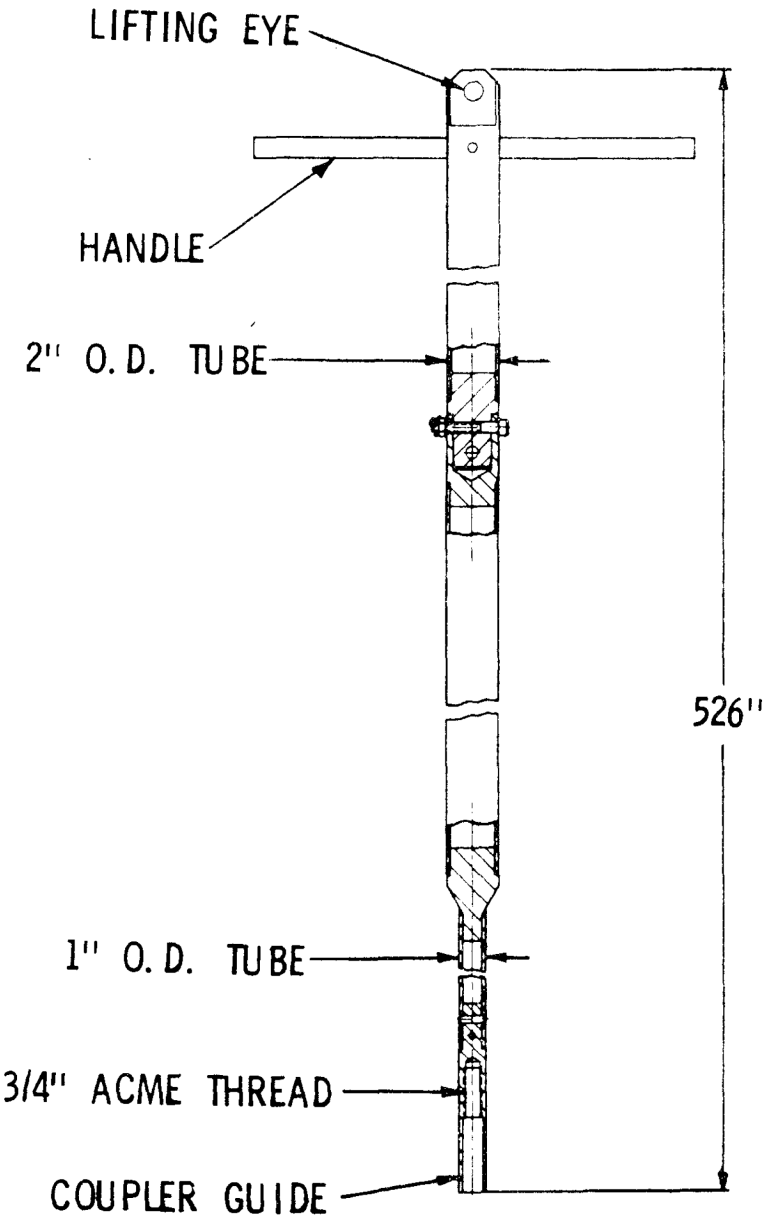


FIGURE 9.8–16 SURVEILLANCE CAPSULE RETRIEVAL TOOL



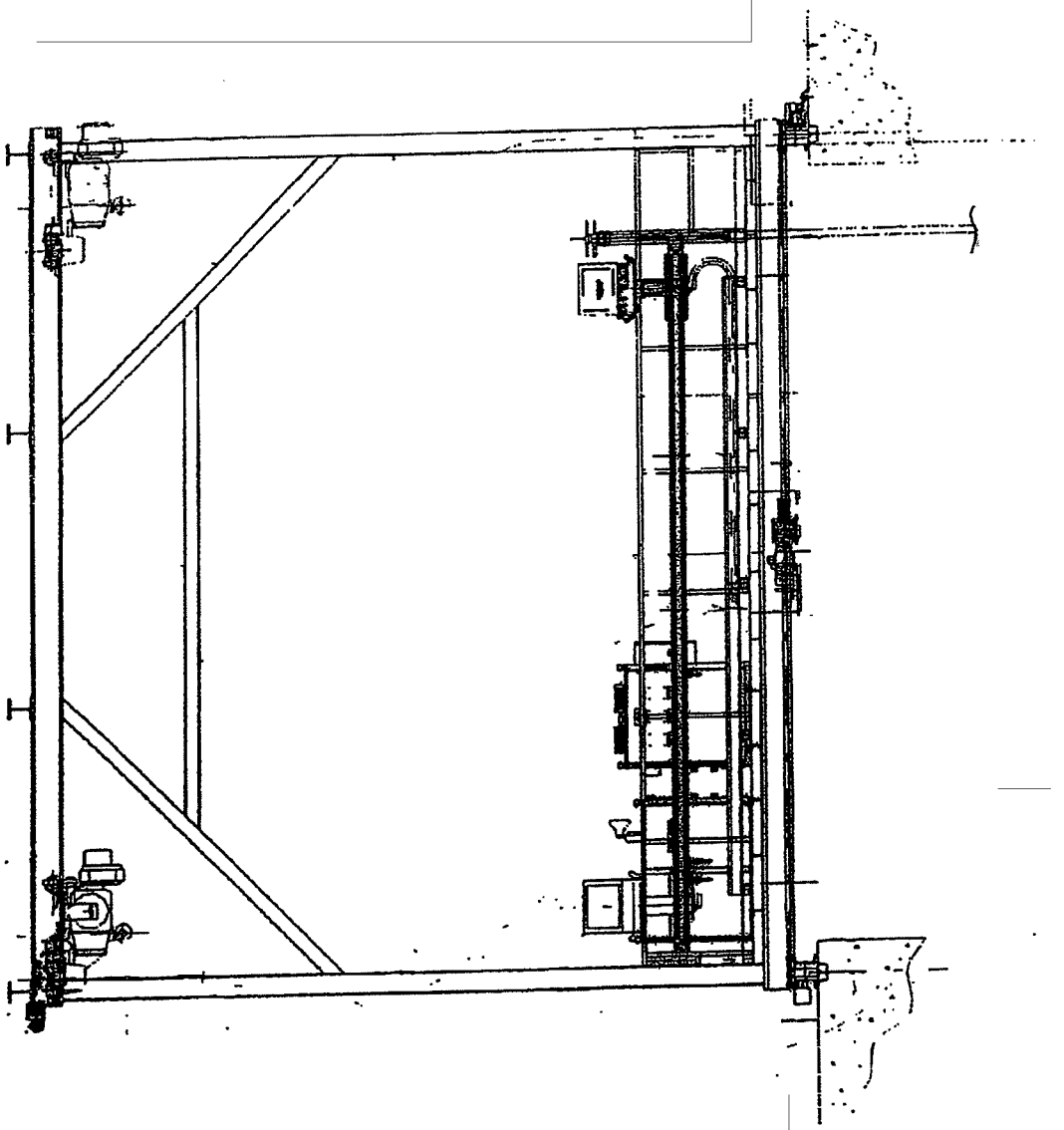


FIGURE 9.8–17 SPENT FUEL POOL PLATFORM CRANE

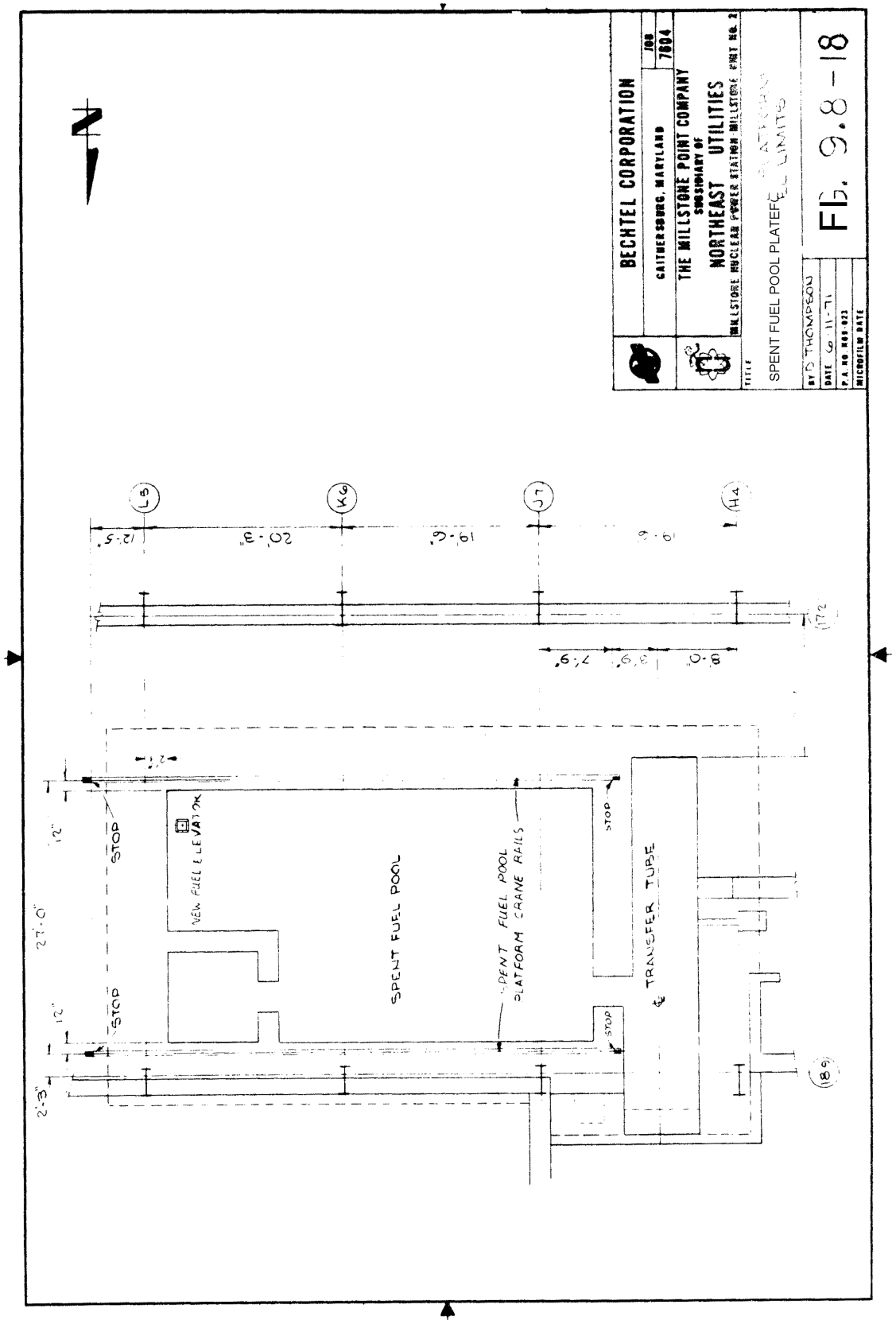


FIGURE 9.8–18 SPENT FUEL POOL PLATFORM CRANE TRAVEL LIMITS

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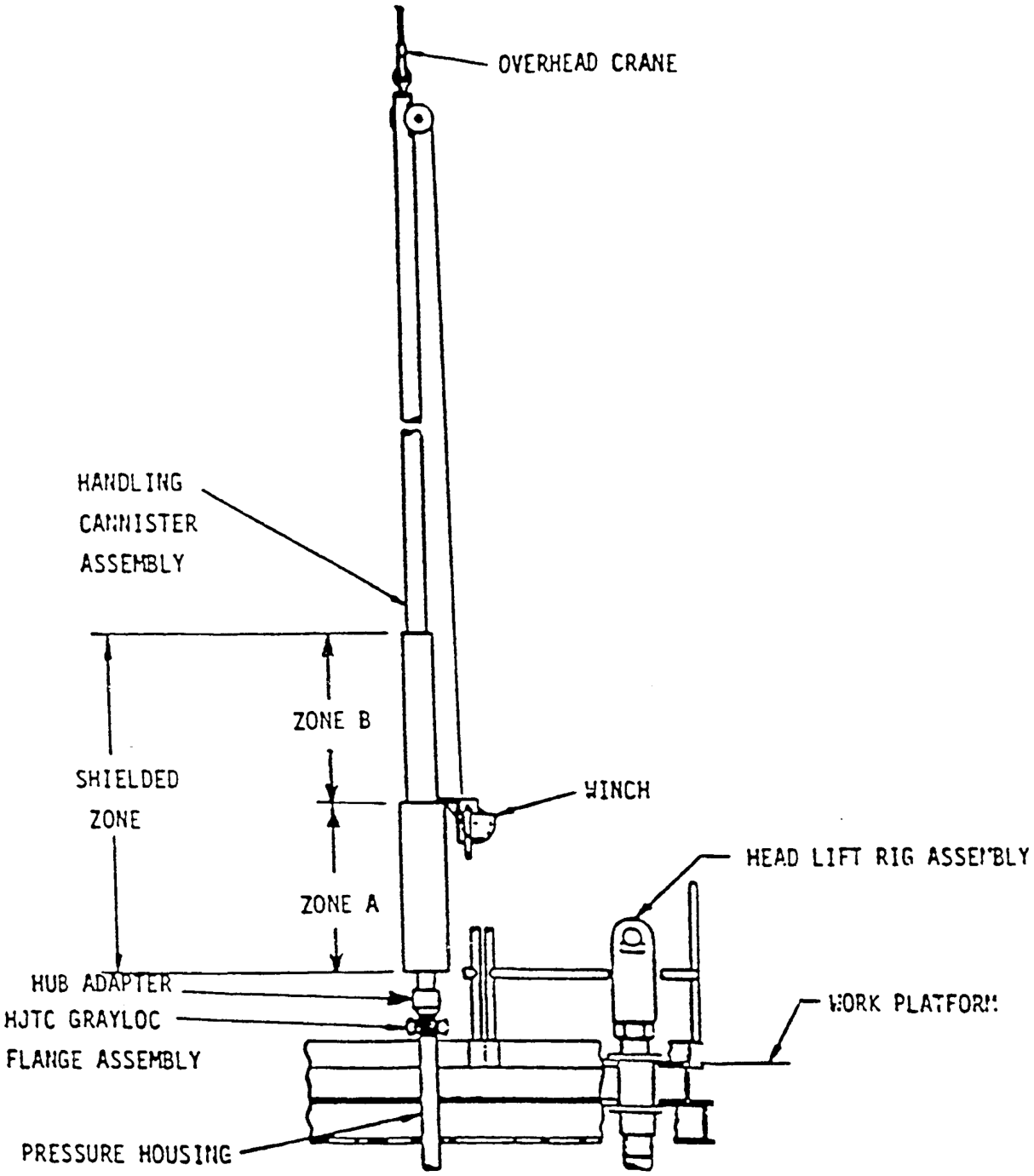
FIGURE 9.8–19 SPENT FUEL CASK CRANE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.8–20 HEATED JUNCTION THERMOCOUPLE HANDLING CANISTER



9.9 PLANT VENTILATION SYSTEMS

9.9.1 DESIGN TEMPERATURE BASES

Outdoor Design Temperatures

Most original Millstone Unit 2 HVAC calculations are based on design conditions for New Haven, Connecticut, specified in the Heating Ventilating Air Conditioning Guide, 1956 published by the American Society of Heating and Ventilating Engineers (ASHVE - now titled ASHRAE, American Society of Heating, Refrigerating, and Air Conditioning Engineers). The summer and winter outdoor design temperatures selected for Millstone Unit 2 calculations to December 3, 1996 were based on values "in common use" per the ASHVE Guide as follows:

Season	Temperature	Source
Winter	0°F dry bulb	ASHVE Guide, Chapter 14 Table 1, Column 9
Summer	95°F dry bulb, 75°F wet bulb	ASHVE Guide, Chapter 15 Table 3, Columns 7 & 8

However, some original HVAC calculations were also based on the ASHRAE Fundamentals Handbook 1972, using an outdoor design DB temperature of 86°F (2.5% design).

The current design basis temperatures for Millstone Unit 2 heating, ventilating, and air conditioning (HVAC) are based on general, long-term, climatic data of averages and extremes obtained from the American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) 1972 Fundamentals Handbook, Design Weather Data for New London, Connecticut. The selected outdoor temperature values have been substantiated by data collected from the Millstone Site meteorology tower during a nineteen year span (1974-1992). Millstone Site Climatological summary of occurrences of outdoor ambient temperatures below 0°F and above 86°F and their durations are shown in Reference 2.3-1, Table 2.3-19, of Section 2.3. The summer and winter outdoor design temperatures are as follows:

Season	Temperature	References
Winter	0°F dry bulb	a) ASHRAE 1972, Weather Data and Design Conditions, page 671
Summer	86°F dry bulb, 75°F wet bulb	a) ASHRAE 1972, Weather Data and Design Conditions, page 671

Typically nuclear power plants use a 99% winter design temperature. This represents the temperature that has been equaled or exceeded by 99% of the total hours in the average winter months of December, January, February (a total 2160 hours). In New London, Connecticut, this is a 4°F (ASHRAE 1972, page 671, column 4, winter 99%). A conservative design winter

temperature of 0°F is selected for Unit 2 and is consistent with the median of the annual extreme minimum in New London (ASHRAE Fundamentals 1972, page 671, column 4). Summer outdoor design temperatures are typically based on 2.5% exceedence. In New London, these temperatures are 86°F DB and 75°F WB (ASHRAE Fundamentals 1972, page 671, summer, columns 6 and 8). These represent values that are exceeded 2.5% of the total summer hours of 2928; i.e., 73 hours, and are used for the summer outdoor design temperatures for Unit 2.

Monthly, seasonal and annual averages and extremes of temperature from 1901 through 1981 at Bridgeport, Connecticut, are shown in Reference 2.3-1, Table 2.3-1, of Section 2.3, and for the Millstone Site from 1974 through 1992 on Table 2.3-19 of the same reference. These extreme low and high temperatures are not used in design but do establish and acknowledge that when the maximum temperature exceedences occur above or below the design temperatures, it is for a short duration of time. The episodes from 1974 through 1992 that occurred below 0°F and above 86°F with the duration for each occurrence are shown in Reference 2.3-1, Table 2.3-19 of Section 2.3.

Indoor Design Temperatures

Indoor design temperatures for the various areas of the plant are based on design outdoor temperatures specified above. Indoor design temperatures excursions may occur coincident with outdoor air temperature excursions for short periods of time. However, temperature excursions are mitigated by thermal inertia effects of heavy concrete structures, below grade walls and other factors. In general, critical areas containing equipment required for safety related functions are monitored by installed instrumentation and alarms and/or inspected on a periodic basis (i.e., Control Room, Diesel Generator rooms, Intake Structure, etceteras.). Abnormal temperature conditions will be monitored, evaluated and corrected by operator actions. Equipment evaluations consider loss of ventilation for up to 24 hours without operator action. After 24 hours, the evaluations consider operator actions to occur to reduce ambient temperatures.

A summary of the indoor design temperatures for the significant areas of the plant are provided in Table 9.9-21.

9.9.2 CONTROL ELEMENT DRIVE MECHANISM COOLING SYSTEM

9.9.2.1 Design Bases

9.9.2.1.1 Functional Requirements

The Control Element Drive Mechanism (CEDM) cooling system functions to maintain a suitable environment within the CEDM shroud to provide an acceptable coil life and maintain an acceptable coil replacement cycle.

9.9.2.1.2 Design Criteria

The following criteria have been used in the design of the CEDM cooling system:

a. The system shall have three redundant, independent subsystems, each having 50 percent of the total required heat removal capability.

b. The system shall have suitable subsystem and component alignments to assure operation of complete subsystem with associated components.

c. Capabilities shall be provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as fans, cooling coils, ductwork, piping, valves and instrumentation to assure the integrity and capability of the system.

f. The components of the CEDM cooling system shall be designed to operate in the environment to which exposed.

9.9.2.2 System Description

9.9.2.2.1 System

The CEDM cooling system is shown schematically in Figure 9.9–1.

The CEDM cooling system consists of three identical, redundant, independent fan-coil units, each capable of removing 0.81×10^6 Btu/hr. The containment air is circulated through the finned-tube-water cooling coils and is supplied by fans to the CEDM shroud. The CEDM shroud is provided with internal baffles to assure proper air distribution over the CEDM coils. The Reactor Building Closed Cooling Water (RBCCW) system (Section 9.4) serves as the cooling medium. Air is supplied via two ducts to the upper cooling plenum from two of the three cooling units located on the missile shield above. The air is distributed through four vertical ducts of the shroud to the lower plenum. The air is discharged through holes in the inner shell of the lower plenum.

9.9.2.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-1. Fan performance curve is shown in Figure 9.9–5.

9.9.2.3 System Operation

Cooling is required for the CEDM whenever the coils are energized (Section 3.3.3). Cooling is not required for safe shutdown of the unit but can be provided to prolong coil life.

9.9.2.3.1 Normal Operation

Air at 95°F is supplied into the shroud to remove the sensible heat from the CEDM coils. The cooling air maintains these CEDM coils below a maximum operating temperature of 350°F. The air is heated sensibly to 120°F and discharged from the shroud.

Total heat removal capabilities are provided by operating any two of the three CEDM coils. The switchover from one fan-coil unit to the other is accomplished by remote manual control from the control room. A damper on each fan discharge prevents short circuiting of air through the idle unit. The air temperature entering the CEDM is monitored by the computer (Section 7.5.5). Motor trip alarms are provided for all fans.

9.9.2.3.2 Abnormal Operation

During abnormal operation, the CEDM cooling system functions as required to maintain the environment within the shroud. Cooling is required for a short time only following a scram of the control rods. Whenever the rods are scrammed, the CEDM coils are de-energized. The CEDM cooling system functions to remove residual heat until the CEDM coils are cooled to ambient temperatures. The CEDM coil temperatures are monitored on the computer.

The cooling of the CEDM coils during abnormal conditions is not critical to shutdown. However, this operation is provided to maintain an acceptable CEDM coil life.

9.9.2.4 Availability and Reliability

9.9.2.4.1 Special Features

Any two of the three CEDM cooling units is capable of maintaining the design environment within the CEDM shroud. Any one of the three units is capable of maintaining the shroud ambient temperature below 145°F which is the recommended maximum design temperature. Therefore, shutdown is required only upon failure of all three of the CEDM cooling units.

The components of the CEDM cooling system are designed to operate in an environment of 120°F, atmospheric pressure, 100 percent relative humidity and 5×10^6 rad of accumulated (40 years) radiation dose.

The CEDM cooling units are located on the reactor vessel missile shield and, therefore, are not exposed to flooding or credible missiles.

9.9.2.4.2 Tests and Inspection

The vaneaxial fans are similar to fans which are rated in accordance with Air Moving and Conditioning Association (AMCA) Standard 211A. The water cooling coils are tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and Tubular Exchanger Manufacturers Association (TEMA), Class C. The ductwork is designed and

fabricated in accordance with Sheet Metal and Air Conditioning Contractors National Association (SMACNA) Standard, “High Velocity Duct Construction Standards.”

The CEDM cooling system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13.

The CEDM cooling system is not accessible for inspection or maintenance except during shutdown.

9.9.3 CONTAINMENT AND ENCLOSURE BUILDING PURGE SYSTEM

9.9.3.1 Design Bases

9.9.3.1.1 Functional Requirements

The purge system functions to maintain a suitable environment for personnel access into the containment or enclosure building. The purge system provides fresh air ventilation or heating whenever required.

9.9.3.1.2 Design Criteria

The following criteria have been used in the design of the purge system:

a. The containment and enclosure building purge system shall be designed to permit periodic inspection of important components such as fan, motor, belt, coil, ductwork, piping and valves to assure the integrity and capability of the system.

b. The purge containment isolation valves shall be designed with leak detection capabilities.

c. The components of the system shall be designed to operate in the environment to which exposed.

9.9.3.2 System Description

9.9.3.2.1 System

The purge system is shown schematically in Figure 9.9–1.

The purge system can be lined up to either containment or the enclosure building.

The purge system is designed to provide adequate fresh air for the containment or the enclosure building by using supply fan F-23. However, the purge rate is balanced to maintain a negative pressure in the area being purged. Its associated steam heating coil, X-43, is sized to temper 0°F outside air to 70°F for personnel comfort, maintaining a desired environment atmosphere temperature of approximately 60°F, which is consistent throughout the plant for unmanned areas.

The purge system consists of an air handling unit, located in the auxiliary building, which supplies filtered and tempered air to either the containment or the enclosure building. All the purge system, except interior ducts and two isolation valves, will be located outside the containment. Branch ducts and isolation dampers are provided for ventilating the enclosure building. Auxiliary steam is provided as the heating medium when required.

9.9.3.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-2. Fan performance curve is shown in Figure 9.9–6.

9.9.3.3 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 Section 7.3.2.2.c). However, automatic isolation of the purge valves is not credited in the fuel handling accident analyses.

9.9.3.3.1 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 (Section 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 Section 6.5.3.4.

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.

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

The containment isolation valves (2-AC-4, 2-AC-5, 2-AC-6, 2-AC-7) on the purge system are locked closed, pneumatically isolated and electrically isolated during all modes of operation

except cold shutdown and refueling. This is accomplished by closing the instrument air isolation valves and pulling the control power fuses for each of the valves. The associated instrument air isolation valves and fuse blocks are then locked. Once electrically isolated, the valves do not have control room indication.

Following a Containment Isolation Actuation Signal (CIAS) the supply fan, if operating, and associated isolation damper, if open, are isolated. 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 maintained at 55°F minimum. Ventilation and auxiliary heating in the enclosure building is an operator option for personnel comfort and not essential for unit operation.

9.9.3.4 Availability and Reliability

9.9.3.4.1 Special Features

All components of the purge system are designed to operate in their respective environments. Components inside containment are designed to operate in an environment of 120°F, atmospheric pressure, 100 percent relative humidity and 5×10^6 rads of accumulated radiation dose (40 years). All components located outside the containment are designed for operation in an environment of 110°F, 100 percent relative humidity and atmospheric pressure.

9.9.3.4.2 Tests and Inspections

The purge air handling unit fan is similar to fans which are rated in accordance with AMCA Standard 211A. The heating coil is of the steam distributing type and provided with adequate condensate drainage capacity to prevent freezing. The steam coil is tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and TEMA, Class C. The ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.” The purge butterfly valves are in accordance with ASME Code, Section III, Class 2. The purge system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13. The purge ventilation equipment and exterior containment isolation valves are accessible for periodic tests, inspection and maintenance. The ductwork and valves inside the containment are accessible during unit shutdown. Purge valve elastomers and shaft packing are adjusted and/or replaced based on 10 CFR 50, Appendix J, Type C test results.

The containment purge isolation valve seats are designed to withstand the pressure and temperature transients associated with the Loss of Coolant Accident (LOCA).

9.9.4 CONTAINMENT AUXILIARY CIRCULATION SYSTEM

9.9.4.1 Design Bases

9.9.4.1.1 Functional Requirements

The containment auxiliary circulation system functions to maintain a uniform temperature by mixing the containment atmosphere.

9.9.4.1.2 Design Criteria

The components of the containment auxiliary circulation system shall be designed to operate in the environment to which exposed.

9.9.4.2 System Description

9.9.4.2.1 System

The containment auxiliary circulation system is shown schematically in Figure 9.9–1. The containment auxiliary circulation system consists of ductwork from the containment dome region and two, two speed vaneaxial fans. The fans are provided with two-speed motors to reduce horsepower requirements during leak rate testing conditions (Section 5.2.8). The system exhausts air from the upper regions of the containment through the ductwork and discharges near the inlets of the containment air recirculation units (Section 6.5). This operation provides a thoroughly mixed containment atmosphere and uniform air temperatures.

9.9.4.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-3. Fan performance curve is shown in Figure 9.9–7.

9.9.4.3 System Operation

The containment auxiliary circulation system operates during normal operation and is capable of operation during leak rate testing conditions.

9.9.4.3.1 Normal Operation

The containment auxiliary circulation fans are started manually from the control room. During normal operation, these fans operate at the high speed condition to provide a uniform containment temperature by mixing.

The containment auxiliary circulation system is capable of approximately 75 percent of the total flow of the containment air recirculation and cooling system (Section 6.5). Air from the upper regions of the containment is discharged near the operating floor level where it is mixed. This

mixture is distributed throughout the lower regions of the containment by the containment air recirculation and cooling system (Section 6.5).

The containment auxiliary circulation system operates during containment purge operations (Section 9.9.2) to provide mixing.

The fans can operate at the low speed during the elevated pressure conditions of the leak rate testing (Section 5.2.8) conditions. This low speed operation is provided to reduce motor horsepower requirements. Fan operation is monitored by motor trip alarms.

9.9.4.4 Availability and Reliability

9.9.4.4.1 Special Features

The components of the containment auxiliary circulation system are designed to operate in the normal containment environment and under leak rate testing conditions. The normal environment conditions are 120°F, atmospheric pressure, 100 percent relative humidity and 40 year accumulated radiation dose of 5×10^6 rad. Leak rate testing environmental conditions are 90°F and 62.1 psig.

The containment auxiliary circulation system is not essential for normal operation. Therefore, loss of both subsystems will not require a unit shutdown.

9.9.4.4.2 Tests and Inspection

The vaneaxial fans are similar to fans which are rated in accordance with AMCA Standard 211A. The ductwork is designed and fabricated in accordance with SMACNA Standard, “Low Velocity Duct Construction Standards.”

The containment auxiliary circulation system undergoes an acceptance test prior to startup. The containment auxiliary circulation system is accessible only during shutdown for inspection and maintenance.

9.9.5 CONTAINMENT PENETRATION COOLING SYSTEM

9.9.5.1 Design Bases

9.9.5.1.1 Functional Requirements

The containment penetration cooling system functions to limit the concrete temperature to 150°F around all containment penetration sleeves which contain systems operating at temperatures greater than 150°F.

9.9.5.1.2 Design Criteria

The following criteria have been used in the design of the containment penetration cooling system:

a. The system shall have two redundant fans each having 100 percent of the total required heat removal capability.

b. The components of the containment penetration cooling system shall be designed to operate in the environment to which exposed.

c. The system shall be designed to permit periodic inspection of important components, such as fans and ductwork to assure the integrity and capability of the system.

9.9.5.2 System Description

9.9.5.2.1 System

The containment penetration cooling system is shown schematically in Figure 9.9–2. The containment penetration cooling system consists of two full capacity redundant vaneaxial fans and associated system ductwork. The containment penetration cooling system supplies enclosure building air to any containment penetration which has a normal operating temperature greater than 150°F. Each of these penetration sleeves is provided with a baffle plate in the horizontal plane (see Figure 5.2–8). Cooling air is supplied in the bottom half of the sleeve, forced down the penetration to the flued head where the air is deflected back along the upper portion of the sleeve and discharged back into the enclosure building. The temperature of the concrete around the penetration sleeve will be maintained below 150°F for all penetrations during normal operating conditions.

9.9.5.2.2 Components

The major system components and associated fabrication and performance data are listed in Figure 9.9–4. Fan performance curve is shown in Figure 9.9–8.

9.9.5.3 System Operation

The containment penetration cooling system operates at any time normal penetration cooling is required.

9.9.5.3.1 Normal Operation

The containment penetration cooling system is started manually by hand switches located in the control room.

Cooling air is provided from the enclosure building to cool the penetration sleeves. Adequate cooling can be provided by air below 120°F. Cooling air is available for the enclosure building from the purge system (Section 9.9.2), if required, to maintain the ambient temperature below 120°F. Cooling air at a maximum temperature of 120°F is supplied to the penetrations at the rates indicated in Table 9.9-5.

The performance of the containment penetration cooling system is monitored by temperature indication in each main steam and main feedwater penetration. Fan operation is monitored by motor trip alarms. Interlocks are provided to prevent short circuiting of air through idle equipment.

9.9.5.4 Availability and Reliability

9.9.5.4.1 Special Features

The containment penetration cooling system is provided with two full capacity fans. Each fan has the capability of maintaining the concrete temperature around the sleeve below 150°F. Following the unlikely loss of penetration cooling, a maximum temperature of 390°F may be tolerated for 120 days without appreciable loss of strength of the concrete (Section 5.1.3). The components of the containment penetration cooling system are designed to operate in an environment of 120°F, atmospheric pressure and 100 percent relative humidity.

9.9.5.4.2 Tests and Inspection

The vaneaxial fans are similar to fans which are rated in accordance with the AMCA Standard 211A. The system ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.” The containment penetration cooling system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13. Equipment is located in controlled access areas but can be inspected periodically.

9.9.6 RADWASTE AREA VENTILATION SYSTEM

9.9.6.1 Design Bases

9.9.6.1.1 Functional Requirements

The radwaste ventilation system is a Non-QA system, consisting of supply fan F-16 in combination with the main exhaust fans F-34A/B/C. It functions to provide a suitable environment for the equipment and fresh air ventilation for personnel within the potentially radioactive areas of the auxiliary building. The areas serviced by this system consist of the auxiliary building elevations (-)45 foot 6 inches, (-)25 foot 6 inches, and (-)5 foot 6 inches in its entirety, the general open area at elevation 14 foot 6 inches (i.e., area by: elevator, MCC's B51 and B61), and the closed rooms (142A, 150) located adjacent to the truck bay area of the auxiliary building at elevation 14 foot 6 inches.

9.9.6.1.2 Design Criteria

The following criteria have been used in the design of the radwaste ventilation system:

a. The system shall be designed to permit periodic inspection of important components, such as fan, motor, belt, coil, filters, ductwork, piping and valves to assure the integrity and capability of the system.

b. The components of the radwaste ventilation system are designed to operate in the environment to which exposed.

c. The potentially radioactive areas of the auxiliary building shall be served by a separate ventilation system from the clean areas.

d. The potentially radioactive areas of the auxiliary building shall be maintained at a negative pressure.

9.9.6.2 System Description

9.9.6.2.1 System

The radwaste ventilation system is shown schematically in Figure 9.9–3.

The radwaste area ventilation system consists of a supply fan F-16 providing 100%, once through, tempered outside air through a steam heating coil to all levels of the auxiliary building containing potentially radioactive areas, with the exception of the spent fuel area which has its own ventilation system (Section 9.9.8). Air flow rates are established to maintain the areas at or below its maximum design temperature of 110°F and steam coils are provided to maintain a minimum temperature of 55°F.

The air is drawn from the potentially radioactive areas of the auxiliary building via the main exhaust system, fans F-34A/B/C, (Section 9.9.9) through the radwaste system HEPA filter unit L-26 and is discharged into the Unit 2 stack.

The auxiliary building’s potentially radioactive areas are maintained at slightly negative pressure within their defined pressure boundaries. This negative pressure is induced and maintained within these boundaries by exhausting more air, via the main exhaust fans F-34A/B/C, than the supply air flow rate provided by fan F-16. The directional air flow between these radioactive areas is maintained in the direction of potentially higher radioactivity.

9.9.6.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-6. Fan performance curve is shown in Figure 9.9–9.

9.9.6.3 System Operation

The radwaste ventilation system is required for building ventilation during unit operation and during shutdown operation.

9.9.6.3.1 Normal Operation

The radwaste ventilation system is manually started from a local control hand switch in the control room mechanical equipment area. The system logic (Figure 9.9–10) requires two main exhaust fans (Section 9.9.9) to be operating prior to initiation of the supply fan F-16.

The supply fan F-16 distributes fresh outside air to the potentially contaminated areas at all levels of the auxiliary building. A steam heating coil is provided for tempering outside air when required. A flow switch is provided on the supply unit discharge to monitor air flow and alarm low flow conditions. A temperature switch is provided at the heating coil outlet to monitor temperature and alarm low temperature conditions. Two (2) safety related steam leak detection temperature detectors (RTD's) are located in the ductwork downstream of the heating coils to isolate auxiliary steam to the Auxiliary Building (Section 7.10).

Outdoor air is supplied at approximately 86°F during summer operations to limit the auxiliary building ambient temperature to 110°F. During winter operations, the air is tempered to maintain a minimum auxiliary building temperature of 55°F in the areas with storage tanks containing boron solution to prevent boron precipitation. However, booster coils are provided for elevation 14 feet 6 inches floor level to maintain a desired temperature of 70°F within the dressing area for personnel comfort.

The exhaust from the radwaste areas is filtered through particulate and HEPA filters and is discharged by the main exhaust fans through the Unit 2 stack. This is described in Section 9.9.9. Radiation monitoring is described in Section 7.5.6.

9.9.6.4 Availability and Reliability

9.9.6.4.1 Special Features

To assure an overall slight negative pressure within the potentially radioactive areas of the auxiliary building, the supply fan, F-16, can be started only if two main exhaust fans F-34A/B/C (Section 9.9.9) are in operation. Upon the loss of the supply fan, F-16, the main exhaust fans keep operating, thus maintaining a negative pressure within these areas. However, the supply fan, F-16, is returned to service as soon as practicable. Upon the loss of two main exhaust fans, a main exhaust duct pressure switch trips off supply fan F-16.

Two automatically closed dampers, in a series arrangement, are provided at all penetrations into the Enclosure Building Filtration Region (EBFR) (Section 6.7). These dampers are pneumatically actuated and designed to fail in a closed position. All components of the radwaste ventilation system are designed to operate in an environment of 110°F, atmospheric pressure and 100 percent relative humidity.

Two safety related temperature detectors are installed in ductwork downstream of the steam heating coil to detect potential steam leaks. Actuation of either of these temperature detectors will cause isolation of the auxiliary steam supply to the Auxiliary Building in isolation valves located in the Turbine Building. This condition will also be alarmed in the control room.

9.9.6.4.2 Tests and Inspection

The centrifugal fans are similar to fans which are rated in accordance with AMCA Standard 211-A. The steam heating coils are tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and TEMA, Class C. The ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.”

The radwaste ventilation system undergoes an acceptance test prior to start-up. The test procedure is described in Chapter 13.

All components of the radwaste ventilation system are accessible for inspection and maintenance.

9.9.7 NONRADIOACTIVE AREA VENTILATION SYSTEM

9.9.7.1 Design Bases

9.9.7.1.1 Functional Requirements

The nonradioactive area ventilation system is a non-QA system, consisting of supply fan F-17, recirculating fan F-19, and exhaust fan F-26 for the East 480-volt switchgear room, and exhaust fan F-112A & F-112B for the remaining areas. It functions to provide a suitable environment for the equipment and fresh air ventilation for personnel within the clean areas of the auxiliary building, and the East & West turbine building cable vaults. However, there is no heating coil associated with this system as there is sufficient heat rejected by the electrical equipment within these areas. The maximum design temperature in these areas is 104°F and the minimum is 55°F for personnel comfort, which is consistent throughout the plant for unmanned areas, with the exception of the battery rooms, which is 60°F.

The nonradioactive areas of the auxiliary building consist of the East 480 volt switchgear room (elevation 36 foot 6 inches), the main cable vault (elevation 25 foot 6 inches), the East & West DC switchgear rooms as well as the East & West battery rooms (elevation 14 foot 6 inches), and the HVAC equipment room (elevation 36 foot 6 inches), which has no ventilation system. The East & West turbine building cable vaults, elevation 45 foot, are opened to the auxiliary building main cable vault (elevation 25 foot 6 inches). The auxiliary building nonradioactive areas are physically separated from the potentially radioactive areas. Other than the East 480 volt switchgear room, which is maintained at a slightly negative pressure, there is no pressurization requirement for the remaining nonradioactive areas.

9.9.7.1.2 Design Criteria

The following criteria have been used in the design of the nonradioactive ventilation system:

a. The system shall be designed to permit periodic inspection of important components, such as fan, motor, belt, filters, ductwork, piping and valves to assure the integrity and capability of the system.

b. The components of the nonradioactive ventilation system are designed to operate in the environment to which exposed.

9.9.7.2 System Description

9.9.7.2.1 System

The nonradioactive area ventilation system is shown schematically in Figure 9.9–3.

The clean areas of the auxiliary building as well as the East & West turbine building cable vaults are served by the nonradioactive ventilation system which consists of supply fan F-17, recirculating fan F-19, and exhaust fans F-26 for the East 480 volt switchgear room and battery rooms exhaust fans F-112A & F-112B for the remaining areas.

The battery rooms are ventilated by drawing main cable vault air through the battery rooms while also drawing a small amount from the DC switchgear rooms into the battery rooms with roof exhaust fans F-112 A & F-112B which discharge to atmosphere. All rooms are separated by fire dampers.

9.9.7.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-7. Fan performance curves are shown in Figures 9.9–11 through 9.9–14.

9.9.7.3 Systems Operation

The nonradioactive area ventilation system is required during normal and shutdown operations.

9.9.7.3.1 Normal Operation

The nonradioactive area ventilation system supply fan, F-17, is interlocked with the East 480 volt switchgear room exhaust fan F-26 via a room thermostat and time delay relays. These fans start at 75°F and reset at 72°F, with a 20 minute time delay. This interlock scheme ensures that the exhaust fan F-26 starts before supply fan F-17. The control scheme is designed to limit the number of fan starts to three per hour by running the fans F-17/F-26 for a minimum time of 20 minutes per start. This system is designed to maintain the East 480 Volt switchgear room below 104°F, based on a summer outdoor design temperature of 86°F. During winter operation there is sufficient heat generated by the electrical equipment to maintain the room temperature above 55°F for personnel comfort, which is consistent throughout the plant for unmanned areas.

Because the East 480 volt switchgear room is the dominant heat load area, the thermostat controlling supply fan F-17 is located within this room. During warm weather, these areas served

by fan F-17 get adequate cooling since the controlled area has a higher demand; during cold weather, these areas temperatures may be lower than the dominant area. Although this is not a concern for the cable vault areas, the minimum desired battery room temperature is 60°F. The battery rooms are provided with HI-LO temperature alarms. Historically, since the steam coil was removed from fan F-17 in 1992, temperature in the battery rooms has not been a problem. The impact on the DC switchgear rooms temperature, caused by the small amount of fresh air drawn through it, is insignificant.

A recirculating fan, F-19, equipped with a chilled water coil X-40, and located in the upper 4160 volt AC switchgear room, elevation 56 foot 6 inches, provides extra cooling for the main cable vault, as well as the East and West turbine building cable vaults.

9.9.7.4 Availability and Reliability

9.9.7.4.1 Special Features

The East & West battery rooms are interconnected by ductwork to assure venting following the failure of either 50 percent air flow capacity exhaust fans F-112A or F-112B, which is adequate to maintain room temperatures.

All components of the nonradioactive ventilation system are designed to operated in an environment of 110°F, atmospheric pressure, and 100 percent relative humidity.

9.9.7.4.2 Tests and Inspection

The centrifugal and vaneaxial fans are similar to fans that are rated in accordance with AMCA Standard 211-A. The ductwork is designed and fabricated in accordance with SMACNA Standard, “Low Velocity Duct Construction Standards.”

The nonradioactive ventilation system undergoes an acceptance test prior to startup.

The nonradioactive system is accessible for periodic tests and inspection.

9.9.8 ENGINEERED SAFETY FEATURES ROOM AIR RECIRCULATION SYSTEM

9.9.8.1 Design Bases

9.9.8.1.1 Functional Requirements

The Engineered Safety Features Room Air Recirculation System (ESFRARS) functions to maintain a suitable environment for the electric motor drivers of the safety injection and containment spray pumps during post accident operation.

9.9.8.1.2 Design Criteria

The following criteria have been used in the design of the ESFRARS:

a. The system shall have two redundant, independent subsystems, each having 100 percent of the total required heat removal capability.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as fans, coils, ductwork, piping and valves to assure the integrity and capability of the system.

f. The system shall be designed to permit appropriate periodic functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, the performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The components of the system shall be designed to operate in the most severe post-accident environment as described in Section 6.1.

9.9.8.2 System Description

9.9.8.2.1 System

The ESFRARS is shown schematically in Figure 9.9–2. The automatic initiation logic for ESFRARS is described by the combination of: (a) the initiation logic diagram for the Safety Injection Actuation Signal (SIAS) in Figure 7.3–1, and (b) the tabulation of the equipment that is actuated by SIAS in Figures 7.3–2A through 7.3–2D.

The ESFRARS consists of two redundant, independent subsystems, each capable of maintaining the required temperature in one engineered safety features pump room (Section 6.1.4.1). Each ESFRARS subsystem consists of a fan and finned tubed heat exchanger. Each safeguard pump room contains one full capacity ESFRARS fan and coil unit. The third High Pressure Safety Injection (HPSI) pump room is served by both fan and coil units.

Each ESFRARS fan and coil unit is capable of cooling two HPSI pumps (including the swing pump in “C” cubicle), one Low Pressure Safety Injection (LPSI) pump and one containment spray

pump. The ESFRARS is designed to limit the maximum ambient temperature to 177.9°F. All components can withstand the ambient conditions.

9.9.8.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-8. Fan performance curve is shown in Figure 9.9–15.

9.9.8.3 System Operation

The ESFRARS is provided to maintain a suitable environment for the safety injection and containment spray equipment following a LOCA.

During normal plant operations, the Radwaste Ventilation System (F-16), Section 9.9.5 services the ESF pump rooms. However, during operation of the shutdown cooling system, the ESFRARS may be operated to supplement the cooling provided by fan F-16.

9.9.8.3.1 Emergency Conditions

The ESFRARS is automatically initiated by the safety injection actuation signal (SIAS) (Section 7.3). Room air is circulated over the water cooling coils by the vaneaxial fans and discharged back into the room. A minimum air flow of 3,650 CFM is diverted to the third high pressure safety injection pump room from each recirculation unit fan (F-15A and F-15B).

The ESFRARS is aligned with the respective emergency power source (Section 8.3) as the associated safety features equipment served. The RBCCW system (Section 9.4) serves as the heat sink.

System operation is monitored by temperature and flow indication of the water side. Indication is provided in the control room. Fan operation is monitored by motor trip alarms. High room temperature is alarmed in the control room.

9.9.8.4 Availability and Reliability

9.9.8.4.1 Special Features

The components of the ESFRARS are designed to the general criteria including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physical separation of duplicate equipment as described in Section 6.1.

The ESFRARS fan and coil units are completely redundant, physically separated in water tight pump rooms and powered by separate emergency power sources. A failure mode analysis for the ESFRARS is given in Table 9.9-16.

Each ESFRARS subsystem is designed to remove the heat given off by two HPSI pumps (including the swing pump in the “C” cubicle), one LPSI pump, and one containment spray pump.

However, since the LPSI pumps are not required following the recirculation mode (Section 6.3), the fan coil units have additional cooling capabilities.

The ESFRARS is provided to maintain an ambient temperature of no greater than 177.9°F for the electric motor drivers of the engineered safety feature pumps. The safety injection and containment spray pump motors can withstand exposure to the ambient conditions.

All components of the ESFRARS are designed to operate in an environment of 177.9°F, atmospheric pressure and radiation level of at least 10^6 rad. Electric motors are designed for 100 percent relative humidity.

9.9.8.4.2 Tests and Inspection

One of the two vaneaxial fans is rated in accordance with AMCA Standard 211-A. The water cooling coils are tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and TEMA, Class R. The ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.”

Provisions are incorporated into the system design for on-line testing capabilities. Each ESFRARS subsystem is tested separately from the redundant subsystem. The ESFRARS is automatically started by manually initiating the SIAS. Fan operation is indicated by motor trip alarms.

The ESFRARS undergoes a preoperational test prior to startup. The test procedure is discussed in Chapter 13.

The components of the ESFRARS are accessible for periodic inspection and maintenance.

9.9.9 FUEL HANDLING AREA VENTILATION SYSTEM

9.9.9.1 Design Bases

9.9.9.1.1 Functional Requirements

The fuel handling area ventilation system is a non-QA system, consisting of supply fan F-20 in combination with the main exhaust fans F-34A/B/C and a recirculating fan F-140 (coil X-191). It functions to provide a suitable environment for the equipment and fresh air ventilation for personnel within the fuel handling area of the auxiliary building. The fuel handling area consists of the open area at elevation 38 foot 6 inches, the mezzanine on the East side, and the open lower area known as the truck bay area.

9.9.9.1.2 Design Criteria

The following criteria have been used in the design of the fuel handling ventilation system:

a. The system is designed to permit periodic inspection of important components, such as fan, motor, belt, oil, filters, ductwork, piping and valves, to assure the integrity and capability of the system.

b. The components of the fuel handling ventilation system are designed to operate in the environment to which exposed.

c. The fuel handling area which is part of the auxiliary building shall be served by a separate ventilation system.

d. The fuel handling area shall be maintained at a slightly negative pressure.

9.9.9.2 System Description

9.9.9.2.1 System

The fuel handling area ventilation system is shown schematically in Figure 9.9–3.

The fuel handling area ventilation system is designed to provide adequate ventilation in the spent fuel pool area and to prevent cross contamination with surrounding areas. The maximum design temperature in the fuel handling area is 110°F, and the minimum temperature maintained at 55°F, which is consistent throughout the plant for unmanned areas. However, booster coils are provided to maintain a desired temperature of 70°F for personnel comfort (e.g., dressing area).

The fuel handling ventilation system consists of a supply fan, F-20, providing 100 percent outside fresh air, tempered through a steam coil, to the spent fuel area at elevation 38 feet 6 inches.

A self contained air conditioning unit F-140 (coil X-191), located in the spent fuel pool area, provides cooling whenever operations in the spent fuel pool area are required during extreme warm weather. The air conditioner discharges through ductwork along the south wall of the spent fuel pool. The ductwork has been sized to accommodate a second air conditioning unit of similar size.

The air is drawn through return registers located on the north side wall, immediately above the spent fuel pool, to ensure that the negative pressure within the fuel handling area of the auxiliary building is at its strongest across the spent fuel pool surface area where the radioactive gaseous release is most likely to occur. The air is exhausted at a greater rate than it is supplied to ensure an overall negative pressure within the fuel handling area of the auxiliary building.

During normal operation, air is drawn by the main exhaust fans F-34/A/B/C through HEPA filter unit L-27. Prior to the handling of irradiated fuel, the exhaust air control logic may be aligned to the auxiliary exhaust system (AES).

Ventilation of the fuel handling area during handling of irradiated fuel may be provided by main exhaust or the auxiliary exhaust system (AES), but neither system is required. While these systems are used and fuel handling area boundary integrity is set, then any radioactive effluent is

discharged via a filtered and monitored pathway. The integrity of the fuel handling area boundary is not a requirement while main exhaust or AES ventilation is in use. In the event that neither ventilation system is available, then suitable radiological monitoring is recommended per the Millstone Effluent Control Program.

As discussed in the Millstone Unit 2 Environmental Report and FSAR Section 14.7, an accident is not credible unless fuel is being handled or a heavy object has been moved over the spent fuel pool.

The radiation monitoring system in the fuel handling area consists of four (4) gamma sensitive plastic scintillation, detector assemblies equally spaced above the spent fuel pool area with local indication plus visual and audible alarms.

Alarm circuitry of the readout module located in the control room will provide contact closure (2-out-of-4 matrix) upon high radiation for automatic operation of the auxiliary exhaust system. The automatic operation of the auxiliary exhaust system is not credited in the fuel handling accident analysis.

Alarm set points are adjustable and are normally set for 50 mr/hr.

9.9.9.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-9. The fan performance curve is shown in Figure 9.9–16. The EBFS is described in Section 6.7.

9.9.9.3 System Operation

The fuel handling area ventilation is available during normal and shutdown operations.

9.9.9.3.1 Normal Operation

The fuel handling area ventilation system is manually started from the control room. The system logic requires one main exhaust fan (Section 9.9.9) to be operating prior to initiation of the supply fan (F-20).

During normal operation, it may be necessary to move heavy equipment over the spent fuel pool (i.e., moving fuel assembly, fuel cask), in which case the AES may be manually initiated. Manual initiation of AES is not credited in the fuel handling accident analyses.

The supply fan F-20 distributes fresh outside air to the operating floor level at elevation 38 feet 6 inches. A steam heating coil is provided for tempering outside air when required. A flow switch is provided on the supply unit discharge to monitor air flow and alarm flow conditions.

Outdoor air is supplied at approximately 86°F during summer operation to limit the fuel handling area maximum temperature to 110°F. During winter operation, the air is tempered to maintain a

temperature at 55°F, which is consistent throughout the plant for unmanned areas. However, supplemental building heat is provided by steam unit heaters to maintain a desired temperature of 70°F for personnel comfort (e.g,. dressing area).

During normal operation, air is pulled across the spent fuel pool surface by the main exhaust fans F-34A/B/C (Section 9.9.9) through the HEPA filter unit (L-27) prior to discharge to the Unit 2 stack. The effluent is monitored for radiation (Section 7.5.6).

The air conditioning unit F-140 is manually started and stopped locally as required by the operator to maintain comfortable conditions for the personnel working in the area.

9.9.9.3.2 Emergency Operations

There is no requirement for operation of the fuel handling area ventilation system prior to movement of irradiated fuel or a cask in the spent fuel pool. Nor is there any requirement for operation of any ventilation system to mitigate a fuel handling accident, or a cask drop accident in the fuel handling area. Post-accident doses attributable to a fuel handling or cask drop accident are within the criteria identified by Regulatory Guide 1.183 and 10 CFR 50.67 and do not credit either the main exhaust or AES ventilation systems.

9.9.9.4 Availability and Reliability

9.9.9.4.1 Special Features

During normal plant operation, to assure a negative pressure within the fuel handling area of the auxiliary building, the supply fan F-20 can be started only if one main exhaust fan F-34A/B/C (Section 9.9.9) is operating. Upon a loss of the supply fan F-20, ventilation is still maintained, although at a lower rate by infiltration induced by an increased negative pressure generated by main exhaust fans. However, the supply fan F-20, is returned to service as soon as practicable.

Upon the loss of all three main exhaust fans F-34A/B/C, a main exhaust duct pressure switch trips-off supply fan F-20.

Round ductwork is provided for additional strength for the AES. Generally, if space permits, round ductwork is provided for seismic Class 1 requirements. This facilitates seismic analysis of the system. The components of the EBFS are described in Section 6.7.

All components of the normal fuel handling ventilation system are designed to operate in an environment of 110°F, atmospheric pressure and 100 percent relative humidity.

9.9.9.4.2 Tests and Inspection

The centrifugal fans are similar to fans which are rated in accordance with AMCA Standard 211-A. The steam heating coils are tested in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and TEMA, Class C. The ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.”

The normal ventilation system undergoes an acceptance test prior to startup, while the AES undergoes a preoperational test. The test procedure is described in Chapter 13.

The AES is incorporated with provisions for online testing. The AES is manually initiated in the control room by operator action. The EBFS is automatically isolated from the EBFR and aligned with the fuel handling area. The normal ventilation is isolated from the fuel handling building.

System operation is monitored by flow and filter differential pressure indication. Damper opening and system alignment is monitored by positioned indication in the control room.

The components of the fuel handling ventilation system are accessible for periodic inspection and maintenance.

9.9.10 MAIN EXHAUST VENTILATION SYSTEM

9.9.10.1 Design Bases

9.9.10.1.1 Functional Requirements

The main exhaust ventilation system is a Non-QA system, consisting of exhaust fans F-34A/B/C. It functions to filter the exhaust from all potentially radioactive areas of the unit.

9.9.10.1.2 Design Criteria

The following criteria have been used in the design of the main exhaust system:

a. The system shall be designed to permit periodic inspection of important components, such as fan, motor, belt, coil, filters, ductwork, piping and valves, to assure the integrity and capability of the system.

b. The components of the main exhaust system are designed to operate in the environment to which exposed.

c. The potentially radioactive areas of the auxiliary building shall be served by a separate ventilation system from the clean areas.

d. The potentially radioactive areas of the auxiliary building shall be maintained at a negative pressure in respect to the clean areas.

9.9.10.2 System Description

9.9.10.2.1 System

The main exhaust system is shown schematically in Figures 9.9–1 and 9.9–2.

The main exhaust ventilation system is a Non-QA system and is designed to exhaust air from the areas of the auxiliary building served by the radwaste supply fan F-16, the spent fuel handling area supply fan, F-20 and the containment/enclosure building purge fan, F-23. Exhaust air from these areas is processed through HEPA filter units prior to discharge through the unit 2 stack.

The main exhaust ventilation system consists of three filter units (L-25, L-26, L-27), three centrifugal fans F-34A/B/C, and a pressure relief damper 2-AC-59. Each filter unit is associated with a respective area and is designed for the full system flow. Each main exhaust fan is sized for 32,000 cfm. However, during start-up it was accepted that the exhaust fans could not generate the design flow and consequently, supply fans F-16, F-20, F-23 serving the areas covered by the main exhaust fans are balanced to assure that the exhaust air flow is greater than the supply air flow to maintain a negative pressure in these areas. Normally, two fans are required for the radwaste and fuel handling areas with the third fan on standby. During purging operations or enclosure building ventilation, all three exhaust fans are operating.

9.9.10.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-10. Fan performance curve is shown in Figure 9.9–18.

9.9.10.3 System Operation

The main exhaust ventilation system is required during normal and shutdown operations.

9.9.10.3.1 Normal Operation

The main exhaust ventilation system is manually started from the control room. Normally two of the three fans, F-34A/B/C are initiated for normal operation. Three fans are required to operate during purging or enclosure building ventilation operations. The third fan serves as backup for the two operating and provides redundancy in system design.

Air is exhausted from the respective areas in the auxiliary building at a greater rate than it is supplied to induce a negative pressure in these areas. The exhaust air is processed through the respective HEPA filter unit to remove airborne particulate matter. The exhaust duct from each of the three areas is provided with process radiation monitors (Section 7.5.6) to alarm high radiation levels.

Flow switches are provided at each fan discharge to alarm low flow conditions. A counterweight backdraft damper is provided at each exhaust fan discharge to minimize recirculation through an idle fan. The damper position for the air pressure damper 2-AC-59, and for each HEPA filter unit (L-25, L-26, L-27) isolation damper, is monitored in the control room.

9.9.10.4 Availability and Reliability

9.9.10.4.1 Special Features

Two of the three main exhaust fans F-34A/B/C are required for normal operation. Although purging containment or enclosure building ventilation requires the third fan, this operation is discontinued upon the loss of any exhaust fans. Purging is not essential for unit operation or for shutdown and can be discontinued at any time to assure availability of exhaust fans for the auxiliary building.

Provisions for outside makeup air are provided through damper 2-AC-59 to act as a pressure relief damper whenever the pressure in the main exhaust duct exceeds a preset limit switch (automatic function) and to act as a flow balancing damper whenever an auxiliary building HEPA filter unit L-26/L-27 is taken out of service for maintenance, testing or filter replacement. Outside air is available at a maximum rate of 20,000 cfm (automatic function) to prevent the main exhaust plenum from exceeding its design negative pressure of (-)6 inches wg. By procedure, a manual function, whenever the radwaste ventilation system HEPA filter unit, L-26 is isolated supply fan F-16 is taken out of service, only one exhaust fan is allowed to operate and the makeup air damper 2-AC-59 is manually controlled to open at its minimum flow. This operation is necessary because one exhaust fan's capacity exceeds the spent fuel handling fan, F-20, rated at approximately 20,000 cfm. The minimum flow is preset to match the actual exhaust fan capacity with the actual fuel handling HEPA filter L-27 rated flow to maintain approximately the same negative pressure within the fuel handling area.

All the MES fans will trip on receipt of a Channel 1 CIAS actuation signal. This trip signal can be overridden to restart the fans to restore the ventilation if a CIAS signal is still present. This trip signal is not a credited engineered safety feature (ESF) equipment actuation. The trip will minimize the consequence of the single failure of dampers 2-AC-11 (Section 6.7.4.1.2).

By procedure, with the fuel handling HEPA filter unit L-27 out of service, supply fan F-20 is deenergized, makeup damper 2-AC-59 is opened to its maximum setting providing approximately 20,000 cfm and two main exhaust fans are activated. The maximum makeup flow is preset to match the actual exhaust fans capacity with the actual radwaste HEPA filter L-26 rated flow to maintain approximately the same negative pressure within the radwaste areas of the auxiliary building.

Outside makeup air capabilities are not required for the purge exhaust scenario since the requirements for the HEPA filter unit L-25 are within one exhaust fan capacity.

Each ventilation filter unit is provided with a differential pressure control damper to maintain a constant pressure drop across each unit as the filters get dirty. This provides the means to balance air flow for operations among parallel HEPA filter units L-25, L-26 and L-27.

To assure a negative pressure within the auxiliary building, the main exhaust fans must be started prior to starting any of the supply fans F-16, F-20, or F-23. The exhaust fans can maintain the specific area negative pressure upon the loss of its respective supply fan. However, upon the loss

of one or more main exhaust fans F-34A/B/C, supply fans F-16, F-20 and F-23 will trip off based on their respective pressure switches related setting.

The discharge duct from the non-safety related Chemistry Laboratory Exhaust Ventilation Fans, F-165 and F-166 is connected to the inlet plenum of Main Exhaust Fans F-34A/B/C. This duct is equipped with an air operated isolation damper, 2.HV-710, which is controlled by a solenoid valve designed to close the damper when a low or no flow condition exists in the plenum. The damper also closes on loss of instrument air.

All components of the main exhaust ventilation system are designed to operate in an environment of 110°F, atmospheric pressure, and 100 percent relative humidity.

9.9.10.4.2 Tests and Inspections

The centrifugal fans are similar to fans which are rated in accordance with AMCA Standard 211-A. The ductwork is designed and fabricated in accordance with SMACNA Standard, “High Velocity Duct Construction Standards.”

The HEPA filters are periodically tested in accordance with Regulatory Guide 1.140 and ANSI N510-1975, Testing of Nuclear Air-Cleaning Systems.

The main exhaust system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13.

All components of the main exhaust system located outside the containment are accessible for inspection and maintenance. Components within containment are accessible only during shutdown.

9.9.11 CONTROL ROOM AIR CONDITIONING SYSTEM

9.9.11.1 Design Bases

9.9.11.1.1 Functional Requirements

The control room air conditioning system functions to maintain a suitable environment for personnel and for safety related control and electrical equipment.

9.9.11.1.2 Design Criteria

The following criteria have been used in the design of the control room air conditioning system:

a. The system shall have two redundant, independent subsystems, each having 100 percent of the total required heat removal capability. However, there are some common supply and return ductwork and dampers.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure system operation with either on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as fans, motors, coils, compressor, filters, piping, valves, instrumentation and ductwork, to assure the integrity and capability of the system.

f. The control room air conditioning system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components; (2) the operability and performance of the active components of the system; and (3) the operability of the system as a whole. Under conditions as close to the design as practical, performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, and the transfer between normal and emergency power sources.

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The components of the control room air conditioning system shall be designed to operate in the most severe post-accident environment in which exposed.

i. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

9.9.11.2 System Description

9.9.11.2.1 System

The control room air conditioning system is shown schematically in Figure 9.9–4A.

The control room air conditioning system consists of two full capacity, completely independent air handling and mechanical refrigeration subsystems with the exception of some common ductwork and dampers. Each control room air conditioning subsystem is a single zone system. The system has the capability of ventilating with outside air while cooling, using mechanical refrigeration.

Each subsystem is provided with a bypass through the control room filtration system (CRFS) consisting of particulate, HEPA and charcoal filters and fan. The physical properties of the CRFS

charcoal filters are listed in Table 9.9-11. Each CRFS unit is capable of filtering 2500 cfm of control room air or for an Emergency Fresh Air Intake Mode, to introduce an adequate supply of outside air into the system after filtering.

Outside air is not provided for pressurizing the control room because of the potential radioactivity during the post-accident condition. Outside makeup air is avoided to minimize possibilities of inducing contamination into the control room. Outside air is introduced over the long term post-accident case only to provide fresh air for personnel safety.

Control room pressure is maintained at a relatively neutral pressure during normal plant operations by air balancing the HVAC system. When the HVAC system is in recirculation mode (accident condition), the control room in-leakage rate is limited by Technical Specification limits.

The control room air conditioning system is designed to maintain a suitable environment in the control room for operating personnel and safety related control equipment. The control room is maintained at 78°F in the summer and 72°F in the winter.

The control room air conditioning system is equipped with full capacity redundant fans, filters, and mechanical refrigeration equipment, plus the necessary dampers and controls for automatically switching to full recirculation for post-accident operation. The control room area system performance is continually monitored with alarms for high radiation, fan failure and excessive pressure drop through filters. The control room operator has remote, manual control for selecting damper position, backup fan and filter operation to ensure satisfactory control room conditions following an accident.

The dose in the Unit 2 control room, resulting from a Design Basis Accident (DBA), has been determined as follows. The fission product release model given in Regulatory Guide 1.183 is assumed. All activity is assumed to be in the containment atmosphere. The dose to operating personnel in the control room as a result of inhalation, submersion and shine following a LOCA is described in Section 14.8.4.

Airborne radiation in the control room is negligible since any leakage out of the containment is into the enclosure building filtration region (Section 6.7) where it is filtered by charcoal filters and released through the Millstone stack. In addition, the control room is provided with an air conditioning system which may be operated in a closed, complete recirculation mode with provisions for filtering through charcoal filters following a high radiation level.

9.9.11.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-11. Fan performance curves are shown in Figures 9.9–19 through 9.9–22.

Figure 9.9–4 indicates that a fan unit has been provided between the supply plenum and the control room (coordinates A-8). This blower is part of the process radiation monitor for the supply air duct. This is not part of the process air conditioning system.

9.9.11.3 System Operation

9.9.11.3.1 Normal Operation

The control room air conditioning system operates during all modes of operation and shutdown. The flow rate associated with normal operation of the control room air conditioning system is given in Table 9.9-11. The normal flow rate is approximately 14,800 cfm.

One hand switch starts the supply air handling unit, opens the outside air dampers and places the temperature control system in control of system operation. A second hand switch starts the exhaust fan and opens the associated isolation damper. The temperature control system satisfies the room cooling requirements by providing mechanical refrigeration. System operation is monitored by control room temperature indication in the control room. System alignment is monitored by valve position indication on the control boards. Flow balance and neutral pressure is maintained by directing some flow to the cable vault, equivalent to the fresh air intake.

Smoke detectors are provided in the return ductwork to automatically isolate the supply unit and initiate purging operations. The control room is purged by operating the exhaust fans and discharging to atmosphere. Discharge dampers to other areas are automatically closed. Both control room air conditioning subsystems are served by the smoke detection system. Fire dampers are provided at all ductwork penetrations through firewalls.

9.9.11.3.2 Emergency Operation

Normally, the method of conditioning the air is controlled by the automatic temperature control system. In the event of a LOCA emergency safeguards system generates an EBFAS, which automatically shifts the control room air conditioning system to a complete recirculation mode of operation in which outside air is not introduced into the system and all outside air dampers are closed. In addition, in the event of a fuel handling accident in the Spent Fuel Pool area, an AEAS is generated which automatically shifts the system to the complete recirculation mode. The analysis does not credit AEAS or EBFS during a fuel handling or cask drop accident. The automatic control system is capable of cooling using the mechanical refrigeration. Portions of the system air or outside air, can be manually bypassed through the CRFS charcoal filters for cleanup prior to supplying air into the conditioned space. The smoke detection system is overridden by the complete recirculation mode of operation to prevent malfunctions during post-accident conditions.

System operation is monitored by temperature indication. Process and area radiation monitors are provided in the room supply air duct and control room to indicate and alarm high radiation levels. Operation of the CRFS is monitored by filter bank differential pressure and temperature indication. Fan operation is monitored by motor trip alarms.

In the event of a LOCA or a fuel handling accident, the control room air conditioning system is automatically switched to the isolation/ recirculation mode. Tests show that the unfiltered in-leakage is less than 200 scfm.

A fresh air make-up system will not be used to maintain a positive pressure differential with respect to the external environment or the adjacent internal spaces at any time during the normal or emergency modes of operation.

The control room air conditioning system mode of operation includes an automatic isolation of the system to the complete recirculation mode and automatic initiation of the bypass filtering operation. This automatic switchover to the complete recirculation mode and filtering mode is initiated by the EBFAS or the AEAS.

The post-accident mode of operation is a closed cycle with air intakes and outlets isolated. The control room atmosphere is exhausted from the space, filtered, and cooled as required and returned to the space. Outside air is not introduced into the system unless required for personnel safety.

9.9.11.4 Availability and Reliability

9.9.11.4.1 Special Features

The components of the control room air conditioning system are designed to engineered safety feature requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physical separation of duplicate equipment, as described in Section 6.1.

Each air conditioning subsystem is capable of maintaining a suitable environment within the control room. Each system is designed for the normal control room cooling load which is greater than the cooling requirements under post-accident operation. Each system is completely independent, including the control and filtration systems with the exception of some common ductwork and dampers. Common components such as dampers are isolated during post-accident operation. Control inputs to these devices are overridden. Each subsystem is powered by a separate emergency source (Section 8.3). A failure mode analysis for the control room air conditioning system is given in Table 9.9-17. Although there are common plenums, all ductwork is considered a passive component not subject to a single failure mode.

The charcoal filter elements within the CRFS are analyzed to ensure adequate residual heat-removal capabilities following any single failure. The analysis concludes that the maximum temperature calculated, based on a radioactive filter inventory which was conservatively assumed to be ten (10) times greater than the maximum inventory calculated resulting from a design basis accident at the site, was less than 212°F (100°C). This is substantially below the charcoal ignition temperature, thus filter bed isolation should not constitute a fire hazard. Temperature indication is provided to alert personnel of excessive charcoal bed temperature.

All control room air conditioning system fans and filters are remote from the control area and are not exposed to fire hazards. The atmosphere within the control room is maintained as constant as possible during the post-accident recirculation mode.

Isolation dampers are provided at each fan discharge to prevent short circuiting of air through idle units.

Snow screens are located on the roof around the Control Room Make-up Inlet Housing, the Control Room Condenser Fan Housings F-36A and F-36B, and the Control Room Exhaust Housing.

The shielding design limits dose rates in the control room to less than 0.5 mrem/hr and does not exceed 5 rem for the duration of the DBA.

All components of the control room air conditioning system located outside the conditioned space are designed to operate in an environment of 110°F atmospheric pressure and 100 percent relative humidity. Typically, safety related components within the conditioned space are designed to operate in an environment of 104°F (40°C) and 10 to 95 percent relative humidity.

9.9.11.4.2 Tests and Inspection

The centrifugal and vaneaxial fans are similar to fans which are rated in accordance with AMCA Standard 211A. The condenser and compressor are rated in accordance with Air Conditioning and Refrigeration Institute (ARI) Standard 410-64 and 520-68, respectively. The evaporator coils are rated in accordance with American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) Standard 33-64. Refrigeration equipment is manufactured in accordance with ANSI B9.1, Safety Code for Mechanical Refrigeration. Refrigeration piping is designed, fabricated and tested in accordance with ANSI B31, 5-1966, Refrigeration Piping Systems. Filters are described in Section 6.7.4.2.

The ductwork is designed and fabricated in accordance with SMACNA Standard, “Low Velocity Duct Construction Standards.”

The control room air conditioning system is incorporated with provisions for online testing. Each subsystem is tested independently for operation of associated components. The subsystem in operation is tested by manually switching to the complete recirculation mode. System alignment and valve positions are monitored in the control room by position indication while system performance is monitored by temperature indication. The associated filtration system is manually initiated for alignment with the control room by-passed air and for fresh outside air. System alignment and valve position are monitored in the control room by position indication.

The redundant air conditioning subsystems are operated alternately to provide assurance of operability.

The control room air conditioning system undergoes a preoperational test prior to startup. The procedure is described in Chapter 13.

All components of the control room air conditioning system are accessible for periodic inspection and maintenance.

9.9.12 DIESEL GENERATOR VENTILATION SYSTEMS

9.9.12.1 Design Bases

9.9.12.1.1 Functional Requirements

The normal diesel generator room ventilation system (Fan F-27) functions to maintain a suitable environment for equipment and plant operating personnel during normal operation and shutdown conditions. The emergency diesel generator ventilation (DGV) system (Fan F-38A/F-38B) functions to maintain a suitable environment for equipment during emergency conditions.

9.9.12.1.2 Design Criteria

The following criteria have been used in the design of the diesel generator room ventilation system:

a. The system shall be designed to permit periodic inspection of important components, such as fan, motor, belt, coil, and ductwork, to assure the integrity and capability of the system.

b. The components of the system shall be designed to operate in the environment to which exposed.

The following criteria have been used in the design of the emergency DGV system:

a. The system shall have an independent ventilation subsystem for each Emergency Diesel Generator (EDG) room (2), having 100 percent of the total required heat removal capability.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as fans and ductwork to assure the integrity and capability of the system.

f. The DGV system shall be designed to permit appropriate periodic pressure and functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, performance shall be demonstrated of

the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system and transfer between normal and emergency power sources.

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The components of the DGV system shall be designed to operate in the most severe post-accident environment to which it is exposed.

9.9.12.2 System Description

9.9.12.2.1 System

The normal non-safety diesel generator room ventilation system (F-27), and the QA-safety DGV systems (F-38A/B) are shown schematically in Figure 9.9–4.

The normal non-safety diesel generator room ventilation system (F-27), which serves both diesel generator rooms, consists of a heating and ventilation supply air unit and louvered exhaust.

The diesel generator rooms are served, during normal operation, by the normal non-safety diesel generator room ventilation system (F-27), which supplies tempered air to these areas. During winter conditions approximately 80 percent of the air flow is recirculated to minimize heating requirements. These rooms are vented to atmosphere through exhaust louvers.

During emergency operation, DGV fan (F-38A/B) consisting of a direct drive, in-line, vaneaxial fan is provided for each diesel generator room to provide a suitable operating environment for the diesels during required conditions. The DGV system is provided with a modulating recirculation damper 2-HV-257A/B which is controlled by a room thermostat to maintain a suitable environment for personnel and equipment during winter conditions. Temperature alarms are provided to indicate low and high room temperature conditions.

9.9.12.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-12. Fan performance curves are shown in Figures 9.9–23 and 9.9–24.

9.9.12.3 System Operation

The normal non-safety diesel generator room ventilation system (F-27) operates during all modes of unit operation and during shutdown. The QA Category 1 DGV systems (F-38A/B) operate during emergency operations.

9.9.12.3.1 Normal Operation

The normal non-safety diesel generator room ventilation system (F-27) is manually initiated by operator action from a locally mounted control panel.

The system is designed to maintain a minimum room temperature of 55°F during winter conditions coincident with 0°F outside ambient air temperature, and to maintain room temperature below 120°F during the summer conditions coincident with 86°F outside ambient air temperature. A temperature sensing element is provided in each diesel generator room to alarm high and low room temperature conditions.

9.9.12.3.2 Emergency Operation

When required, the emergency DGV fans (F-38A/B) are automatically started, and the exhaust dampers (2-HV-255A/B) are opened by the same logic that initiates the diesel generators (Section 8.3).

The DGV system is designed to maintain the room temperature below 120°F during emergency operation by providing outside cooling air. Summer conditions require 100 percent outside air. During winter conditions, a modulating recirculation damper (2-HV-257A/B), controlled by a room thermostat, limits the outside air intake to maintain a suitable environment for personnel and equipment.

The DGV fans (F-38A/B) are interlocked with the diesel generator such that the fans operate at any time the diesel generator is operating. The modulating circulation dampers (2-HV-257A/B), are pneumatically controlled and designed to fail in the closed position. The exhaust damper (2-HV-255A/B) is pneumatically controlled and designed to fail in the open position.

The inlet and exhaust ductwork are missile protected.

The diesel generator room ventilation system F-27 (Seismic Class 2) is connected to the inlet of the DGV fan F-38B. This connection to the DGV inlet (F-38B) is Seismic Class 1 up through the first isolation damper 2-HV-253B on the nonvital fan F-27. Some recirculation flow is assumed for F-38B via this path. Fan F-27 and its isolation damper 2-HV-253B have no automatic controls, and are manually controlled by their individual local switches.

The DGV systems F-38A/B have an inherent short circuiting air flow due to leakage through the recirculation damper 2-HV-257A/B and due to the exhaust louvers proximity to the intake louvers.

Since Train “B” DGV system has interaction with the non-QA ventilation system, an analysis conservatively evaluated Train “B” for recirculation air flow. Results show that the amount of recirculation due to damper 2-HV-253B remaining open, the non-vital ductwork being ruptured, and the proximity of the intake louvers to the exhaust louvers may reach 25 percent of design air flow without exceeding the EDG room temperature limits of 120°F. This analysis also conservatively bounds Train “A.”

9.9.12.4 Availability and Reliability

9.9.12.4.1 Special Features

The components of the DGV system are designed to engineered safety feature requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physical separation of duplicate equipment as described in Section 6.1.

The availability of the DGV system is assured by locating duplicate equipment in separate diesel generator rooms. Each DGV fan is powered by the associated diesel generator located in the respective room. The DGV system is designed with a specific component failure mode to assure the availability of air cooling during the worst case. A failure mode analysis for the DGV system is given in Table 9.9-18.

Operation of the normal diesel generator room ventilation system (Fan F-27) is not vital for plant operation since the diesels are not normally operating. However, low/high temperature alarms are provided to alert plant personnel to take remedial action. Room temperature limitations of 55°F minimum and 120°F maximum are imposed to assure a suitable environment for the diesel generators.

The components of the normal diesel generator room ventilation system (Fan F-27) are specified to operate in an environment of 110°F and 100 percent relative humidity. The components of the emergency DGV system (Fan F-38A/F38B) are designed to operate in an environment of 120°F and 60 percent relative humidity. The DGV fan motor is designed to operate in an environment of 120°F and 100 percent relative humidity.

9.9.12.4.2 Tests and Inspection

The centrifugal and vaneaxial fans are similar to fans which are rated in accordance with AMCA Standard 211-A. One of the DGV fans is rated in accordance with the above. The electric heating coils are built in accordance with the National Electric Code and Underwriters Laboratory requirements. The ductwork is designed and fabricated in accordance with SMACNA Standard, “Low Velocity Duct Construction Standards.”

The DGV system is incorporated with provisions for online testing. The subsystem is tested coincident with the respective diesel generator (Section 8.4.1) but separately from the redundant subsystem. The subsystem is automatically initiated by starting the respective diesel generator manually. Subsystem operation and alignment are indicated by visual inspection.

The DGV system undergoes a preoperation test prior to startup and the diesel generator room system is acceptance tested. Test procedures are described in Chapter 13.

The DGV systems are accessible for periodic maintenance and inspections.

9.9.13 TURBINE BUILDING VENTILATION SYSTEM

9.9.13.1 Design Bases

9.9.13.1.1 Functional Requirements

The turbine building ventilation system functions to maintain a suitable environment for equipment and personnel within the turbine building.

9.9.13.1.2 Design Criteria

The following criteria have been used in the design of the turbine building ventilation system:

a. The system shall be designed to permit periodic inspection of important components, such as fans and ductwork to assure the integrity and capability of the system.

b. The components of the turbine building ventilation system shall be designed to operate in the environment to which exposed.

9.9.13.2 System Description

9.9.13.2.1 System

The turbine building ventilation system is shown schematically in Figure 9.9–4. The turbine building ventilation system is designed to maintain the turbine building environment between 55 and 110°F. The turbine building is provided with a forced air ventilation system using outside air for the removal of building heat. Recirculation of building air through the ventilation system is provided for cold weather heating. Unit heaters are provided throughout the building to provide heating for shutdown during cold weather.

The turbine building ventilation system consists of nine supply air fans; F-101A through -101G, supplemental supply fan F-102, and heater drain pumps' motors cooling supply fan F-143, 10 roof exhaust fans F-111A through -111J and lube oil room and access control area exhaust fan F-124. Transfer grilles are provided between the operating floor and lower elevations to enhance air distribution.

9.9.13.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-13. Fan performance curves are shown in Figures 9.9–25 and 9.9–27.

9.9.13.3 System Operation

9.9.13.3.1 Normal Operation

The turbine building ventilation system is manually started by operator action at locally mounted control panels. Turbine building supply fans F-101A through -101G, provide 100 percent outside air during periods of warm weather to limit the turbine building ambient temperature to 110°F. Supplemental supply fan F-102 is operated to supply additional outside air to the steam generator feed pump's and southeast turbine bay area, and fan F-143 is operated to provide additional outside air to the heater drain pumps' motors.

Air is exhausted from the turbine bay by roof exhaust fans F-111A through F-111J and from the lube oil room and access control area by fan F-124.

During cold weather operation, recirculation is provided by turbine building supply fans F-101A through F-101G. Each of the seven fans can supply approximately 20 percent outside air and 80 percent return air to maintain a minimum temperature of 55°F for personnel comfort without supplemental heating. Unit heaters supplied by auxiliary steam can be operated to maintain 55°F when the unit is shutdown during periods of cold weather. Supply and exhaust fans are operated as necessary.

9.9.13.4 Availability and Reliability

9.9.13.4.1 Special Features

There are no requirements to maintain the turbine building at any specific pressure under any modes of operation.

All ductwork penetrations through fire walls are provided with fire dampers. The turbine building is provided with vent area through the ventilation equipment at 1 square foot of vent area per 100 square feet of floor area. This vent area is provided as a means of smoke removal since the ventilation equipment may be out of service.

The components of the turbine building ventilation system are designed to operate in an environment of 110°F and atmospheric pressure.

9.9.13.4.2 Tests and Inspections

The vaneaxial fans are similar to fans which have been rated in accordance with AMCA Standard 211-A. All fans are provided with steel wheels and housings. The ductwork is designed and fabricated in accordance with SMACNA Standards, "Low Velocity Duct Construction Standards."

The turbine building ventilation system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13.

The turbine building supply and exhaust fans are accessible for periodic tests and inspections.

9.9.14 ACCESS CONTROL AREA AIR CONDITIONING SYSTEM*

*NOTE: The area is no longer an access control area. It was converted to office space consisting of offices, lunch and locker rooms and hallways.

9.9.14.1 Design Bases

9.9.14.1.1 Functional Requirements

The air conditioning system functions to maintain a suitable environment for personnel comfort within the area.

9.9.14.1.2 Design Criteria

The following criteria have been used in the design of the area air conditioning system:

a. The system shall be designed to permit periodic inspection of important components, such as fans and ductwork to assure the integrity and capability of the system.

b. The components of the air conditioning system shall be designed to operate in the environment to which exposed.

9.9.14.2 System Description

9.9.14.2.1 System

The air conditioning system is shown schematically in Figure 9.9–4.

The access control area air conditioning system is provided for comfort cooling and heating of the access control area, the instrument shop and selected areas in Unit 1 such as the lunch room and office spaces. The HVAC system is designed to maintain an environment of 78°F and 50 percent relative humidity in the conditioned space.

The system consists of self-contained air conditioning unit which houses the supply fan, chilled water coil, and filters. Chilled water is provided by the Auxiliary Chilled Water System. Steam distributing coils are provided for heating during winter operation.

The air handling unit supply fan delivers ventilation air to the various rooms which include office space, locker and lunch rooms, and the hallways. Air is distributed and recirculated through ductwork. Fresh air is mixed with return air. Excess air is exhausted either by the main exhaust fans (F-34A/B/C) and discharged through Unit 2 stack or by exhaust fan F-124 which discharges directly to the outside.

9.9.14.2.2 Components

The major system components and associated fabrication and performance data are listed in Table 9.9-14. Fan performance curve is shown in Figure 9.9–28.

9.9.14.3 System Operation

The access control area air conditioning system operates during unit operation and during shutdown.

9.9.14.3.1 Normal Operation

The air conditioning system is manually initiated by operator action at a locally mounted control panel. The system operation is automatically controlled by the temperature control system.

Minimum outside air is provided through penthouse louvers located on the turbine building. The conditioned air is recirculated through the unit to conserve refrigeration and heating requirements.

Air flow which is equivalent to the outside air makeup is exhausted without recirculation.

During the winter heating cycle, steam heat is available by the steam distributing coil.

9.9.14.4 Availability and Reliability

9.9.14.4.1 Special Features

The air conditioning system is designed to provide personnel comfort. Loss of the system will not require shutdown. Fire dampers are provided at all penetrations through fire walls.

The components of the air conditioning system are designed to operate in an environment of 110ºF, atmospheric pressure and 100 percent relative humidity.

9.9.14.4.2 Tests and Inspection

The centrifugal fan is similar to fans which are rated in accordance with AMCA Standard 211A. The condenser and compressor are rated in accordance with ARI Standard 410-64 and 520-68, respectively. The evaporator coils are rated in accordance with ASHRAE Standard 33.64. Refrigeration equipment is manufactured in accordance with ANSI B9.1, “Safety Code for Mechanical Refrigeration.” Refrigeration piping is designed, fabricated and tested in accordance with ANSI B31.5-1966, “Refrigeration Piping Systems.”

The ductwork is designed and fabricated in accordance with SMACNA Standard, “Low Velocity Duct Construction Standards.”

The air conditioning system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13.

All components of the access control area air conditioning system are accessible for periodic inspection and maintenance.

9.9.15 BALANCE OF UNIT

9.9.15.1 Design Bases

9.9.15.1.1 Functional Requirements

Ventilation systems for the balance of the unit function to provide a suitable environment for equipment and ventilation for operating personnel.

9.9.15.1.2 Design Criteria

The following criteria have been used in the design of the systems:

a. The system shall be designed to permit periodic inspection of important components, such as fans and ductwork to assure the integrity and capability of the system.

b. The components of the systems shall be designed to operate in the environment to which exposed.

9.9.15.2 System Description

9.9.15.2.1 System

Non safety-related Heating/Ventilation & Air Conditioning (HVAC) systems are provided for the intake structure and other general areas of the plant (i.e., radiation protection facility, telephone rooms, computer rooms, etc.) to maintain a year round environment suitable for personnel and equipment. However, the only areas containing safety related equipment are the main equipment room of the intake structure, which houses the safety related service water system, as well as other non safety related equipment such as the circulating water pumps, traveling screens and the Old Computer Room which houses the safety related Inadequate Core Cooling Monitoring System (ICCMS) as well as other non safety-related equipment including CISCO Network Switches that relay data from the Station Transformers and RCP vibration monitors to the PPC.

During normal operation, the temperature of the intake structure main equipment room is a minimum of 45°F, and a maximum of 104°F, with a design basis minimum temperature of 40°F. Upon loss of ventilation exhaust fans F-114A/B/C, the maximum nominal design temperature of this area of the intake structure is 130°F. In the event of such a loss of forced ventilation, two passive nonmechanical roof ventilators X-219A and X-219C maintain the main equipment room at or below 130°F through natural circulation. The non safety related equipment located within the main equipment room of the intake structure is designed for 104°F while the safety related equipment is qualified for a nominal temperature of 130°F.

During normal operation, the temperature of the Old Computer Room is nominally maintained at 70°F. The majority of room heat gain is from interior walls and the floor separating adjacent plant spaces that are at higher temperatures given normal plant operating conditions. A comparatively small amount of heat gain is due to equipment within the room. As a result, no heating is required in colder weather due to heat loss through exterior walls and the ceiling/roof. Upon loss of the air conditioning system (X199/198), under worst case accident conditions, the room temperature will rise to 117.4°F. This results in internal ICCMS cabinet temperatures less than their maximum allowable temperature of 122°F.

9.9.15.2.2 Components

The major system components and associated fabrication and performance data for the main equipment room of the intake structure and Old Computer Room are listed in Table 9.9-15.

9.9.15.3 System Operation

These non-safety related HVAC systems maintain a year round environment suitable for personnel and equipment during normal and shutdown plant operations. Since the main equipment room of the intake structure and the Old Computer Room of the Auxiliary Building are the only areas containing safety related equipment, the other plant areas will not be discussed in the subsequent subsections.

9.9.15.3.1 Normal Operation

In the summer mode, the ventilation/cooling is provided to the main equipment room by roof exhaust fans F-114A/B/C which draw outside air through the four 8 foot by 10 foot louvered openings in the west wall of the intake structure, past the equipment being cooled, then out through the exhaust fan discharge. In the winter mode, the wall louvers are closed, thermostatically controlled electric heaters operate as needed during times of colder outdoor temperatures, and the exhaust fans are manually cycled on/off as necessary to maintain main equipment room temperatures in the range of 45°F to 104°F.

The exhaust fans F-114A/B/C are manually started by operator action at locally mounted control switches.

The Old Computer Room is cooled by a split air conditioning system that is automatically controlled by a thermostat nominally set to 70°F. Air is circulated within the room by Air Handler X-198 which distributes air to four ceiling outlets. Air conditioning is required all year round primarily due to the heat gain through interior walls and the floor.

9.9.15.4 Availability and Reliability

9.9.15.4.1 Special Features

A loss of the intake structure ventilation/cooling system, which is of concern only in the summer mode of operation, will not require a plant shutdown, since passive roof ventilators X-219A and

X-219C, in conjunction with the opened wall louvers, provides natural circulation which maintains the space at or below 130°F. This natural circulation results from a draft being created between the louvers in the west wall of the intake structure, past the service water pumps, and up through the roof ventilators. All safety related components in the main equipment area of the intake structure are qualified for operation in a 130°F environment.

A loss of heating in the intake structure main equipment room, either through reduced heat load (during shutdown conditions) or loss of the space heaters, which is of concern only in the winter mode of operation, could cause freezing conditions that could affect the operability of the service water strainer backwash control system. Therefore, safety related temperature controllers are installed, which automatically initiate the service water strainer backwash cycle if the main equipment room of the intake structure falls to 40°F or below.

A loss of air conditioning in the Old Computer Room will not result in exceeding the qualified operating temperature of 122°F for all safety related components in this area.

9.9.15.4.2 Tests and Inspection

The roof exhaust fans (F-114A/B/C) are similar to fans which have been rated in accordance with AMCA Standards.

The Intake Structure systems underwent acceptance tests prior to startup. Test procedures are described in Chapter 13.

The components of the systems are accessible for maintenance and inspection.

9.9.16 VITAL SWITCHGEAR VENTILATION SYSTEM

9.9.16.1 Design Bases

9.9.16.1.1 Functional Requirements

The vital switchgear ventilation system function to maintain a suitable environment for the safety related electrical equipment during normal operation, loss of off site power and post-accident conditions.

9.9.16.1.2 Design Criteria

The following criteria have been used in the design of the vital switchgear ventilation system:

a. Each redundant switchgear group shall have an independent subsystem having 100 percent of the total required heat removal capability.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components such as fans, coils, ductwork, piping and valves to assure the integrity and capability of the system.

f. The system shall be designed to permit appropriate periodic functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, the performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The components of the system shall be designed to operate in the most severe post-accident environment as described in Section 6.1.

9.9.16.2 System Description

9.9.16.2.1 System

The vital switchgear ventilation system is shown on Figure 9.9–3, 9.9–4 and 9.9–31.

The vital switchgear ventilation system consists of independent subsystems, capable of removing 100% of the heat generated in the associated vital electrical equipment room. The vital AC switchgear rooms containing the 4160 volt and 6900 volt electrical equipment and the west 480 volt unit load center are each cooled by separate closed cycle air subsystems sized for 100% of the room cooling requirements under normal plant operation and emergency conditions such as Loss of Coolant Accident “LOCA” and High Energy Line Break “HELB”. Therefore, since each subsystem is normally operating, and the normal cooling requirements exceed the cooling requirements for emergency operation, no system realignment is necessary for emergency conditions.

Each of the above subsystems consist of fan-coil units utilizing service water as the ultimate heat sink. The original room temperature was limited to 40°C (104°F) for the vital AC switchgear rooms. However, engineering analysis responsive to NRC Generic Letter 89-13 “Service Water System Problems Affecting Safety-Related Equipment, July 18, 1992,” identified the need to raise the room temperature limitation for the upper and lower 4160/6900 volt switchgear rooms. As a result, the upper and lower 4160/6900 volt rooms have been qualified for a room temperature limitation of 50°C (122°F).

The subsystem for the upper switchgear room is contained within the upper switchgear room. The subsystem for the lower switchgear room is located in an adjacent cable vault. This affords protection for both subsystems in the event of a high energy pipe rupture. Service water supply is provided with isolation valves located outside the rooms. Water detectors within the rooms' coffer dam will cause the corresponding isolation valve to close automatically to prevent flooding.

The east 480 volt unit load center is located in the auxiliary building and is cooled during normal and emergency operations by 100% outside air supply systems. The room temperature for both the east and west 480 volt unit load center rooms is limited to 40°C (104°F).

The east and west vital DC switchgear rooms are provided with closed cycle air subsystems utilizing mechanical refrigeration to maintain the ambient conditions within these areas. Cooling is normally provided by nonvital chilled water. The room temperature is limited to 40°C (104°F).

The MCC B51 and B61 enclosures are provided with self-contained air conditioning units. (Vital MCC Coolers Number A/C-3 (B51) and A/C-4 (B61)) which maintain the ambient temperature inside the enclosures below 122°F during normal plant operation and emergency conditions such as Loss of Coolant Accident "LOCA" and High Energy Line Break "HELB." No system realignment is necessary during emergency conditions.

9.9.16.2.2 Components

The major system components and associated performance data are listed in Table 9.9-19.

9.9.16.3 System Operation

9.9.16.3.1 Normal Operation

The subsystems serving the two 4160 volt and 6900 volt rooms and the east and west 480 volt unit loadcenters operate during normal operation and are actuated by room thermostats. Normal cooling for the east 480 loadcenter is provided by the nonradioactive ventilation system. (See Section 9.9.6.) Air is supplied and exhausted through the roof louvers provided for the nonradioactive ventilation system and the vital switchgear east 480V room ventilation system.

The vital DC switchgear ventilation system operates during normal unit operation. Chilled water is aligned for normal cooling from the auxiliary chilled water system (see Section 9.9.16). Valving and components are initiated by remote manual operation.

The Vital Coolers A/C-3 (MCC B51) and A/C-4 (MCC B61) operate during normal plant operation and are supplied with electrical power from the same MCC it serves. Each cooler is provided with an integral temperature controller which cycles the unit, as required, to maintain the temperature inside the enclosure below 122°F.

9.9.16.3.2 Emergency Operation

The vital AC switchgear room ventilation system will operate during post accident or loss of off site power conditions, depending on the cooling requirements as determined by the corresponding room thermostats.

The vital DC switchgear ventilation system, if not already running, is automatically initiated by the SIAS or by a loss of AC power. The vital chilled water system automatically assumes the emergency alignment upon SIAS or loss of normal power. To maintain room cooling, the chilled water supply valve to each room's heat exchanger is secured open and the fans run continuously.

The Vital Coolers A/C-3 (MCC B51) and A/C-4 (MCC B61) operate during emergency conditions such as Loss-of-Coolant Accident "LOCA" and High Energy Line Break "HELB" to limit the temperature inside the enclosures below 122°F. Following a loss of normal power the Vital Coolers, which are powered by the MCC it serves, are automatically supplied with safety related power via the emergency diesel generators.

9.9.16.4 Availability and Reliability

9.9.16.4.1 Special Features

The components of the vital switchgear ventilation system are designed to engineered safety feature requirements including seismic response, as described in Section 6.1.

The availability of the system is assured by providing duplicate, full-capacity equipment, each subsystem physically separated from its counterpart. The system is powered by separate, emergency buses.

The equipment in the 6900 volt and 4160 volt rooms is designed to operate in an environment of 122°F and 100% relative humidity.

9.9.16.4.2 Tests and Inspections

The centrifugal and vaneaxial fans are rated in accordance with AMCA Standard 211A. The ductwork is designed and fabricated in accordance with SMACNA Standard, "High Velocity Duct Construction Standards" or "Low Velocity Duct Construction Standards," as appropriate.

The vital switchgear ventilation system is incorporated with provisions for online testing. Each subsystem is tested independently for operation of associated components. The subsystem in operation is tested by manually switching to the emergency mode. System alignment and valve positions are monitored on the local control panel by position indication while system performance is monitored by temperature indication.

The vital switchgear ventilation system undergoes a preoperational test prior to startup. The procedure is described in Chapter 13.

All components of the vital switchgear ventilation system are accessible for periodic inspection and maintenance.

9.9.17 AUXILIARY CHILLED WATER SYSTEM

9.9.17.1 Design Bases

9.9.17.1.1 Functional Requirements

The auxiliary chilled water system functions to provide a source of chilled water to maintain a suitable environment for portions of the safety related electrical equipment during normal plant operations and to provide a source of chilled water for other non-safety related plant cooling requirements.

9.9.17.1.2 Design Criteria

The auxiliary chilled water system has been designed to the following criteria:

a. Two identical mechanical refrigeration units provide a source of chilled water during normal plant operation.

b. The auxiliary chilled water system isolates automatically from the safety related chilled water system upon receipt of a SIAS.

c. Connections to the safety related chilled water system have been designed to Seismic I requirements.

9.9.17.2 System Description

9.9.17.2.1 System

The auxiliary chilled water system is shown on Figure 9.9–31. The auxiliary chilled water system consists of two identical mechanical refrigeration units which can be operated either singularly or in parallel to provide a source of chilled water during normal plant operation. Cooling of the mechanical refrigeration units is provided by the Turbine Building Closed Cooling Water (TBCCW).

9.9.17.2.2 Components

The major system components and associated performance data are listed in Table 9.9-20.

9.9.17.3 System Operation

9.9.17.3.1 Normal Operation

The auxiliary chilled water system (see Figure 9.9–31) operates during normal operation with the access control area temperature being controlled by a temperature controller.

9.9.17.3.2 Emergency Operation

Auxiliary Chilled Water System is automatically isolated following a SIAS, loss of off site power, or loss of control air.

9.9.17.4 Availability and Reliability

9.9.17.4.1 Special Features

The availability of the system is assured by providing duplicate equipment to provide the necessary operating flexibility. The mechanical refrigeration units are liberally sized so that one unit can handle the expected heat loads. As additional need for chilled water develops due to plant design changes both units may be required to operate.

9.9.17.4.2 Tests and Inspections

The mechanical refrigeration units are rated in accordance with the ARI Standard 590-62.

All components of the auxiliary chilled water system are accessible for periodic inspection and maintenance.

9.9.18 VITAL CHILLED WATER SYSTEM

9.9.18.1 Design Basis

9.9.18.1.1 Functional Requirements

The Vital Chilled Water System functions to provide a source of chilled water to maintain a suitable environment for portions of the safety related electrical equipment during loss of off site power and post-accident conditions.

9.9.18.1.2 Design Criteria

The following criteria have been used in the design of the Vital Chilled Water System:

a. Each redundant vital switchgear group shall have an independent vital chilled water subsystem having 100 percent of the loss of off site power or post-accident heat removal capability.

b. The systems shall have a suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components such as cylinder heads, tubing, piping, and valves to assure the integrity and capability of the system.

f. The system shall be designed to permit appropriate periodic functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, the performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

g. The system shall be designed to the general criteria as described in Section 6.1.

h. The components of the system are designed to operate in a mild environment. Harsh environmental conditions may result from a turbine building fire or a high-energy line break event. However, operability of chillers X-169A/B is not required to mitigate these events.

9.9.18.2 System Description

9.9.18.2.1 System

The Vital Chilled Water System is shown on Figure 9.9–31. The Vital Chilled Water System consists of two chiller subsystems, both of which are designed to engineered safety features (ESF) requirements to provide a source of chilled water to the Vital Switchgear Ventilation System during loss of off site power and post-accident conditions.

9.9.18.2.2 Components

The major system components and associated performance data are listed in Table 9.9-21.

9.9.18.3 System Operation

9.9.18.3.1 Normal Operation

Two Vital Chilled Water subsystems, X-169A/B are in standby mode during normal operation. Vital Switchgear Ventilation chilled water is supplied by the Auxiliary Chilled Water System, X-196A/B during normal operation.

9.9.18.3.2 Emergency Operation

Two Vital Chilled Water subsystems are automatically initiated by a SIAS or by a loss of normal AC power to provide a source of chilled water to the Vital Switchgear Ventilation System.

9.9.18.4 Availability and Features

9.9.18.4.1 Special Features

The availability of the system is assured by providing duplicate equipment. The two independent mechanical refrigeration units are sized such that each unit can handle its entire loss of off site power or post-accident condition load. The refrigeration units are powered by separate emergency buses.

The Vital Chilled Water System is designed with specific failure modes to assure emergency system alignment.

9.9.18.4.2 Tests and Inspections

The mechanical refrigeration units are rated in accordance with the ARI Standard 590.

All components of the Vital Chilled Water System are accessible for periodic inspection and maintenance.

The condenser and evaporator sections of the vital chillers are stamped in accordance with American Society of Mechanical Engineers (ASME) Code Section VIII. Refrigeration equipment is manufactured in accordance with American National Standards Institute (ANSI) B 9.1, Safety Code for Mechanical Refrigeration. Refrigeration piping is designed, fabricated, and tested in accordance with ANSI B 31.5, Refrigeration Piping Systems.

TABLE 9.9-1 CONTROL ELEMENT DRIVE MECHANISM COOLING SYSTEM

Fan (F-13A/B/C)

Type	Vaneaxial
Flow (cfm)	31,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	25
Code	NEMA MG-1
Seismic Class	II

Coil (X-34A/B/C)

Type	Water (RBCCW)
Heat removal capability, each (Btu/hr)	8.10×10^5 (nominal rating)
Air temperature entering/leaving (°F)	120/95 (nominal rating)
Water temperature entering/leaving (°F)	85/95 (nominal rating)
Water flow rate (gpm)	165 (nominal rating)
Code	ASME Section VIII TEMA, Class C
Seismic Class	II

Ductwork

Material/Type	IAW Specification 526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-2 CONTAINMENT AND ENCLOSURE BUILDING PURGE SYSTEM

Fan (F-23)

Type	Centrifugal
Flow (cfm)	32,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating	15
Code	NEMA MG-1
Seismic Class	II

Coil (X-43)

Type	Steam distributing
Heat transfer (Btu/hr)	2.42×10^6 (nominal rating)
Air temperature entering/leaving (°F)	0/70 (nominal rating)
Code	ASME Section VIII TEMA, Class C
Seismic Class	II

Particulate Filters

Quantity	Media (field cut)
Type	Throwaway

Ductwork

Material/Type	IAW Specification M-52
Standard	SMACNA
Seismic Class	II

TABLE 9.9-2 CONTAINMENT AND ENCLOSURE BUILDING PURGE SYSTEM (CONTINUED)

Piping

Material	ASTM A-333 Grade 6
Wall (inches)	0.375
Fittings	Butt-welded except at flanged equipment
Flanges	Carbon steel, ANSI 150 lb rating
Code	ANSI B31.7, Class II
Seismic Class	I

Valves (2AC-4/5/6/7)

Type	Wafer-type butterfly
Class	150 lb
Line size (inches)	48
Code	ASME Section III, Class 2, 1971
Seismic Class	I

TABLE 9.9-3 CONTAINMENT AUXILIARY CIRCULATION SYSTEM

Fan (F-24A/B)

Type	Vaneaxial
Flow (cfm)	75,000/37,500 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction (2 speed)
Horsepower rating (hp)	60/30
Code	NEMA MG-1
Seismic Class	II

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-4 CONTAINMENT PENETRATION COOLING SYSTEM

Fan (F-37A/B)

Type	Vaneaxial
Capacity (cfm)	4,320 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	5
Code	NEMA MG-1
Seismic Class	II

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-5 PENETRATION AIR COOLING REQUIREMENTS

Penetration Number	System	Air Required (cfm)
19	Main steam	1000
20	Main steam	1000
15	Main feedwater	360
16	Main feedwater	360
22	Steam generator blowdown	300
23	Steam generator blowdown	300
10	Shutdown cooling	300
2	Letdown	100
65	Steam generator blowdown sample	200
72	Steam generator blowdown sample	200
21	Reactor containment and pressurizer sample	200 4320

TABLE 9.9-6 RADWASTE VENTILATION SYSTEM

Fan (F-16)

Type	Centrifugal
Flow (cfm)	40,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	40
Code	NEMA MG-1
Seismic Class	II

Coil (X-37)

Type	Steam distributing
Heat transfer (Btu/hr)	2.72×10^6 (nominal rating)
Air temperature entering/leaving (F)	0/60
Code	ASME Section VIII, TEMA Class C
Seismic Class	II

Filters

Quantity per unit	Media (field cut)
Type	Throwaway

Ductwork

Material/Type	IAW Specification M-526
Seismic Class	II

TABLE 9.9-6 RADWASTE VENTILATION SYSTEM (CONTINUED)

Booster Coils (X-56 / 57 / 58)

	X-56	X-57	X-58
Type	Steam distributing		
Air entering temperature (F)	55	55	55
Code	ASME Section VIII, TEMA Class C		
Air leaving temperature (F)	70	80	70 (nominal rating)
Air flow (cfm)	550	3510	645 (nominal rating)
Seismic Class	II		

TABLE 9.9-7 NONRADIOACTIVE VENTILATION SYSTEM

Fan F-17

Type	Centrifugal
Flow (cfm)	25,750 (nominal rating)
Code	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	20
Code	NEMA MG-1
Seismic Class	II

Particulate Filters

Quantity	Media (field cut)
Type	Throwaway

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA

East 480 Loadcenter Room Normal Exhaust Fan F-26

Type	Vaneaxial
Flow (cfm)	17,500 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	10
Code	NEMA MG-1
Seismic Class	II

TABLE 9.9-7 NONRADIOACTIVE VENTILATION SYSTEM (CONTINUED)

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

Battery Room Roof Exhaust Fans F112A/F112B

Type	Roof exhaust
Flow (cfm)	4125 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	3
Code	NEMA
Seismic Class	II

Cable Vault Transfer Fans F-123

Type	Propeller
Flow (cfm)	1,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	1/8
Code	NEMA MG-1
Seismic Class	II

Ductwork

None	(Fan mounted in wall only)

TABLE 9.9-7 NONRADIOACTIVE VENTILATION SYSTEM (CONTINUED)

Cable Vault Air-Recirculation Fan (F-19)

Type	Centrifugal
Flow (cfm)	16,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	15
Code	NEMA
Seismic Class	II

Coil (X-40)

Type	Chilled water
Heat transfer (Btu/hr)	154,000 (nominal rating)
Code	ASME Section VIII, TEMA Class C
Seismic class	II

Particulate Filter

Quantity	12
Type	Throwaway

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-8 ENGINEERED SAFETY FEATURES ROOM AIR RECIRCULATION SYSTEM

Fans (F-15A/B)

Type	Vaneaxial
Capacity (cfm)	18,650 (minimum)
Standard	AMCA 211-A
Seismic class	I

Motor

Type	Induction
Horsepower rating (hp)	15
Code	NEMA MG-1
Seismic Class	I

Coils (X-36A/B

Type	Water (RBCCW)
Heat removal capability, each (Btu/hr)	Note 1
Air temperature entering/leaving (°F)	Note 1
Water temperature entering/leaving (°F)	Note 1
Water flow rate (gpm)	60 (minimum)
Code	ASME Section VIII TEMA, Class R
Seismic Class	I

Ductwork

Material/Type	IAW Specification M-506
Standard	SMACNA
Seismic Class	I

Note 1: Calculation 96-ENG-1529M2 provides a summary of performance data for various RBCCW inlet temperatures including a post-accident temperature transient.

TABLE 9.9-9 FUEL HANDLING VENTILATION SYSTEM

Fan (F-20)

Type	Centrifugal
Flow (cfm)	18,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	15
Code	NEMA
Seismic Class	II

Coil (X-41)

Type	Steam distributing
Heat transfer (Btu/hr)	164×10^6 (nominal rating)

Coils

Air temperature entering/leaving (°F)	0/85
Code	ASME Section VIII, TEMA Class C

Particulate Filters

Quantity	12
Type	Throwaway .

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-9 FUEL HANDLING VENTILATION SYSTEM (CONTINUED)

Air Conditioner (F-140)

Type	Self Contained
Capacity	15 Ton
Standard	ARI 520
Seismic Class	II

Sub-components:

Evaporator (X-191)	Direct expansion (Refrigerant R-22)
Condenser	Water cool (TBCCW)
Compressor	Hermetic (Refrigerant R-22)

Motor:

Type	Induction
Horsepower rating (hp)	2
Code	NEMA

Particulate Filters:

Quantity	2
Type	Throwaway

TABLE 9.9-10 MAIN EXHAUST SYSTEM

Fan (F-34 A/B/C)

Type	Centrifugal
Flow (cfm)	32,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower rating (hp)	40
Code	NEMA
Seismic Class	II

Filter Housing (L-25, L-26, L-27)

Housing material standard	ASTM A-415
Frame material standard	ASTM A-36
Seismic Class	II

Particulate Filters	Quantity (per unit)
L-25	32
L-26	44
L-27	20

Type (prefilter)	Throwaway

HEPA Filters	Quantity (per unit)
L-25	32
L-26	44
L-27	20

Type	High efficiency, dry
Rate air flow per filter at 1 in WG	1,000 cfm
Standard	MIL-STD-282

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

TABLE 9.9-11 CONTROL ROOM AIR CONDITIONING SYSTEM

Supply Fan (F-21A/F-21B)

Type	Centrifugal
Flow (cfm)	16,460 (nominal rating)
Standard	AMCA 211-A
Seismic Class	I

Motor

Type	Induction
Horsepower rating (hp)	15
Code	NEMA
Seismic Class	I

Coil - Evaporator (X-42A / X-42B)

Type	Direct expansion
Heat transfer (Btu/hr)	432,000 (nominal rating)
Standard	ASHRAE Standard 33-64
Seismic class	I

Particulate Filters

Quantity (per unit)	10
Type (prefilter)	Throwaway
Quantity (per unit)	10
Type (primary)	High efficiency (bag)

Compressor (F22A / B)

Type	Hermetic
Power input (kW)	44
Standard	ARI 520-68
Seismic Class	I

Condenser

Type	Air cooled
Seismic Class	I

TABLE 9.9-11 CONTROL ROOM AIR CONDITIONING SYSTEM (CONTINUED)

Fan (F-36A / F-36B)

Type	Centrifugal
Flow (cfm)	23,900 (nominal rating)
Standard	AMCA 211-A
Seismic Class	I

Motor

Type	Induction
Horsepower rating (hp)	20
Code	NEMA
Seismic Class	I

Coil - Condenser

Type	Refrigerant condenser (R-22)
Heat transfer (Btu/hr)	588,000
Standard	ARI Standard 410-64
Seismic Class	I

Particulate Filters

Quantity (per unit)	16
Type	Throwaway

Exhaust Fan (F-31A/F-31B)

Type	Vaneaxial
Flow (cfm)	16,460 (nominal rating)
Seismic Class	I
Standard	AMCA 211-A

Motor

Type	Induction
Horsepower rating (hp)	7.5
Code	NEMA
Seismic Class	I

TABLE 9.9-11 CONTROL ROOM AIR CONDITIONING SYSTEM (CONTINUED)

Control Room Filtration Fan (F-32A/F-32B)	
Type	Centrifugal
Flow (cfm)	2500 (±10%)
Standard	AMCA 211-A
Seismic Class	I
Motor	
Type	Induction
Horsepower rating (hp)	5
Code	NEMA
Seismic Class	I
Filtration Unit (L-30A/B)	
Housing Seismic Class	I
Particulate Filters	
Quantity per unit	2
Type	Throwaway
HEPA Filters	
Quantity (per unit)	2
Type	High Capacity
Rate air flow per filter at 1.30 in WG	1500 CFM
Rate air flow per filter at 0.87 in WG	1000 CFM
Standard	MIL-STD-282
Charcoal Filters	
Quantity per unit (trays)	6
Charcoal type	CNN-816 activated coconut shell
Rated flow per unit	2500 CFM
Standard	ANSI N-509
Refrigeration Piping	
Standard	ANSI B9.1, ANSI B31.5 (1966)
Seismic Class	I
Ductwork	
Material/Type	IAW Specification M-506
Standard	SMACNA
Seismic Class	I

TABLE 9.9-12 DIESEL GENERATOR VENTILATION SYSTEMS

Fan (F-27)	
Type	Centrifugal Class
Flow (cfm)	1000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II
Motor	
Horsepower rating (hp)	1.0
Code	NEMA
Seismic Class	II
Coil (X-44)	
Type	Electric
Heat transfer (kW Btu/hr)	18kW / 61,4000 Btu/hr (nominal rating)
Filters	
Quantity	1
Type	Throwaway
Ductwork (Non-Vital)	
Material / Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II
DGV Fan (F-38A / F-38B)	
Type	Vaneaxial
Flow (cfm)	31,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	I
Motor	
Horsepower rating (hp)	20
Code	NEMA
Seismic Class	I
Ductwork (Vital)	
Material/Type	IAW Specification M-506
Standard	SMACNA
Seismic Class	I

TABLE 9.9-13 TURBINE BUILDING VENTILATION SYSTEM

Supply Fan (F-101A through G)	
Type	Vaneaxial
Capacity (cfm)	44,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	20
Code	NEMA
Seismic Class	II
Supply Fan (F-102)	
Type	Vaneaxial
Flow (cfm)	44,400 (nominal rating)
Code	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	25
Code	NEMA
Seismic Class	II
Roof Exhaust Fans (F-111A through J)	
Type	Roof exhaust
Flow (cfm)	35,400 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	5
Code	NEMA
Seismic Class	II

TABLE 9.9-13 TURBINE BUILDING VENTILATION SYSTEM (CONTINUED)

Exhaust Fan (F-124)	
Type	Centrifugal (inline)
Flow (CFM)	1,500 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	1
Code	NEMA
Seismic Class	II
Ductwork	
Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II
Supply Fan (F-143)	
Type	Inline
Flow (cfm)	15,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	5
Code	NEMA
Seismic Class	II
Ductwork	
Material / Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II
TDAFW Exhaust Fan (F-158)	
Type	Roof exhauster
Flow (CFM)	1,000 (nominal rating)
Standard	AMCA 211-A
Seismic Class	II

TABLE 9.9-13 TURBINE BUILDING VENTILATION SYSTEM (CONTINUED)

Motor

Type	Induction
Horsepower rating (hp)	1/4
Code	NEMA
Seismic Class	II

TABLE 9.9-14 ACCESS CONTROL AREA AIR CONDITIONING SYSTEM

NOTE: The area is no longer used as an access control area. It was converted to office space consisting of offices, lunch and locker rooms, and walkways.

Fan (F-116)

Type	Centrifugal
Flow (cfm)	15,550 (nominal rating)
Code	AMCA 211-A
Seismic Class	II

Motor

Type	Induction
Horsepower Rating	20
Code	NEMA-G
Seismic Class	II

Cooling Coil (X-85)

Type	Chilled Water
Heat Transfer (Btu/hr)	420,000 (nominal rating)
Air Temperature Entering (°F)	80.7 DB, 66.6 WB (nominal rating)
Air Temperature Leaving (°F)	60 DB, 57 WB (nominal rating)

Cooling Coil

Code	ASME Section VIII, TEMA Class C
Seismic Class	II

Heating Coil (X-167)

Type	Steam distributing
Heat transfer (Btu/hr)	418,000 (nominal rating)
Code	ASME VIII
Seismic Class	II

TABLE 9.9-14 ACCESS CONTROL AREA AIR CONDITIONING SYSTEM (CONTINUED)

Particulate Filters

Quantity (per unit)	8
Type (prefilter)	Throwaway
Quantity (per unit)	8
Type (primary)	High efficiency (bag)

Ductwork

Material/Type	IAW Specification M-526
Standard	SMACNA 9l
Seismic Class	II

TABLE 9.9-15 BALANCE OF UNIT VENTILATION SYSTEM

Intake Structure Fan (F-114A/B/C)

Type	Roof exhauster
Flow (CFM)	30,000
Standard	AMCA 211-A
Seismic Class	II
Motor	
Type	Induction
Horsepower rating (hp)	7.5
Code	NEMA
Seismic Class	II

Intake Structure Ventilators (X-219A/C)

Type	Passive, Non-Mechanical
Flow Rate	Variable
Code Requirements	ANSI/AWS D1.1 (1997) ANSI/AWS D1.3 (1989)
Seismic Class	II
Ductwork	
Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II

Old Computer Room

Air Conditioner	
Type	Split system
Capacity (Ton)	4 (nominal rating)
Standard	ARI 520
Seismic Class	II
Subcomponents	
Evaporator Unit (X-198)	
Type	Direct expansion (refrigerant R-410A)

TABLE 9.9-15 BALANCE OF UNIT VENTILATION SYSTEM (CONTINUED)

Motor	
Type	Induction
Horsepower rating (hp)	0.75
Code	NEMA
Seismic Class	II
Condenser Unit (X-199)	
Type	Air cooled
Compressor	
Type	Scroll Hermetic
Ductwork	
Material/Type	IAW Specification M-526
Standard	SMACNA
Seismic Class	II
New Computer Room	
Air Conditioner (F-144A/B)	
Type	Split system
Capacity (Btu/hr)	218,000 (nominal rating)
Standard	ARI 520
Seismic Class	II
Subcomponents	
Evaporator Unit (X-200/X-201)	
Type	Direct expansion (refrigerant R-22)
Motor	
Type	Induction
Horsepower rating (hp)	7.5
Code	NEMA
Seismic Class	II
Condenser Unit (X-202/X-203)	
Type	Air cooled
Compressor	
Type	Hermetic

TABLE 9.9-15 BALANCE OF UNIT VENTILATION SYSTEM (CONTINUED)

Particulate Filter	
Quantity (per unit)	8
Type	Throwaway
Ductwork	
None	

SAJE Fans

	F-55A	F-55B
Type	Centrifugal	Centrifugal
Flow (cfm)	3204 (nominal rating)	1240 (nominal rating)
Standard	NAFM	NAFM
Seismic Class	II	II
Motor type	Induction	Induction
Motor horsepower rating (hp)	50	10
Motor Code	NEMA	NEMA
Motor Seismic Class	II	II
Ductwork Material / Type	IAW Specification M-526	IAW Specification M-526
Ductwork Standard	SMACNA	SMACNA
Ductwork Seismic Class	II	II

TABLE 9.9-16 ENGINEERED SAFETY FEATURES ROOM AIR RECIRC. SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Cooling coils	Tube or header rupture	Room temperature indication and flow indication on tube side	Loss of cooling	Unit taken out of service	One subsystem out of service	Redundant system available in redundant room. Each room contains a combination of a minimum safety features.
Fan	Fails to operate	Status lights from the control room	Loss of air flow	Same as above	Same as above	Same as above.
Dampers	Damper mispositioned	Room temperature alarm	Loss of cooling	Same as above	Same as above	Dampers are manually secured in position to provide proper flow to all rooms.

TABLE 9.9-17 CONTROL ROOM AIR CONDITIONING SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION AND QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Supply fan (F-21A & F-21B)	Fails to operate	Status light on C25A/B	Loss of air flow	Unit out of service	Redundant system operable	Each unit has 100% capacity
Evaporator coil (X-42A & X-42B)	Rupture	Room temperature indicator	Loss of cooling	Same as above	Same as above	Same as above
Exhaust fan (F-31A & F-31B)	Fails to operate	Status light on C25A/B	Loss of air flow	Same as above	Same as above	Same as above
Filtration fan (F-32A & F-32B)	Fails to operate	Status light on C25A/B	Loss of filtered air flow	Same as above	Same as above	Same as above
Condenser fan (F-36A & F-36B)	Fails to operate	Room temperature indicator	Loss of cooling capacity	Same as above	Same as above	Same as above
Compressor (F-22A & F-22B)	Fails to operate	Room temperature indicator	Loss of cooling capacity	Same as above	Same as above	Same as above
Supply outside air damper (2-HV-202)	Fails to close	Status light on C25A	None	None	Redundant damper 2-HV-495 closed	Safe position fails closed
F-21A discharge damper (2-HV-203A)	Fails to operate	Status light on C25A	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity

TABLE 9.9-17 CONTROL ROOM AIR CONDITIONING SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
F-21B discharge damper (2-HV-203B)	Fails to operate	Status light on C25B	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity
F-21A discharge duct fire damper (2-HV-204A)	N/A See remark	N/A	N/A	N/A	N/A	Fire damper is deemed a passive component
F-21B discharge duct fire damper (2-HV-204B)	N/A See remark	N/A	N/A	N/A	N/A	Fire damper is deemed a passive component
F-31A & F-31B common return duct fire damper (2-HV-205)	N/A See remark	N/A	N/A	N/A	N/A	Fire damper is deemed a passive component
F-31A discharge damper (2-HV-206A)	Fails to operate	Status light on C25A	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity
F-31B discharge damper (2-HV-206B)	Fails to operate	Status light on C25B	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity
Exhaust damper to cable vault (2-HV-207)	Fails to close	Status light on C25A	None	None	Redundant damper 2-HV-497 closed	Safe position fails closed

TABLE 9.9-17 CONTROL ROOM AIR CONDITIONING SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Exhaust damper to atmosphere (2-HV-208)	Fails to close	Status light on C25A	None	None	Redundant damper 2-HV-496 closed	Safe position fails closed
Return air volume damper (2-HV-209)	N/A See remark	N/A	N/A	N/A	N/A	Fixed damper passive component
F-32A/B Common intake duct, air damper (2-HV-210)	Fails to close	Status light on C25A	None	None	Manual action	Damper normally open. Needs to close only during emergency fresh air mode.
F-32A/B Common outside air damper for emergency fresh air mode (2-HV-211)	Fails to close	Status light on C25A	None	None	Redundant damper 2-HV-495 closed	Safe position fails closed
F-32A discharge damper (2-HV-212A)	Fails to operate	Status light on C25A	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity
F-32B discharge damper (2-HV-212B)	Fails to operate	Status light on C25B	Air flow affected	Unit out of service	Redundant system operable	Each unit has 100% capacity

TABLE 9.9-17 CONTROL ROOM AIR CONDITIONING SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Filter cross connect. damper (2-HV-213)	N/A See remark	N/A	N/A	N/A	N/A	PERMANENTLY CLOSED
F-32A & F-32B fresh air intake volume damper (2-HV-234)	N/A See remark	N/A	N/A	N/A	N/A	FIX DAMPER PASSIVE COMPONENT
F-21A/B or F-32A/B outside air damper (2-HV-495)	Fails to close	Status light on C25B	None	None	Redundant dampers to dual path 2-HV-202 & 2-HV-211	Safe position fails closed
Exhaust damper to atmosphere (2-HV-496)	Fails to close	Status light on C25B	None	None	Redundant damper 2-HV-208 closed	Safe position fails closed
Exhaust damper to cable vault (2-HV-497)	Fails to close	Status light on C25B	None	None	Redundant damper 2-HV-207 closed	Safe position fails closed
All control room ductwork	N/A See remark	N/A	N/A	N/A	N/A	Ductwork is seismic Class-1 and deemed a passive component
Fresh Air Intake Volume Damper (2-HV-725)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component

TABLE 9.9-17 CONTROL ROOM AIR CONDITIONING SYSTEM FAILURE MODE ANALYSIS (CONTINUED)

COMPONENT IDENTIFICATION AND QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
F-32A Discharge Volume Damper (2-HV-726A)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component
F-32B Discharge Volume Damper (2-HV-726B)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component
F-21A Discharge Volume Damper (2-HV-727A)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component
F-21B Discharge Volume Damper (2-HV-727B)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component
Exhaust Volume Damper to Cable Vault (2-HV-728)	N/A See remark	N/A	N/A	N/A	N/A	Fixed Damper Passive Component

TABLE 9.9-18 DIESEL GENERATOR VENTILATION SYSTEM FAILURE MODE ANALYSIS

COMPONENT IDENTIFICATION & QUANTITY	FAILURE MODE	METHOD OF DETECTION & MONITOR	DETRIMENTAL EFFECT ON SYSTEM	CORRECTIVE ACTION	RESULTANT SYSTEM STATUS	REMARKS
Fans (2) F-38A/B	Fails to start	Status lights in control room	One subsystem inoperable	Repair fan	Redundant system operable	One diesel generator adequate for LOCI operation
Louvers	Clogging of the louvers	None	Same as above	Cleaning the louvers	Same as above	Same as above
Modulating dampers 2-HV-257A/B	Mechanical failure	None	Partial loss of outside air cooling	Repair operator / damper	Same as above	Backup system available in the other generator room
Exhaust dampers 2-HV-255A/B	Mechanical failure	None	Loss of air cooling	Repair operator / damper	Same as above	Same as above
All vital ductwork	N/A See remark	N/A	N/A	N/A	N/A	Ductwork is a passive component and seismic Class-1

TABLE 9.9-19 VITAL SWITCHGEAR VENTILATION SYSTEM COMPONENT DESCRIPTION

AC Switchgear Air Handling Unit (4.16 KV and 6.9 KV Rooms)

	Lower Room at EL. 31'-6"	Upper Room at EL. 56'-6"
Fan	Fan F-134	Fan F-133
Type	Centrifugal	Centrifugal
Flow, CFM	7200 minimum	6000 minimum
Standard	AMCA 211-A	AMCA 211-A
Seismic Class	I	I
Cooling Coil	Coil X182	Coil X183
Type	Service Water	Service Water
Heat transfer (Btu/hr)	192,763	166,879
Code	ASME Section VIII	ASME Section VIII
Seismic class	I	I
Motor	Fan F-134	Fan F-133
Type	Induction	Induction
Horsepower rating (hp)	5	5
Code	NEMA	NEMA
Seismic Class	I	I
Particulate Filter		
Quantity	8	6
Type	Throwaway	Throwaway
Ductwork		
Material/Type	IAW Spec. M-526	IAW Spec M-526
Standard	SMACNA	SMACNA
Seismic Class	I	I

AC Switchgear Fans (480-Volt Unit Loadcenter Rooms, East and West)

	West Room	East Room Exhaust	East Room

TABLE 9.9-19 VITAL SWITCHGEAR VENTILATION SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Fan		Fan F-142	Supply Fan F-52
Type	Vaneaxial	Propeller	Vaneaxial
Flow (CFM)	18,300 minimum	23,000 (nominal rating)	21,000 minimum
Standard	AMCA 211-A	AMCA 211-A	AMCA 211-A
Seismic Class	I	I	I
Cooling Coil (West Room)	Coil X-181A/B		
Type	Service Water		
Heat transfer (Btu/hr)	295,641 (nominal rating)		
Code	ASME Section VIII		
Seismic Class	I		
Motor			
Type	Induction	Induction	Induction
Horsepower rating (hp)	25	10	25
Code	NEMA	NEMA	NEMA
Seismic Class	I	I	I
Particulate Filter			
Quantity	9	None	None
Type	Throwaway		
Ductwork			
Material/Type	IAW Spec. M-506	None	IAW Spec. M-506
Standard	SMACNA		SMACNA
Seismic Class	I	N/A	I
D-C Switchgear Air Handling Unit			
Fans F-54A/B			
Type	Centrifugal		
Flow (cfm)	3500 (minimum)		
Standard	AMCA 211-A		
Seismic Class	I		

TABLE 9.9-19 VITAL SWITCHGEAR VENTILATION SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Motor	
Type	Induction
Horsepower rating (hp)	5
Code	NEMA
Seismic Class	I
Particulate Filter	
Quantity (per unit)	2
Type (prefilter)	Throwaway
Quantity (per unit)	2
Type (primary)	High efficiency (bag)
Ductwork	
Material/Type	IAW Spec. 506
Standard	SMACNA
Seismic Class	I
Vital MCC B51 and B61 Coolers A/C-3, A/C-4	
Type	Self-contained A/C unit
Power Supply	480V/3 Ph/60 Hz
Evaporator Fan Data	
Type	Centrifugal
Air Flow	≈500 cfm
Motor HP	1/3
Code	NEMA MG1
Seismic Class	1
Condenser Fan Data	
Type	Propeller
Motor HP	1/2
Code	NEMA MG 1
Seismic Class	1
Evaporator Coil Data	
Type	Direct Expansion

TABLE 9.9-19 VITAL SWITCHGEAR VENTILATION SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Cooling Capacity	24,000 Btu/hr
Standard	ARI 410
Seismic Class	1
Condenser Coil Data	
Type	Air cooled
Standard	ARI 410
Seismic Class	1
Compressor Data	
Type/Refrigerant	Hermetic / R-22
HP nominal	2
Standard	ARI 520
Seismic Class	1

TABLE 9.9-20 AUXILIARY CHILLED WATER SYSTEM COMPONENT DESCRIPTION

Reciprocating Chilled Water Units (X-196A/B) Non-QA

Condenser Flow Rate	277.4 gpm (nominal rating)
Evaporator Flow Rate	183.6 gpm (nominal rating)
Condenser Temperature Rise	10°F (nominal rating)
Evaporator Temperature Drop	12°F (nominal rating)
Heat Removal Rate	91.8 tons (nominal rating)

Chilled Water Pumps (P-149A/B/C)

Type	Centrifugal
TDH	140 feet (nominal rating)
Flow	220 gpm (nominal rating)

TABLE 9.9-21 VITAL CHILLED WATER SYSTEM COMPONENT DESCRIPTION

Reciprocating Chilled Water Units (X-169A/B)

Type	Refrigerant (R-22)
Rating (capacity)	15 Ton (nominal rating)
Standard	ARI 590
Seismic Class	I

Evaporator

Type	Direct expansion, refrigerant
Heat transfer (Btu/hr)	194,000 (nominal rating)

Condenser

Type	Water cooled
Heat transfer (Btu/hr)	234,400 (nominal rating)

Compressor

Type	Semi-hermetic

Chilled Water Pumps (P-122A/B)

Type	Centrifugal
Total Dynamic Head (ft)	108 (nominal rating)
Flow (gpm)	35 (nominal rating)

TABLE 9.9-22 INDOOR DESIGN TEMPERATURES FOR SIGNIFICANT PLANT AREAS

Zone No. / Zone Description6	Maximum Indoor Design Temperature (°F)	Minimum Indoor Design Temperature (°F)
Containment Building Elevation		
General Containment Areas, (Normal Operation Mode 1-4)	120	N/A
General Containment Areas, (Shutdown Mode 5, 6)	120	40
(EQ-C06) Pressurizer Block House (Interior), Elevation 14 foot 6 inches & 38 foot 6 inches	150	N/A
Auxiliary Building		
(EQ-A15) East 480 Volt Load Center, Elevation 36 foot 6 inches	104	32
(EQ-A35) East Vital DC Switchgear Room, Elevation 14 foot 6 inches	104	32
(EQ-A36) West Vital DC Switchgear Room, Elevation 14 foot 6 inches	104	32
(EQ-A42) Control Room, Elevation 36 foot 6 inches	100 * *per TS 3/4.7.6	N/A
(EQ-A28) Emergency Diesel Generator Room “A”, Elevation 14 foot 6 inches	120 (EDG's operating)	N/A (EDG's operating)
	120 (EDG's in standby)	55 (EDG's in standby)
(EQ-A29) Emergency Diesel Generator Room “B”, Elevation 14 foot 6 inches	120 (EDG's operating)	N/A (EDG's operating)
	120 (EDG's in standby)	55 (EDG's in standby)

TABLE 9.9-22 INDOOR DESIGN TEMPERATURES FOR SIGNIFICANT PLANT AREAS (CONTINUED)

Zone No. / Zone Description6	Maximum Indoor Design Temperature (°F)	Minimum Indoor Design Temperature (°F)
(EQ-A02, A03, & A04) ESF Rooms, Elevation (-) 45 foot 6 inches	177.9 (Safety related equipment operating)	N/A (Safety related equipment operating)
	110 (Safety related equipment in standby)	40 (Safety related equipment in standby)
(EQ-A24A & A-24B) MCC B51 & B61 Enclosures, respectively, Elevation 14 foot 6 inches	122	32
(EQ-A27A, B & C) Fuel Handling Area, Elevation 14 foot 6 inches & 38 foot 6 inches	110	40
(EQ-A37) West Battery Room “B” Elevation 14 foot 6 inches	104	60
(EQ-A41) East Battery Room “A” Elevation 14 foot 6 inches	104	60
EQ-A01A, A01A1, A01B, A03A, A05, A06, A08, A14A, A16, A20, A23, A30, A31, A38, A39 Radwaste Potentially Radioactive Areas	110	40
(EQ-A11) Waste Gas Decay Tanks Room, Elevation (-)25 foot 6 inches	110	55
(EQ-A12) Waste Gas Compressors Room, Elevation (-)25 foot 6 inches	110	55
(EQ-A13) Waste Gas Surge Tank Room, Elevation (-)25 foot 6 inches	110	55
(EQ-A14) Boric Acid and Refueling Water Purification Areas, Elevation (-)5 foot 0 inches	110	55

TABLE 9.9-22 INDOOR DESIGN TEMPERATURES FOR SIGNIFICANT PLANT AREAS (CONTINUED)

Zone No. / Zone Description6	Maximum Indoor Design Temperature (°F)	Minimum Indoor Design Temperature (°F)
(EQ-A21) West Boric Acid Evaporator Room, Elevation (-)5 foot 0 inches	110	55
(EQ-A22) East Boric Acid Evaporator Room, Elevation (-)5 foot 0 inches	110	55
(EQ-A24) Boric Acid Batch Tank and Chemical Storage Area, Elevation 14 foot 6 inches	110	55
(EQ-A40) Cable Vault in Auxiliary & Turbine Buildings, Elevation 25 foot 6 inches	104	32
(EQ-A09) Charging Pumps, Elevation (-)25 foot 6 inches	110	40
(EQ-A52) Auxiliary Building/CRACS Mechanical Equipment Room, Elevation 36 foot 6 inches	110	32
(EQ-A49) Auxiliary Building Fan Room (Open to Auxiliary Building Truck Bay), Elevation 36 foot 6 inches	110	32
(EQ-A27D) Enclosure Building Filtration System Equipment Room, Elevation 14 foot 6 inches	110	40
(EQ-A07) Coolant Waste Tanks Room Elevation (-)25 foot 6 inches / (-)5 foot 6 inches	110	40
(EQ-A10) Spent Resin Tank Room, Elevation (-)25 foot 6 inches	110	55
Enclosure Building		

TABLE 9.9-22 INDOOR DESIGN TEMPERATURES FOR SIGNIFICANT PLANT AREAS (CONTINUED)

Zone No. / Zone Description6	Maximum Indoor Design Temperature (°F)	Minimum Indoor Design Temperature (°F)
(EQ-A50) MS Isolation Valve Room, East Penetration Area, Elevation 38 foot 6 inches	120	40
(EQ-A51) MS Isolation Valve Room, West Penetration Area, Elevation 38 foot 6 inches	120	40
(EQ-A26A & A50A) Enclosure Building Upper & Lower Areas, respectively, Elevation 14 foot 6 inches & 38 foot 6 inches	120	40
(EQ-A17) East Piping Penetration Room, Elevation (-)25 foot -6 inches & (-)5 foot 6 inches	120	40
(EQ-A18) West Piping Penetration Room, Elevation (-)5 foot 6 inches	135	40
(EQ-A19) West Piping Penetration Room, Elevation (-)25 foot 6 inches	120	40
(EQ-A25) West Electrical Penetration Room, Elevation 14 foot 6 inches	130	40
(EQ-A26) East Electrical Penetration Room, Elevation 14 foot 6 inches	130	40
Turbine Building		
EQ-T10, T12, T13, A51A General Area All Levels, Bulk Average Temp.	110	40
(EQ-T04) West 480-Volt Load center, Elevation 36 foot 6 inches	104	40

TABLE 9.9-22 INDOOR DESIGN TEMPERATURES FOR SIGNIFICANT PLANT AREAS (CONTINUED)

Zone No. / Zone Description6	Maximum Indoor Design Temperature (°F)	Minimum Indoor Design Temperature (°F)
(EQ-T07) Lower 4160-Volt Switchgear Room, Elevation 36 foot 6 inches	122	40
(EQ-T09N) Turbine Driven Auxiliary Feedwater (TDAFW) Pump Room, Elevation 1 foot 6 inches	135 (TDAFW Pumps in service)	N/A (TDAFW Pumps in service)
	110 (TDAFW Pumps in standby)	40 (TDAFW Pumps in standby)
(EQ-T09S) Motor Driven Auxiliary Feedwater (MDAFW) Pump Room, Elevation 1 foot 6 inches	135 (MDAFW Pumps in service)	N/A (MDAFW Pumps in service)
	110 (MDAFW Pumps in standby)	40 (MDAFW Pumps in standby)
(EQ-T05) West Cable Vault Room, Cable Vault ∇, Elevation 45 foot 6 inches	104	32
(EQ-T06) East Cable Vault Room, Cable Vault ∣∇, Elevation 45 foot 6 inches	104	32
(EQ-T11) Non-Vital Turbine Battery Room, Elevation 31 foot 6 inches	110	60
Intake Structure		
(EQ-I01) Main Area (large open space), Elevation 14 foot 0 inches	130	40

FIGURE 9.9–1 P&ID CONTAINMENT AND ENCLOSURE BUILDING VENTILATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–2 VENTILATION P&IDS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–2 VENTILATION P&IDS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–2 VENTILATION P&IDS (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–2 VENTILATION P&IDS (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–3 P&ID AUXILIARY BUILDING VENTILATION SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–3 P&ID AUXILIARY BUILDING VENTILATION SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–3 P&ID AUXILIARY BUILDING VENTILATION SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–4 P&ID HVAC SYSTEM TURBINE BUILDING, INTAKE STRUCTURE, WAREHOUSE AND DIESEL GENERATOR ROOMS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–4A P&ID CONTROL ROOM AIR CONDITIONING SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–5 CONTROL ELEMENT DRIVE MECHANISM (CEDM) FAN PERFORMANCE CURVE

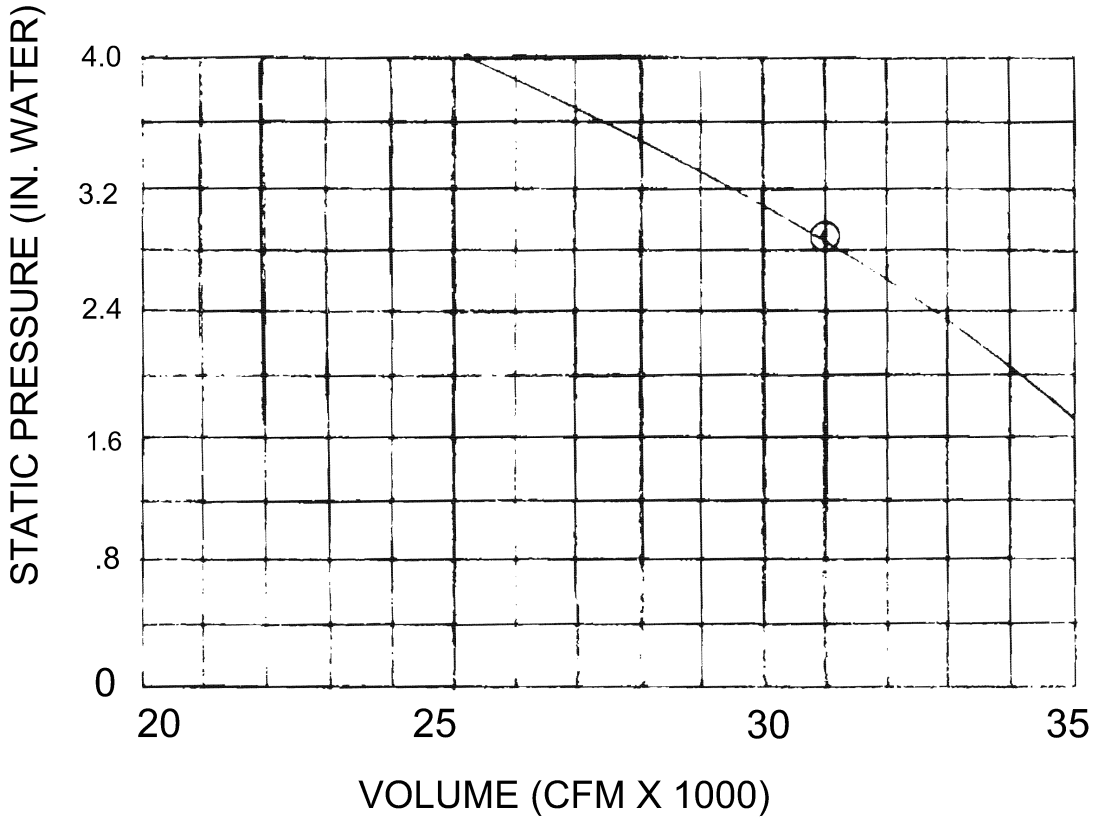


FIGURE 9.9–6 CONTAINMENT PURGE FAN PERFORMANCE CURVE

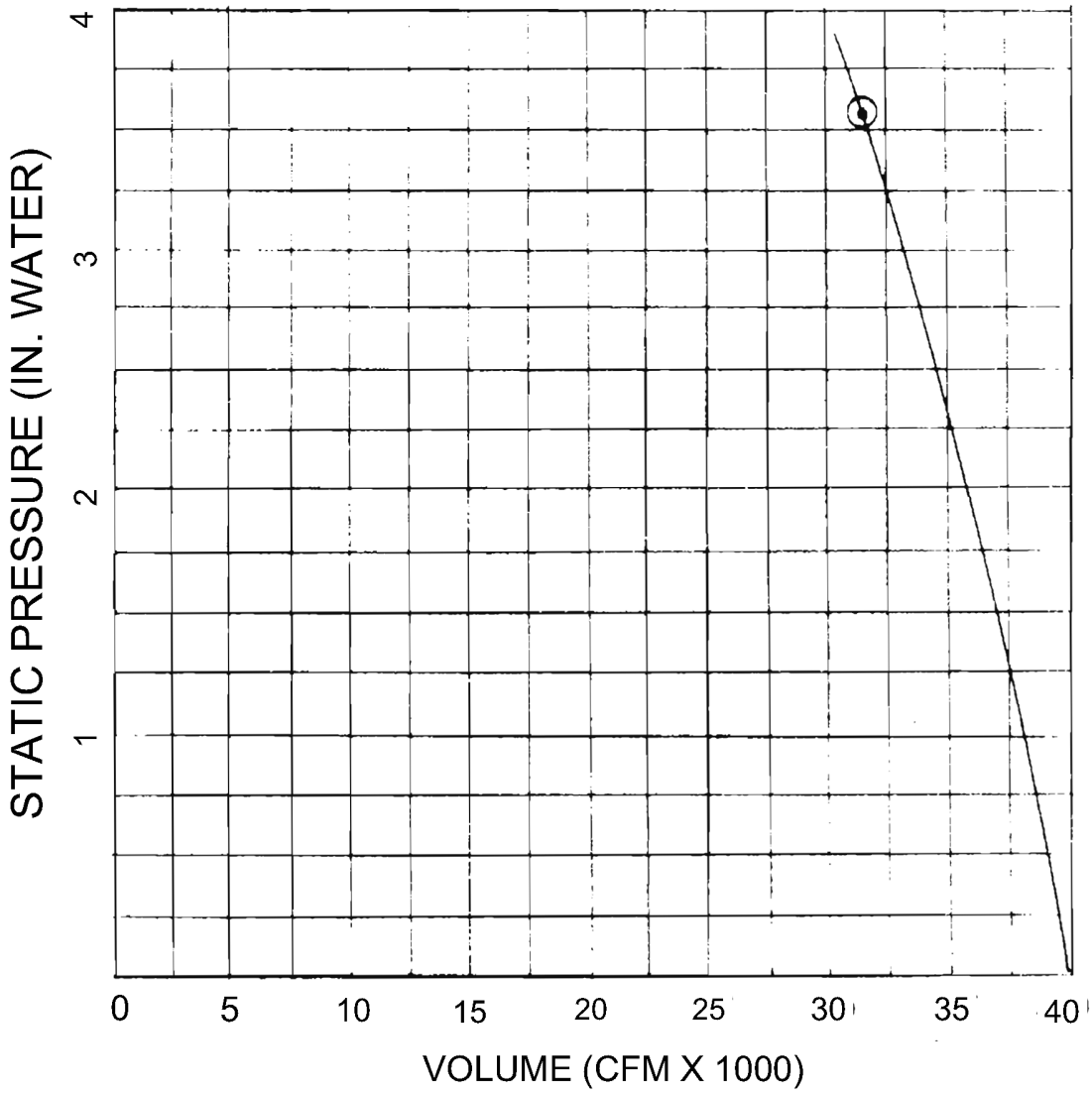


FIGURE 9.9–7 CONTAINMENT AUXILIARY CIRCULATION FAN PERFORMANCE CURVE

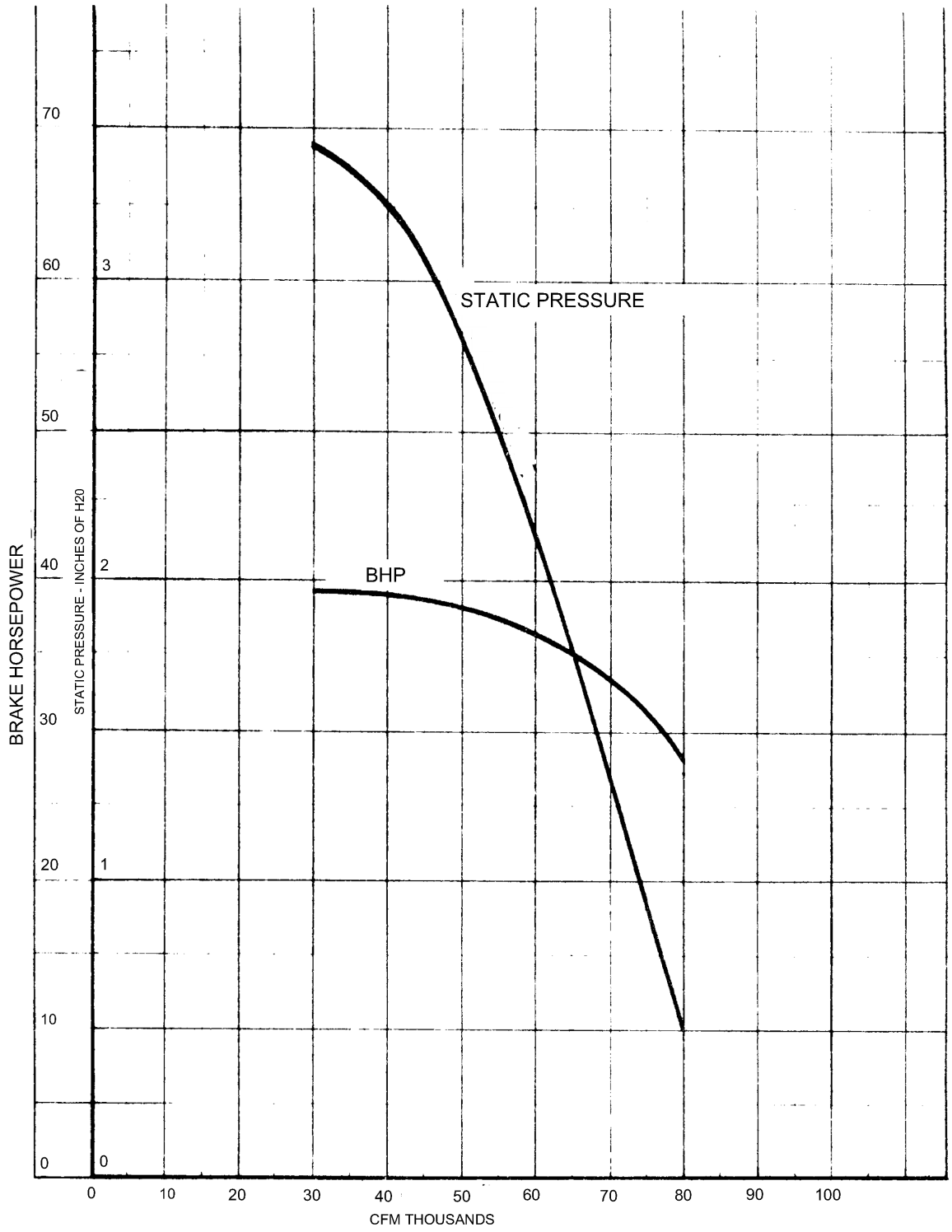


FIGURE 9.9–8 CONTAINMENT PENETRATION COOLING FAN PERFORMANCE CURVE

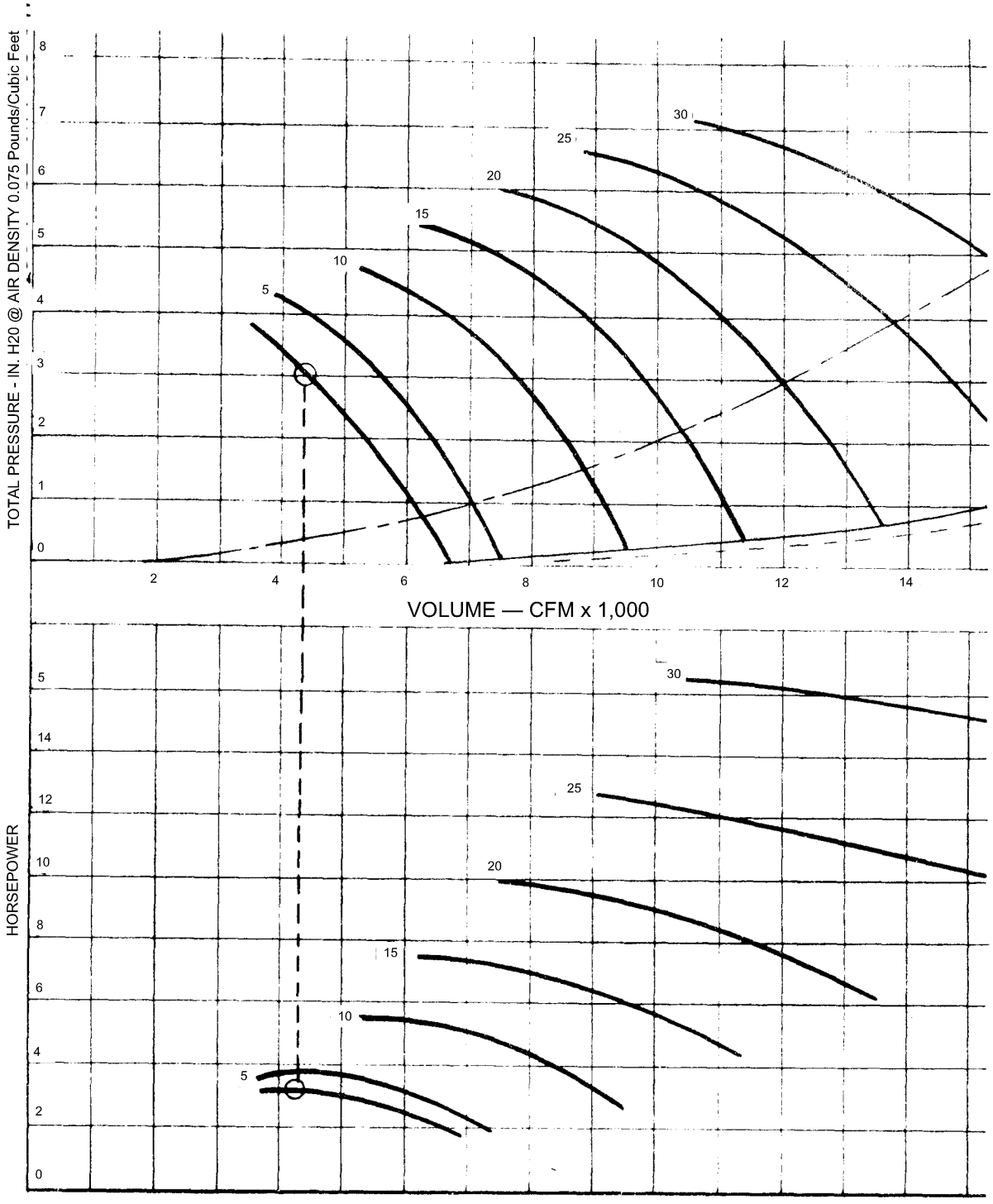


FIGURE 9.9–9 RADWASTE VENTILATION FAN PERFORMANCE CURVE

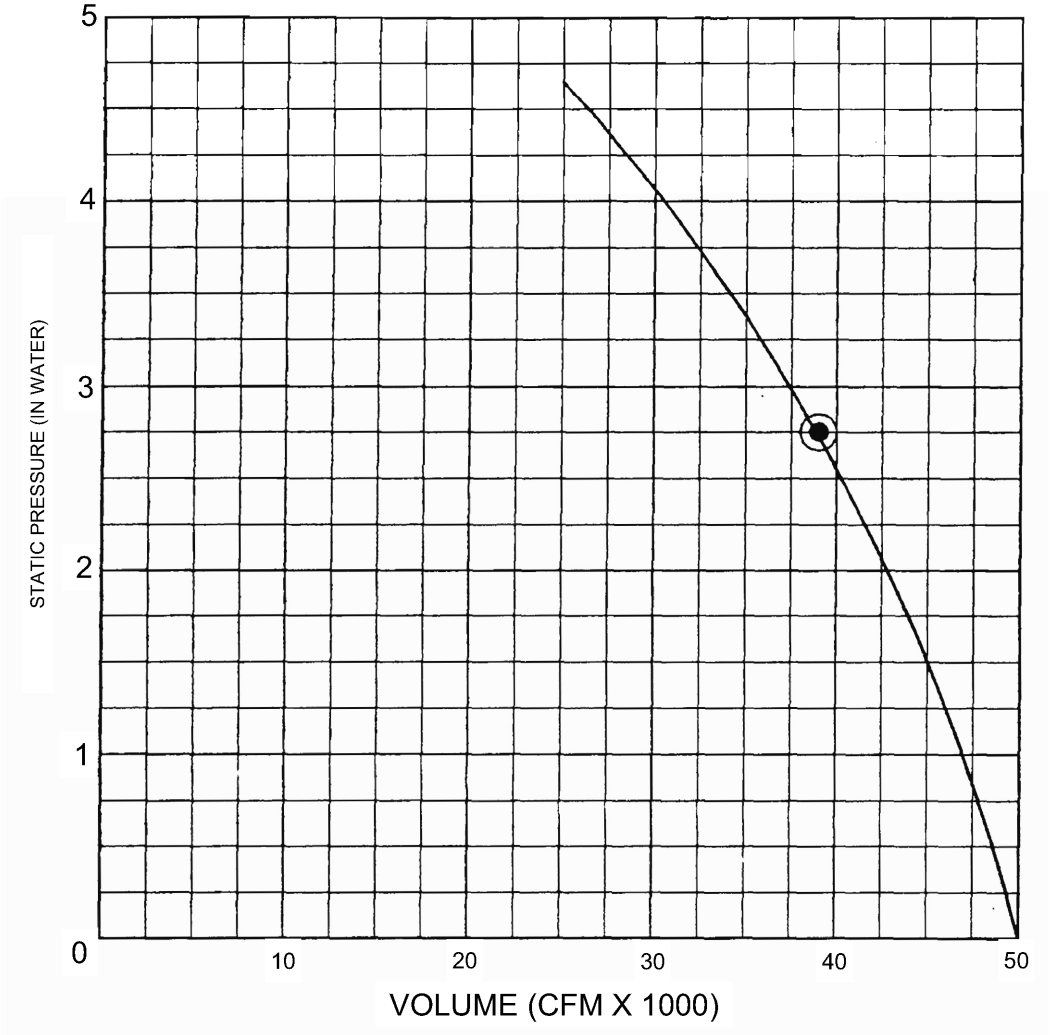


FIGURE 9.9–10 LOGIC DIAGRAM RADWASTE AREA SUPPLY FAN

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.9–11 NON-RADIOACTIVE SUPPLY FAN PERFORMANCE CURVE

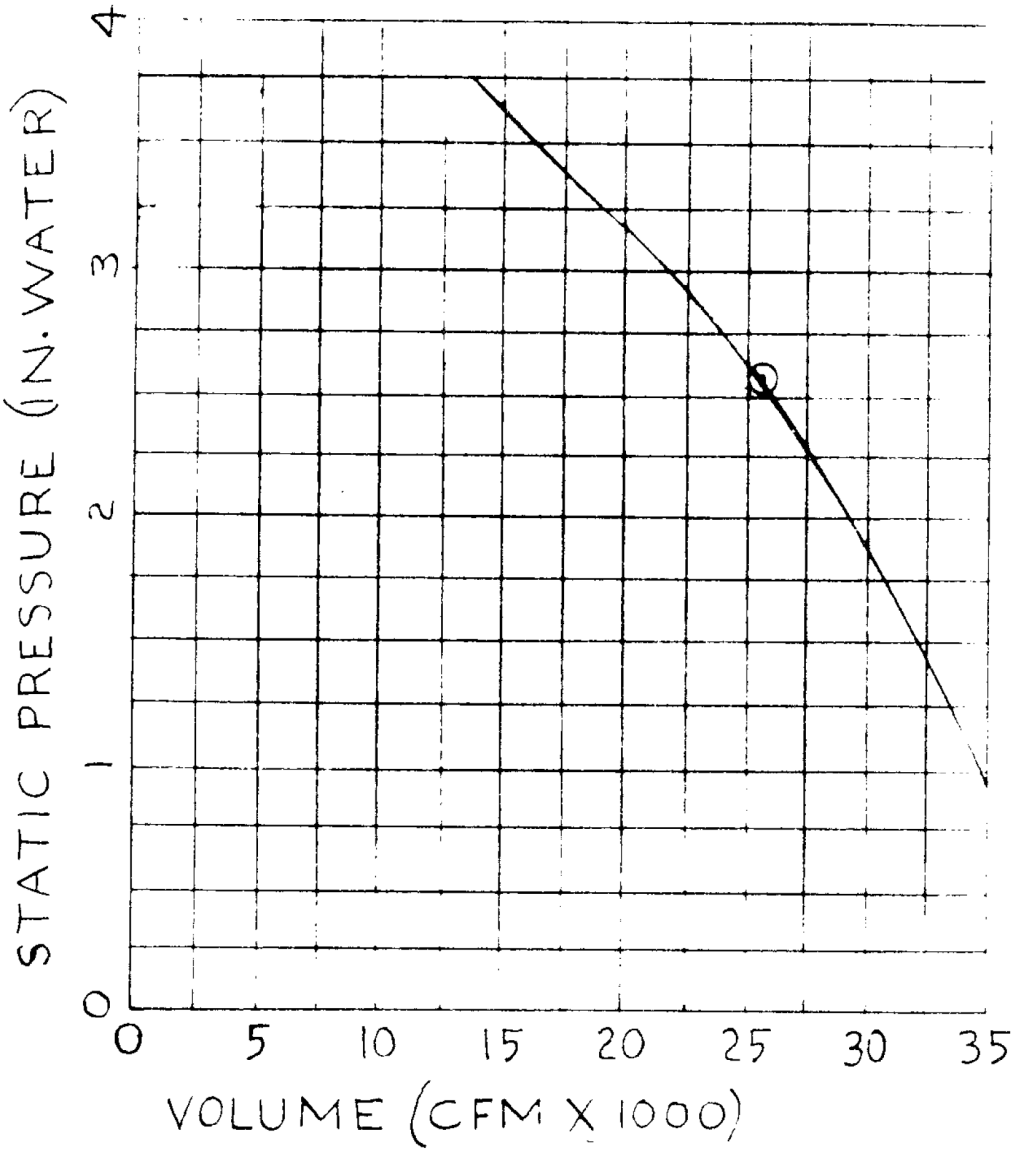


FIGURE 9.9–12 NON-RADIOACTIVE EXHAUST FAN PERFORMANCE CURVE

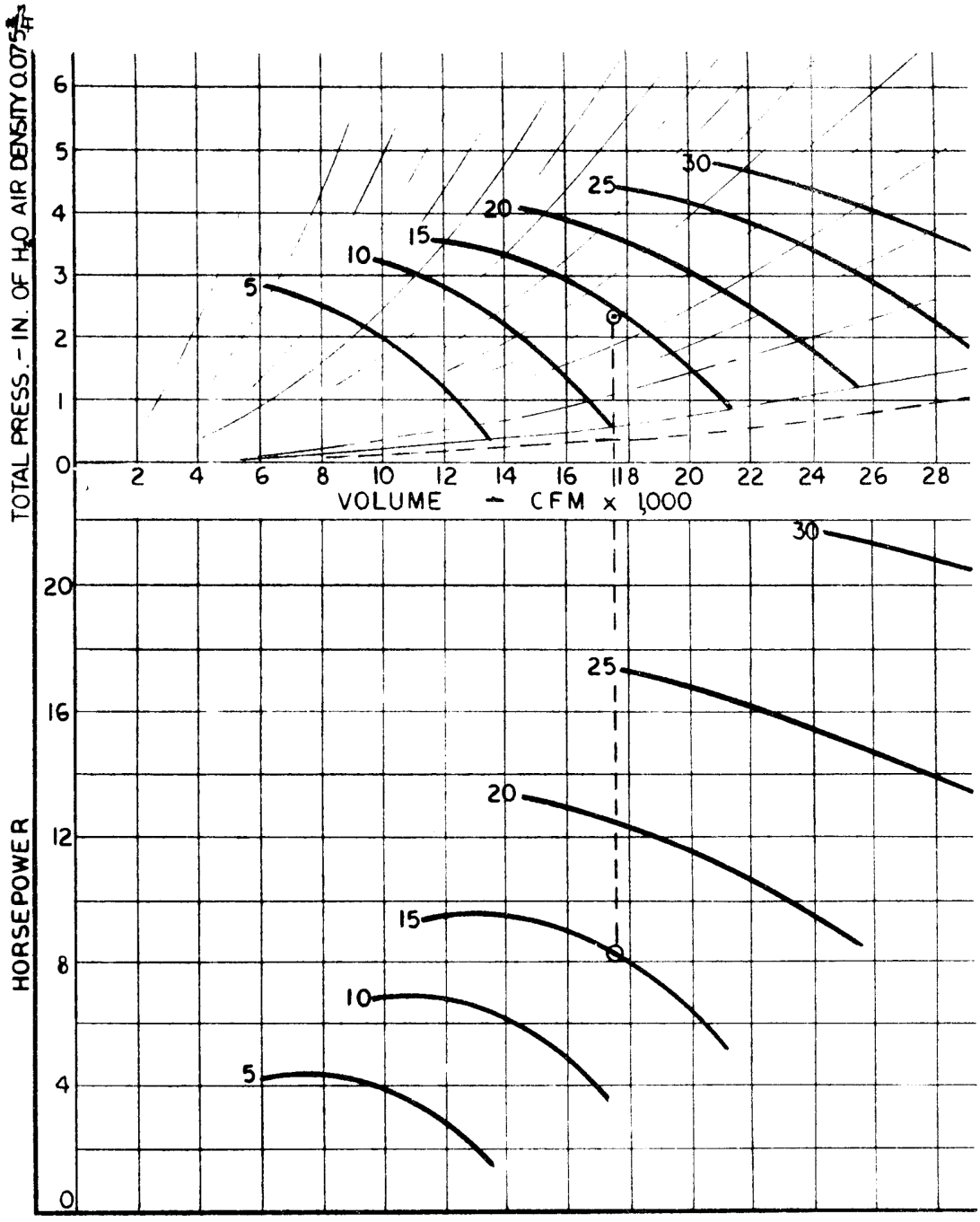


FIGURE 9.9–13 BATTERY ROOM EXHAUST FAN PERFORMANCE CURVE

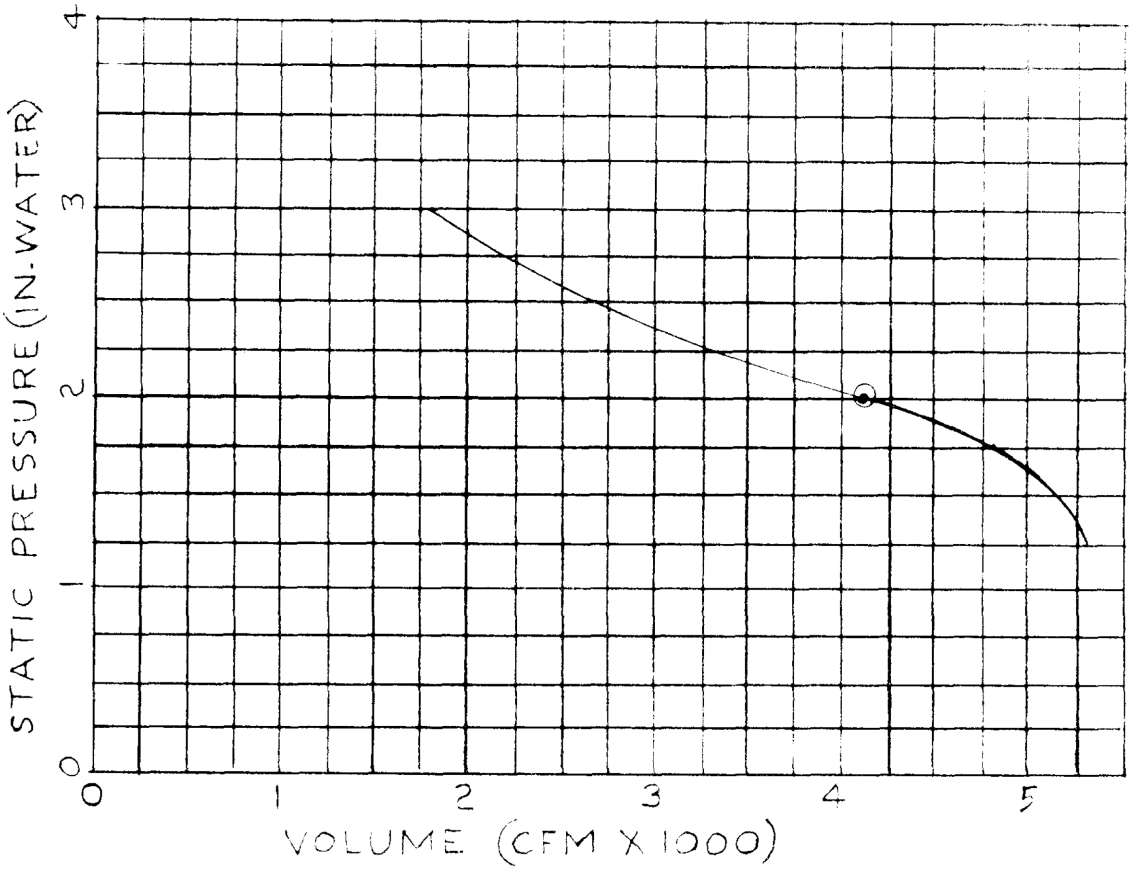


FIGURE 9.9–14 CABLE VAULT TRANSFER FAN PERFORMANCE CURVE

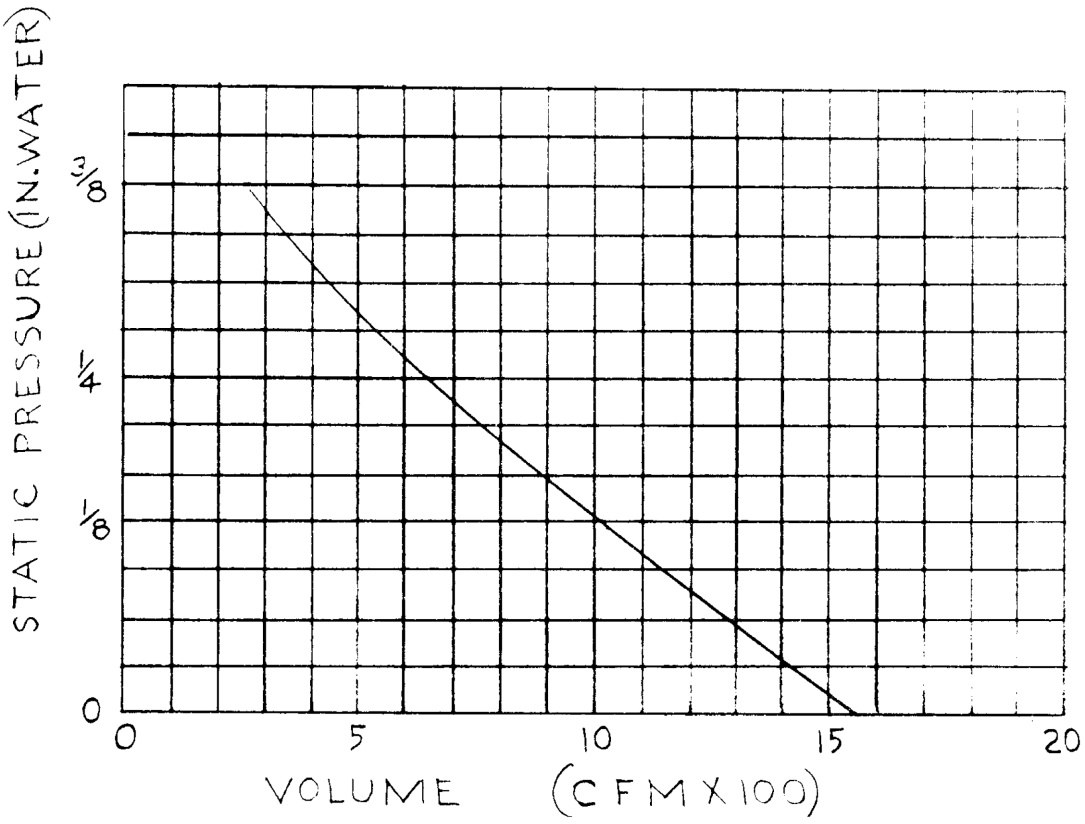


FIGURE 9.9–15 ENGINEERED SAFETY FEATURES ROOM FAN PERFORMANCE CURVE

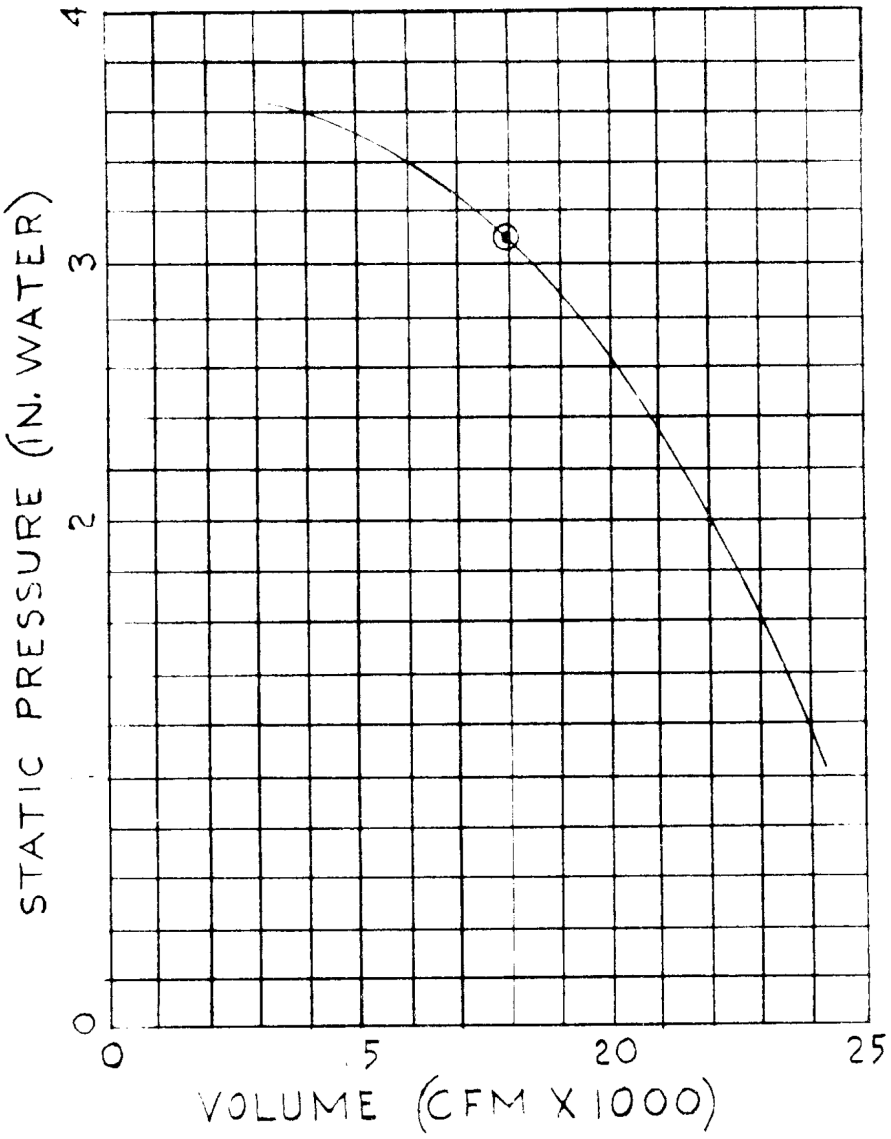


FIGURE 9.9–16 FUEL HANDLING SUPPLY FAN PERFORMANCE CURVE

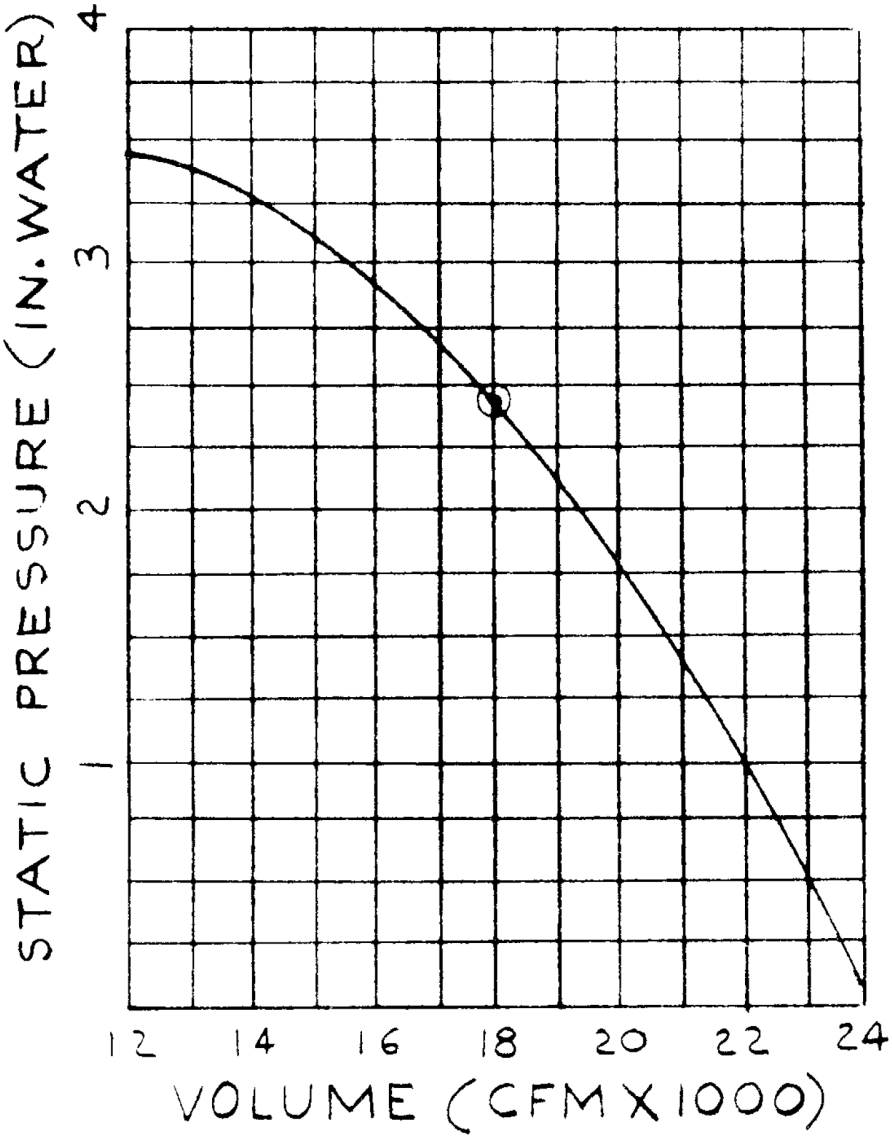


FIGURE 9.9–17 DELETED

FIGURE 9.9–18 MAIN EXHAUST FAN PERFORMANCE CURVE

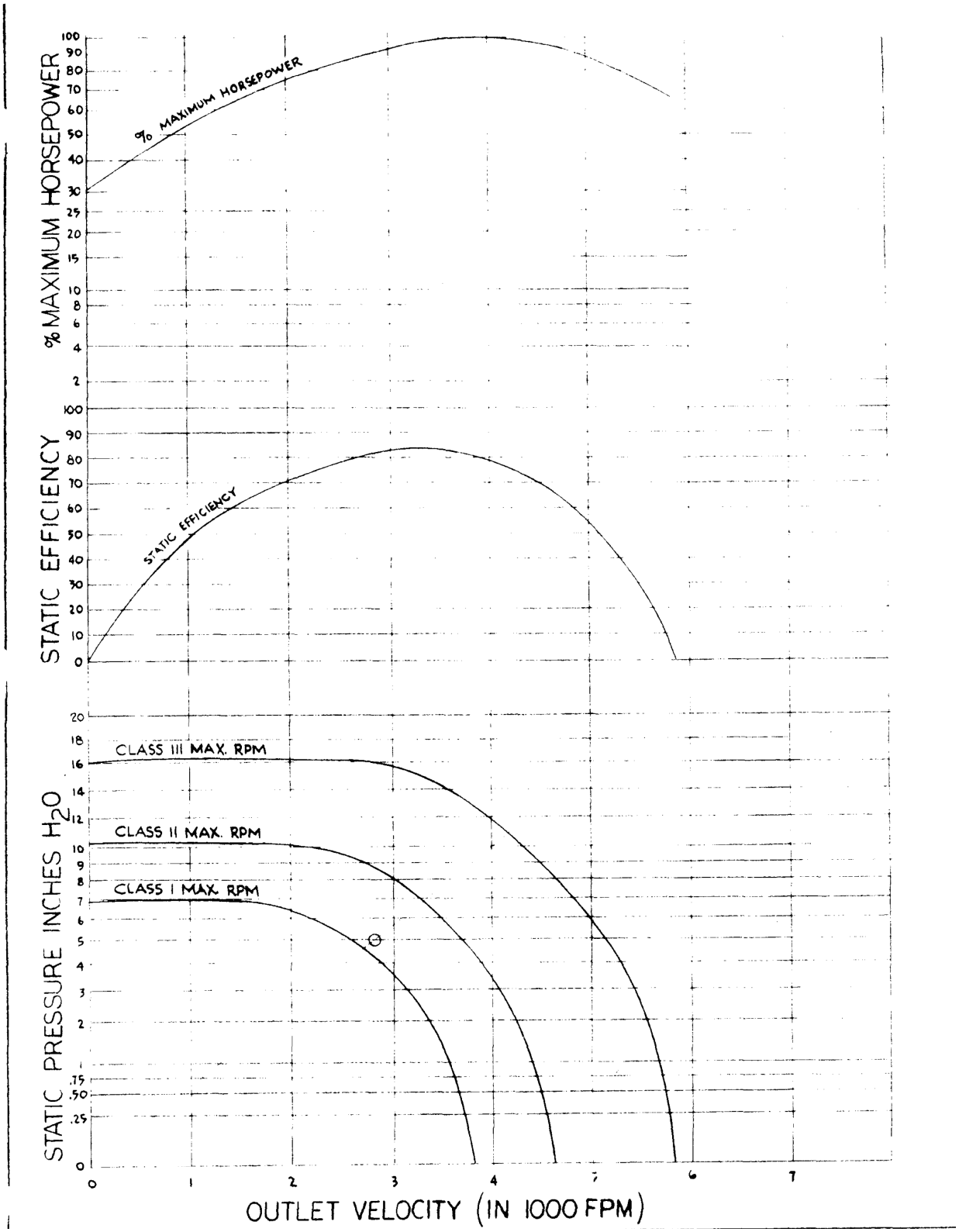


FIGURE 9.9–19 CONTROL ROOM AIR CONDITIONING FAN PERFORMANCE CURVE

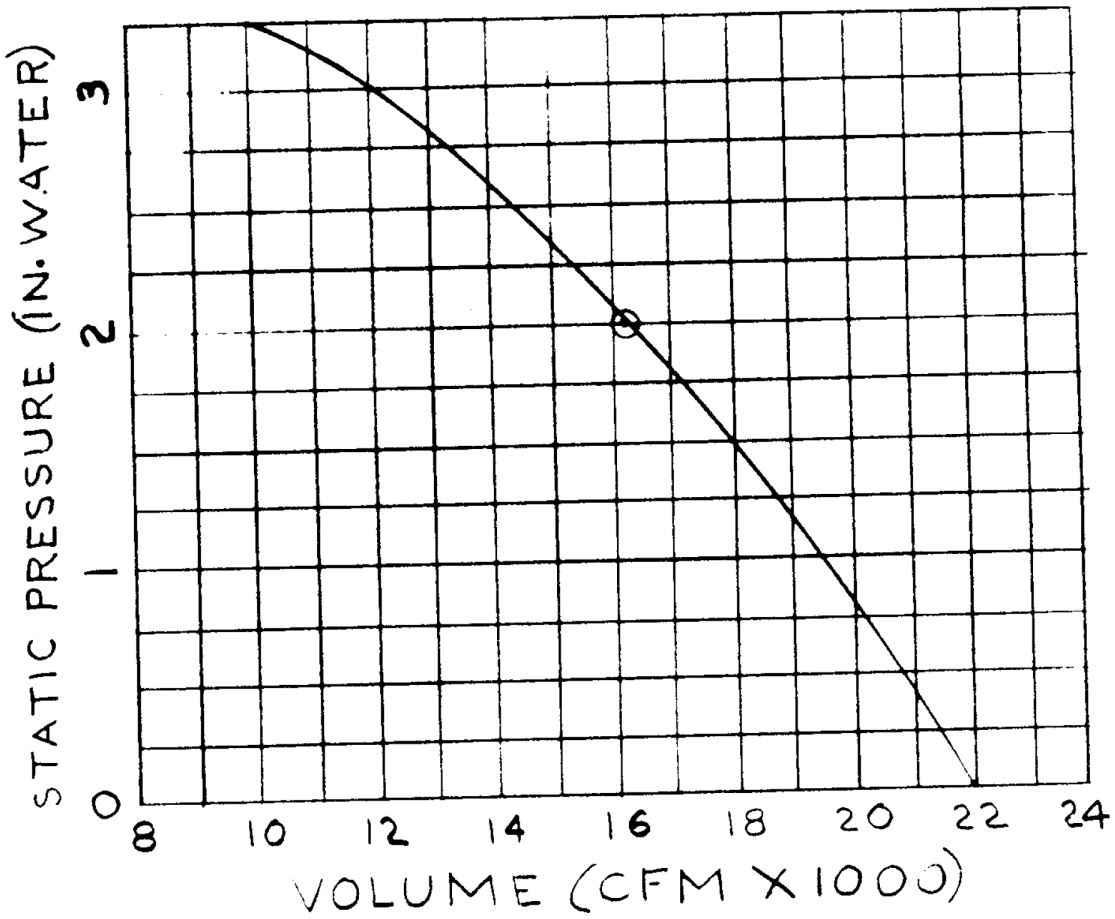


FIGURE 9.9–20 AIR COOLED CONDENSER FAN PERFORMANCE CURVE

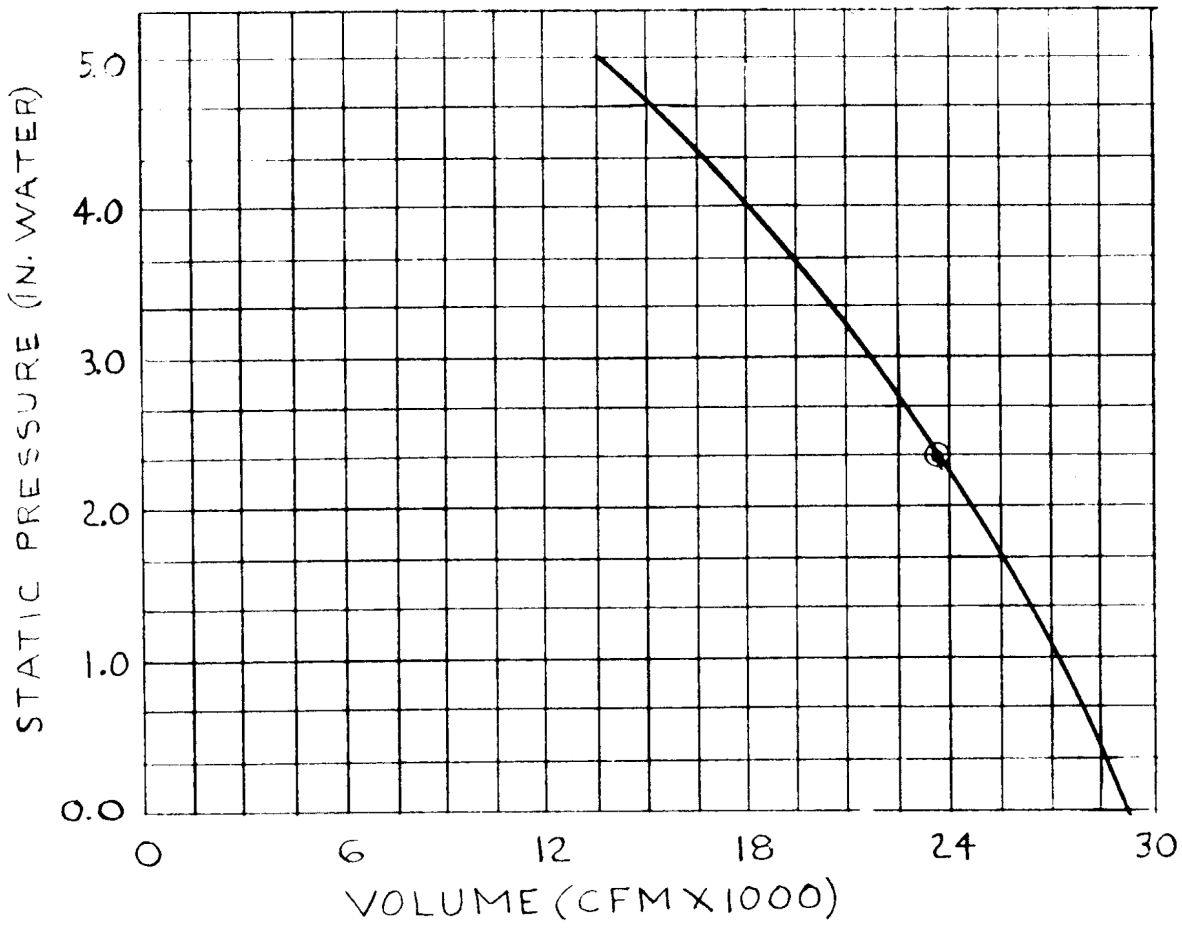


FIGURE 9.9–21 CONTROL ROOM EXHAUST FAN PERFORMANCE CURVE

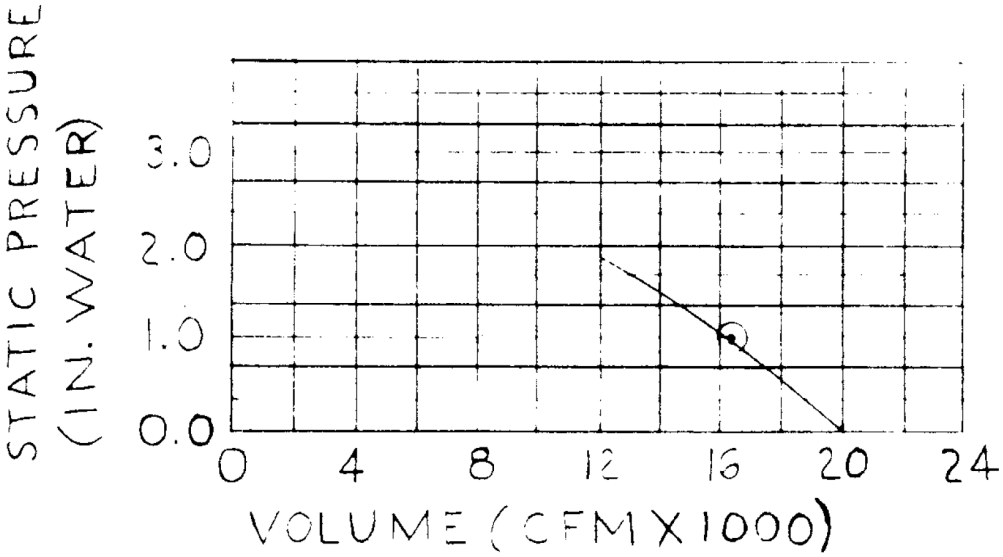


FIGURE 9.9–22 CONTROL ROOM FILTRATION SYSTEM FAN PERFORMANCE CURVE

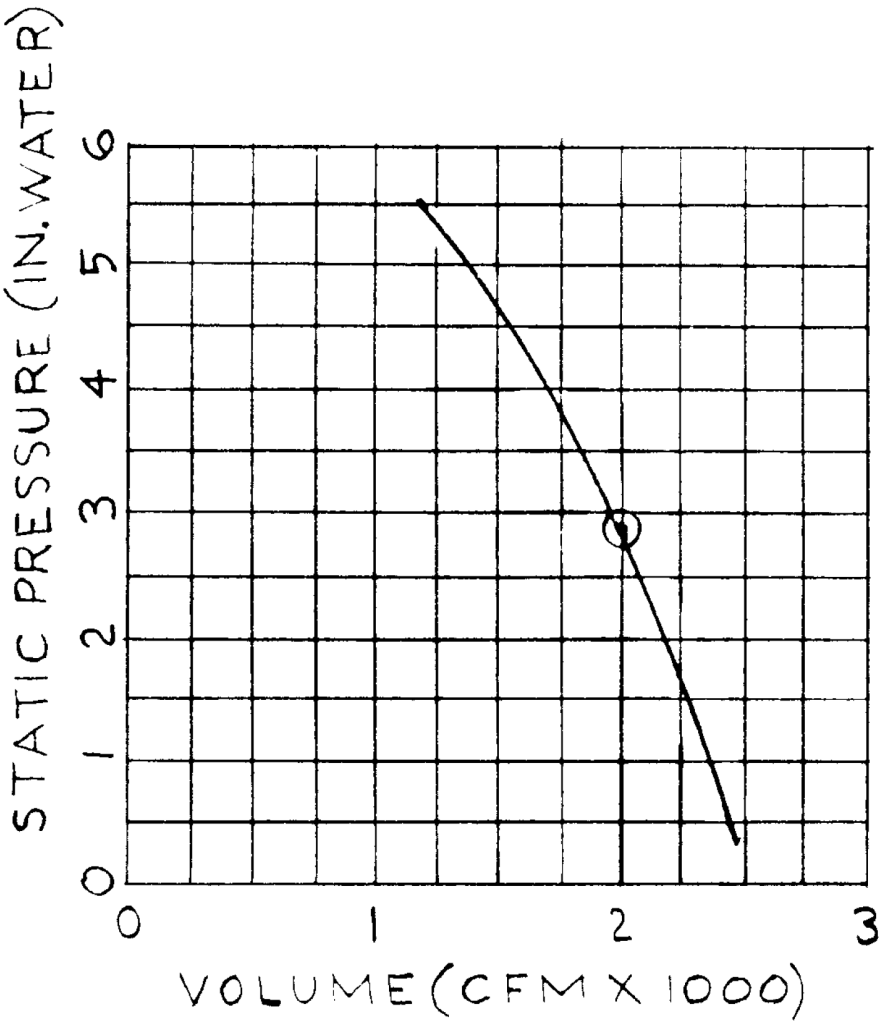


FIGURE 9.9–23 DIESEL GENERATOR ROOM SUPPLY FAN PERFORMANCE CURVE

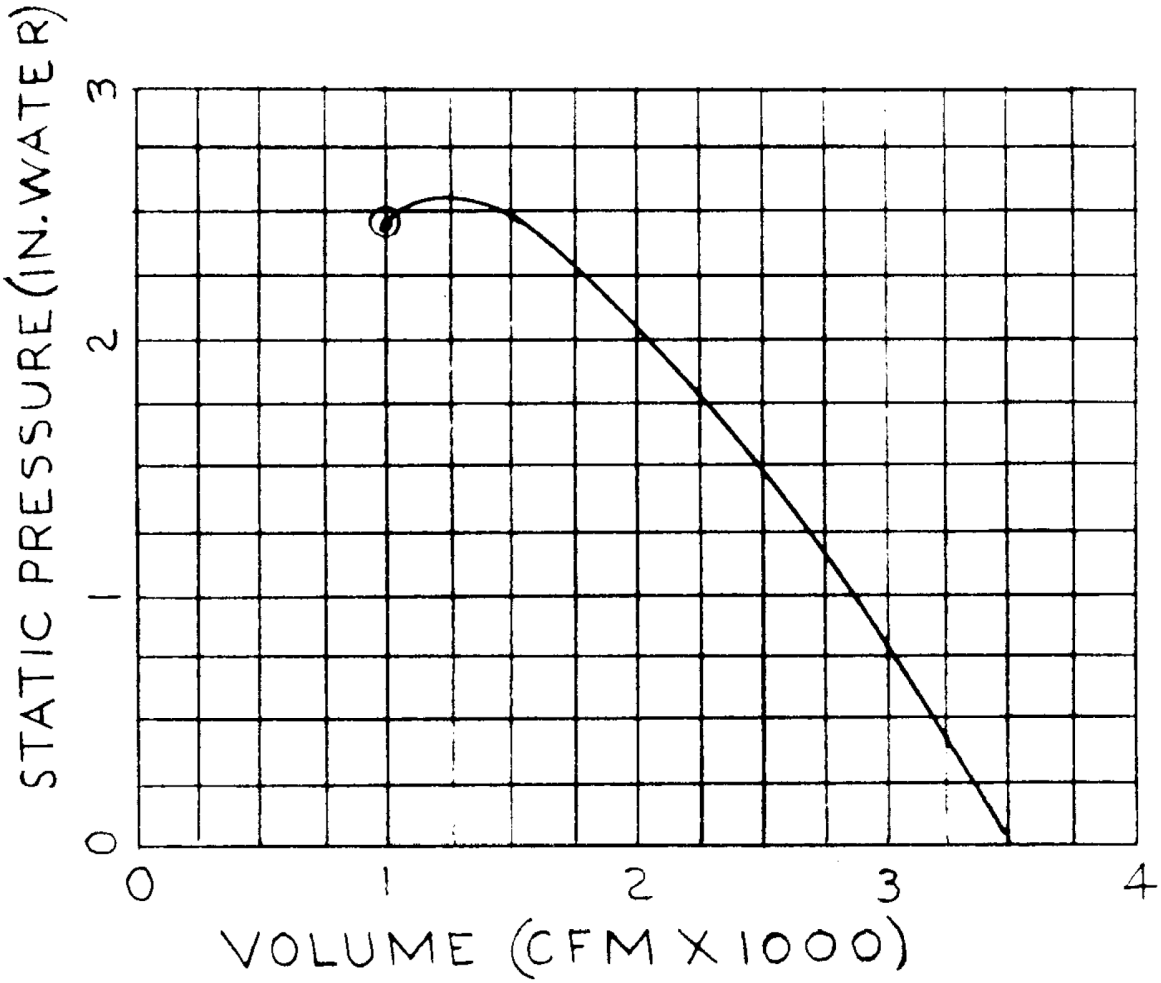


FIGURE 9.9–24 DIESEL GENERATOR FAN PERFORMANCE CURVE

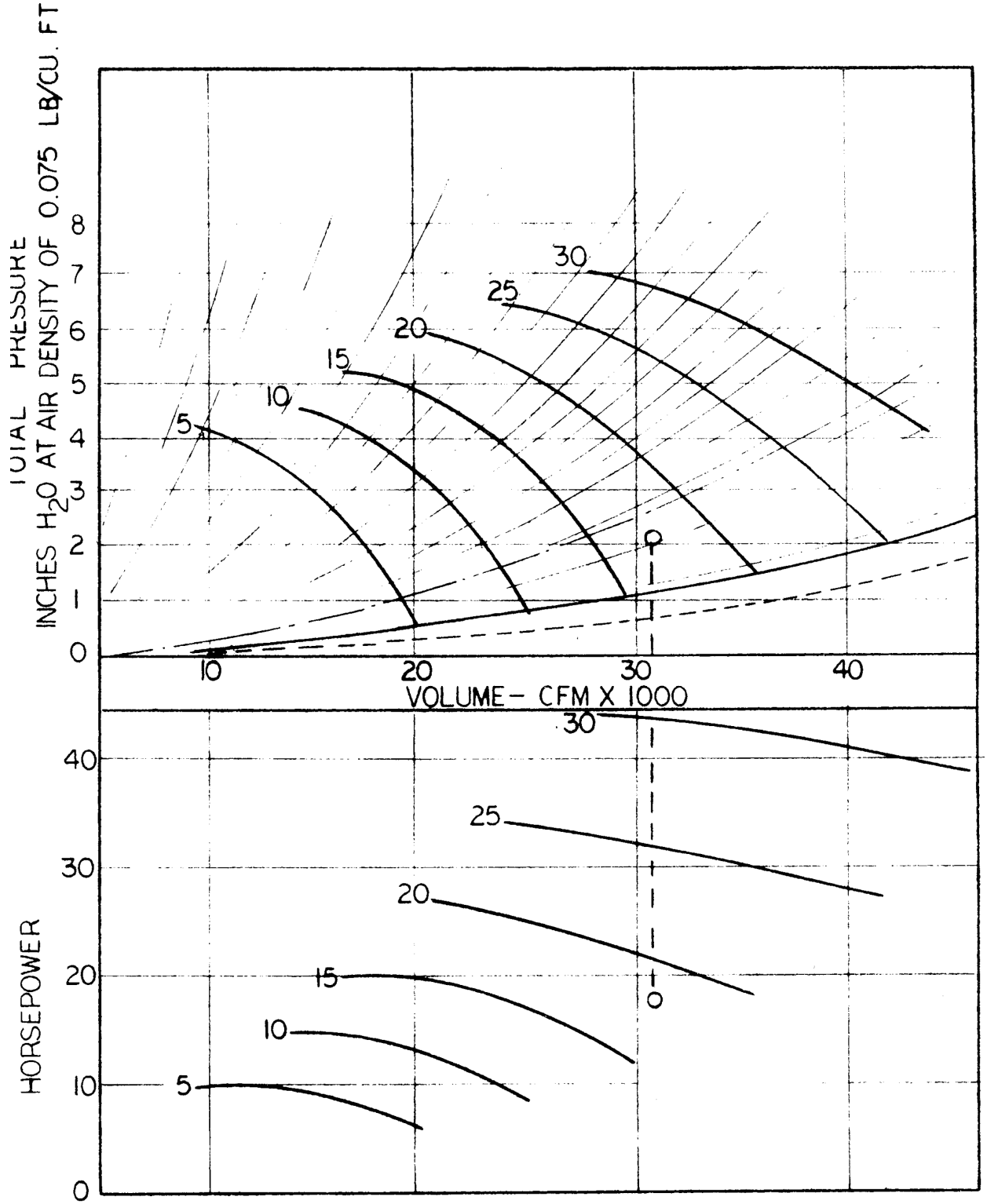


FIGURE 9.9–25 TURBINE BUILDING SUPPLY FAN PERFORMANCE CURVE

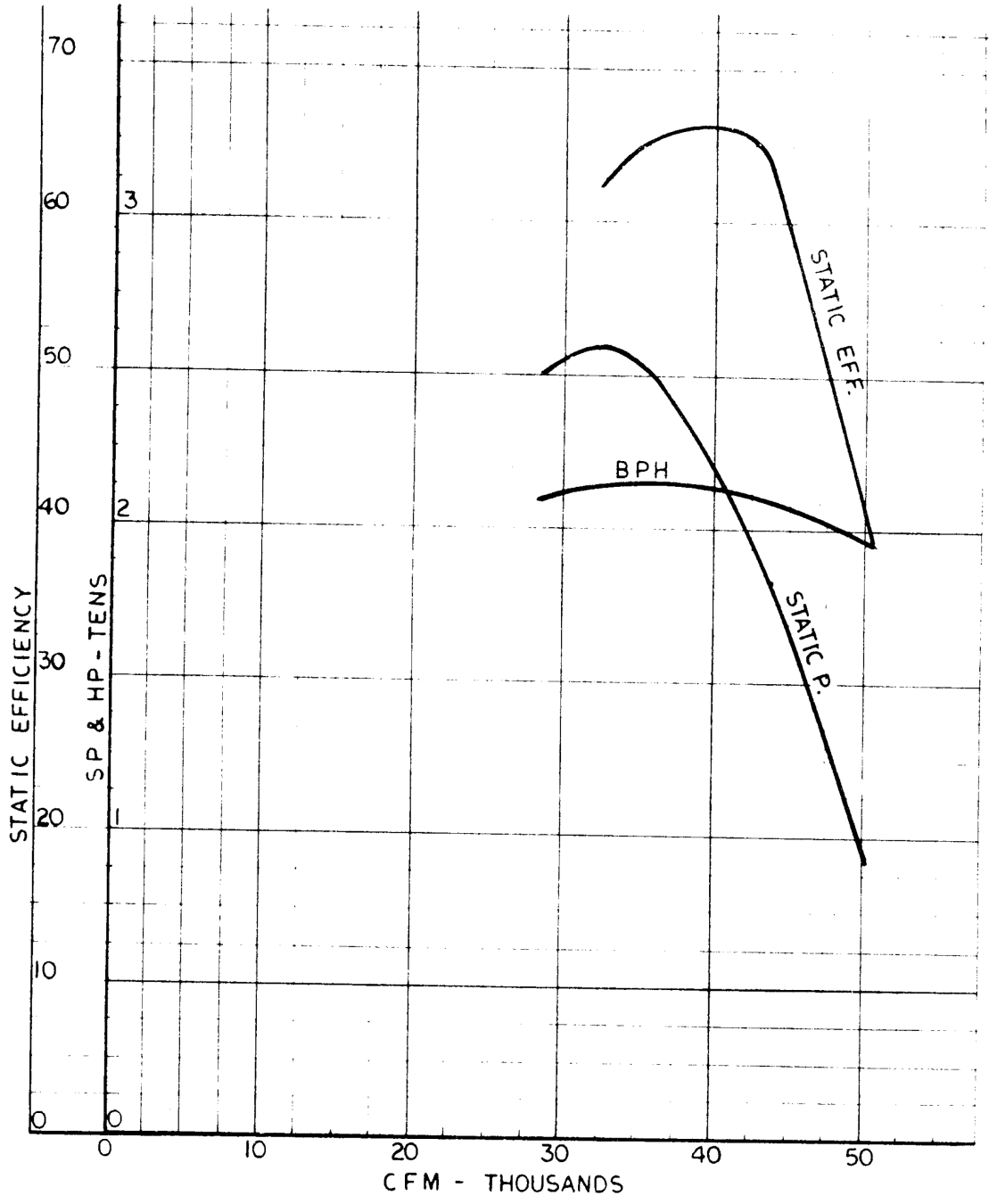


FIGURE 9.9–26 TURBINE BUILDING EXHAUST FAN PERFORMANCE CURVE

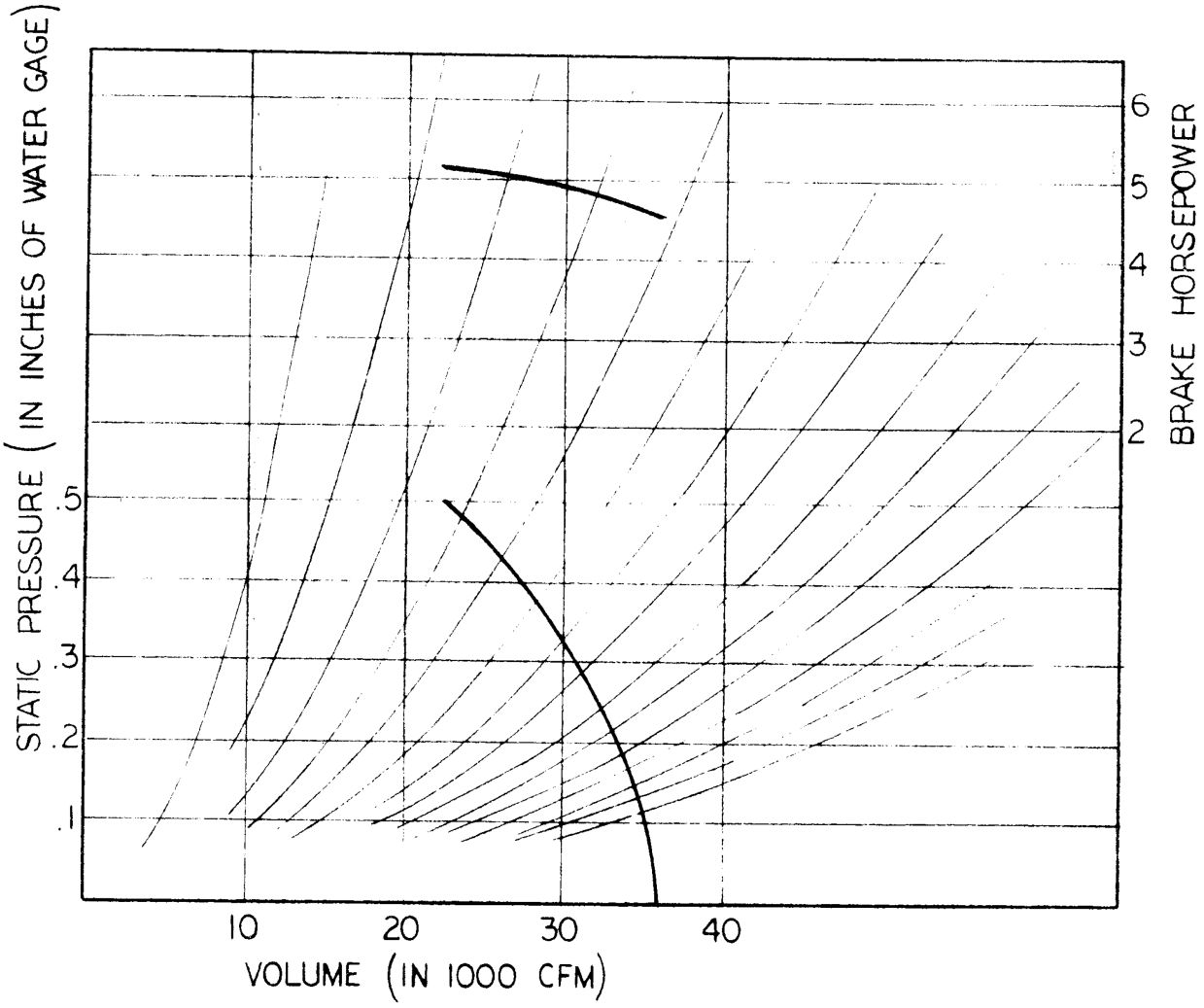
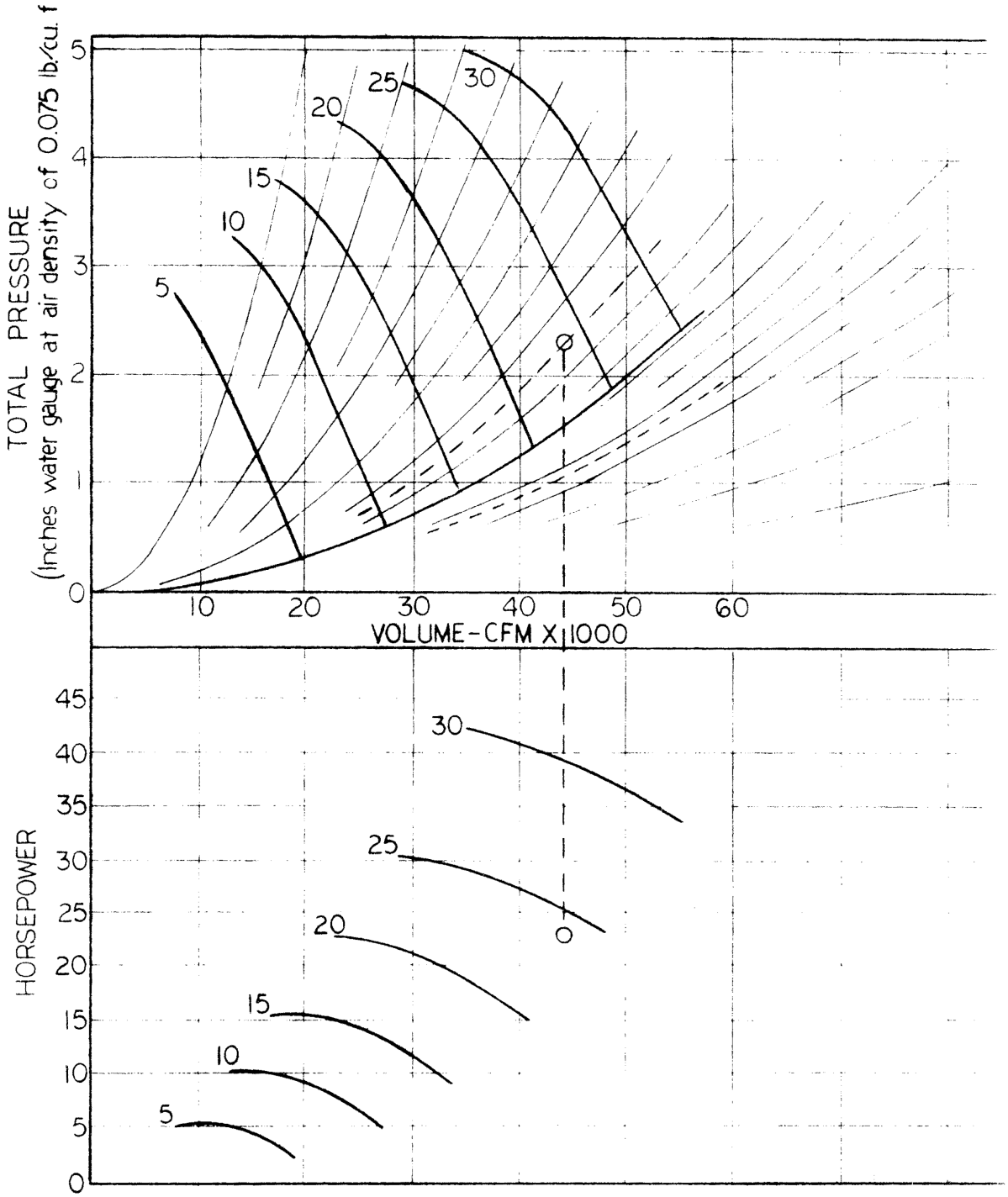


FIGURE 9.9–27 ELECTRICAL ROOM SUPPLY FAN PERFORMANCE CURVE



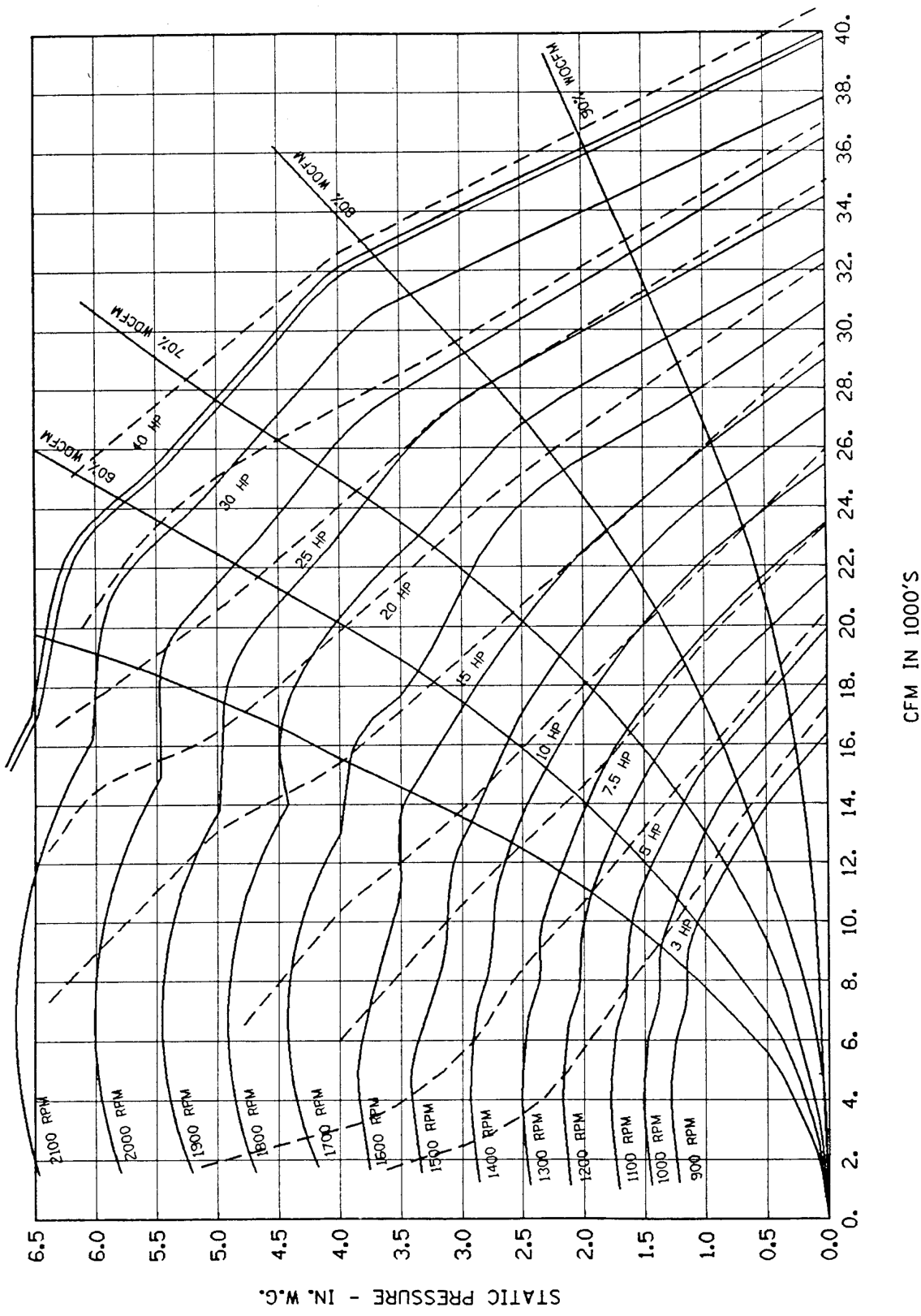


FIGURE 9.9–28 ACCESS CONTROL AREA AIR CONDITIONING FAN PERFORMANCE CURVE

FIGURE 9.9–29 DELETED

FIGURE 9.9–30 DELETED

FIGURE 9.9–31 P&ID TURBINE BUILDING, INTAKE STRUCTURE, WAREHOUSE AND DIESEL GENERATOR ROOM CHILLED WATER SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.10 FIRE PROTECTION SYSTEM

The Millstone Nuclear Power Station Fire Protection Program Manual, (Reference 9.10-1), has been developed to ensure that any single fire will not cause an unacceptable risk to public health and safety, will not prevent the performance of necessary safe shutdown functions, and will not significantly increase the risk of radioactive release to the environment.

A Fire Protection Program has been established at Millstone Unit 2. This program establishes the fire protection policy for the protection of structures, systems, and components important to the safety of the plant and the procedures, equipment, and personnel required to implement the program. For details, refer to the Station Procedure for the Millstone Fire Protection Program (Reference 9.10-4).

The Fire Protection Program is under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position. Refer to the Fire Protection Program Manual.

9.10.1 DESIGN BASES

To achieve and maintain a high level of confidence for the Fire Protection Program, it has been organized and is administered using the defense in depth concept. The defense-in-depth concept assures that if any level of fire protection fails, another level is available to provide the required defense. In fire protection terms, this defense in depth concept consists of the following levels:

a. Preventing fires from starting.

b. Early detection of fires that do start and controlling and/or extinguishing them quickly so as to limit their damage.

c. Designing the safety system so that if a fire should start in spite of the fire prevention program, and if it should burn for a considerable period of time in spite of fire suppression activities, it will not prevent the safe shutdown of the plant.

None of these levels can be perfect or complete, but strengthening any one level can compensate in some measure for weaknesses, known or unknown, in the others.

9.10.2 SYSTEM DESCRIPTION

9.10.2.1 Site Water Supply System

The underground fire protection water supply consists of a 12 inch cement lined iron pipe in a loop arrangement around Millstone Units 1, 2, and 3. Post-indicating type valves in the piping loop permit partial pipeline isolation without interrupting service to the entire system.

The supply system services individually valved lines feeding fixed pipe water suppression systems (sprinklers, water spray, and standpipes) throughout the plant. Hydrants are located on an

approximate 250 foot spacing arrangement or are otherwise strategically located for fire fighting purposes. Hydrant hose houses, equipped for 2.5 and 1.5 inch hose stream service, are located near the hydrants.

The Millstone Station Fire Water Loop is supplied from two 250,000 gallon ground level suction tanks. The tanks are automatically filled through a domestic water line fed from the city water system. This line is a 12 inch city feed with a 10 inch meter, bypass line, and two backflow preventers. The city supply is capable of refilling the fire water tanks in eight hours. Valves on the interconnected tank suction lines provide the capability to manually isolate each tank in the event of failure of a tank or its piping system. The fire protection water supply system is independent of service and sanitary water piping systems.

Two fire pump houses contain the station's three fire pumps each rated at 2,000 gpm at 100 psi. All three pumps can take suction from either or both tanks and have individual connections to the underground supply system. All three fire pumps have separate control panels supplied from separate power supplies. Pump running and trouble signals for P-82 are available in the Unit 3 Control Room. All pumps and fire water tank level alarms are in the Unit 3 Control Room.

The Unit 3 (Building 123) Fire Pumphouse contains the electric fire pump (M7-8), the diesel driven fire pump (M7-7), and the 50 gpm electric jockey pump (M7-11). The Unit 2 (Building 124) Fire Pumphouse is a separate structure adjacent to Building 123 and contains the other electric fire pump (P-82). These pumphouses are adjacent and independent of each other with the exception of sharing a common barrier.

System operation is such that the 50 gpm electric jockey pump (M7-11) maintains system pressure by automatically starting when line pressure drops to approximately 105 psig and will run until pressure reaches 120 psig as indicated by a line pressure switch. An electric interlock between the jockey pump and the M7-7 and M7-8 pump exists which stops the jockey pump when either pump starts. A hydro-pneumatic tank is provided in the system to prevent short cycling of the jockey pump. The electric driven fire pump (P-82) is driven by an AC motor from the 480V load center Bus 22D. This pump is activated by a pressure switch set at 95 psig. In the event this switch or pump fails to operate and line pressure continues to drop, the electric pump (M7-8) is activated by a separate pressure switch set at 85 psig. This pump is driven by an AC motor powered from the MP3 480V Load Center Bus 32Q. In the event this switch or pump fails to operate, or system demand overwhelms this capacity and line pressure continues to drop, the Diesel Driven Fire Pump (M7-7) is activated by a separate pressure switch set at 75 psig. The diesel driven fire pump is electrically independent with its own self contained redundant battery system for starting. A battery charger is provided for maintaining the batteries charged.

The P-82 motor driven pump automatically stops after system pressure is maintained at 100 psi for 5 minutes. Once started, the M7-7 and M7-8 fire pumps remain in operation until manually stopped.

If a major fire in any location of the Millstone Unit 2 site should occur, the combined water tanks 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 of the three station fire pumps.

The site fire water storage tanks are provided with freeze protection via electric closed loop water heaters.

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.

9.10.2.1.1 Water Suppression Systems

1. Sprinkler and Waterspray Systems

   Sprinkler and waterspray systems, provided in various areas of the plant where in-situ combustible loading warrants such protection, have been designed using the guidance of National Fire Protection Association (NFPA) Standard Number 13, for the Installation of Sprinkler Systems or NFPA Standard Number 15 for Waterspray Fixed Systems.

   Water systems are provided in the following design arrangement:

   - Automatic wet-pipe sprinkler;
   - Automatic fixed waterspray; and
   - Automatic preaction sprinkler.

   The individual system details and general locations are indicated in the Fire Hazards Analysis (FHA) (Reference 9.10-2).

2. Automatic Wet Pipe Sprinkler Systems

   Wet-pipe sprinkler systems have closed sprinkler heads, and an alarm check valve or alarm flow switch. All systems are provided with an outside screw and yoke (OS&Y) isolation valve between the supply connections and system distribution piping.

3. Automatic Fixed Waterspray Systems

Fixed waterspray systems are of the automatic design. All systems have a deluge valve located between the supply header and the distribution piping. An OS&Y isolation valve or post indicating valve (PIV) is used on all systems. Upon actuation of the deluge valve, water flows into the distribution piping and discharges from all spray heads.

Automatic operation is initiated by a single zone heat detection circuit installed in the hazard area.

4. Automatic Preaction Sprinkler Systems

Preaction type sprinkler systems are of the automatic design. All systems have a deluge/preaction valve located between the supply header and the distribution piping. Where appropriate, distribution piping is supervised by monitoring air pressure. An OS&Y isolation valve or PIV is used on all systems. Upon actuation of the deluge/preaction valve, water flows into the distribution piping. Upon actuation of each individual sprinkler head, water is discharged. Automatic operation is initiated by single zone heat detection circuit installed in the hazard area.

9.10.2.1.2 Gaseous Suppression Systems

1. Carbon Dioxide Systems

The Main Generator Exciter enclosure is protected with a total flooding carbon dioxide system. This system is automatically discharged by activation of two heat detectors provided in the enclosure. The system can also be manually activated by use of a manual pull station. The carbon dioxide suppression systems are designed using the guidance of NFPA12.

2. Halon Systems

Total flooding Halon 1301 fire suppression systems are provided for the Old Computer Room (A-26) subfloor and the New Computer Room (A-27), Z1 and Z2 DC Switchgear Rooms.

The Halon suppression systems are designed using the guidance of NFPA 12A, Halon 1301 Suppression Systems.

9.10.2.1.3 Portable Suppression Capabilities

1. Hose Stream Coverage

Hose stream coverage is available to all fire areas of the plant from stand pipe connections to fixed 1.5 inch hose stations or by use of 2.5 inch diameter hose with gated wye connections available from outside hose houses.

Hose station locations are listed in the Millstone Unit 2 Technical Requirements Manual (TRM) (Reference 9.10-6).

2. Portable Extinguishers

Selection and placement of portable fire extinguishers are in accordance with the intent of the guidelines of NFPA Standard Number 10, Standard for Portable Fire Extinguishers. All extinguishers utilized are Underwriters Laboratories (UL) listed.

9.10.2.1.4 Fire Detection and Alarm Systems

The fire detection and alarm systems installed in the plant are designed in general compliance with NFPA Standard Number 72D, Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems.

Fire detection systems are used for early warning detection and in some cases may have the capability to actuate water suppression systems.

Detection devices consist of rate-of-rise, fixed temperature, line type, and rate compensated heat detectors, and smoke detectors. Smoke detectors employ either the ionization or photoelectric principle. Specific application of these detectors in each fire area is detailed in the FHA.

In general, the installation of detector units is in accordance with the intent of the guidelines set forth in NFPA Standard Number 72E, Standard on Automatic Fire Detectors.

Fire/smoke detectors, as with waterflow indicators, CO_2 and Halon actuation indicators and valve tamper devices are arranged to transmit signals to local alarm panels and a water suppression system control panel, if applicable. Actuation signals are also transmitted through the local alarm panels to control panels in the Control Room. Trouble signals for these devices are transmitted in a similar manner.

The alarm system also monitors other miscellaneous fire protection system (FPS) features such as CO_2 system trouble, Halon system trouble, and preaction sprinkler system air pressure.

9.10.2.1.5 Ventilation Systems and Smoke Removal

Removal of the products of combustion from any specific plant area requires the use of the normal plant ventilation system. Millstone Unit 2 relies mainly on power venting to satisfy the recommended smoke and heat venting guidelines set forth. No ventilation system was designed for the sole purpose of exhausting smoke or corrosive gases. In all areas where airborne activity could exist, the ventilation is monitored prior to release.

The ventilation and filtration systems of safety related or potential radiation release areas are discussed in detail for the cable spreading area, switchgear rooms, battery rooms, 480V load centers, containment, intake structure, auxiliary building, fuel handling area, auxiliary feedwater

pump room, control room, engineered safety features room, diesel rooms, and enclosure building in FSAR Section 9.9, "Plant Ventilation Systems".

9.10.3 SAFETY EVALUATION AND FIRE HAZARDS ANALYSIS

9.10.3.1 Evaluation Criteria

An evaluation of the overall Fire Protection Program as indicated by the FHA, found that the program does provide reasonable assurance that a fire will not cause an unacceptable risk to the public health and safety. The fire protection program accomplishes this by assuring a fire will not prevent the performance of necessary safe shutdown functions and will not significantly increase the risk of radioactive release to the environment. Therefore, the Fire Protection Program meets the basic requirements of General Design Criteria (GDC) 3 and 5. Appendix A to Branch Technical Position (BTP) APCSB 9.5-1, “Guidelines for Fire Protection for Nuclear Power Plants,” provides the implementing criteria for GDC 3 and gives the general guidelines used to review Millstone Unit 2. Appendix A to BTP APCSB 9.5.1 provides the guidelines acceptable to the NRC staff for implementing the following criteria:

a. GDC 3 (10 CFR 50, Appendix A) - Fire Protection.

b. Defense-in-Depth Criterion: For each fire hazard, a suitable combination of fire prevention, fire detection and suppression capability, and ability to withstand safely the effects of a fire is provided. Both equipment and procedural aspects of each are considered.

c. Single-Failure Criterion: No single active failure shall result in complete loss of protection of both the primary (fixed installed systems) and backup fire suppression capability (standpipe/extinguishers).

d. Fire Suppression System Capacity and Capability: Fire suppression capability is provided, with capacity adequate to extinguish any fire that can credibly occur and have no significant adverse effects on equipment and components important to safety.

e. Backup Fire Suppression Capability: Total reliance for fire protection is not placed on a single automatic fire suppression system. Appropriate backup fire suppression capability is provided in the form of portable fire extinguishers or hose stations.

f. Acceptability of Manual Fire Suppression: When it can be shown that safe-shutdown capability is independent of any credible fire, manual fire fighting capability is sufficient to protect safety-related systems.

In addition to the specific guidance of the BTP, the evaluation considered the adequacy of the Fire Protection Program on the effects of potential fire hazards throughout the plant based on sound fire protection engineering practices and judgments.

9.10.3.2 Fire Hazard Analysis Methodology

Fire Protection was evaluated by conducting an FHA of individual fire areas and fire zones within the plant.

A detailed description of the analysis method is given in the following summary:

1. Plant design features related to fire safety were determined. These include the overall plant layout, type, and location of combustible materials, type of construction and its fire resistant characteristics, fire detection, and fire suppression systems, separation distance, etc.

2. Areas containing equipment and components important to safety were identified. These areas and adjacent areas with fire hazard potential were subdivided into fire areas and zones within areas on the basis of existing fire barrier boundaries and other logical physical divisions or equipment groupings. For each fire area/zone, the following were determined:

    a. Total heat potential (Btu/ft^2) in the area/zone, assuming total combustion of cable insulation, oil, charcoal, and other identifiable combustibles including transients.

    b. Fire severity is determined by the material burned and its rate of burning. To evaluate the fire resistance needed for any fire barrier, a fire severity (duration) is developed for each area. Severity is measured in terms of temperature and fire duration. Once the total heat potential (Btu/ft^2) of an area or zone has been computed and has been corrected to be equivalent to wood, an equivalent fire severity may be determined.

        To determine an equivalent fire severity, Table 5-9B, Estimated Fire Severity for Offices and Light Commercial Occupancies, NFPA, Fire Protection Handbook, Fifteenth Edition, has been adopted for this analysis.

    c. Safety related equipment and safe shutdown equipment and systems in the area.

    d. Fire detection and suppression systems.

    e. Fire area/zone boundaries.

3. For each area/zone, the adequacy of existing fire detection and fire suppression systems was evaluated considering the combustibility of materials, potential ignition sources, and the concentration of combustible materials.

4. Plant features were evaluated that impact directly or indirectly on the plant fire brigade's ability to reach and effectively fight credible fires.

The specific analysis results for each fire area or zone are provided in the FHA.

The NFPA Standard Time-Temperature Curve is representative of the severity of a fire completely burning out of brick, wood-joisted building and its contents. The curve has been adopted by the American National Standards Institute (ANSI) and the American Society for Testing and Materials (ASTM) in ANSI/ASTM E 119, Standard Methods of Fire Tests of Building Construction and Materials. This curve has been used in the fire hazard analysis to evaluate fire duration.

9.10.3.3 Fire Areas and Zones

The plant arrangement is divided into fire areas and fire zones for the purposes of conducting the FHA. Fire areas are defined as plant areas bounded by fire rated assemblies of either three hour rated construction or lesser fire resistance as specifically identified and justified in the FHA. For this analysis a fire area may also be defined by physical separation of combustible materials and built in fire suppression features which will contain a fire to an area of origin. Examples of built in features include water curtains and gaseous fire suppression systems.

Fire zones are zones within fire areas that are used to more thoroughly describe the fire area. Fire zones may or may not be bounded by fire-rated construction.

For the purposes of the FHA, the fire areas and fire zones were divided in accordance with the original divisions in the 1977 FHA. These divisions provide a clear indication of the combustibles within a location and their effect on equivalent fire severity. The Millstone 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 Assurance 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.

As conditions warrant, remedial actions include compensatory measures to ensure adequate 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.

The responsibilities for Appendix R coordination (including related technical reviews and maintenance of Appendix R Compliance Reports), conduct of FHAs, engineering and design for FPSs, development of specific fire fighting procedures, inspection programs, maintenance, training (including formal Fire Brigade and Fire Watch training), drills and risk management are discussed in Reference 9.10-1.

9.10.5.2 Fire Brigade and Training

The Millstone Site Fire Brigade consists of a minimum of a Shift Leader and four Fire Brigade personnel. The Fire Brigade Shift Leader is qualified to provide direction and support concerning plant operations and priorities for their assigned unit. When a specific unit does not have a Fire Brigade Shift Leader, a Fire Brigade Advisor from that unit is required in addition to the five member Fire Brigade.

Members of the Fire Brigade are trained by the Nuclear Training Department. A list of Fire Brigade training programs is located in Reference 9.10-5.

Site Fire Brigade personnel are responsible for responding to all fires, fire alarms, and fire drills. To ensure availability, a minimum of a Shift Leader and four Fire Brigade personnel shall remain

in the owner controlled area and shall not engage in any activity which would require a relief in order to respond to a fire (e.g., continuous fire watch).

If assistance is needed to fight a fire, additional equipment and manpower is supplied by the off site local fire departments. Within a 5 mile radius of the plant there are numerous local volunteer fire companies. Letters of commitment to supply public fire department assistance have been obtained from these fire companies.

The Shift Leader coordinates the Site Fire Brigade activities, and ensures proper communications and coordination of support for the local fire department chief or officer in charge once on site, and other on site activities (e.g., Radiological Protection & Chemistry, and Security).

Nuclear Training coordinates with the Site Fire Marshal and periodically familiarizes local fire department personnel with the Station's layout and fire fighting equipment. The Site Fire Marshal coordinates with the Site Fire Brigade and all Unit Shift Managers, informing them of the status of the site fire protection equipment, should equipment become inoperable or unavailable.

Fire Protection drills shall be planned and critiqued by Nuclear Training and members of the management staff responsible for plant fire protection. Performance deficiencies of the Fire Brigade or of individual Fire Brigade personnel shall be remedied by scheduling additional training for the Site Fire Brigade or individuals.

9.10.5.3 Quality Assurance

The QA Program has been applied via the Fire Protection Program Manual to the FPSs, components, and programs providing fire detection and suppression capabilities to those areas of the plant that are important to safety.

9.10.6 SAFETY SHUTDOWN DESIGN BASES

Paragraph 50.48(b) of 10 CFR 50, which became effective on February 17, 1981, requires all nuclear plants licensed to operate prior to January 1, 1979 to comply with specific portions of Section III of Appendix R to 10 CFR 50, in addition to any previous fire protection Safety Evaluation Reports (SERs). One of the applicable sections, Section III.G, requires that fire protection features be provided for those systems, structures, and components important to safe shutdown. These features must be capable of limiting fire damage so that:

1. One train of systems necessary to achieve and maintain hot shutdown conditions from either the Main Control Room or the Emergency Control Station(s) is free of fire damage; and

2. Systems necessary to achieve and maintain cold shutdown from either the Main Control Room or the Emergency Control Station(s) can be repaired with 72 hours.

Section III.L of Appendix R and Generic Letters 81-12 and 86-10 provide additional guidance on the NRC Staff's requirements for this safe shutdown capability.

Section III.G established “design basis protective features” as criteria for the protection of safe shutdown capability and offered the following compliance alternatives to operating plant licensees:

1. Separation of redundant safe shutdown systems outside of containment by either three hour rated barriers, one hour rated barriers plus automatic suppression and detection, or 20 feet of horizontal separation free of intervening combustibles or hazards plus automatic suppression and detection (Section III.G.2, 10 CFR 50 Appendix R)

2. An alternate or dedicated shutdown system and fixed suppression (Section III.G.3, 10 CFR 50 Appendix R)

3. A combination of measures that are shown by analysis to provide an equivalent level of protection (10 CFR 50.48(c)[6] exemption request)

9.10.6.1 Safety Functions

Specific safe shutdown functions necessary to satisfy Appendix R criteria are as follows:

1. Reactor Reactivity Control Function

2. Reactor Coolant Makeup Control Function

3. Reactor Heat Removal Function

4. Process Monitoring Function

5. Supporting Functions

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.

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 off site power and availability of off site 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.

Safe Shutdown Systems

The following safe shutdown systems were identified and analyzed to determine which portions needed to remain operational and which needed to be isolated and remain isolated to meet the safe shutdown functions.

- Auxiliary Feedwater (AF)
- Reactor Coolant
- Main Steam
- Safety Injection
- Chemical and Volume Control

- Service Water
- Reactor Building Closed Cooling Water (RBCCW)
- Condensate Storage Tank (CST)
- Diesel Generator (DG)
- Shutdown Cooling
- Ventilation and Area Coolers
- Emergency AC and DC Electrical Distribution

Millstone Unit 2 10 CFR 50 Appendix R Compliance Review (Reference 9.10-3) describes the inherent features of the system which are required to achieve hot/cold shutdown and their performance during the design basis fire in each of the Appendix R fire areas.

Safe Shutdown Components

A list of safe shutdown components was developed based on the identified safe shutdown systems. Reference 9.10-3 provides a list of all safe shutdown equipment and instrumentation with their fire area locations.

9.10.6.3 Safe Shutdown Analysis

Based on the identified safe shutdown systems and components, a list of all associated power and control cables were identified and their routing documented on a fire area basis.

After completion of the above tasks and the evaluation of the various scenarios for shutting down the plant, a worst case failure analysis was developed for each fire area. The worst case fire in each area is considered a fire that disables all unprotected equipment and systems in that area. The results of the safe shutdown analysis for each fire area is provided in the safe shutdown matrix provided in Reference 9.10-3. If safe shutdown cables exist in a fire area, a circuit failure analysis was performed to determine the impact, if any, on safe shutdown. In some instances manual operator actions were used to mitigate or replace the function of cable assumed to be lost in the worst case fire. The matrices also address required operator corrective actions which have to be performed to allow shutdown for the worst case fire within any fire area. Reference 9.10-3 also provides a summary list of safe shutdown methods by fire area.

Circuit Failure Analysis

A circuit failure analysis was performed when equipment/components, or cables for a component, are in a fire area and:

1. The equipment/component is required to change position or be operable to perform its function, or

2. Spuriously the equipment/component actuates thereby preventing the system from performing its required function.

The circuit failure analysis evaluated the effects of fire induced failures (hot shorts, shorts-to-grounds, and open circuits) on that portion of the circuit that is located in the fire area in accordance with Generic Letters 81-12, 86-10, and Information Notice 84-09.

Associated Circuit Analysis

Associated circuits are considered as those cables (safety-related, nonsafety-related, Class 1E or Non-1E) that have physical separation less than that required by Section IIIG.2 of Appendix R and are associated with safe shutdown circuits in one of three ways:

- Common bus
- Spurious signals
- Common enclosure

To determine the interaction of associated circuits with shutdown systems, a fire area approach was utilized in which components and cables of components were identified for each fire area. Types of cables analyzed were power, control, and instrument cables.

Protection between associated circuits of concern and shutdown circuits are addressed below for each concern:

1. Common Bus Concern

   Safety and nonsafety buses for Millstone Unit 2 were reviewed for those circuits which had a common power source with shutdown equipment, to ensure that the power source supply of the shutdown circuits are electrically protected.

   a. For the power sources of concern, the trip characteristics of primary and secondary electrical protection devices were reviewed to insure proper coordination.

   b. The NRC's Generic Letter 86-10, question 5.3.8, "Short Circuit Coordination Studies," requests that the licensees consider the impact of high impedance faults on the shutdown capability. Simultaneous high impedance faults (below the trip point for the breaker on each individual circuit) were considered for all fire areas and area addressed in the shutdown matrices of Reference 9.10-3.

c. A program has been established to ensure that the overall coordination is maintained.

2. Spurious Operation Concern

Where equipment/components were subject to spurious operation which would affect the capability to safely shut down, various means of isolation and/or manual actions are utilized, such as prefire opening of circuit breakers, isolation of control circuits by transfer or isolation switches, or de-energizing and manually operating valves post fire. Valve 2-SI-651 has the disconnect switch in its power circuit locked open during plant heatup, while in Mode 4 prior to proceeding to Mode 3. The disconnect switch is not re-closed (except for periodic testing/maintenance) until the plant attains Mode 4 again. Valves 2-MS-65A and 2-MS-65B have their MCC Power enabled at lockable disconnect switches ("on" position to operate) during plant start up while in Mode 2 or 3 prior to proceeding to Mode 1. The disconnect switches are not put back to the "on" position (except for periodic testing/maintenance) or until the plant attains Mode 2, 3, 4, 5 or 6 again. Valve 2-MS-202 has a disconnect switch at its MCC that is "off" during plant start up while in Mode 4 prior to proceeding to Mode 3. The disconnect keys are kept in the control room and the disconnect switches are not put back in the "on" position (except for periodic testing/maintenance) until the plant attains Mode 4, 5 or 6 again.

Spurious Signal Concern, Current Transformers

A current transformer study was performed to determine the applications of where secondary circuits of current transformers were routed between fire areas and were susceptible to fire-induced open circuit failures. Two generic applications resulted from the review, in one case, current transformers were used for protective relaying, and in the other case, current transformers were used for remote reading of current loads. The type JCS-0 current transformers are utilized at both the 4160V level and the 480V level. Manufacturer data on testing of both the current transformers and their secondary circuit wiring has indicated that under open circuit conditions, caused by fire, damage will not result in a breakdown of either component.

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 makeup 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 10 CFR 50, 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 10 CFR 50 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 10 CFR 50 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 the requirement of Section III.O that requires the Reactor Coolant Pump Oil Collection System be designed to withstand the Safe Shutdown Earthquake (SSE). The exemption was based on the RCP lubricating oil system being seismically qualified with oil leaks not expected as a result of an SSE. Also, the oil collection system is designed such that dropping of components during a SSE should not cause loss of operability of safety-related equipment nor cause a fire, the systems would not degrade safety features within containment.

4. Exemption from 10 CFR 50 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 10 CFR 50 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 foot 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 10 CFR 50 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 3 cross tie capability.

7. Exemption from 10 CFR 50 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 3 cross tie capability. 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 10 CFR 50, Appendix R, III.J requirement to provide eight hour battery powered emergency lighting for access/egress to Unit 3 Alternate AC diesel generator (alternate AC source for fires in specifically identified Unit 2 Appendix R fire areas) based on the use of yard security lighting.

9.10.7 REFERENCES

9.10-1 Millstone Nuclear Power Station Fire Protection Program Manual.

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 Nuclear Fleet Administrative Procedure, CM-AA-FPA-100, “Fire Protection/Appendix R (Fire Safe Shutdown) Program”

9.10-5 Millstone Nuclear Power Station, Station Procedure TQ-1, Personnel Qualification and Training

9.10-6 Millstone 2 Technical Requirements Manual

FIGURE 9.10–1 DELETED

FIGURE 9.10–2 P&ID DOMESTIC WATER (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.10–2 P&ID DOMESTIC WATER (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.11 COMPRESSED AIR SYSTEM

9.11.1 DESIGN BASES

9.11.1.1 Functional Requirements

The compressed air system provides a reliable supply of clean, dry, oil free, air for the pneumatic instruments, operators and controls. In addition, the system provides the necessary air requirements for normal maintenance.

9.11.1.2 Design Criteria

The following criteria have been used in the design of the compressed air system:

a. The instrument air system shall have two subsystems, each capable of providing the total air requirements of the unit.

b. The instrument air system shall have suitable subsystem and component alignments to assure operation of a complete subsystem with associated components.

c. Capabilities shall be provided to assure instrument air system operation with on site power (assuming off site power is not available) or with off site electrical power.

d. The instrument and station air system shall be designed to permit periodic inspection of important components, such as air compressors, receiver tanks, dryers, after filters, piping, and valves to maintain the integrity and capability of the system.

9.11.2 SYSTEM DESCRIPTION

9.11.2.1 System

The compressed air system is shown schematically in Figure 9.11–1 and 9.11–2. The system incorporates two full capacity and one double capacity non lubricated compressors for instrument air, each having a separate inlet filter, aftercooler and moisture separator. The two full capacity Ingersoll-Rand instrument air compressors each have separate dedicated air dryer/filter assemblies and an air receiver. The double capacity instrument air compressor has a separate, dedicated air receiver and instrument air dryer and filter assembly. All three compressor subsystems supply the instrument air header which then divides into branch lines supplying the intake structure, condensate storage tank area, turbine building, main steam dump valves, containment structure, auxiliary building, enclosure building, diesel generator rooms, primary water storage tank area, and refueling water storage tank area.

One full capacity station air compressor with an inlet filter, aftercooler, and moisture separator discharges to the station air receiver. The receiver outlet header is divided into branch lines supplying the intake structure, condensate storage tank area, turbine building, containment structure, auxiliary building, primary water storage tank area, refueling water storage tank area, and the condensate polishing facility. Station Air also supplies Instrument Air to the condensate polishing facility via a cross-tie with a dedicated air dryer located in the condensate polishing facility. A system cross-tie between Unit 3 service air and Unit 2 station air headers is also provided. In addition, manual cross ties are provided which allow the instrument air compressors to be used to supplement the station air supply during periods of heavy use.

An emergency backup cross tie from the station air header automatically supplies air to the instrument air system if instrument air pressure falls below 85 psig. Local controls isolate station air loads when station air is acting as the emergency backup supply to the instrument air system.

A remotely operated manual cross tie within the containment can provide station air to the containment instrument air branch line if the line pressure drops below 80 psig. In addition, a third instrument air receiver is located in the containment structure to provide a 30 minute supply of air in the event that both station and instrument branch lines fail.

The design of the compressed air system is based on an estimated station air requirement of 600 scfm and a minimum instrument air requirement of 217 scfm.

Table 9.11-2 gives information on the gas receiver tanks with the exception that tank rupture is not considered because the receiver tanks have been designed, per ASME Section VIII, for pressures greater than the maximum supply pressures, and each individual tank is provided with an ASME Section VIII safety relief valve. As shown on the above-referenced table, all the gas receiver tanks are in compliance with Section 1910.169 of OSHA.

9.11.2.2 Components

The major components and associated fabrication and performance data are listed in Table 9.11-1.

9.11.3 SYSTEM OPERATION

9.11.3.1 Normal Operation

A continuous supply of instrument air is provided to regulate the controllers and controlled devices. Either of the following configurations is acceptable and can be considered normal operation: The double capacity instrument air compressor supplies both the instrument air and station air systems through available cross tie valves and the other instrument air compressors are on automatic standby, or the station air compressor and one instrument air compressor operate and the other instrument air compressors are on automatic standby.

9.11.3.2 Emergency Conditions

The station or instrument air system can be utilized to provide makeup air for the containment during the long term post-accident hydrogen purge operation (Section 6.6). The instrument and station air systems are connected to the containment air monitoring system containment penetrations (Figure 9.9–2). A globe valve and flow element on each connection provide manual flow regulation.

Air-operated containment isolation valves are designed to assume the safe position by spring action following a loss of instrument air pressure at the valve operators or electric power to the valve operator solenoid valves. Generally, containment isolation valves for nonessential process systems are designed to fail closed and remain tightly closed. Valves for essential processes are designed to fail open to assure proper system alignment. Valves which must close and be capable of opening as the hydrogen purge valves are provided with accumulator tanks to assure an air supply. Valves required to maintain a fixed position are designed to fail “as is.” Where identified, inaccessible valves that must be capable of operating after a design basis loss-of-coolant accident are also provided with backup air bottles to assure an air supply.

9.11.4 AVAILABILITY AND RELIABILITY

9.11.4.1 Special Features

The power supply for the compressor motors is the normal distribution system. The power supply for the full capacity instrument air motors is backed up by the emergency diesel generators with each compressor on a separate channel (Section 8.3). The double capacity compressor has nonvital power supplies and is not relied upon for emergency operation.

Instrument air is primarily used as motive power for valve actuation and is not used in any reactor indication, control, or protective circuit. The overall unit is designed to assure that valve failures occurring upon loss of air are consistent with the capability to maintain the unit in a safe condition.

9.11.4.2 Tests and Inspections

Each component is inspected and cleaned prior to installation in the system. Instruments are calibrated during testing, and automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during operational testing. The system is operated and tested initially with regard to flow paths, flow capacity and mechanical operability.

Flow orifices, thermometers and pressure gauges are provided for local system monitoring.

The compressed air system undergoes an acceptance test prior to startup. The test procedure is described in Chapter 13.

TABLE 9.11-1 COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION

Instrument Air Compressor - Mark Number F-3D

Quantity	One
Type	Rotary Screw, oil-free type
Capacity (acfm) at 115 psig	640 at 100 °F, 14.7 psia, 40 % RH
Design discharge pressure (psig)	125
Cooling medium	Air cooled
Motor	150 hp, 460 V, 3 phase, 60 HZ, variable speed
Seismic Class	II
Safety Relief Valve Setpoint	165 psig

Instrument Air Receiver - Mark Number T-51A

Type	Vertical
Design pressure (psig)	135 psig at +20 to 500°F
Actual volume (feet)	150
Code	ASME Section VIII
Seismic Class	II
Safety Relief Valves (2)	
Setpoint, Capacity	135 psig, 3196 cfm at 60°F each

Instrument Air Dryer - Mark Number T-52C/D

Type	Heatless - dual tower - purge type - desiccant
Quantity	370 lbs of desiccant per tower (2)
Capacity (acfm) at 115 psig	650 at 123°F, 100% R.H. inlet conditions
Outlet moisture content with saturated air inlet	-40°F DP at line pressure
Seismic Class	II
Safety Relief Valve Setpoint	135 psig
Design Pressure (psig)	150 psig at 450°F

TABLE 9.11-1 COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Instrument Air System

Air Compressor - Mark Number F-3E, F-3F

Type	Rotary-screw, oil free type
Quantity	2
Capacity (scfm)	237
Discharge pressure (psig)	100
Cooling medium	Air cooled
Motor	60 hp, 460 volt, 3 phase 60 Hz
Seismic Class	II

Air Receiver - Mark Number T-51

Manufacturer	Ingersoll-Rand
Type	Vertical
Quantity	1 per 2 compressors
Design pressure (psig)	125
Actual volume (feet)	160
Code	ASME Section VIII
Seismic Class	II

Piping and Valves

Seismic Class	II

Station Air System

Air Compressor - Mark Number F-2

Type	Horizontal nonlubricated reciprocating type
Quantity	1
Capacity (scfm)	630
Discharge pressure (psig)	100
Cooling medium	Water, 16.7 gpm

TABLE 9.11-1 COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Motor	150 hp, 460 volt, 3 phase, 60 Hz
Seismic Class	II

Aftercooler and Moisture Separator - Mark Number X-27

Type	Shell and tube
Quantity	1
Code	ASME Section VIII
Seismic Class	II

Air Receiver Mark Number T-50

Type	Vertical
Quantity	1
Design pressure (psig)	125
Actual volume (feet)	160
Code	ASME Section VIII
Seismic Class	II

Piping and Valves

Seismic Class	II

Instrument Air Dryer - Mark Number T-52E1/E2

Type	Dual tower, heated blower purge, dessicant, regenerative.
Dessicant/Vessel	43.25 inches of activated Alumina.
Capacity (scfm) at 100 psig	300 at 100°F inlet air temperature.
Outlet moisture content	-40°F DP at line pressure.
Purge Rate	24 scfm for one hour out of four hour cycle with cooldown feature enabled.
Seismic Class	II
Safety Relief Setpoint	165 psig

TABLE 9.11-1 COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Instrument Air Dryer - Mark Number T-52F1/F2

Type	Dual tower, heated blower purge, dessicant, regenerative.
Dessicant/Vessel	43.25 inches of activated Alumina.
Capacity (scfm) at 100 psig	300 at 100°F inlet air temperature.
Outlet moisture content	-40°F DP at line pressure.
Purge Rate	24 scfm for one hour out of four hour cycle with cooldown feature enabled.
Seismic Class	II
Safety Relief Setpoint	165 psig

Condensate Polishing Facility Instrument Air Dryer - Mark Number T-IAS-DRYER1

Type	Dual tower, heatless, desiccant, regenerative.
Dessicant/Vessel	40 inches of activated Alumina.
Capacity (scfm) at 100 psig	90 at 100°F inlet air temperature.
Outlet moisture content	-40°F DP at line pressure.
Purge Rate	24 scfm for one hour out of four hour cycle with cooldown feature enabled.
Seismic Class	Non-seismic
Safety Relief Setpoint	165 psig

Instrument Air Receiver - Mark Number T51E

Type	Vertical
Design Pressure (psig)	155 at 400°F
Actual volume (gallons / feet)	1060 / 142
Code	ASME Section VIII
Seismic Class	II
Safety Relief Valve Setpoint / Capacity	135 psig / 250 scfm

TABLE 9.11-1 COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Instrument Air Receiver - Mark Number T51F

Type	Vertical
Design Pressure (psig)	155 at 400°F
Actual volume (gallons / feet)	1060 / 142
Code	ASME Section VIII
Seismic Class	II
Safety Relief Valve Setpoint / Capacity	135 psig / 250 scfm

TABLE 9.11-2 GAS RECEIVER TANKS SAFETY EVALUATION

TANK NAME	TANK NUMBER	DESIGN PRESSURE (PSIG)	OPERATING PRESSURE (PSIG)	MAXIMUM GAS SUPPLY PRESSURE (PSIG)	LOCATION	PROTECTIVE MEASURES TO PREVENT TANK FAILURES	DEVIATIONS FROM OSHA SECTION 1910.169
Station air receiver tank	T-50	125	100	100	FSAR Figure 1.2–5 Zone (D-5)	Gas supply pressure is 25 psi less than design pressure of tank, tank has relief valve set at 125 psig.	None
Instrument air receiver tank	T-51	125	110	110	FSAR Figure 1.2–5 Zone (E-6)	Gas maximum supply pressure is 15 psi less than design pressure of tank, tank has relief valve set at 125 psig.	None
Containment instrument air receiver tank	T-89	125	100	100	FSAR Figure 1.2–6 Zone (E-8)	Gas supply pressure is 25 psi less than design pressure of tank, tank has relief valve set at 125 psig.	None

TABLE 9.11-2 GAS RECEIVER TANKS SAFETY EVALUATION (CONTINUED)

TANK NAME	TANK NUMBER	DESIGN PRESSURE (PSIG)	OPERATING PRESSURE (PSIG)	MAXIMUM GAS SUPPLY PRESSURE (PSIG)	LOCATION	PROTECTIVE MEASURES TO PREVENT TANK FAILURES	DEVIATIONS FROM OSHA SECTION 1910.169
Instrument air receiver tank	T-51A	135	120	120	FSAR Figure 1.2–5 Zone (C-12)	Gas maximum supply pressure is 15 psi less than design pressure of tank; tank has two relief valves set at 135 psig. Note - factory hydrostatic tested at 203 psig.	None
Instrument air receiver tank	T-51E	155	110	129	FSAR Figure 1.2–5 Zone (C-10)	Gas maximum supply pressure is 26 psi less than design pressure of tank, tank has relief valve set at 135 psig.	None
Instrument air receiver tank	T-51F	155	110	129	FSAR Figure 1.2–5 Zone (C-10)	Gas maximum supply pressure is 26 psi less than design pressure of tank, tank has relief valve set at 135 psig.	None

FIGURE 9.11–1 P&ID INSTRUMENT AIR SYSTEM COMPRESSORS F-3E AND F-3F

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.11–2 P&ID INSTRUMENT AIR SYSTEM COMPRESSORS F-3D

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.12 WATER TREATMENT SYSTEM

9.12.1 DESIGN BASES

9.12.1.1 Functional Requirements

The function of the common makeup water treatment system is to provide makeup, demineralized, deoxygenated water for Millstone Unit 2 and Unit 3 nuclear steam supply system (NSSS) and its auxiliary systems, and to provide demineralized makeup water to Millstone Unit 2 and Unit 3 secondary systems.

9.12.1.2 Design Criteria

The following criteria have been used in the design of the common water treatment system:

a. The vendor makeup water demineralizer section of the water treatment system shall provide up to 400 gpm to meet the demands of Units 1, 2, & 3.

b. The water treatment system shall be designed to permit periodic inspection of important components such as pumps, demineralizers, tanks, filters, valves and piping to ensure the integrity and capability of the system.

c. The water treatment components shall be designed to operate within the environment to which they are exposed.

9.12.2 SYSTEM DESCRIPTION

9.12.2.1 System

The water treatment system is shown schematically in Figures 9.12–1 and 9.10–2.

Domestic (city) water is supplied to vendor demineralization equipment through a back flow prevention valve to prevent chemical contamination of the domestic water supply. The common water treatment facility supplies deoxygenated water to the Unit 2 condensate storage tank via the separate eight inch supply header and to all other tanks in Units 2 and 3 via the cross tie header.

Primary make-up water is deoxygenated before it is transferred to the PWST.

To prevent freezing, the water temperature in the outdoor PWST is maintained at 50°F minimum by means of a circulation system which uses a primary circulation pump and a primary water heat exchanger.

Millstone Unit 2 NSSS and its auxiliary systems requiring demineralized, deoxygenated makeup water obtain it from the PWST. The demineralized, deoxygenated water is transferred from the PWST to the components requiring makeup water by one of two primary water transfer pumps (PWTP). During normal operation, one of the PWTP will always be in operation whether makeup

water is required or not. When makeup water is not required, 40 gpm will be recirculated back to the PWST.

The Primary Water Heat Exchanger return line to the PWST is a suction connection that provides a make-up water source alternative. This suction connection is a defense-in-depth design feature that is available for coping with an extended loss of AC power (ELAP) event. The location of this BDB PWST FLEX suction connection is shown on Figure 9.12–1.

The common water treatment system is sized to provide the makeup water requirements for Millstone Units 1, 2 & 3.

Millstone Unit 2 secondary systems requiring demineralized makeup water obtain it from the CST (Section 10.4.5.3). Millstone Unit 3 systems and other Millstone Unit 2 systems obtain makeup water via the cross-tie header.

The common water treatment system contracted capacity is 576,000 gpd (400 gpm) of demineralized water. The contracted capacity is based on normal start-up steam generator blowdown requirements of any one of the Units to meet chemistry specifications.

During periods when the makeup water requirements exceed the normal makeup water flowrates for either Unit, the common water treatment facility has the capability to divert additional water as required.

The design chemical analysis of demineralized water from the vendor water treatment system is as follows:

Silica, maximum (ppb)	10
Conductivity, maximum (umhos/cm)	0.1
Dissolved oxygen, maximum (ppm)	0.1
Sodium, maximum (ppb)	1
Chloride, maximum (ppb)	1
Sulfate, maximum (ppb)	2
Calcium, maximum (ppb)	2
Magnesium, maximum (ppb)	2
Aluminum, maximum (ppb)	10
Total organic carbon (ppb)	50

9.12.2.2 Components

A description of the water treatment system components is given in Table 9.12-1.

9.12.3 SYSTEM OPERATION

9.12.3.1 Normal Operation

During normal unit operation, continuous blowdown from both steam generators requires the common water treatment facility to provide makeup water to the CST. Makeup water flow is manually aligned to the CST, and can be concurrently aligned to any of the other tanks in Unit 2 or Unit 3.

If the makeup water requirement exceeds the normal operating flow rates for either Unit, such as chemistry holds during startup, the common water treatment facility has the capability to divert the additional water as needed.

The operation of the water treatment system is controlled and monitored with the following instrumentation:

a. Pressure indication on the discharge of each pump.

b. Level controllers, indicators, and high/low alarms on the PWST.

c. Temperature-indicating controller and high/low alarm on primary water storage.

d. Flow switch and flow controller on the primary water pump discharge.

9.12.4 AVAILABILITY AND RELIABILITY

9.12.4.1 Special Features

Utilization of vendor supplied and operated make-up water equipment allows for changes in that equipment to remain current with water treatment technology and changing water supply conditions. Key components are supplied with sufficient redundancy to allow for equipment breakdowns, and flexible connections between components allows for rapid change outs and versatility for changing system demands. Additionally, off-site demineralizer regenerations eliminates neutralization and discharge of regenerate waste.

Outside make-up water supply lines are insulated with redundant heat tracing to prevent line freeze-up.

The makeup water from the PWST is supplied to the various systems by redundant PWTPs.

Pressure vessels are provided with relief valves for overpressure protection.

A Back Flow Preventor (BFP) prevents contaminants from entering the domestic (city) water supply.

9.12.4.2 Tests and Inspections

The pressure vessels in the Unit 2 water treatment system are inspected and tested per American Society of Mechanical Engineers (ASME) Code Section VIII and are furnished with an ASME Code Stamp and Reports.

Each component is inspected and cleaned prior to installation into the system. The water treatment system undergoes an acceptance test prior to startup; the detailed test procedure is described in Chapter 13.

Major components of the system, such as pumps, tanks and heat exchangers, are accessible for periodic inspection during normal operation.

Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during preoperational testing.

The system is operated and tested initially with regard to flow paths, flow capacity and mechanical operability.

Local sample points are provided for obtaining samples for laboratory testing of the water quality during normal operation.

TABLE 9.12-1 WATER TREATMENT SYSTEM COMPONENTS

Domestic Water Storage Tank	
Manufacturer	Richmond Engineering Co
Type	Vertical
Quantity	1
Design pressure (psig)	15
Design temperature (°F)	100
Volume (gal)	275
Materials Shell and head	ASTM A-285 Gr C
Code	ASME Section VIII & IV
Seismic Class	II
Primary Water Storage Tank	
Manufacturer	Richmond Engineering
Type	Vertical
Quantity	1
Design pressure	Atmospheric
Design temperature (°F)	100
Net capacity (gal)	150,000
Material	SA-240, Type 304
Code	AWWA-D100
Seismic Class	II
Primary Water Circulation Pump	
Pump manufacturer	Ingersoll-Rand
Model	3 x 2 x 6 VOC
Type	Inline centrifugal
Quantity	1
Capacity (gpm)	100
Head (feet)	25
Net Positive Suction Head required / available (feet)	3 / 25
Materials	
Casing	ASTM A-296 Gr CF8M
Impeller	ASTM A-296 Gr CF8M
Shaft	ASTM A-276 - Type 316

TABLE 9.12-1 WATER TREATMENT SYSTEM COMPONENTS (CONTINUED)

Motor manufacturer	General Electric
Motor	1.5 hp, 460 V, 60 Hz, 1750 rpm
Codes	Motor: NEMA Pumps: Hydraulic Institute Standards
Seismic Class	II

Primary Water Storage Tank Heat Exchanger

Manufacturer	Perfex Corporation
Model	AE4
Type	Horizontal, U tube
Quantity	1
Design duty (Btu/hr)	540,000
Heat transfer area (square feet)	6.1
Design pressure (psig)	Shell side: 75 tube side: 75
Design temperature (°F)	Shell side: 350 tube side: 325
Material	
Shell	ASTM A-106 Gr B
Tubes	ASTM A-249 Gr TP304
Tube sheet	ASTM A-240 Type 304
Channel	ASTM A-312 Type 304
Codes	TEMA Class C ASME Section VIII
Seismic Class	II

Primary Water Transfer Pumps

Pump manufacturer	Ingersoll-Rand
Model	3 x 1-1/2 x 8 VOC
Type	Inline centrifugal
Quantity	2
Capacity P22A (gpm)	200
Capacity P22B (gpm)	188
Head (feet)	190
Net Positive Suction Head required / available (feet	10 / 37)
Materials	
Casing	ASTM A-296 Gr CF8M
Impeller	ASTM A-296 Gr CF8M
Shaft	ASTM A-276 Type 316

TABLE 9.12-1 WATER TREATMENT SYSTEM COMPONENTS (CONTINUED)

Motor manufacturer	General Electric
Motor	20 hp, 460 V, 60 Hz, 3 phase, 3550 rpm
Codes	Motor: NEMA Pump: Hydraulic Standards Institute
Seismic Class	II
Piping, Fittings and Valves	
Standard	ANSI B31.1.0, Standards for Power Piping
Seismic Class	II

FIGURE 9.12–1 P&ID DIAGRAM WATER TREATMENT SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

9.13 AUXILIARY STEAM SYSTEM

9.13.1 DESIGN BASES

9.13.1.1 Functional Requirements

The auxiliary steam system functions to maintain a suitable environment for equipment and plant operating personnel during normal operation and shutdown conditions. The Unit 2 auxiliary steam system receives steam from Unit 3 via a crosstie between the Units. Because the Unit 3 auxiliary steam system operates at 150 psig and the Unit 2 auxiliary steam system operates at 50 psig, a pressure reducing valve station, including isolation and relief valves are installed. In addition a condensate return line from the Unit 2 auxiliary feedwater surge tank is routed to the Unit 3 condensate system. Condensate is routed to the auxiliary boiler deaerator, when the Unit 3 auxiliary boilers are supplying auxiliary steam, or to the Unit 3 condensate surge tank when Unit 3 main steam is supplying auxiliary steam.

9.13.1.2 Design Criteria

The following criteria have been used in the design of the auxiliary steam system:

a. The system shall be designed to permit periodic inspection of important components, such as piping and valves to assure the integrity and capability of the system.

b. The components of the system shall be designed to operate in the environment to which exposed.

The following criteria have been used in the design of the Auxiliary Steam Detection and Isolation (ASDI) system (Section 7.10):

a. The system shall have a redundant, safety related independent detection and isolation subsystems.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system isolation function with onsite power (assuming offsite power is not available) or with offsite electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as temperature detectors and automatic isolation valves to assure the integrity and capability of the system.

f. The ASDI system shall be designed to permit appropriate periodic functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, performance shall be demonstrated of the full operational sequence that activates applicable portions of the detection and isolation system.

g. The components of the ASDI system shall be designed to operate in the most severe auxiliary steam line break environment to which it is exposed.

9.13.2 SYSTEM DESCRIPTION

9.13.2.1 System

The auxiliary steam and ASDI are shown schematically in Figure 9.13–1.

The auxiliary steam system is provided for building heating and freeze protection for outdoor water storage tanks. The site fire water storage tanks are provided with electric closed loop water heaters.

Steam for the auxiliary steam system is provided via a crosstie connection to Unit 3. Because the Unit 3 auxiliary steam system operates at 150 psig and the Unit 2 auxiliary steam system operates at 50 psig, a pressure reducing valve station, including isolation and relief valves are installed. The pressure reducing valve station consists of two pressure reducing valves installed in parallel and supplied with upstream and downstream isolation valves. The isolation valves are located at the crosstie between the Unit 3 auxiliary steam system supply to Unit 2. The pressure reducing valve station is located in the Condensate Polishing facility in line 3-ASS-008-15-4 downstream of isolation valve 2-ASS-3. This pressure reducing valve station regulates Unit 3 auxiliary steam system mass flow to Unit 2. The pressure reducing valve station is comprised of a two inch pressure reducing valve, four inch pressure reducing valve, isolation valves, pressure sensing line, and a dual operated controller. This configuration allows the two inch valve to regulate auxiliary steam system mass flow from between 0 and 8,999 lb/hr as long as the downstream pressure sensing line indicates an operating pressure of 50 psig or higher. When the downstream operating pressure falls below 50 psig, the four inch pressure valve will modulate to allow up to the additional required 35,682 lb/hr auxiliary mass flow required for Unit 2 use.

A safety relief valve located downstream of the pressure reducing valve station provides downstream pressure protection for Unit 2 system piping and connected equipment.

In addition a condensate return line from the Unit 2 auxiliary feedwater surge tank is routed to the Unit 2 condensate system. Condensate is routed to the auxiliary boiler deaerator, when the Unit 3 auxiliary boilers are supplying auxiliary steam, or to the Unit 3 condensate surge tank when Unit 3 main steam is supplying auxiliary steam.

9.13.2.2 Auxiliary Steam Detection and Isolation System

The ASDI System is described in Section 7.10.

9.13.3 SYSTEM OPERATION

9.13.3.1 Normal Operation

The auxiliary steam system is manually initiated by operator action from a locally mounted control panel.

Operation of the auxiliary steam system is classified as non-nuclear safety consistent with ANSI/ ANS 59.2 1985, Safety Criteria for HVAC Systems Located Outside Primary Containment.

9.13.4 AVAILABILITY AND RELIABILITY

9.13.4.1 Tests and Inspections

The ASDI system is incorporated with provisions for online testing (Section 7.10).

FIGURE 9.13–1 P&ID AUXILIARY STEAM AND CONDENSATE (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.13–1 P&ID AUXILIARY STEAM AND CONDENSATE (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.13–1 P&ID AUXILIARY STEAM AND CONDENSATE (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 9.13–1 P&ID AUXILIARY STEAM AND CONDENSATE (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

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Safety Analysis Report

Chapter 10: Steam and Power Conversion System

CHAPTER 10—STEAM AND POWER CONVERSION SYSTEM

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NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

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10.4–2	P&ID Condensate System (25203-26005 Sheet 1)

CHAPTER 10 – STEAM AND POWER CONVERSION SYSTEM

10.1 SUMMARY DESCRIPTION

The components of the steam and power conversion system are designed to electromechanically produce electrical power using steam from the steam generators. The steam is condensed, and returned to the steam generators as heated feedwater. The gaseous, dissolved and particulate impurities are maintained within the level acceptable to the steam generators.

The power conversion system includes the turbine generator, main condenser, air ejector and steam packing exhauster, turbine steam dump and bypass system, and the feedwater pumping and heating system. The heat rejected to the main turbine condenser is transferred to the circulating water system.

Steam is generated in two generators and supplied to the high pressure section of the turbine. Steam leaving the high pressure turbine passes through moisture separators and reheaters prior to entering the low pressure sections of the turbine. A portion of the turbine steam is extracted for feedwater heating. The moisture separator drains, reheater drains, and drains from the top three highest pressure feedwater heaters are pumped into the feedwater stream. The drains from the remaining lower three feedwater heaters are cascaded to the condenser.

Steam exhausted from the low pressure turbines is condensed and deaerated in the condenser. The condensate pumps take suction from the condenser hotwell, delivering the condensate through the turbine steam packing exhauster, the condensate polishing demineralizers, the air ejector condenser, and five stages of low pressure feedwater heaters to two turbine-driven steam generator feedwater pumps whose suction and discharge are headered together. Steam generator feedwater pumps discharge feedwater through high pressure heaters into each steam generator. The condensate and feedwater heating equipment is duplicate, half-capacity and arranged in parallel trains from the air ejectors to the steam generators.

The turbine cycle utilizes all the steam being generated by the steam generators. The turbine steam dump and bypass system is provided to discharge steam directly to the condenser during load transient and turbine trip. If the condenser is not available, steam is discharged to the atmosphere through atmospheric dump valves and/or main steam safety valves.

The Steam Generator blowdown liquid is normally discharged to the circulating water outlet, and approximately one-third of the steam generator blowdown liquid is flashed to steam and is discharged to the environment via the Unit 2 steam generator blowdown tank vent on the enclosure building roof. When in use, the blowdown quench tank condenses the flashed steam, and non-condensibles are vented to the gaseous waste processing system. Steam generator blowdown from the sample line prior to the blowdown tank is continuously monitored for radioactivity. Blowdown stop, blowdown tank drain, and blowdown quench tank drain valves are closed when high radioactivity is detected. Residual quantities in the blowdown system can be manually diverted to the radioactive waste processing system.

TABLE 10.1-1 DELETED

FIGURE 10.1–1 DELETED BY FSARCR 03-MP2-007]

FIGURE 10.1–2 DELETED BY FSARCR 03-MP2-007

10.2 TURBINE GENERATOR

10.2.1 DESIGN BASES

10.2.1.1 Functional Requirements

The turbine generator converts steam energy to mechanical energy and mechanical into electrical energy. The turbine generator is capable of receiving steam with up to 2715 MW thermal energy from the steam generators. The total energy available is converted to 935 MW electrical energy, and the remainder is rejected to the condenser. A closed regenerative turbine cycle heats the condensate and feedwater and returns it to the steam generators.

10.2.1.2 Design Criteria

The following criteria have been used in the design of the turbine-generator:

a. The turbine generator shall have the capacity to utilize all the steam produced by the steam generators at full power except that required for auxiliary use.

b. The system shall have suitable means of controlling the speed, load, pressure and flow for startup, normal operations and all emergencies such that plant equipment is operated safely and in accordance with the prescribed mode.

c. The system shall be designed to permit periodic testing of the important components such as main steam stop and control valves, combined intermediate valves, extraction no-return valves, and emergency overspeed trip systems. See Section 10.2.4.2 for a detailed description.

10.2.2 SYSTEM DESCRIPTION

10.2.2.1 System

The General Electric turbine generator consists of the turbine, generator, exciter, controls, moisture separator/reheaters, turbine lubricating oil, gland seal, stator system winding cooling water system, generator hydrogen sealing system and electrohydraulic control system. The moisture separation package and reheat tube bundles for the moisture separator/reheaters have been replaced with equipment manufactured by the Senior Engineering Company. The General Electric pressure vessels have been retained.

The 1,800 rpm turbine includes one double flow high pressure turbine and two double flow low pressure turbines. Exhaust steam from the high pressure turbine passes through two parallel moisture separators/reheaters before entering the two low pressure turbines. The low pressure turbines exhaust steam to the main turbine condenser.

The turbine generator utilizes an electrohydraulic control (EHC) system which governs the speed, load, and flow for startup, operation and shutdown. The EHC unit trips the turbine on any of the

conditions listed in Section 7.4.6.3.3. Low hydraulic oil pressure signals are supplied from the turbine EHC system to the reactor protective system to trip the reactor. The turbine first-stage pressure is fed to the reactor regulating system as a load reference.

The turbine lubricating oil system supplies lubrication for the bearings. A bypass stream of the turbine lubricating oil flows continuously through an oil conditioner at a rate sufficient to process the oil inventory once every six hours.

The generator shaft is provided with oil seals to prevent hydrogen leakage. The generator field excited by an alternator located on the stub shaft of the generator.

The turbine generator supervisory instrumentation is provided for operational analysis and malfunction diagnosis.

10.2.2.2 Components

The major system components and associated performance data are listed in Table 10.2-1.

10.2.3 SYSTEM OPERATION

The turbine generator is designed to normally operate as a base-loaded machine.

10.2.3.1 Startup

The turbine generator is started at 900 psia (controlled) steam generator pressure. Startup times and initial speeds are functions of the turbine metal temperature. When the machine is at rated speed, it is synchronized with the electrical 345 kV network and then manually connected to the system.

10.2.3.2 Normal Operation

The turbine generator is operated and controlled by the EHC system which combines the turbine generator speed error signals with the generator load signal to produce the steam flow demand. The steam flow demand accurately positions the turbine steam flow control valves to maintain the required speed and thereby satisfy the generator load. The turbine generator is capable of accommodating reactor maximum maneuvering limitations of a power change at a rate of five percent of full power per minute or up to a step change of 10 percent of full power.

10.2.4 AVAILABILITY AND RELIABILITY

10.2.4.1 Special Features

The turbine generator is equipped with an electrohydraulic control system. It is highly reliable and employs components and subsystems of proven high reliability and a completely redundant speed control subsystem (including speed pickups and logic). Logic is processed in electronic and hydraulic channels.

Valve opening actuation is provided by a 1500 psig hydraulic system which is totally independent of the bearing lubrication system, and utilizes a fire resistant fluid. Valve closing actuation is provided by springs and steam forces upon the reduction or relief of fluid pressure. The system is designed to respond to a loss of fluid pressure for any reason, and leads to turbine inlet valve closing and consequent turbine generator shutdown.

Because of the extreme importance of guarding against excessive overspeed, three lines of defense are provided. These consist of:

a. First, during normal operation turbine overspeed is precluded by the governing action of the electro-hydraulic control system by modulating the control and intercept valves. Speed sensing for this control system is provided by three magnetic pickups in conjunction with a toothed wheel on the main turbine shaft.

b. The second line of defense against turbine overspeed is the primary overspeed trip system, utilizing the same three magnetic pickups used to provide the control system governing action.

c. The third line of defense is provided by the emergency overspeed trip system. This is an independent electrical tripping function in which a set of three magnetic pickups, independent from the primary speed signal magnetic pickups, sense the speed of the toothed wheel on the main turbine shaft. The primary overspeed trip and emergency overspeed trip systems constitute two separate and independent means of protecting the turbine against an overspeed condition.

10.2.4.1.1 Overspeed Protection Inherent in Normal Operating Control System

Under normal operation, turbine overspeed is precluded by the governing action of the electrohydraulic control system. Speed sensing for this control system is provided by three magnetic pickups in conjunction with the toothed wheel on the main turbine shaft. The median value of the three speed signals is used for control and also for the primary overspeed protection logic. The circuitry and software for these control signals is isolated from, and independent of, the emergency overspeed trip circuitry and software. Failure of two of the three primary speed control signals results in a turbine trip.

10.2.4.1.2 Primary Overspeed Trip

The turbine protective system consists of a highly reliable trip manifold assembly along with the EHC system software and hardware. The trip manifold assembly consists of two independent hydraulic circuits arranged in parallel. Each circuit includes three Emergency Trip Devices (ETDs) arranged hydraulically for two-out-of-three voting logic. The ETD solenoids, which are de-energized to trip, can be de-energized by either the primary or emergency overspeed protection systems using independent circuits. If two ETDs in the same hydraulic circuit are de-energized, the EHC fluid trip supply header is isolated and the emergency trip system header is depressurized.

The primary overspeed trip system utilizes the primary speed pickup signals. The median value of the three signals is compared to the overspeed trip setpoint of 109 percent of rated speed. When the setpoint is reached, the primary trip relays de-energize all six ETDs in the trip manifold assembly, removing emergency trip system hydraulic pressure. This causes the fast closure of the main stop valves and the slower control valves which are in series in the high pressure stage turbine inlet. Also, the trip system pressure removal causes the rapid closure of the intercept and reheat stop valves which are in series in the inlet to the low- pressure stage of the turbine.

10.2.4.1.3 Emergency Overspeed Trip

The emergency overspeed trip system utilizes the emergency speed pickup signals. The median value of the three signals is compared to the emergency overspeed trip setpoint of 109.5 percent of rated speed. When the setpoint is reached, the emergency trip relays de-energize all six ETDs in the trip manifold assembly, thereby removing the emergency trip system hydraulic pressure and activating the fast closure of all the turbine steam valves previously mentioned.

The emergency overspeed trip system magnetic pickups, I/O modules, software and processors are independent from the components used for control and primary overspeed protection.

10.2.4.1.4 Testing

Testability during operation is provided for both the control system and the two overspeed protection systems. The test features provide coverage of the initiating, tripping, and controlling devices.

10.2.4.1.5 Valves

Two valves in series are used in the high pressure stage inlet to the turbine.

a. Control valves are normally controlled by the redundant speed control system and (tripped) closed rapidly when pressure in the emergency trip fluid system is removed by the redundant trip valves.

   The operation of each control valve, including the operating and fast closing devices, can be tested during normal operation.

b. Main stop valves are held open by the hydraulic pressure of the emergency trip fluid system and tripped closed rapidly upon removal of the pressure.

   The operation of each stop valve including its fast closing device can be tested during normal operation.

Stop valves of the steam-sealed design have been used on over 650 General Electric steam turbines since 1948. There have been no reports of a main stop valve failing to close, when required, to protect the turbine. Impending sticking has been disclosed by means of testing so that a planned shutdown could be made to make necessary corrections. This almost always involves

removal of the oxide layer which builds up in the stem and bushing. This oxide buildup does not occur on a low-temperature nuclear application.

Combined stop and intercept valves are furnished to prevent the energy stored in reheaters or moisture separators from accelerating the turbine generator to excessive overspeed. These two independently operated valves are arranged in series in one valve body.

a. The intercept valves are normally wide open but are closed by the speed control system upon a moderate speed increase and are tripped closed rapidly upon removal of the pressure in the emergency trip fluid system.

   The intercept valves are also provided with a stem seal. Each intercept valve, including its fast closing devices, can be tested during normal operation.

b. Reheat stop valves are normally open but are closed rapidly upon removal of the pressure in the emergency trip fluid system. They are also provided with a steam seal. Each reheat stop valve, including its fast closing devices, can be tested during normal operation.

Thus, there are two independent valves for defense against overspeed in each steam admission line to the turbine. The normal speed control system modulates one of them to prevent overspeed and the overspeed trip systems close both of them on a higher overspeed. All valves are testable during operation and the fast closing feature of any valve is fully operative while the valves are being tested.

Where necessary to provide adequate overspeed protection resulting from the energy stored in extraction lines, positive closing nonreturn valves are fitted and actuated indirectly by the emergency trip fluid system. Station piping, heater, and check valve systems are reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

The described design, inspection and testing features adequately preclude the possibility of a destructive overspeed condition from occurring.

10.2.4.2 Tests and Inspection

Equipment, instruments and controls are regularly inspected to ensure proper functioning of the system.

a. Test the main stop valves and combined intermediate valves fully closed by sequence testing at the EHC control panel.

b. Close extraction check valves, equipped with air-operated closing mechanism, part way by operating the test levers.

c. Test the fully closed control valves by sequence testing at the EHC control panel.

d. Test the alarm on the oil tank level gage.

e. Check the hydraulic thrust wear detectors.

f. Test the overspeed trip systems operation by using the controls on the EHC panel.

g. Deleted by MP-PACKAGE-FSC-MP2-UCR-2013-013.

h. Test power load unbalance circuits.

i. Deleted by MP-PACKAGE-FSC-MP2-UCR-2013-013.

j. Deleted by MP-PACKAGE-FSC-MP2-UCR-2013-013.

k. Lift pump test.

l. Check automatic starting of turbine motor-driven oil pumps.

m. Deleted by MP-PACKAGE-FSC-MP2-UCR-2013-013.

n. Check main stop and control valve tightness.

o. Test automatic pump starting.

p. Deleted by MP-PACKAGE-FSC-MP2-UCR-2013-013.

The moisture separator reheaters are tested periodically for tube leaks.

The detectable flaw size in the rotating members of the turbine generator is several orders of magnitude smaller than any critical crack which might develop from normal operation and could possibly result in a rotor failure.

10.2.5 BULK HYDROGEN STORAGE FACILITY

The bulk hydrogen storage facility is located outside, on the west side of Unit 2 Turbine building. A firewall is provided between the Turbine building and the Hydrogen Storage Facility. The cylinders are isolated from personnel by a chain link fence. “No Smoking” signs and “Hydrogen Flammable Gas - No Open Flames” signs are posted per NFPA requirements. The H_2 bulk supply header is run above ground from H_2 bulk storage area to the Unit 2 Turbine Building west wall. An excess flow check valve is installed on the hydrogen supply piping outside the Turbine building. The H_2 distribution headers inside Unit 2 are run as follows:

1. Headers are located to prevent physical damage to pipe.

2. Headers are located as safe distances from equipment that present a fire hazard to H_2.

3. Headers are run through well-ventilated areas.

4. H_2 piping is provided with guard pipes.

The equipment supplied with H_2 is the volume control tank and the main turbine generator. The following protective measures are provided to prevent fires and explosions during operation:

1. Volume Control Tank

   During normal operation N_2 and H_2 are supplied to the volume control tank pressure-reducing valves at a maximum pressure of 75 psig. If a tank leak occurs there is adequate ventilation to exhaust the H_2 to the atmosphere. During purging, N_2 is used to purge that tank to the gaseous waste processing system.

2. Main Turbine Generator

   During normal operation, H_2 at 60 psig is used to cool the turbine generator rotor. To prevent H_2 from leaking through the generator shaft seal glands into the turbine building, a shaft oil sealing system is provided. During purging of the H_2 from the generator, CO_2 is used to prevent fires and the mixture of H_2 and CO_2 is exhausted through a vent line to the outside atmosphere.

TABLE 10.2-1 TURBINE GENERATOR COMPONENT DESCRIPTION

Turbine

Steam generator power, rated design/(MWth)	2715
Throttle steam pressure, rated design (psia)	870
Main steam moisture content, max (%)	2.0
kW output, rated-design	935, 338
Makeup (%)	0
Turbine backpressure, (inches Hg abs)	2.0
Points of extraction	6

Note: The MP2 original licensed power was 2607 MWth Steam Generator Power. License up-rating took place during 1979 to stretch power of 2715 MWth Steam Generator Power.

Generator

Rating (kVA)	1,011,000
Power factor	0.9
Voltage (volts)	24,000
Hydrogen pressure (psig)	60

10.3 MAIN STEAM SUPPLY SYSTEM

10.3.1 DESIGN BASES

10.3.1.1 Functional Requirements

The main steam system (MSS) will perform the following functions:

a. Deliver steam from the steam generators to the turbine generator from warmup to valve wide open flow and pressure.

b. Provide steam for turbine gland seals, steam jet air ejectors (SJAE), moisture separators reheaters, and steam generator main and auxiliary feedwater pump (AFP) turbines.

c. Transfer heat generated by the nuclear steam supply system (NSSS) to the condenser or atmosphere in the event the turbine generator is out of service.

10.3.1.2 Design Criteria

The following design criteria have been used in the design of the main steam supply system:

a. The steam line leaving each steam generator is equipped with isolation valves located outside the containment. The isolation valves are designed to function so as to isolate a steam generator in the event of a rupture of the steam piping at any point, and to maintain at least one steam generator as a heat sink to remove reactor decay and sensible heat.

b. Each main steam line is provided with spring-loaded safety valves upstream of the isolation valves. The total relieving capacity is in excess of 100 percent steam flow at 2700 MWt rated thermal power. The safety valves discharge to the atmosphere.

c. A steam generator AFP turbine is supplied with steam from either main steam line upstream of the isolation valves.

d. The main steam supply system is provided with an automatically actuated steam dump system and turbine bypass to control steam pressure (hence, reactor coolant temperature) at hot standby zero load operation and to remove reactor coolant system (RCS) stored energy following a turbine trip. Atmospheric dump valves (ADV) from each steam generator are provided to limit or control secondary pressure whenever the condenser is out of service. Either steam dump system to the condenser or the atmospheric dump is capable of cooling the plant to the point of shutdown cooling manual initiation.

e. The main steam piping, from the steam generators up to and including the main steam isolation safety valves and steam dump to atmosphere valves, is designed to withstand a Class I seismic disturbance.

f. The main steam piping from the main steam isolation valves (MSIV) to the turbine stop valves is designed to with stand dynamic loading resulting from rapid closure of the turbine stop valves.

An analysis was performed on the 34 inch Atwood and Morrill main steam isolation swing check valve under the dynamic loads associated with postulated steam line breaks.

The valve disc kinetic energy was calculated by determining its velocity at the time of impact with the seat. The forces acting on the disc due to steam pressure were integrated with time to determine the terminal velocity. The pressures were calculated assuming a full area break on the main steam line under maximum flow conditions. The flow conditions were evaluated at different degrees of disc closure and the effects of choked flow were considered.

The inner portion of the disc and the point of impact on the body casting go plastic during impact. However, the zones of plastic deformation at any instant of time are surrounded by a sufficient amount of material in the elastic stress range to prevent failure.

The stresses were calculated using the SAAS computer program. The SAAS was a state-of-the-art program at the time the analysis was performed. The SAAS program is a finite element code that calculates the time history of stresses due to impact on an elastic basis.

Although this analysis proved that a failure of the MSIV disc would not occur, several modifications were made by Atwood and Morrill to preclude the possibility of local surface failures at the farthest point from the disc arm. The modifications consisted of increasing the seating area and a redesign of the disc from a flat configuration to a shallow spherical shell configuration.

10.3.2 SYSTEM DESCRIPTION

10.3.2.1 System

The main steam supply system is shown in Figure 10.3–1. The main steam piping consists of two parallel 34 inch OD lines and terminates at the 34 inch cross tie header which manifolds to the four turbine stop valves.

Each steam line has eight spring-loaded safety valves, set to sequentially discharge to the atmosphere. The spring-loaded safety valves have setpoints ranging from 985 psig to 1035 psig, and the total relieving capacity for all valves on all of the steam lines is 12.7×10^6 lb/hr, which is 108 percent of the total secondary steam flow of 11.8×10^6 lb/hr at 100 percent rated thermal power. See Section 4.3.2 for more details.

The atmospheric steam dump system consists of one automatically actuated dump valve for each steam generator which exhausts to the atmosphere. The total atmospheric dump system capacity is 15 percent of 2700 MWt rated thermal power. The turbine bypass and dump-to-condenser system consists of four automatically actuated valves which exhaust to the condenser. It has a total capacity of approximately 40 percent of the 2700 MWt rated thermal power. The system control valves are arranged for automatic operation but can be manually operated. More details may be found in Section 7.4.5.

The MSIV assemblies are located outside of and adjacent to the containment building. Each consists of an air-operated trip valve adjacent to a free swinging, reverse flow check valve and a motor-operated bypass valve.

Each air-operated trip valve is equipped with a separate auxiliary air accumulator to assure an air supply for its operation if the normal air supply fails.

A pressure switch alarms in the control room in the event of loss of air supply to the accumulator. The mechanism has a spring-assist feature which closes if there is a loss of air in the accumulators or if it is vented by a solenoid pilot. The trip valve component of the assembly protects the steam generator and reactor from damage in the event a rupture occurs in the main steam line downstream of the isolation valve.

Closing of the isolation valve within a maximum of six seconds after a command signal is initiated prevents rapid flashing and blowdown of water stored in the shell side of the steam generator, thus avoiding a rapid uncontrolled cooldown of the RCS. As an auxiliary function, the isolation valves also prevent the release of the contents of the secondary sides of both steam generators in the event of a rupture in one main steam line upstream of the valves. During normal operation these valves remain open. Low steam generator pressure or high containment pressure causes a steam generator isolation signal to energize the closing mechanism of the valves to stop the steam flow through the main and bypass valves.

Four pressure transmitters on each steam generator are monitored by four independent bistables. Upon low pressure in either steam generator and 2-out-of-4 coincidence bistable trip signals, both air-operated MSIVs are closed.

Automatic closure of the air-operated MSIVs via the low steam generator pressure signal can be manually blocked to permit orderly startup and shutdown of the unit (see Section 7.3). The high containment pressure signal cannot be blocked.

One 4-inch line is provided from each main steam line upstream of the MSIV to feed the AFP turbine.

Each steam supply line from the respective main steam line to the AFP turbine is equipped with a check valve and a remote (main control room) manual motor-operated gate valve for isolation to prevent the blowdown of the isolated steam generator in the event of a steam line break at the other steam generator. Each of the remote, manual, motor-operated gate valves has an upstream drain line with a manual isolation valve. This valve can be opened for draining the upstream side

of a gate valve while an operator is in attendance. These valves will otherwise be maintained in the “locked closed” position while containment integrity is required.

Seismic Class I design requirements are placed on the main steam line, bypass piping, branch lines to auxiliary feedwater pumps and to the atmospheric dump valves, up to and including the isolation trip valve outside the containment. This includes the piping up to and including the main steam safety valves, the atmospheric dump valves and the steam generators. All downstream components are designed to seismic Class II requirements.

10.3.2.2 Components

Table 10.3-1 describes the major components of the main steam system.

10.3.3 SYSTEM OPERATION

10.3.3.1 Startup

Steam produced by the steam generator is bypassed from the turbine generator to the condenser to maintain steam generator pressure and temperature at 900 psia and 532°F. Steam flow is increased according to the turbine metal temperature-steam temperature differentials. Steam flow to the steam generator feed pump (SGFP) turbines is changed from high to low pressure and the condenser air removal mode changed to SJAE operation from mechanical pump operation, according to moisture separator pressure and main steam availability, respectively.

10.3.3.2 Normal Operation

Main steam is led to the turbine stop valve inlet headers from each steam generator through separate 34-inch OD main steam lines to the turbine with no steam bypassed and the generator producing steam as required by the turbine generator. Feedwater flow is controlled by the feedwater control valve constant differential characteristic which adjusts feedwater flow; therefore, steam flow is according to steam demand.

10.3.3.3 Emergency Condition

During a reactor-turbine trip from full power, the four (4) steam dump to condenser valves (CDVs) and the two (2) main steam dump to atmosphere valves (ADVs) are fully open within four (4) and five (5) seconds, respectively, to avoid lifting of the safety valves. These limiting stroke times include any control circuit processing time. In the event of a loss of condenser vacuum, the steam dump to condenser valves close automatically. The ADVs open at a preset steam pressure of 920 psig to exhaust the steam generated by decay heat. The atmospheric dump capacity, equivalent of 15 percent of 2700 MWt (rated thermal power), is sufficient to cool the RCS to 300°F when shutdown cooling is initiated. A flow equivalent to 55 percent of 2700 MWt is sufficient to prevent lifting of the main steam safety valves. As the RCS approaches the shutdown cooling temperature, the steam pressure also decreases and if the bypass valve capacity is not sufficient to continue the desired rate of cooldown, the operator can partially open the steam dump to condenser or the ADVs to obtain the desired rate of cooldown.

Under normal operation, the ADV will not open. A steam dump valve is assumed to open inadvertently for the accident analysis Sections 14.1.3 and 14.1.4 address increase in steam flow events that bound the inadvertent opening of an ADV. The inadvertent operation of these valves is unlikely.

Assuming a double-end break of an individual auxiliary steam supply line upstream of the common header and check valve with the reactor at rated thermal power (2700 MWt), (see Figure 10.3–1), the check valves will prevent one steam generator from blowing down and the other steam generator will blowdown at the rate of 133 pounds per second causing the reactor power to reach 105 percent of full power. This condition will continue until the operator stops the feedwater to the steam generator associated with the break and shuts down the plant. No additional protective action is required.

Assuming the line breaks in either one of the 4 inch auxiliary steam supply lines downstream of the check valves or on the common header and with the reactor at 100 percent power, the steam flow through each of the 4 inch lines is 133 pounds per second. The reactor power will increase to 109 percent of full power and will stay at this new steady state condition. Each steam generator will continue to blowdown through the break until the operator closed the remote-manual, motor operated isolation valves in the 4 inch lines, at which time the reactor will return to its initial power level. No other protective action is required.

The feeding of the steam generators using AFW combined with the dumping of steam to the environment through the atmospheric steam dump valves may be employed to provide long-term cooling during a LOCA. This would be required post-LOCA when the shutdown cooling system cannot be placed into operation or during simultaneous hot and cold leg injection. During a LOCA event, an aggressive cooldown rate is to be initiated and maintained, using the atmospheric steam dump valves, within 1 hour after the start of the accident and at a minimum rate of 40°F/hr, until steam removal from the steam generators becomes limited by the fully open atmospheric steam dump valves.

10.3.4 AVAILABILITY AND RELIABILITY

10.3.4.1 Special Features

The main steam piping, the isolation valves, the atmospheric dump, the main steam safety valves and the steam generators are designed in accordance with seismic Class I requirements.

Three self-operating valve systems, using separate components, protect the steam generator from overpressure; however, safety valves alone can do the job. Each system operates as given in Table 10.3-3.

10.3.4.2 Tests and Inspection

See Section 5.2.7.4.2 and Technical Specifications Section 4.7.1.5.

TABLE 10.3-1 MAJOR COMPONENTS OF MAIN STEAM SYSTEM

Main Steam Isolation Valves		
Description	Main steam swing check valves, 600 pound ANSI, carbon steel, butt-welded ends	Main steam swing disc trip valves, air cylinder operated 600 pound ANSI, butt-welded ends, carbon steel.
Manufacturer	Atwood & Morrill	Atwood & Morrill
Size	34 inch	34 inch
Codes and Standards	ANSI B31.1.0[a], B16.10, B16.25 MSS-SP-25, -55, -61	Draft ASME Code for pumps and valves for nuclear power, November 1968; including 1970 Addenda, Code Case 1427
Material		
Body	Cast steel ASTM A216 Gr WCB	Cast steel ASTM A-216-Gr WCB
Disc	Cast steel ASTM A216 Gr WCB	Cast Steel ASTM A216 Gr WCB
Shaft	Stainless steel ASTM A276 Type 410	Steel bar ASTM A-276 Type 410
Design pressure / temperature	1000 psig/600°F	1000 psig/600°F

a. Except weld reinforcement requirements of Code Case 83 shall apply.

Three inch motor-operated bypass valve assemblies provided for each 34 inch valve assembly. The applicable codes and material are similar to those for the main steam swing disc trip valves.

Main Steam Dump to Atmosphere				
Description		600 pound ANSI, cast steel, butt-welded ends, diaphragm operated.		
Manufacturer		Copes-Vulcan		
Size		8 inch		
Codes and Standards		ASME Code for Pumps and Valves for Nuclear Power Class II; ASME Code Case 1427; ANSI B16.5, -B16.10, -B16.25; MSS-SP-6, -25, -45, -53, -55, -61, -66.		
Material				
Body		Cast steel ASTM A-216 Gr WCB		
Plug		Stainless steel ASTM A276 Type 420		
Stem		Stainless steel ASMT A-276 Type 316-B		
Conditions and Characteristics				
Quick opening time (seconds)				5 or less

Inlet pressure (no-load steam generator		
pressure less piping losses) (psig)		896
Temperature (°F)		533
Differential pressures (psi)		829
Maximum differential pressure (psi)		985
Total combined flow rate (2 valves)(lb/hr)		1,597,590
Modulating service stroke time, max (seconds)		10

Turbine Bypass to Condenser

Description	600 pound ANSI, cast steel, butt-welded ends, pneumatically operated piston actuator		
Size	12 inch inlet x 14 inch outlet		
Codes and Standards	ANSI B16.10, B16.25, B31.1; MSS-SP-25, -45, -53, -55, -61, -66; ASME B16.34; ISA S75.01		
Material			
Body	Cast steel ASTM A-216 Gr WCB		
Plug	Stainless steel Commercial ASTM - A479 - 410		
Stem	Stainless steel ASTM A-276 - 410		
Conditions and Characteristics		Dump mode	Bypass mode
Quick opening time (sec)		4 or less	N/A
Inlet pressure (psig)		822	879
Outlet pressure (psig)		155	155
Steam state		Dry, saturated	Dry, saturated
Total combined flow rate (lb/hr)		4,850,000	590,150
Modulating service, max (lb/hr)		4,850,000	590,150
Modulating service, stroke time (sec)		10 max	10 max

Piping - See Table 10.4-1.

Main Steam Safety Valves - See Section 4.3.2 for details.

TABLE 10.3-2 OMITTED

TABLE 10.3-3 OPERATING DETAILS FOR VALVE SYSTEMS PROTECTING STEAM GENERATIONS OF OVERPRESSURE

Valve Description	Number Of Valves	Actuating Inlet Pressure (psia)	Mode
Steam dump bypass to condenser	4; one bypass	900	Quick open; modulated on reactor T_{ave}
Atmospheric dump	2; one for each steam generator	920	Pressure controller
Main steam safety valves	16; 8 on each line	1000,1005 1015,1025 1035,1045 and 1050	Pressure controlled spring-loaded

FIGURE 10.3–1 P&ID MAIN STEAM FROM GENERATORS (25203-26002 SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.3–1 P&ID MAIN STEAM GENERATOR BLOWDOWN SYSTEM (25203-26002 SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.3–1 P&ID MAIN STEAM TURBINE AND REHEATER 1A (25203-26002 SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.3–1 P&ID MAIN STEAM TURBINE (25203-26002 SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSERS

10.4.1.1 Design Bases

The purpose of the condenser is to provide a heat sink for the turbine exhaust steam, main steam dump and bypass and other flows. It also deaerates and provides storage for the condensate.

The main condenser is designed for the following conditions:

Condenser duty (Btu/hr)	6059×10^6
Condenser pressure (inch Hg absolute)	2.07
Circulating water temperature	
Inlet (°F)	60.8
Outlet (°F)	84.8
Flow (gpm)	522,500 total
Heat transfer area (square feet)	485,260 total
Design pressure	
Shell (inches Hg vacuum)	29.5
Waterbox (psig)	25

The condenser is designed to accept up to 40 percent of the design main steam flow (vwo) through the steam dump and bypass valves while maintaining turbine back pressure below 5 inches. Hg absolute without exceeding the allowable turbine exhaust temperature.

The condenser hotwell has four minutes of vwo flow storage capacity.

The condenser is designed to deaerate the condensate properly, to provide condensate of the required quality and to remove noncondensable gases and air inleakage from the condensing steam.

10.4.1.2 System Description

During planned operation, steam from the low pressure turbine is exhausted directly downward into the condenser shells through exhaust openings in the bottom of the turbine casings. The condenser consists of two sections, each section serving one double-flow, low pressure turbine section. The condenser also serves as a heat sink for several other sources, such as exhaust steam from feed pump turbines, cascading heater drains, air ejector condenser drain, steam packing exhauster condenser drain, feedwater heater shell operating vents and condensate pump suction vents.

During abnormal conditions, the condenser is designed to receive steam dump and bypass steam and feedwater heater and moisture separator/reheaters drain tank high level dump(s).

The condenser is a single-pass, single-pressure deaerating type with divided water boxes. The condenser is supported on the turbine foundation mat with expansion joints provided between each turbine exhaust opening and the steam extraction connections in the condenser shells. Condenser shells are connected by a pressure equalizing duct.

Dearation in the condenser removes normal inleakage of air plus noncondensable gases contained in the turbine steam. The noncondensable gases are then collected in the air cooling section of the condenser, from which they are removed by the mechanical vacuum pump at startup and by the steam jet air ejectors (SJAE) during normal operation.

10.4.1.3 Safety Evaluation

The condenser is designed to store a sufficient volume of condensate to provide a three minute effective retention time of the condensate for radioactive decay in the event of tube leakage of the steam generator.

10.4.1.4 Tests and Inspection

The condenser is built in accordance with Heat Exchanger Institute (HEI) standards for the steam surface condensers. The water boxes shall be tested in accordance with these standards. After erection, the condenser will be filled with water to check tube and other leaks.

10.4.1.5 Instrumentation

Condenser vacuum and temperature are monitored continuously. Necessary alarm and trip signals are provided for high pressure and exhaust hood temperatures. The condenser hotwell level is controlled by level controllers. Conductivity elements in each effluent line detect any inleakage of the circulating water into the condenser steam space.

Redundant conductivity elements are provided, one in each condenser outlet line to detect in leakage of circulating water into the feedwater system. Each conductivity cell output is continuously recorded. High conductivity will be alarmed in the Control Room.

10.4.2 MAIN CONDENSERS EVACUATION SYSTEM

10.4.2.1 Design Bases

The main condenser evacuation system is designed to remove a total of 80 scfm of noncondensable gases in accordance with HEI standards.

10.4.2.2 System Description

The main condensers evacuation system, as shown in Figure 10.4–1, includes two SJAE units complete with inter- and after-condensers which remove air and noncondensable gases from the main condenser. A mechanical vacuum pump is provided for use during startup.

Main steam, reduced in pressure by an automatic steam-pressure reducing station, is supplied as the driving medium to the twin element two-stage air ejectors. Air ejector condenser cooling is provided by the main condensate flow. Air inleakage and noncondensable gases are removed from the condenser and discharged to the Millstone stack which is continuously monitored for radioactivity.

Two full capacity mechanical vacuum pumps remove air and noncondensable gases from the condenser during startup and when the desired rate of air and gas removal exceeds the capacity of the air ejectors. The discharge from the vacuum pump is routed to the Millstone stack.

10.4.2.3 Safety Evaluation

A radiation monitor is provided in the air ejector vent line to the Millstone stack to detect primary to secondary leakage in the steam generators. Upon a high radiation signal an alarm is actuated and checks of the monitor alarm and possible leakage sources and rates are instigated.

10.4.2.4 Tests and Inspection

The air ejector inter- and after-condensers are hydrostatically tested at the manufacturer’s plant in accordance with HEI standards. Mechanical vacuum pumps and air ejectors first and second stages are performance tested at the factory for the design capacity.

10.4.2.5 Instrumentation

All necessary instrumentation is provided for normal and abnormal operations.

10.4.3 TURBINE GLAND SEALING SYSTEM

10.4.3.1 Design Bases

The turbine gland seal system provides a means of sealing the turbine shaft and valve stem packing against air leakage into the cycle and to prevent steam leakage into the Turbine Room.

10.4.3.2 System Description

The turbine gland sealing system, consists of a main steam source, steam seal pressure regulator, steam seal header, one full capacity steam packing exhauster with two full capacity blowers and the associated piping and valves.

During planned operation, one blower is operating while the other is on standby. The exhaust blower discharges gland air leakage to the Millstone stack. The steam packing exhauster is cooled by main condensate flow and is drained to the main condenser.

10.4.3.3 Safety Evaluation

The radiation monitor for the Millstone stack would alarm any high radioactive releases from this system.

10.4.3.4 Tests and Inspection

The gland seal condenser is designed in accordance with American Society of Mechanical Engineers (ASME) Section VIII and is hydrostatically tested.

10.4.4 CIRCULATING WATER SYSTEM

This system is described in Section 9.7.1.

10.4.5 CONDENSATE AND FEEDWATER SYSTEM

10.4.5.1 Design Bases

10.4.5.1.1 Functional Requirement

The function of the condensate and feedwater system is to process the condensate and provide the required amount of feedwater to the steam generators. The condensate and feedwater system provides feedwater heating and maintains the required quality of feedwater.

The auxiliary feedwater system (AFWS) is comprised of two full-capacity subsystems to satisfy two safety functional requirements. One subsystem consists of two motor driven, automatically initiated AFW pumps. The second subsystem consists of one turbine-driven pump.

The first safety functional requirement is to provide cooling to either steam generator, on low-steam generator level, with or without the loss of normal AC power along with the most limiting single failure. The arrangement of these two independent subsystems ensures that the auxiliary feedwater system can accomplish this functional requirement.

The second functional requirement is that AFW must be delivered to the steam generators independently of AC power in the event of a station blackout scenario. The turbine-driven AFW pump performs this second function.

10.4.5.1.2 Design Criteria

The following criteria have been used in the design of the condensate and feedwater system:

a. The system shall provide 110 percent of the valve wide open flow at 935 psig at the steam generator feedwater nozzles.

b. The system shall maintain the required volume and quality of water in the cycle.

c. The system shall process and provide a controlled path for the fluids in the cycle.

d. The system shall provide sufficient feedwater for the cold shutdown of the RCS.

The design of the feedwater piping system had been modified to ensure that the loop seals were provided in the piping as close to the steam generator feedwater inlet nozzles as possible. These modifications had been provided in lieu of performing analyses using dynamic forcing functions associated with the draining of feedwater lines to qualify the piping system design.

The design of the feedwater piping has also been modified to the interface requirement for the design associated with steam generators installed in 1991. The modified feedwater design retains the water hammer mitigation designs described in the previous paragraph but accommodates the 1991 steam generator's lower primary deck (below the normal water level), lower feedwater sparger ring, top discharge J-tubes, all welded thermal sleeve and a goose neck inlet design.

The following criteria have been used in the design of the AFWS:

a. The system shall have two redundant, independent subsystems, each capable of performing the functional requirements. The turbine-driven pump and associated piping is one subsystem. The two motor driven pumps and associated piping is the second subsystem.

b. The system shall have suitable subsystem and component alignments to assure operation of a complete subsystem with associated components.

c. Capabilities shall be provided to assure the system operation with onsite power (assuming offsite power is not available) or with offsite electrical power.

d. A single failure of an active component in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components such as pumps, valves, piping and tanks to assure the integrity and capability of the system.

f. The system shall be designed to the general criteria as described in Section 6.1.

g. The AFWS is automatically initiated upon detection of low steam generator level.

10.4.5.2 System Description

The condensate and feedwater system is shown in Figure 10.4–2.

The condensate pumps take condensate from the condenser hotwells and pump it through the condensate polishing demineralizers and the steam packing exhauster. The flow is then split into two streams. Each stream passes through an ejector condenser, an external drain cooler and sixth- and fifth-stage feedwater heaters located in the condenser neck and three low pressure feedwater heaters and one high pressure heater.

The SGFPs are located in the cycle between the low pressure heaters and the high pressure heater.

To maintain the proper feedwater chemistry a secondary chemical feed system is provided for injecting hydrazine as an oxygen scavenger and ethanolamine (ETA) to control pH. The rate of chemical injection is controlled to maintain chemistry within the desired limits. The point of chemical injection during normal operation is in the condensate return header from the Condensate Polishing Facility (CPF), upstream of the bypass valve. An alternate point of injection, when the CPF is bypassed, is in the CPF discharge header downstream of the CPF bypass valve and upstream of the steam jet air ejectors.

Feedwater regulating and control is described in Section 7.4.

10.4.5.3 Auxiliary Feedwater System

The AFWS, shown in Figure 10.4–2, is designed to provide feedwater for the removal of sensible and decay heat, and to cool the primary system to 300°F in case the main condensate and SGFP are inoperative due to loss of normal electric power sources or the main steam. A single motor-driven pump is capable of adequately removing decay heat. This has been demonstrated by a best estimate analysis of the loss of feedwater event (Reference 10.4-1). The best estimate analysis is performed to demonstrate the automatic start of both motor driven auxiliary feedwater pumps on low steam generator level satisfies the automatic auxiliary feedwater initiation requirements of NUREG-0737 item II.E.1.2. Automatic start of the turbine driven auxiliary feedwater pump is not required. The suction condition yields a net positive suction head (NPSH) in excess of that required. The AFWS may also be used for normal system cooldown to 300°F. The AFWS supplies feedwater from the condensate storage tank (CST) to the steam generators for evaporation and heat absorption. Reactor decay heat and sensible heat are transferred to the steam generators by natural circulation of the reactor coolant if power is not available for the reactor coolant pumps (RCP).

In order to perform its safety-related function (per Section 14.2.7) assuming single failure, the auxiliary feedwater system (AFWS) is comprised of two full capacity subsystems. One subsystem consists of two motor driven AFW pumps, that are automatically connected to the diesel generators in the event of a loss of offsite power. The second subsystem consists of one turbine-driven pump that is independent of AC power and may be started by operator action. The turbine-

driven pump has a maximum capacity of 600 gpm at 2437 feet design head (TDH) and the two motor driven pumps have a constant 300 gpm capacity each at 2437 feet TDH.

The two motor driven auxiliary feedwater pumps (AFP) received their power from redundant Facility 1E 4160VAC breakers. The 4160 VAC breakers receive control power from redundant 125VDC feeds, Facility Z1 125 VDC panel DV10 and Facility Z2 125 VDC panel DV20. The speed adjuster motor and the steam inlet valve for the turbine-driven AFP normally receive their power from Facility Z2 125 VDC panel DV20. During design basis accidents coincident with a single failure that could disable one motor driven AFP and the turbine-driven AFP simultaneously, the steam inlet valve and the speed adjuster motor are capable of being transferred to Facility Z1 125 VDC panel DV10. In the event of a loss of Facility Z2 125 or a loss of DC panel DV20, the transfer is accomplished by switching the position of two key-locked isolation switches on panel C-05. The associated wiring from panel C-05 is routed in dedicated Z5 conduit to panel C-21, panel C-10 and ultimately to the steam inlet valve and the speed adjuster motor.

The auxiliary feedwater pumps (AFP) are located in two separate pump rooms at elevation 1 foot 6 inches in the Turbine Building. The Turbine Building is protected from potential flooding by the flood wall system as shown in Figure 2.5–18 of the FSAR. Access to the first room which houses the two motor driven AFPs is by stairs leading down from the ground floor at elevation 14 feet 6 inches. The enclosure over the pump room stairwell serves as a protective barrier against direct water streams into the Pump Room due to a possible overhead pipe failure. The second room which houses the turbine-driven AFP is a vault physically separated from the motor driven AFP room by a reinforced concrete wall. The only access means to this room is through a water-tight fire door.

Separate floor drain and sump systems are provided for each pump room. A high-level alarm system which is monitored in the Control Room for each sump will indicate excessive flooding in the room. The power supply to the sump pump in the motor driven AFP room and to the ‘B’ sump pump in the steam-driven AFP room is fed from vital power supply, MCC B61.

Directly above each pump and on the ground floor of the Turbine Building is a removable hatch serviced by a monorail and hoist system for equipment removal and emergency access. The AFP and piping in both rooms are Seismic Class I.

The AFPs can each be connected to a separate diesel generator by means of a separate control switch located on the main control board with the capability to manually start each pump from the control room.

The CST capacity is 250,000 gallons. A nitrogen blanketing system is utilized to limit oxygen intrusion into the stored condensate inventory. A nitrogen sparger system is available to lower the oxygen concentration in the tank. A redundant relief protection scheme, including both breather valves and rupture disks, is provided to prevent an over-pressure or vacuum situation. At operational low level, the volume of water available for decay heat removal and cooldown of the nuclear steam supply system (NSSS) is 165,000 gallons. This is sufficient to cool down the RCS to less than 300°F following a total loss of offsite power. The 165,000 gallons are also adequate to

remove decay heat alone for more than 10 hours, leaving a normal no-load water level in the steam generators at the end of 10 hours after initiation of the cooldown cycle. The 165,000 gallons includes unusable tank volume due to discharge nozzle pipe elevation above tank bottom, plus an allowance for vortex formation. CST is equipped with a recirculation heating system to prevent freezing within the tank during cold weather. The heating system includes a recirculation line that takes suction inside and near the bottom of the tank, penetrates the tank above the concrete protected region, is pumped through a heat exchanger and returns to the top of the tank. The suction line inside the tank is provided with a siphon breaker to prevent siphoning of the tank following a postulated pipe break. The siphon breaker is located at a height that will protect the minimum required tank water level to assure suction for the CST heating system, and to minimize drainage following a postulated pipe rupture.

The CST is missile protected by a concrete wall extending to a height which will provide adequate water for a safe shutdown. The exposed portion of the steel tank is designed to withstand the tornado wind pressure as described in Section 5.7.3.1.4 of the FSAR. Failure in the upper portion of the tank due to missile impact will be local and since the tank is fabricated of ductile steel, any fragmentation is unlikely. However, to ensure proper water flow from the tank in the event of a postulated free falling fragment of steel, the tank discharges are protected by screens which will prevent blockage.

The portion of the steel liner which is protected by the concrete wall could be damaged locally by a missile entering through the upper portion of the tank. In the event of local damage to the steel tank wall below the top of the exterior concrete wall, significant water loss is not possible as the steel shell is anchored to the concrete with vertical angles having a spacing of 24 inches. Any minor water leakage will be confined to the interface between the steel and concrete.

The concrete wall, which extends to a height of 25 feet, ensures that a minimum of 205,800 gallons of water is available for a safe shutdown of the plant. The concrete wall extends above the operational low-water level corresponding to 165,000 gallons.

The steam generated during decay heat removal and cooldown during an electric power failure will be discharged by the atmospheric dump valves (ADV), except for that used by the turbine-driven AFP. The turbine-driven pump operates reliably as long as there is steam pressure in excess of 50 PSIG in one of the steam generators.

The three AFPs are individually controlled from Control Room Panel CO-5 or from the Hot Shutdown Panel C-21 in the Turbine Building. In addition, the steam-driven AFP can also be controlled from the fire shutdown panel C-10 in the Turbine Building. The electric-driven pumps may be either automatically actuated or manually actuated. The steam-driven AFP can only be manually actuated. In the event of a loss of Facility Z2 125 VDC power or a loss of panel DV20, the steam-driven AFP speed adjuster motor and steam inlet valve can be swapped to Facility Z1 125 VDC power by switching two key-locked isolation switches on panel C-05.

For automatic actuation, each pump and its associated flow control valve have two switches on each panel. The first switch, the automatic permissive, either allows or blocks the automatic start of the respective pump. This auto permissive switch has three positions:

- Pull to lock, which blocks the automatic signal.
- Reset, which resets the automatic function.
- Start, which will start the electric AFPs and open the flow control valves.

The second switch selects the mode of operation of the flow control valve associated with the pump.

The three modes of selection are:

- Normal, which allows the valve to open fully for an automatic actuation.
- Override, which allows manual control of the valve position following an automatic actuation.
- 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.

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.

The AFW system contains suction and discharge connections that facilitate portable diesel engine driven AFW pump deployment and CST replenishment. These conditions are defense-in-depth design features that are available for coping with an extended loss of AC power (ELAP) event. The location of these BDB AFW Flex suction and discharge connections are shown on Figure 10.4–2, Sheet 3.

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 operation at low level in the condenser hotwell. Table 10.4-1 describes the detail design conditions. The third (standby) pump starts automatically on the low pressure signal from the condensate discharge header.

10.4.5.4.2 Feedwater Heaters

Two parallel trains with five low pressure heaters and one high pressure heater are provided. Drain coolers are provided for all stages of feedwater except Number 3 heater from which the drains are pumped forward, and all except Number 6 heater drain cooler sections are integral with the heaters. All heaters have 439 stainless steel tubes except Numbers 2, 3, and 4 which have 304 series SS. Table 10.4-1 provides more details.

10.4.5.4.3 Heater Drain Pumps

Heater drain pumps are motor driven centrifugal types, vertical, canned suction units. Two 55 percent capacity pumps are provided at an elevation that allows adequate NPSH at all loads. Table 10.4-1 lists the details.

10.4.5.4.4 Steam Generator Feed Pumps

Each SGFP is a horizontal turbine-driven centrifugal unit. Two 55 percent capacity pumps operate in series with the condensate pumps. Recirculation control valves are provided in the pump discharge lines to provide automatic recirculation of a portion of the feedwater flow to the main condenser. This assures that a minimum safe flow is maintained through the pumps. Table 10.4-1 lists the characteristics.

10.4.5.4.5 Steam Generator Feed Pump Turbine Drives

Each feedwater pump is driven by an individual steam turbine. The turbine drives are of the dual admission type and are equipped with stop and control valves. Under normal operating conditions, the turbine drives run on the low pressure crossover steam. Main steam is used during plant startup, low load or transient conditions when crossover steam is not available or is of insufficient pressure. The turbines operate at 2.5 inches Hg absolute back pressure and exhaust to the main condenser.

10.4.5.4.6 Condensate Polishing Facility

When in use, the Condensate Polishing Facility (CPF) receives condensate from the condenser hotwell after it passes through the Steam Packing Exhauster. After processing, condensate is returned at the air ejectors. The CPF consists of seven parallel flow demineralizers. Full condensate flow can be accommodated by six demineralizers allowing for regeneration of the seventh without reducing power. The CPF can be completely bypassed. The CPF is used to maintain the secondary system water chemistry within the limits of plant procedures.

10.4.5.5 Safety Evaluation

The components of the condensate and feedwater system are conventional and of a type that has been extensively used in fossil-fueled and other nuclear plants.

The plant can carry 55 percent load with one-half of the feedwater heaters out of service and isolated from the system, with only one condensate pump in operation and with only one SGFP in service.

The components of AFWS are designed to the general requirements including seismic response as described in Section 6.1. The components are protected from missile damage and pipe whip by physical separation of duplicate equipment as described in Section 6.1. The pumps are completely redundant. Each subsystem is capable of providing the required amount of feedwater for performing the cold shutdown of the RCS.

The total inventory of the condensate and feedwater system is approximately 209,000 gallons divided between the components as follows:

1. Condenser (shell) = 80,100 gallons

2. Condensate and Feedwater piping = 55,500 gallons

3. Moisture Separator/Reheater and Drain Tanks = 3,200 gallons

4. Feedwater Heaters = 37,000 gallons

5. Steam Generator = 33,200 gallons

The maximum flowrate to each steam generator is approximately 14,100 gpm during normal plant operation.

The potential for failures in the condensate and feedwater system are minimized by designing the piping for pressures well above any pressures possible in the system, with the low pressure heaters designed for the maximum shut-off head of the condensate pumps and the high pressure heaters designed for the SGFP shut-off head, see FSAR Section 10.4.5 and Table 10.4-1.

Failures in the condensate system can be readily detected by equipment performance alarms in the Control Room indicating the malfunction to the operator. The type of alarm depends on the severity and location of the failure and allows the operator the option of either isolating the failed component and reducing load or complete shutdown of the plant. However, as with any gross failure in this system, the alarms on the steam generator would ultimately indicate decreased flow, loss of pressure and low liquid level resulting in a possible reactor trip by the reactor protection system (RPS). The type of alarms are dependent on the location of the pipe rupture or equipment failure and are summarized below:

1. Failures in the system between the condenser and the condensate pumps would result in gradual flooding of the condenser pit setting off the condenser pit sump high-level alarm, set at 6 inches below the top of the sump. If the condenser hotwell level continues to decrease, a low level alarm is registered in the Control Room.

2. Failures in the equipment or piping between the condensate pumps and the SGFPs would result in low condensate pump discharge pressure and low SGFP suction pressure alarms in the Control Room. In addition, the steam generator would register the alarms indicated previously (i.e., low flow, pressure and liquid level).

3. Failures in the equipment or piping between the SGFP discharge and the feedwater containment isolation valves would result in a low pressure alarm indicated in the feedwater header and low-level alarms for the steam generator.

4. Failures in the piping between the containment isolation valves and the steam generator would be indicated by the low flow, pressure and water level alarms in the steam generator.

The entire system can be brought to rest, with the termination of all flow, within several seconds from the time the condensate and SGFPs are tripped. Static head would be the only driving force

for water escaping through leaks at lower elevations. In the event that the entire system inventory were drained to the Turbine Building ground floor, equipment and floor drains located at low spots in the floor would drain the water to the condenser pit sumps and prevent any accumulation on the ground floor. Feedwater piping in the Enclosure Building is either routed through areas void of essential equipment or provided with restraints in other areas as required so that any ruptures occurring do not affect safe operation of the plant. Therefore, any postulated failures in the condensate and feedwater system can be detected and controlled quickly, and in no way affect the operation of essential equipment.

To increase the original design basis safety margin for a main stream line break (MSLB) inside containment, the MSI signal was modified to actuate on either high containment pressure or low steam generator pressure. The modified MSI signal provides a more rapid isolation signal for feedwater and main steam isolation during a steam line break within containment. Also, the feedwater block valves, feedpump discharge valves, feedwater regulating and bypass valves as well as the feed pumps are provided with redundant MSI trip and closure signals for ensuring feedwater isolation.

10.4.5.6 Tests and Inspection

Equipment, instruments, and controls are regularly inspected to ensure proper functioning of systems. The motor driven pumps and controls are given preoperational tests after erection and before plant warmup.

The AFWS can be tested during normal operation by recirculating an AFP flow of 50 gpm to the CST.

10.4.6 STEAM GENERATOR BLOWDOWN SYSTEM

10.4.6.1 Design Bases

The function of the steam generator blowdown system (SGBS) is to withdraw and reject the system fluid at a rate which will maintain contaminate concentrations in the steam generator water below equipment operating limits. The system has three modes of operation: sampling blowdown: condenser leakage: and startup blowdown.

10.4.6.2 System Description

The SGBS is shown in Figure 10.3–1. Blowdown and quench tanks are provided. The blowdown tank has a capacity of two percent of the vwo main steam flow and is used during startup and when condenser tube leakage occurs. The quench tank has a blowdown capacity of 10 gpm and will normally be in service at all times when the blowdown tank is not in service.

Tube sheet blowdown nozzles are provided on each steam generator. The blowdown tank vents to the atmosphere and the quench tank vents to the gas processing system. Normally, both drain to the condenser cooling water discharge. The quench tank is provided with a closed recirculating pump and cooler cycle for quench water supply.

The steam generator blowdown is automatically isolated in response to the CIAS and the low-low steam generator water level trip (AFAIS) that activates the auxiliary feedwater system.

A radioactivity monitor is located in the combined sampling line from both steam generator blowdown lines. When a high radioactivity level is detected, an alarm is annunciated in the Control Room and the blowdown stop and drain valves are closed automatically. The blowdown system residual inventory is manually diverted to the Aerated Liquid radioactive waste system. The 10 gpm quench tank blowdown may be resumed; if so, the blowdown liquid is diverted to the Clean Liquid radioactive waste system.

10.4.6.3 Safety Evaluation

Failure analysis of system components is given on Table 10.4-2.

10.4.6.4 Tests and Inspection

The steam generator blowdown tank and quench tank are designed in accordance with ASME Section VIII and seismic Class II requirements. The performance of the blowdown system is ascertained by an inplace test.

10.4.7 REFERENCES

10.4-1 “Millstone Unit Number 2 FSAR Chapter 10 Loss of Normal Feedwater Flow Transient with Reduced Auxiliary Feedwater Flow,” EMF-98-049, Rev. 0, Siemens Power Corporation, August 1998.

TABLE 10.4-1 MAJOR COMPONENTS OF THE CONDENSATE AND FEEDWATER SYSTEM

Steam Generator Feed Pumps		
Type	Double suction, diffuser type, single stage, vertical split, horizontal centrifugal	
Quantity	2	
Capacity each (gpm)	15,000	
Head differential tdh (feet)	2,100	
Material		
Case	ASTM A217, Gr C5	
Impeller	ASTM A296, Gr CA 15	
Shaft	AISI 410 Ht & SR	
Driver	Condensing, nonextracting, dual admission, horizontal steam turbine	
Codes	ASME Section VIII and IX, Standards of the Hydraulic Institutes, NEMA, ANSI	
Manufacturer	Ingersoll Rand	
Steam Generator Auxiliary Feed Pumps		
Type	Horizontal, split case, multistage, centrifugal	
Quantity	1 turbine driven,	2 motor driven
Capacity each (gpm)	600	300
Head (feet)	2437	2437
Material		
Case	ASTM A 216, Gr WCB	
Impeller	ASTM A 48, Class 30 for P9B; ASTM A 217, Grade CA15 for P9A, P4	
Shaft	ASTM A 582, Type 416 or ASTM A 276 Type 410	
Driver	Single-state, noncondensing steam turbine (for turbine-driven) GE motor, 350 Hp, 4160 volt, 60 Hz, 3 phase, 3600 rpm (for motor-driven pumps)	

TABLE 10.4-1 MAJOR COMPONENTS OF THE CONDENSATE AND FEEDWATER SYSTEM (CONTINUED)

Seismic requirements Codes	Class I
	ASME Code for Pumps and Valves for Nuclear Power, Class II NEMA Standard SM 20-1958 Hydraulic Institute
Manufacturer	Ingersoll Rand
Condensate Pumps	
Type	Vertical centrifugal
Quantity	3
Capacity each (gpm)	9200
Head (ft)	1050
Material	
Case	ASTM A-48, CL 30
Impeller	ASTM A-296, Gr CA-15
Shaft	ASTM A-276, Type 416
Driver	E1 motor, 3000 Hp 6600 volts, 60 Hz, 3 phase, 900 rpm
Codes	ASME Sections VIII and IX, Standards of the Hydraulic Institute, NEMA ANSI
Manufacturer	B&W Canada
Heater Drain Pumps	
Type	Vertical Centrifugal
Quantity	2 (per unit)
Capacity each (gpm)	4300
Heat (feet)	1000
Material	
Case	ASTM A-217, Gr C-5

TABLE 10.4-1 MAJOR COMPONENTS OF THE CONDENSATE AND FEEDWATER SYSTEM (CONTINUED)

	Impeller	ASTM A-296, Gr. CA-15
	Shaft	ASTM A-276, Type 410 HT
Driver		E1 motor 1250 Hp, service factor 1.15 4160 volts, 60 Hz, 3 phase, 1800 rpm
Codes		ASME Sections VIII and IX, Standards of the Hydraulic Institute, NEMA, ANSI
Manufacturer		Byron-Jackson

Condenser

Type	Two shell, single-pass with divided water boxes, surface condenser
Design duty (Btu/hr)	6.059×10^9
Heat transfer area (ft^2)	485,260 total
Design pressure	Shell 29.5 in. Hg. vacuum Water box 25 psig
Material	
Shell	ASTM A-285, Grade C
Tubes	ASTM B 338, Gr. 2 Titanium
Tube sheets	ASTM B-265, Gr. 2 Titanium
Codes	Standards of the Heat Exchange Institute

Feedwater Heaters

Type	Closed, U-tube except drain cooler is straight			
Material	Heater 1 (A,B)	Heater 2 (A,B)	Heaters 3 & 4 (A, B)	Others
Shell	Carbon steel	Carbon steel	Carbon steel	Carbon steel
Tubes	439 SS	304 LSS	Stainless Steel	439 SS
Tube Sheets	Carbon Steel w/Inconel overlay	Carbon Steel	Carbon Steel	Carbon Steel
Codes	ASME Section VIII, Heat Exchange Institute			

TABLE 10.4-1 MAJOR COMPONENTS OF THE CONDENSATE AND FEEDWATER SYSTEM (CONTINUED)

Details on joints, tube-to-tube sheets	Welded & Rolled	Hydraulically Expanded & Rolled	Rolled	Rolled

Heater No.	Design Duty Each (Btu/hr)	Design Pressure (psig)		Design Temperature (°F)		Heat Transfer Area Each (ft^2)
		Shell	Tube	Shell	Tube	
1, A, B	355.215×10^6	450	1700	460	460	20,445
2, A, B	287.518×10^6	225	650	400	400	23,675
3, A, B	154.152×10^6	30 in Hg. & 150	650	370	370	12,163
4, A, B	296.046×10^6	30 in. Hg & 75	650	330	330	17,019
5, A, B	263.347×10^6	30 in. Hg & 50	650	300	300	16,144
6, A, B	198.156×10^6	30 in. Hg & 50	650	300	300	15,839
Drain cooler (6, A, B)	47.550×10^6	30 in. Hg	650	300	300	5,156
Piping						

TABLE 10.4-1 MAJOR COMPONENTS OF THE CONDENSATE AND FEEDWATER SYSTEM (CONTINUED)

Piping System	Applicable Code	Material	Design Pressure (psig)	Design Temperature °F
Main steam	ANSI B31.1.0	ASTM A-144 Gr. KC 70	1000	580
Main steam penetration piping (including atmospheric dump)	ANSI B31.7 C12	ASTM A-155 Gr. KC 70	1000	580
Feedwater	ANSI B31.1.0	ASTM A-106 Gr. C	1600	400-450
	ANSI B31.1.0	ASTM A-106 Gr. B	1100	600
	ANSI B31.1.0	ASTM A-335 Gr. P5	1100-1600	400-600
	ANSI B31.1.0	ASTM A-335 Gr. P22	1100-1600	400-600
Feedwater penetration	ANSI B31.7CL 2	ASTM A-106 Gr. B	1100	600
	ANSI B31.7CL 2	ASTM A-335 Gr. P5	1100	600
	ANSI B31.7CL 2	ASTM A-335 Gr. P22	1100	600
Condensate	ANSI B31.1.0	ASTM A-106 Gr. B or ASTM A-335 Gr. P22	650	400

TABLE 10.4-2 FAILURE ANALYSIS OF STEAM GENERATOR BLOWDOWN SYSTEM COMPONENTS

1.	Steam Generator Blowdown line	Break at 2 inch diameter line inside containment	1) Containment pressure and Radiation monitoring. 2) Containment sample loss alarm	Plant shutdown	Radioactivity release only to the containment which can be held and processed through filters.
2.	Steam Generator Blowdown piping	a) Outside the containment	1) Area radiation monitoring	Close control valve HV4246 & HV4248 or Plant shutdown	Radioactivity release to the Auxiliary Building is monitored as discharged to the environment through the EBFS and liquid collected in the Drain System.
		b) Break of Blowdown piping downstream valve HV4246 or 4248 in the Enclosure Building	2) Aerated waste sump level alarms.		
3.	Radiation Monitors	Radiation monitoring failure with no mechanical piping failure	Radiation monitoring failure alarm at control room.	Close control valves HV4246, HV4248, HV4245, and HV4578 and manual sampling	

FIGURE 10.4–1 P&ID CONDENSER AIR REMOVAL WATER BOX PRIMING AND TURBINE BUILDING SUMP (25203-26012)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.4–2 P&ID CONDENSATE SYSTEM (25203-26005 SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.4–2 P&ID CONDENSATE SYSTEM (25203-26005 SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.4–2 P&ID CONDENSATE STORAGE AND AUXILIARY FEED (25203-26005 SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 10.4–2 P&ID CONDENSATE STORAGE AND AUXILIARY FEED (25203-26005 SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 11: Radioactive Waste Processing and Radiation Protection Systems

CHAPTER 11—RADIOACTIVE WASTE PROCESSING AND RADIATION PROTECTION SYSTEMS

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CHAPTER 11 – RADIOACTIVE WASTE PROCESSING AND RADIATION PROTECTION SYSTEMS

11.1 RADIOACTIVE WASTE PROCESSING SYSTEMS

11.1.1 GENERAL

The purpose of the radioactive waste processing systems is to provide controlled handling of all radioactive waste, proper discharging of radioactivity in liquid and gaseous effluents, and proper packaging for shipment of solid waste containing radioactivity from Millstone Unit 2.

On occasion, the Unit will generate liquid radioactive waste that cannot practicably be processed in the liquid radwaste system. The station may process this waste outside the Unit in compliance with state and federal regulations, and in accordance with the Radiological Effluent Control Program outlined in the Administrative Section of the Technical Specifications (e.g., Unit 1 evaporator or shipped off site for processing).

11.1.2 DESIGN BASES

11.1.2.1 Functional Requirements

These systems are required to ensure that the general public and plant personnel are protected against exposure to radioactive material in accordance with the regulations of 10 CFR Part 20, and the recommended guidelines of 10 CFR Part 50, Appendix I. Under normal plant operating conditions, the radioactive waste processing systems are required to limit radionuclide release concentrations to unrestricted areas to less than the maximum permissible concentrations as specified in 10 CFR Part 20, Section 1302, and Appendix B.

Interim onsite storage facilities accept waste from Millstone Units 1, 2 and 3. Information regarding facility design criteria is presented in Section 11.4 of the Millstone Unit 3 Final Safety Analysis Report.

11.1.2.2 Design Criteria

The radioactive waste processing system is designed in accordance with the following criteria:

a. To protect the general public and plant operating personnel against radiation from materials in accordance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

b. To limit the levels of radioactivity of the effluents in unrestricted areas to as low as reasonably achievable (ALARA).

c. To provide suitable control of the release of radioactive materials in gaseous and liquid effluents during normal plant operation including anticipated operational occurrences.

d. To provide sufficient holdup capacity to retain gaseous and liquid wastes containing radioactive materials.

e. To ensure adequate safety under normal and postulated accident conditions.

f. To provide the capability to permit inspection and testing of appropriate components.

g. To provide suitable shielding for radiation protection.

h. To provide appropriate containment and confinement of radioactive materials.

i. To provide appropriate monitoring capability to detect excessive radiation levels and to monitor effluent discharge paths.

j. To provide suitable processing of liquid and gaseous radwaste generated in accordance with the following operating and design criteria:

    1. Normal operation with expected primary and secondary side activities calculated in accordance with NUREG-0017 Rev. 1 (Reference 11.1-1) using the PWR-GALE code.

    2. Normal operation with design basis reactor coolant activities based on 1% failed fuel in accordance with Technical Specifications.

The only portions of the radioactive waste processing system designated as Seismic Category I are the containment penetration piping and isolation valves. However, the high pressure (150 psig) portions of the gaseous waste system, consisting of the compressors, decay tanks, interconnecting piping and valves (see Figure 11.1–5), have been designed and analyzed for Seismic Category I requirements as given in Section 5.8.

An analysis had been performed, at the time of initial plant licensing, to determine the site boundary doses due to simultaneous failure of the entire radioactive waste processing system, excluding those portions of gaseous waste processing system that were analyzed to Seismic Category I requirements. That analysis was predicated on the original design and operation of the radioactive waste processing system. More recently, this chapter was updated to reflect changes to the radioactive waste processing system design and operation. The results of this update are bounded by those of the original analysis discussed in this subsection.

Failure the of the radioactive waste processing system may release gaseous and liquid wastes into the auxiliary building. The auxiliary building, a Seismic Category I structure, is designed to contain all liquids within the building.

Therefore, the total inventory of radioactivity within the gases contained in the radioactive waste processing system, excluding high pressure portion of gaseous waste system, are assumed to be released.

The total activity is conservatively assumed to be released to the auxiliary building at an instant followed by puff release at ground level to the turbulent wake downwind of plant structures. Under these conditions, the X/Q value of 9.6×10^{-5} sec/m^3 is applicable. The doses at 625 meters were calculated by the method of Safety Guides 3 and 4. Based on the results of the original plant licensing analysis, the site boundary doses due to simultaneous failure of entire radioactive waste processing system, excluding portions of gaseous waste system, are as follows:

Normal Operation Based on 0.1% Failed Fuel

Thyroid dose (rem) - 3.9×10^{-4}

Whole body dose (rem) - 5.04×10^{-2}

Normal Operation Based on 1.0% Failed Fuel

Thyroid dose (rem) - 4.35×10^{-3}

Whole body dose (rem) - 5.78×10^{-1}

Since the doses are approximately equal to or less than the limits given in 10 CFR Part 20, Sections 105 and 106 and Appendix B (version prior to January 1, 1994), a Seismic Noncategory 1 waste processing system is acceptable.

11.1.2.3 System Components

Descriptions of the radioactive waste processing system components are given in Table 11.1-1.

11.1.3 LIQUID WASTE PROCESSING SYSTEM

11.1.3.1 System Descriptions

Clean Liquid Waste Processing System:

Clean liquid waste is normally tritiated, non-aerated, low conductivity liquid waste consisting primarily of reactor coolant letdown and liquid waste collected from equipment leaks and drains and certain valve and pump seal leaks. The clean liquid radioactive waste processing system is shown schematically in Figures 11.1–1 and 11.1–2. The design of the clean liquid radioactive waste processing system is based upon processing of radioactive liquids postulated to be released from the reactor coolant system (RCS) during normal reactor operation with design basis reactor coolant activities. Operating experience of nuclear power plants indicates that Millstone Unit 2 can expect to continue operating with a percentage of fuel failure much less than the postulated design basis of one percent. Nevertheless, the performance of the clean liquid waste processing system during normal operation is based on both expected and design basis primary side activities. Discussed in Appendix 11.A are the methodologies used to determine the expected and design basis radionuclide activity concentrations in the reactor coolant.

The clean liquid waste processing system is designed to support the processing of 14 RCS volumes per year of reactor coolant waste. However, the quantity of wastes to be generated and processed annually by this system is approximately 1,200,000 gallons, based on the assumptions used in the 10 CFR 50 Appendix I analysis. The liquid waste input streams are shown in Figure 11.B–1.

The clean liquid waste processing system is designed for the processing of reactor coolant wastes concurrently with the letdown flow from the chemical and volume control system (CVCS). This mode of operation, at the maximum clean liquid waste processing flow of 132 gpm, sets the maximum system flow rate.

Reactor coolant is diverted to the clean liquid radioactive waste processing system from the CVCS when changes in RCS inventory or boron concentration are necessitated by startups, shutdowns, fuel depletion, etc. Reactor coolant at a rate of 44 gpm to 132 gpm is let down from the CVCS through a filter to reduce insoluble particulate, after which it flows to the clean liquid radioactive waste processing system for further processing.

Sources of clean liquid waste in the containment are collected in the primary drain tank. A heat exchanger and pump are provided for cooling the primary drain tank content as well as the content of the pressurizer quench tank. The contents of these tanks are maintained or cooled to 120°F to minimize the carryover of radioactive moisture to the gaseous waste processing system due to tank venting.

Equipment drains, valve leak-offs, and relief valve discharges from components that are located in the auxiliary building and that contain liquids with dissolved fission gases are collected in the equipment drain sump tank via the closed drain system, as shown in Figure 11.1–3. This design minimizes the uncontrolled release of gaseous radioactivity to the atmosphere.

The liquid contents of both the primary drain tank and the equipment drain sump tank are pumped via the demineralizers to the coolant waste receiver tank, bypassing the degasifier. The demineralizers and degasifier are discussed below:

1. Degasifier

   The degasifier is installed but not used. The following describes original design. Reactor coolant degassing is accomplished by diverting the letdown flow in the CVCS to the degasifier in the clean liquid waste system. Interconnecting piping and valves are provided for degassing the reactor coolant prior to cold shutdown of the reactor for refueling operations. The degasifier is placed in service, prior to cold shutdown, to remove hydrogen, fission product gases, and other dissolved gases from the reactor coolant system liquid and to discharge these gases to the waste gas surge tank in the gaseous waste processing system. (See Section 11.1.4) The degassed liquid is pumped through the degasifier effluent cooler, which is used to lower the temperature of this liquid to 120°F before it is passed through one of two primary demineralizers. This is done to protect liquid radwaste system demineralizer resins against the damage caused by high liquid temperatures. The

degassed reactor coolant is returned to either the volume control tank inlet or to the clean liquid radwaste system primary demineralizers for processing and discharge to the environment.

The packed columnar type degasifier employs internally generated stripping steam for removal of dissolved gases. A nitrogen cover gas, slightly above atmospheric pressure, is maintained within the degasifier during the shutdown mode to prevent air in-leakage and the formation of a potentially explosive hydrogen/oxygen mixture.

The degasifier system is provided with cascading steam controls to minimize the adverse effects of varying feed flow rates and temperatures on the dissolved gas removal performance of the unit. The degasifier operates at a pressure of 5 psig and a temperature of 228°F, and its pumps (both operating at full capacity) are controlled automatically by the degasifier level controls.

2. Demineralizers

The two (2) primary and two (2) secondary demineralizers are of the mixed-bed non-regenerating type, the designs of which are based on an expected operating ion exchange capacity of 12,000 grains of $CaCO_3$. The resin beds for mixed-bed non-regenerating demineralizers are mixtures of cation and anion resins in the H-OH forms. These demineralizers are bypassed automatically upon detection of liquid waste temperatures above 135°F. An additional secondary demineralizer provides the capability to further polish the waste stream, with the capability of placing the two (2) secondary demineralizers in series operation.

The demineralizers of the mixed-bed nonregenerable type are sized for one-year operation between resin replacement. Resin replacement is accomplished remotely by N_2 pressure and/or water sluicing to the spent resin tank. All valves required for resin removal are located either outside the demineralizer compartments if non-radioactive, or inside shielded vaults with extension operators to provide protection of plant personnel if radioactive. The demineralizer effluent flows to one of two coolant waste receiver tanks.

The two receiver tanks provide storage for approximately two RCS volumes (120,000 gal. nominal) of liquid wastes. A nitrogen gas blanket in each of the tanks is automatically maintained above atmospheric pressure to prevent air in-leakage. The nitrogen cover gas is vented to the gaseous waste processing system or displaced into the other receiver tank or monitor tanks as liquid fills the first receiver tank. No flashing occurs in the receiver tanks, and any transfer of hydrogen or fission gases from the liquid to the cover gas occurs via the slow process of molecular diffusion.

The content of the coolant waste receiver tank is sampled prior to processing. The content of the coolant waste receiver tank is then pumped through either secondary demineralizer(s) or a liquid radioactive waste processing system (activated carbon filters and demineralizers) that can be

connected to the Clean Liquid Waste Processing system via connections in the Auxiliary Building Warehouse and then to one of two coolant waste monitor tanks or the aerated waste monitor tank (5,000 gal), which offers a final check on the liquid waste to be released. The total storage capacity of the monitor tanks is approximately one RCS volume (60,000 gal. nominal).

All liquid to be released will be sampled to ensure that the limits set forth in 10 CFR Part 20, Sections 1301 and 1302, and Appendix B are not exceeded. If the activity level is unacceptable after sampling the monitor tank contents, then the liquid is reprocessed through the demineralizers or the liquid radioactive waste processing system. If the activity level is within discharge limits, the liquid is pumped through a final filter, at a rate selectable over a range of 10 gpm to 132 gpm, to the circulating water system where it is diluted with the water in the discharge conduit. The proper rate of liquid release to the discharge conduit is determined by sampling the liquid to be released. The concentrations of the limiting isotopes for release are determined. Based on the circulating water flow rate, the discharge rate is selected such that the releases to unrestricted areas are within permissible concentrations as specified in 10 CFR Part 20, Appendix B. The second isolation valve in the waste discharge header to the circulating water system is provided with a panel mounted manual controller for setting the desired flow rate. The clean liquid waste processing system effluent enters the circulating water system discharge box that runs along the south wall of the auxiliary building, as shown in Figure 1.2–10. The routing of the discharge structure to the quarry is shown on the plot plan, Figure 1.2–2. The circulating water is further diluted by the Long Island Sound tidal flow.

The only liquid discharge path from the clean liquid waste processing system to the environment is through the discharge header, which contains a radiation monitor. The radiation monitor and redundant isolation valves are installed between the waste processing system and the circulating water system. This radiation monitor annunciates in the control room on high radioactivity level and instrument failure, and will automatically close the isolation valves to prevent further discharge. The single radiation monitor serves as a backup system to sampling of the liquid to be released. All liquid to be released is sampled to confirm that the regulations of 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I are met. The radiation monitor is described in Section 7.5.6.

To prevent the addition of waste to a monitor tank, the content of which is being discharged to the circulating water system, a valve interlock system is provided. The pneumatic valves, located in the discharge piping from each tank, can be opened and kept open only if the corresponding valve in the inlet piping to each tank is closed. This arrangement prohibits the addition of wastes to a previously sampled tank whose content is being released.

Aerated Liquid Waste Processing System:

The design of the aerated liquid waste processing system, shown in Figure 11.1–4, is based upon the processing of radioactive liquids (other than those handled by the clean liquid waste processing system) postulated to be generated annually. However, the expected quantity of wastes to be generated and processed annually by this system is approximately 600,000 gallons, based on the assumptions used in the 10 CFR 50 Appendix I analysis.

The aerated liquid waste system design is based upon the processing of wastes with a radioactivity level resulting from normal reactor operation with design basis reactor coolant activity concentrations. The performance of the aerated liquid waste processing system during normal operation is evaluated based upon processing of wastes with radioactivity levels resulting from expected reactor coolant source terms as well as from design basis reactor coolant source terms.

Aerated liquid wastes are collected by the open drains system (shown in Figure 11.1–3) that empties into one of two drain tanks. The system is designed for batch processing. The contents of the drain tanks are monitored by sampling and processed through either a system that contains a filter, and one hard-piped demineralizer or a radioactive liquid waste processing system (activated carbon filters and demineralizers) which can be connected to the Aerated Waste Liquid Processing System via connection points located in the Auxiliary Building Warehouse.

The demineralizer/radioactive liquid waste processing system effluent is collected in the aerated waste monitor tank, or the clean waste monitor tanks which offers a final check on the liquid to be released. If the radioactivity level is within discharge limits, as determined by sampling, the liquid is then pumped, at a rate selected over a range of 10 gpm to 100 gpm, to the circulating water system, where it is diluted with water in the discharge conduit. The second isolation valve in the waste discharge header to the circulating water system is provided with a panel mounted manual controller for setting the desired flow rate. The liquid released from the aerated liquid radioactive waste processing system is monitored continuously for radioactivity. The radiation monitor annunciates in the control room on high radioactivity and instrument failure, and will automatically close two isolation valves in the discharge piping to prevent further discharge. The radiation monitor is discussed in Section 7.5.6. Provisions are furnished for recycling the monitor tank’s content for further processing.

The only liquid discharge path from the aerated liquid waste processing system to the environment is through the discharge pipe containing the radiation monitor. All system leakage, drains, and relief valve flows are collected in the drains system and returned to the aerated liquid waste processing system.

11.1.3.2 System Operation

Clean Liquid Waste Processing Systems:

1. Normal Operation

   Letdown from the RCS is automatically diverted to the radioactive waste processing system on detection of a high volume control tank liquid level. The flow of wastes into the radioactive waste processing system, normally at a rate of 44 gpm, is intermittent due to changes in RCS inventory or boron concentration necessitated by startups, shutdowns, fuel depletion, etc.

   All other liquid wastes input into the clean liquid waste processing system are intermittent due to plant operation. Liquid wastes are collected in either the equipment drain sump tank

(EDST) or the primary drain tank. On high tank level, the equipment drain sump tank contents are pumped to the coolant waste receiver tank. During reactor shutdown for refueling, a greater quantity of waste is anticipated due to draining of system equipment for maintenance or inspection. The operation of the primary drain tank system, for collection of liquid wastes in the containment, is on a batch-type basis. On primary drain tank high tank liquid level, an alarm is annunciated in the main control room to alert the plant operator. The operation of the primary drain tank pumps and opening of the containment isolation valves are initiated manually from the control room. The initiation of the operation from the main control room is required to maintain the tank normal operating temperature of 120°F.

The processing of EDST and letdown liquid wastes up to the coolant waste receiver tanks is on a fully automatic, unattended basis. Processing up to the coolant waste receiver tanks consists of filtration, degasification (when needed), and ion exchange. Degassing the RCS prior to shutdown facilitates subsequent opening of the system.

The processing of wastes downstream of the coolant waste receiver tanks occurs on a batch-type basis. The receiver tank content is sampled to determine the amount of further processing required. A remotely actuated recirculation system is provided for each tank for the purpose of taking a representative sample. Normal processing consists of demineralization or processing though a liquid radioactive waste processing system (activated carbon filters and demineralizers).

The processed wastes are collected in the monitor tanks, which offer a final check on the liquid to be released. Sampling is employed to determine the required discharge rate to ensure releases are within 10 CFR Part 20, Sections 1301 and 1302 and Appendix B and 10 CFR Part 50, Appendix I limits.

2. Abnormal Operation

Abnormal operation of the clean liquid waste processing system may involve reactor operation with greater than expected failed fuel and larger than anticipated volume of liquid waste generated, such as for cold-shutdowns and startups late in the core cycle. Since the clean liquid radioactive waste processing system is designed for reactor operation with design basis reactor coolant activities for processing 14 RCS volumes of waste per year and with sufficient storage capacity, system operation for these abnormal occurrences will be the same as for normal operation described above.

Aerated Liquid Waste Processing Systems:

1. Normal Operation

The aerated liquid waste processing system is operated on a batch type basis. System processing will vary with the volume of waste generated. The typical processing cycle is expected to be 5 to 7 days.

Normal processing consists of filtration and demineralization or processing though a liquid radioactive waste processing system (activated carbon filters and demineralizers). The aerated waste monitor tank provides a final check on the liquid to be released to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR 50 Appendix I limits. All potential releases are sampled prior to discharge to the environment.

2. Abnormal Operation

Abnormal operating occurrences considered are larger than expected volumes of wastes, such as for steam generator blowdown processing. If the waste generation rate exceeds the capacity of the system, interconnecting piping is provided for pumping the wastes to the clean liquid radioactive waste processing system for treatment.

11.1.4 GASEOUS WASTE PROCESSING SYSTEM

11.1.4.1 System Descriptions

The gaseous waste processing system, shown in Figure 11.1–5, processes potentially radioactive hydrogenated waste gases. The system design is based upon the processing of radioactive gases postulated to be released from the RCS during normal operation with design basis reactor coolant activities. The gaseous waste processing system is shown as one of the streams in Figure 11.B–2.

Waste gases flow to the waste gas header and are collected in the waste gas surge tank. When the surge tank pressure increases to approximately 3 psig, one of the two 25 scfm compressors is automatically started by pressure instrumentation located on the tank. The surge tank gases are compressed into one of the six waste gas decay tanks, where gases are stored at a maximum pressure of 150 psig.

The storage capacity of each waste gas decay tank is based on the storage of waste gases expected to enter the system during any two months of normal operation. For all the waste gases entering the system, including those contributed by operational occurrences, the six decay tanks provide adequate storage capacity for a decay time of 90 days. Gases are held in the decay tanks until the radioactivity level has been reduced by decay and the gases are suitable for release. Prior to release, the gases in the decay tanks are sampled to determine compliance with the regulations of 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR Part 50, Appendix I.

The decay tanks discharge through an absolute (i.e., HEPA) filter to the Millstone stack. The discharge pipe contains a radiation monitor and redundant automatic isolation valves. A radiation monitor annunciates in the control room on high radioactivity level and instrument failure and will automatically close the isolation valves to prevent any further release.

The release rate for waste gases is selectable over a range of 10 scfm to 50 scfm. The rate is determined by sampling the content of the decay tank to be discharged. The radionuclide concentrations are correlated with the potential site boundary dose and the 10 CFR Part 20, Appendix B Effluent Concentrations, and the release rate is determined.

The radiation monitor in the waste gas discharge header is provided as a backup system to the sampling of the decay tank contents. The radiation monitor is described in FSAR Section 7.5.6.

The pneumatic valve in the inlet to each decay tank is interlocked with the corresponding valve in the tank discharge piping. The discharge valve can be opened and remains open only when the inlet valve is closed. This feature precludes the inadvertent release of waste gases that are not sampled.

11.1.4.2 System Operation

1. Normal Operation

Waste gases from the sources shown in Figure 11.B–2 are collected in the waste gas surge tank through the waste gas header. As the pressure in the surge tank increases to 3 psig, one compressor is automatically started by pressure instrumentation mounted on the surge tank. If the surge tank pressure continues to increase, the second compressor is automatically started at 5 psig. The compressor discharges the waste gases into one of the six decay tanks selected by the operator. When the decay tank being filled reaches approximately 140 psig pressure, an alarm is annunciated in the control room to alert the operating personnel. The inlet valve to the decay tank is closed remotely by the operator, and one of the five remaining decay tanks is selected to receive the gases.

After an appropriate storage period, and after sampling has confirmed that 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR 50 Appendix I limits are met, the gases are released on a controlled batch basis to the Millstone stack.

2. Abnormal Operation

The gaseous waste processing system, under abnormal conditions (e.g., unexpectedly large volumes of gases) will function in a fashion similar to that for normal conditions. Suitable storage capability is available to store the cover gas from the coolant waste receiver and monitor tanks resulting from back-to-back cold shutdowns and startups late in core life.

11.1.4.3 Gaseous Release Radiological Consequences

Annual design basis and expected releases of radioactive gaseous waste are presented in Table 11.A-9. Gaseous release radiological consequences are presented in Appendix 11.C.

11.1.4.4 Waste Gas System Failure

11.1.4.4.1 General

The limiting accident considered is the postulated and uncontrolled release to the auxiliary building of the radioactive xenon and krypton gases stored in one waste gas decay tank. The credibility of such an occurrence is low since the waste gas system is not subjected to pressures

greater than 150 psig, or large stresses. The result of a rupture of a gas decay tank is analyzed in order that the maximum hazard, which would result from a malfunction in the radioactive waste process system, will be defined.

11.1.4.4.2 Method of Analysis

It is assumed that the tank contains the gaseous activity evolved from degassing one system volume of reactor coolant for refueling. The maximum activity would exist prior to cold shutdown at the end of an operating cycle during which extended operation with one percent defective fuel had occurred. Based on this and neglecting decay after degasification, the noble gas activity in the tank is given in Table 11.1-5.

11.1.4.4.3 Results of Analysis

	Dose (rems)	
Organ	**Exclusion Area Boundary (EAB)**	**LPZ**
Thyroid	--	--
Whole Body	6.4E-01 (1)	6.6E-02 (1)

(1) The current waste gas system failure analysis is based on updated reactor coolant design activity contained in Table 11.A-1. This analysis results in a whole body dose of 3.0E-01 rem at the EAB and 4.0E-02 rem at the LPZ. The current results are bounded by the licensed values listed above.

11.1.4.4.4 Conclusions

If a waste gas decay tank rupture did occur, the dose would be substantially below 10 CFR Part 100 guidelines.

11.1.5 SOLID WASTE PROCESSING SYSTEM

11.1.5.1 System Descriptions

The solid waste processing system is designed to provide controlled handling of spent resins, contaminated filter cartridges, and miscellaneous solid waste. The system is designed for handling solid waste with radioactivity levels resulting from reactor operation with design basis reactor coolant source terms. The sources and expected annual volumes of solid wastes are given in Table 11.1-2.

The estimated isotopic curie inventory for each source of solid waste is given in Table 11.1-3. The activity data in Table 11.1-3 are based on analyses done for the original licensing of Millstone

Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to this FSAR chapter.

Design of the solid waste processing system for handling and disposal of each type of solid waste is as follows:

A. Spent Resins

Spent resins from the radioactive waste processing system demineralizers, CVCS ion exchangers, and spent fuel pool demineralizer are replaced in accordance with plant procedures. Resin replacement is accomplished by sluicing the resins from the hard-piped demineralizers and ion exchangers with nitrogen and/or demineralized water to the spent resin tank. Spent resins generated from other liquid waste processing systems may be sluiced directly to a solid waste processing system for ultimate off site disposal in a shipping cask.

Spent resins are accumulated and stored in the spent resin tank for radioactive decay prior to filling the disposable container located in a shipping cask at elevation (-)45 feet, 6 inches in the auxiliary building. The spent resin tank is sized for storing the total volume of resins resulting from one resin replacement per year per demineralizer. With the storage capacity available, minimum storage time of about six months is expected for resins in the tank. The resins are dewatered by the use of a pump after placement in the disposable container. Alternatively, resins sluiced directly to a solid waste processing system are not stored in the spent resin tank, but rather transferred directly to a disposable container, which is then dewatered via its own solid waste processing system.

Solid waste containers, shipping casks, methods of packaging, and transportation meet applicable federal regulations 10 CFR Part 71and 10 CFR 171-178, and wastes are buried at a licensed burial site in accordance with applicable NRC 10 CFR Part 61. Solid waste treatment design is in compliance with the requirements of 10 CFR Part 20, Sections 1301 and 1302 as it relates to radioactivity in effluents to unrestricted areas.

The spent resin radwaste system is shown in Figure 11.1–7.

B. Contaminated Filter Cartridges

Filter cartridges for all radioactive service filters are normally replaced when the pressure drop across the filter unit exceeds 40 psi. All filters are located in concrete shielded compartments with access provided by a hatch located in the roof of the compartment. (See FSAR Section 11.2 for shielding design.) For the removal of contaminated cartridges, the concrete hatch plug is removed first. Then contaminated cartridges are removed in accordance with plant procedures.

The contaminated cartridges are transferred to shielded containers, which are then stored for ultimate off-site disposal.

C. Miscellaneous Solid Wastes

Contaminated metallic materials and solid objects are placed in disposable shipping containers for transportation to an off site waste-processing facility or disposal site. Miscellaneous compressible wastes, such as contaminated clothing, rags, paper, etc., are transported to the Millstone Radwaste Reduction Facility for compacting.

A temporary storage facility for two high-integrity containers is provided in the container storage area of the auxiliary building. Temporary storage for various containers prior to further disposition can also be provided depending on curie content and container quantity.

Casks used for shipment of filter cartridges and miscellaneous solid wastes are rented as needed.

The containers are designed to prevent loss or dispersal of the contents and to maintain self-shielding properties under normal conditions of transport. Where surface dose rates exceed those allowed by the applicable Department of Transportation (DOT) regulations, the containers are provided with additional shielding. Where the curie content of the containers might exceed the regulatory limit applicable to this type of shipment, special casks licensed by DOT and by NRC, if required, would be used.

The shipment of radioactive waste materials is governed by NRC regulations as set forth in 10 CFR Part 71 and regulations of the U. S. DOT contained in 49 CFR Parts 170 through 178. The packaging and shipping of all waste from the Millstone site is in accordance with these regulations and any other applicable regulations that may come into effect. All loading of radioactive waste containers is performed by qualified personnel and monitored by radiation protection personnel. The transportation is provided by a carrier authorized by the disposal site operator in accordance with NRC and DOT regulations. The material to be shipped from the site is sent to a licensed waste disposal site or to a licensed waste-processing facility.

Table 11.1-4 gives the total estimated annual curie inventory of solid wastes to be shipped from the station for off-site burial. This estimated inventory is based on analyses done for the original licensing of Millstone Unit 2 and is not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

System Operation

1. Normal Operation

Spent resins from the sources listed in Table 11.1-3 are collected in the spent resin tank for radioactive decay, or loaded directly into a shipping container, dewatered using a radioactive solid waste processing system and then sealed and prepared for shipment. The spent resins from the spent resin tank are subsequently loaded into a shipping container and are dewatered by the spent resin shipping cask dewatering pump and portable dewatering pumps. After dewatering, the shipping container and cask are sealed and prepared for shipment.

Prior to a differential pressure of 40 psi being exceeded across any of the filters listed in Table 11.1-3, the filter is taken out of service, and allowed to decay. To change the filter cartridge, the concrete shield plug above the filter is removed. The cartridge is removed in accordance with plant procedures, placed in a shielded area and stored for shipment.

2. Abnormal Operation

No abnormal operations are anticipated.

11.1.6 SYSTEM RELIABILITY AND AVAILABILITY

11.1.6.1 Special Features

The radioactive waste processing system design is based upon the processing of wastes postulated to be generated from reactor operation with design basis reactor coolant activities and the release of these wastes in accordance with the requirements 10 CFR Part 20, Sections 1301 and 1302 and Appendix B. For expected radioactivity levels, sufficient processing capability is available to ensure all releases from the radioactive waste processing system are in accordance with 10 CFR Part 50, Appendix I.

The radioactive waste processing system is designed for as low as reasonably achievable (ALARA) radioactivity releases to the environment. All radioactive waste processing system vents, equipment drains, leakage, valve stem leadoffs, and relief valve discharges are collected by the drains system and reprocessed by the radioactive waste processing system. Those sources containing dissolved fission gases are collected in the equipment drain sump tank and processed as clean liquid waste. All other sources are collected and processed by the aerated liquid waste system. All vent gases containing fission gases are discharged to the gaseous waste system for storage and decay.

Each liquid and gaseous waste processing system is provided with a single discharge path to the environment. Each discharge header is provided with a radiation monitor and redundant isolation valves which are closed on high radiation level or instrument failure to prevent releases not in compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B.

All releases from the radioactive waste processing system to the environment are to be accomplished on a batch basis for suitable control. All wastes are first sampled to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B and 10 CFR Part 50, Appendix I. Adequate sample points are provided in the radioactive waste processing system for suitable control of the processing and to ensure that processing equipment performs as designed. All sampling points in the system are piped to the sampling room for analysis. Some local sampling points are utilized in the aerated and clean liquid waste systems.

To avoid inadvertent releases prior to sampling, the waste gas decay and coolant waste monitor tanks are provided with inlet and outlet valve interlocks. This control provision precludes the draining and filling of the tanks simultaneously.

All storage tanks in the liquid waste system are provided with piping and valves for recirculating the tank contents to obtain a representative sample. The valves are designed for remote operation to protect station operating personnel from radiation.

Sufficient storage capacity is provided for the retention of wastes in the radioactive waste processing system. The three RCS volume storage capacity in the clean liquid radioactive waste processing system is sufficient to allow for back-to-back cold shutdowns and startups up to 75 percent of the equilibrium core cycle. The six decay tanks allow for 60-day storage of all waste gases, including cover gases from the clean liquid radioactive waste processing system tanks.

The radioactive waste processing system is provided with suitable interconnections and sufficient flexibility to allow recirculation of waste for additional processing, if required. Suitable component redundancy is provided to ensure adequate processing.

11.1.6.2 Tests and Inspections

All components of the radioactive waste processing system are nondestructive tested in accordance with the applicable codes and standards as listed in Table 11.1-1. In addition to code requirements, additional testing is required for important components to ensure component integrity and performance.

All pumps in the radioactive waste processing system are manufacturer shop performance tested to demonstrate compliance with design head and capacity requirements.

Prototype filter cartridge testing was conducted by the filter manufacturer confirming a filter efficiency of 92 weight percent when tested with Fine Arizona Air Dust.

The assembled degasifier package was shop gas leak tested to ensure leak-tightness of all components. The degasifier package was subject to a shop performance test to confirm design calculations and performance. Testing to confirm the performance in the removing of dissolved gases was performed with the use of oxygen at feed rate of 60 gpm. Sufficient data were generated with oxygen testing to predict actual performance with respect to hydrogen, krypton and xenon removal. The results of these tests are shown in Figure 11.1–6. The testing also confirmed the capability of the degasifier to perform effective gas stripping under conditions of instantaneous feed rate changes and varying feed temperatures.

All system components are visually inspected and manually adjusted if necessary to ensure correct installation and arrangement. The completely installed system is subject to an acceptance test in accordance with design requirements. The acceptance test is to check and/or calibrate pumps, valves, instrumentation, interlocks, and system operation. In addition, the completely installed system will be checked for environmental considerations, such as correct routing of drains and vent connections, leakage, etc.

Individual system components are located so as to allow access for periodic inspection and testing after component decontamination. Major components are located in individual shielded rooms or compartments to maintain access control.

11.1.7 REFERENCES

11.1-1 NUREG-0017 Rev. 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, PWR-GALE Code”.

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION

CLEAN LIQUID WASTE PROCESSING SYSTEM

Primary Drain Tank Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	250
Design head (TDH) (feet)	150
Design capacity (gpm)	50
NPSH available (feet)	3 to 33
Minimum NPSH required (feet)	3
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor	7.5 hp
Codes and Standards	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Degasifier Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	250
Design head (TDH) (feet)	115
Design capacity (gpm)	132
NPSH available (feet)	2
Minimum NPSH required (feet)	1.5
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Degasifier Pumps (continued)	
Motor (hp)	7.5
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6
Coolant Waste Receiver Tank and Monitor Tank Pumps	
Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	175
Design head (TDH) (feet)	115
Design capacity (gpm)	132
NPSH available (feet)	30
Minimum NPSH required (feet)	10
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor (hp)	7.5
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6
Equipment Drain Sump Tank Pumps	
Type	Vertical wet pit
Quantity	2
Design temperature (°F)	150
Design head (feet)	125
Design capacity (gpm)	50
Material:	
Case	Type 316 stainless steel
Impeller	Type 316 stainless steel
Shaft	Type 316 stainless steel

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Equipment Drain Sump Tank Pumps (continued)	
Motor (hp)	10
Code	Hydraulic Pump Institute
Seismic design class	2
Design integrated radiation dosage (rad)	10^6
Primary Drain Tank and Quench Tank Cooler Pump	
Type	Inline horizontal centrifugal with mechanical seals
Quantity	1
Design temperature (°F)	325
Design head (TDH) (feet)	125
Design capacity (gpm)	100
NPSH available (feet)	3.3–50
Minimum NPSH required (feet)	3
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor (hp)	10
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6
Primary Drain Tank	
Type	Horizontal
Quantity	1
Volume (gallons net)	1,500
Design pressure (psig)	20
Design temperature (°F)	250
Operating pressure (psig)	3
Operating temperature (°F)	120–150
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Coolant Waste Receiver Tanks

Type	Vertical
Quantity	2
Volume (gallons net)	60,000
Design pressure (psig)	15
Design temperature (°F)	150
Operating pressure (psig)	3
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code:	
ASME Section III Class C (1968 Edition including addenda through Summer 1970)	
Seismic design classification	2

Coolant Waste Monitor Tanks

Type	Vertical
Quantity	2
Volume (gallons net)	30,000
Design pressure (psig)	15
Design temperature (°F)	150
Operating pressure (psig)	3
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code:	
ASME Section III Class C (1968 Edition including Addenda through Summer 1970)	
Seismic design classification	2

Equipment Drain Sump Tank

Type	Vertical
Quantity	1
Volume (gallons net)	500
Design pressure (psig	20
Design temperature (°F)	150
Operating pressure (psig)	3

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Equipment Drain Sump Tank (continued)

Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

Degasifier

Type	Packed column utilizing internally generated stripping steam.
Quantity	1
Design pressure (psig)	50
Design temperature (°F)	250
Operating pressure (psig)	5.3
Operating temperature (°F)	228
Design capacity (gpm)	40 to 132
Performance	See Figure 11.1–6

Design codes:

Components containing radioactive material:

ASME Section III Class C (1968 Edition including addenda through summer 1970), ANSI B31.7 Class III, TEMA R.

Other components:

ASME Section VIII, ANSI B31.1

Column:

Packing	One inch Rashig rings.
Height (feet)	4.25
Diameter (feet)	2

Materials:

Components containing radioactive material: ASTM A-240 TP 304

Other components: Carbon steel

Seismic design classification 2

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Coolant Waste– Demineralizers (Primary and Secondary)

Type	Mixed-bed, non-regenerative
Quantity	4 (See Note 3)
Design pressure (psig)	100
Design temperature (°F)	150
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow (gpm)	132
Design code	ASME Section III Class 3 (1971)
Material	ASTM A-240 TP 304
Seismic design classification	2

Coolant Waste–Filters

Type	Disposable cartridge
Quantity	2
Design pressure (psig)	200
Design temperature (°F)	250
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow rate (gpm)	132
Operating flow (gpm)	40–132
Filter rating (micron)	3
Design filter efficiency (%)	80

Design code:

ASME Section III Class C (1968 Edition including Addenda through summer 1970).

Materials:

Vessel	ASTM A-312 TP 304
Internals	TP 304 SS, Micarta
Cartridges	Ethylene propylene

Seismic design classification 2

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Coolant Waste–Piping and Valves

Piping:

Material	ASTM A-312 TP-304 or 316/316L
Design pressure (psig)	50, 100, 150
Design temperature (°F)	150, 250, 300, 350
Joints 2.5 inches and larger	Butt welded except at flanged equipment.
Joints 2 inches and smaller	Socket welded except at flanged equipment **

Codes:

Fabrication	ANSI B31.7 Class III *
Testing and Installation	ASME Section III Class 3 (1971) *, **

Valves: ASTM A-182 F304, F316; ASTM A-351 CF8, CF-8M, ASME SA-479

Ratings:

2.5 inches and larger	150 lb ANSI
2 inches and smaller	600 lb ANSI ***

Code:

ASME Draft for Pumps and Valves for Nuclear Service (1968) *, **

* Portions of the Clean Liquid Waste Processing System have been replaced with piping and piping components designed, constructed and tested to the ANSI B31.1 Power Piping Code with augmented requirements per the guidance in Regulatory Guide 1.143.

** Portions of the Clean Liquid Radwaste System have been designed, constructed and tested to the ANSI B31.1 Piping Code with augmented quality requirements per Regulatory Guide 1.143.

*** 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 600 lb, based on the design conditions of the specific application.

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

AERATED LIQUID WASTE PROCESSING SYSTEM

Aerated Waste Drain and Monitor Tanks

Type	Vertical
Quantity	3
Volume (gallons net)	5,000
Design pressure	Atmospheric
Design temperature (°F)	150
Operating pressure	Atmospheric
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design codes	ASME Section III – ASME Section VIII (Original Fabrication)
Seismic design classification	2

Aerated Waste Demineralizer

Type	Mixed-bed, non-regenerative
Quantity	1
Design pressure (psig)	100
Design temperature (°F)	150
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow (gpm)	132
Normal operating flow (gpm)	50
Design code	ASME Section III Class 3 (1971)
Material	ASTM A-240 TP 304
Seismic design classification	2

Filters

Type	Disposable cartridge
Quantity	2
Design pressure (psig)	200
Design temperature (°F)	250
Operating pressure (psig)	60

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Operating temperature (°F)	120
Design flow rate (gpm)	132
Operating flow (gpm)	40–132
Filter rating (micron)	3
Design filter efficiency (%)	80
Design code:	
ASME Section III Class C (1968 Edition including Addenda through summer 1970).	
Seismic design	2

"A" Aerated Waste Drain Tank Pump

Type	Inline horizontal centrifugal with mechanical seals
Quantity	1
Design temperature (°F)	180
Design head (TDH) (feet)	238
Design capacity (gpm)	90
NPSH available (feet)	27
Minimum NPSH required (feet)	9
Horsepower (hp)	15
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Codes and standards	ASME Section III Class 3 (1971)
Seismic design classification	2
Design integrated radiation dosage (rads)	10^6

"B" Aerated Waste Drain Tank and Monitor Tank Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quality	2
Design temperature (°F)	150
Design head (TDH) (feet)	135
Design capacity (gpm)	50
NPSH available (feet)	27
Minimum NPSH required (feet)	5
Horsepower (hp)	5

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Codes and standards	ASME Section III Class 3 (1971)
Seismic design classification	2
Design integrated radiation dosage (rads)	10^6

Piping and Valves

Piping:	
Material	ASTM A-316 TP 304 or 316L
Design pressure (psig)	50 and 100
Design temperature (°F)	150
Joints:	
2.5 inches and larger	Butt welded except at flanged equipment
2 inches and smaller	Socket welded except at flanged equipment
Codes:	
Fabrication	ANSI B31.7 Class III *
Testing and Installation	ASME Section III Class 3 (1971) *
Valves	
Material	ASTM A-182 F 304, F316, ASME SA-479
Ratings:	
2.5 inches and larger	150 lb ANSI
2 inches and smaller	600 lb ANSI**
Code	ASME Draft for Pumps and Valves for Nuclear Service (1968) *

* Portions of the Aerated Liquid Waste Processing System have been replaced with piping and piping components designed, constructed and tested to the ANSI B31.1 Power Piping Code with augmented requirements per the guidance in Regulatory Guide 1.143.

** 2 inch and smaller Socket-Welding Ends valves were originally specified as an ANSI 600 lb rating. The pressure temperature rating class of valves replaced in accordance with the Dominion Design Control Process is determined based on the design conditions of the application in accordance with applicable Construction Code requirements. Therefore, the pressure temperature rating class of installed replacement valves may be higher or lower than 600 lb, based on the design conditions of the specific application.

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

GASEOUS WASTE PROCESSING SYSTEM

Waste Gas Compressors

Type	Diaphragm, single state
Quantity	2
Capacity (scfm)	25 at 14.7 psia suction pressure
Design discharge pressure (psig)	150
Design pressure (psig)	165

Materials:

Heads - ASTM A-105 Grade 2, with ASTM A-240 TP 304 in contact with waste gases.

Diaphragm	TP 301 stainless steel or 316 stainless steel
Support cylinder	ASTM A-105 Grade 2
Motor (hp)	40
Seismic design classification	See Note 1
Design integrated radiation level (rads)	3.5×10^5

Code:

ASME Section VIII, ASME Draft for Pump and Valves for Nuclear Service, ANSI B31.7.

Waste Gas Surge Tank

Quantity	1
Type	Vertical
Design pressure (psig)	20
Design temperature (°F)	150
Normal operating pressure (psig)	3 to 5
Normal operating temp (°F)	150
Volume (ft^3)	approx 609
Material	ASTM A-240 TP 304
Seismic design classification	2

Code:

ASME Section III Class C (1968 Edition including Addenda through 1970).

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Waste Gas Decay Tank

Type	Vertical
Quantity	6
Design pressure (psig)	165
Design temperature (°F)	150
Normal operating pressure (psig)	5 to 150
Normal operating temp (°F)	120
Volume (ft^3)	582 Specified 627 As-Built
Material	ASTM A-515 Grade 70
Seismic design classification	See Note 1

Code:

ASME Section III Class C (1968 Edition including Addenda through summer 1970).

Waste Gas Filter

Type	Disposable cartridge with HEPA filter and demister
Quantity	1
Design flow (scfm)	50
Normal operating flow (scfm)	5 - 25
Design pressure (psig)	165
Design temperature (°F)	150
Normal operating pressure (psig)	4
Normal operating temp (°F)	120
Design efficiency	99.97% of particles 0.3 micron and larger
Material:	
Filter housing	ASTM A-240 TP 304
Internals	TP 304 stainless steel
Cartridge	Glass fiber
Code	ASME Section III Class 3 (1971)
Seismic design classification	See Note 1

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

Piping and Valves

Piping:	
Material	ASTM A-106 Grade B
Design pressure (psig)	50, 150 and 200
Design temperature (°F)	150, 200 and 300
Joints:	
2.5 inches and larger	Butt welded except at flanged equipment.
2 inches and smaller	Socket welded except at flanged equipment.
Codes:	
Fabrication	ANSI B31.7 Class III
Testing and Installation	ASME Section III Class 3 (1971)
Design seismic classification	2 (See Note 1)
Valves:	
Materials	ASTM A-216, Grade WCB or ASTM A-105
Ratings:	
2.5 inches and larger	150 lb ANSI butt weld ends
2 inches and smaller	3,000 lb ANSI socket weld ends
Code:	ASME Draft for Pump and Valves for Nuclear Service (1968), or ASME Section III Class 3 (1971)

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION (CONTINUED)

SOLID WASTE PROCESSING SYSTEM

Spent Resin Storage Tank

Type	Vertical
Quantity	1
Volume (ft^3)	380
Design pressure (psig)	75
Operating pressure (psig)	35
Design temperature (°F)	175
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

Spent Resin Shipping Cask Dewatering Pump

Type	Horizontal Centrifugal
Quantity	1
Capacity (gpm)	10
Head (feet)	92
Material	Stainless steel

Note 1: The high pressure components of the gaseous waste processing system are designated as Seismic Class 2 equipment. However, the components are designed to meet Seismic Class 1 loadings due to the nature of the equipment service.

Note 2: The supply lines to the solidification concentrates pump were cut in accordance with PDCR 2-54-95.

Note 3: Secondary Demineralizer may be filled with either ion-specific resin or mixed bed resin.

TABLE 11.1-2 SOURCES AND EXPECTED VOLUMES OF SOLID WASTES

Source	Waste Generating Operation	Quantity Per Year	Per Year
Spent resins	One resin replacement per demineralizer/year	351 cu ft	7 *
Contaminated filter cartridges	One cartridge replacement per filter/year	12 cartridges	12 **
Miscellaneous solid wastes	One 55 gallon drum/week	-	52 **

* 50 cubic foot containers

** 55 gallon drums

Aerated Waste hardpiped demineralizer at 42 cu. ft. x 1 = 42 cubic feet

Aerated Waste sluicible portable demineralizers at 15 cubic feet/each x 3 = 45 cubic feet

Spent Fuel Pool demineralizer at 42 cubic feet x 1 = 42 cubic feet

Primary Liquid Radwaste demineralizers at 42 cubic feet x 2 = 84 cubic feet

Secondary Liquid Radwaste demineralizer at 42 cubic feet x 1 = 42 cubic feet

Letdown Ion Exchangers at 32 cubic feet x 3 = 96 cubic feet

351 cubic feet

Note: The data in this Table is not derived from the updated 10 CFR 50 Appendix I analysis. The best source of Solid Waste Volumes can be found in the Annual Radiological Effluent Release Report.

TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE)

ION EXCHANGER AND DEMINERALIZER RESINS

	Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Deborating Ion Exchanger Resin		Primary Clean Liquid Waste Demineralizer Demineralizer Resin		Aerated Liquid Waste Demineralizer Resin		Spent Fuel Pool Clean-up Demineralizer Resin	
Isotope	**(Curies)**		**(Li Removal) (Curies)**		**(Curies)**		**(Curies)**		**(Curies)**		**(Curies)**	
	Normal	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**
Cr-51												
Mn-54												
Mn-56												
Co-58												
Fe-59												
Co-60	4.78×10^{-2}	0.478	9.6×10^{-3}	9.6×10^{-2}	1.2×10^{-3}	1.2×10^{-2}	1.12×10^{-4}	1.9×10^{-3}	8.72×10^{-5}	1.01×10^{-2}	1.23	12.3
Rb-88	1.366	13.66	0.246	2.46	--	--	3.1×10^{-3}	5.22×10^{-2}	2.4×10^{-3}	0.279	67.2	672
Rb-89	3.06×10^{-2}	0.306	5.5×10^{-3}	5.5×10^{-2}	--	--	6.9×10^{-5}	1.17×10^{-3}	5.38×10^{-5}	6.25×10^{-3}	1.69	16.9
Sr-89	13.8	138.0	2.398	23.98	--	--	3.11×10^{-2}	0.525	2.41×10^{-2}	2.80	0.134	1.34
Sr-90	3.052	30.52	0.471	4.71	--	--	6.77×10^{-3}	0.114	5.25×10^{-3}	0.610	6.88×10^{-3}	6.88×10^{-2}
Y-90	--	--	2.24×10^{-2}	0.224	--	--	4.31×10^{-5}	7.26×10^{-4}	3.34×10^{-5}	3.88×10^{-3}	2.69×10^{-2}	0.269
Sr-91	6.9×10^{-2}	0.691	1.26×10^{-2}	0.126	--	--	1.57×10^{-4}	2.65×10^{-3}	1.22×10^{-4}	1.41×10^{-2}	9.39×10^{-2}	0.939
Y-91	--	--	7.71×10^{-2}	0.771	--	--	1.48×10^{-4}	2.5×10^{-3}	1.15×10^{-4}	1.33×10^{-2}	2.93	29.3
Zr-95												
Mo-99	--	--	52.46	524.6	--	--	0.101	1.70	7.82×10^{-2}	9.08	53.5	535
Ru-103	7.84	78.4	1.57×10^{-2}	0.157	--	--	1.51×10^{-2}	0.255	1.17×10^{-2}	1.36	0.109	1.09
Ru-106	2.02	20.2	0.363	3.63	--	--	4.58×10^{-3}	7.72×10^{-2}	3.55×10^{-3}	0.412	6.54×10^{-3}	6.54×10^{-2}

TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) (CONTINUED)

ION EXCHANGER AND DEMINERALIZER RESINS

	Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Deborating Ion Exchanger Resin		Primary Clean Liquid Waste Demineralizer Demineralizer Resin		Aerated Liquid Waste Demineralizer Resin		Spent Fuel Pool Clean-up Demineralizer Resin	
Isotope	**(Curies)**		**(Li Removal) (Curies)**		**(Curies)**		**(Curies)**		**(Curies)**		**(Curies)**	
	Normal	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**
I-129	9.36×10^{-4}	9.36×10^{-3}	1.56×10^{-4}	1.56×10^{-3}	--	--	2.1×10^{-6}	3.54×10^{-5}	1.63×10^{-6}	1.89×10^{-4}	1.9×10^{-6}	1.9×10^{-5}
Te-129	5.42×10^{-2}	0.542	1.09×10^{-2}	0.109	1.3×10^{-3}	1.3×10^{-2}	1.28×10^{-4}	2.15×10^{-3}	9.88×10^{-5}	1.15×10^{-2}	0.662	6.62
I-131	1,680.0	16,800.0	336.0	3,360.0	19.2	192.0	3.91	65.9	3.03	352	105	1,050
I-132	5.04	50.4	1.007	10.07	0.121	1.21	1.19×10^{-2}	0.20	9.19×10^{-3}	1.07	26.9	269
Te-132	49.3	493.0	9.86	98.6	0.944	9.44	0.116	1.95	8.96×10^{-2}	10.4	8.7	87.0
I-133	247.0	2,470.0	4.94	49.4	5.91	59.1	0.496	8.35	0.384	44.6	149	1,490
Cs-134	388.0	3,800.0	517.0	5,170.0	--	--	1.74	29.3	1.35	156	2.64	26.4
I-134	1.06	10.6	0.212	2.12	2.54×10^{-2}	0.254	2.49×10^{-3}	4.2×10^{-2}	1.93×10^{-3}	0.225	16.3	163
Te-134	3.19×10^{-2}	0.319	6.4×10^{-3}	6.4×10^{-2}	7.65×10^{-4}	7.65×10^{-4}	7.5×10^{-5}	1.26×10^{-3}	5.82×10^{-3}	6.76×10^{-3}	0.691	6.91
I-135	35.9	359.0	7.24	72.4	0.869	8.69	8.46×10^{-2}	1.43	6.56×10^{-2}	7.62	71.2	712
Cs-136	--	--	18.12	181.2	--	--	3.48×10^{-2}	0.587	2.7×10^{-2}	3.14	0.672	6.72
Cs-137	1,695.0	16,950.0	1,728.0	17,280.0	--	--	6.58	111	5.10	593	8.44	84.4
Cs-138	--	--	0.14	1.40	--	--	2.69×10^{-4}	4.54×10^{-3}	2.09×10^{-4}	2.42×10^{-2}	18.2	182
Ba-140	4.08	40.8	0.743	7.43	--	--	9.27×10^{-3}	0.156	7.19×10^{-3}	0.835	0.161	1.61
La-140	0.469	4.69	9.37×10^{-2}	0.937	--	--	1.08×10^{-3}	1.82×10^{-2}	8.38×10^{-4}	9.74×10^{-2}	0.154	1.54
Pr-143	3.49	34.9	0.697	6.97	--	--	8.05×10^{-3}	0.136	6.24×10^{-3}	0.725	0.154	1.54
Ce-144	26.2	262.0	5.24	52.4	--	--	6.04×10^{-2}	1.02	4.68×10^{-2}	5.44	0.109	1.09

TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) FILTER CARTRIDGES

	Chemical and Volume Control System Filter Cartridges		Clean Liquid Waste Filter Cartridges		Aerated Liquid Waste Filter Cartridges		Spent Fuel Pool Clean-up Filter Cartridges		Reactor Vessel Head Decontamination System Filter Cartridges	
Isotope	(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)	
	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum
Cr-51	0.209	13.05	0.001	0.043	--	0.452	0.003	0.185	0.039	0.039
Mn-54	0.935	0.67	0.002	0.002	0.004	0.024	0.004	0.003	0.028	0.028
Mn-56	--	0.486	--	0.002	--	0.018	--	--	--	--
Co-58	64.0	107.0	0.123	0.347	0.192	3.702	0.50	0.84	4.78	4.78
Fe-59	0.192	0.069	0.001	0.001	--	0.002	0.002	0.001	0.022	0.022
Co-60	0.025	61.0	--	0.166	--	1.766	0.068	0.142	0.533	0.533
Br-84										
Rb-88										
Rb-89										
Sr-89										
Sr-90										
Y-90										
Sr-91										
Y-91										
Zr-95	--	--	--	--	--	--	--	--	--	--
Mo-99										
Ru-103										
Ru-106										
I-129										
Te-129										
I-131										

TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) FILTER CARTRIDGES (CONTINUED)

	Chemical and Volume Control System Filter Cartridges		Clean Liquid Waste Filter Cartridges		Aerated Liquid Waste Filter Cartridges		Spent Fuel Pool Clean-up Filter Cartridges		Reactor Vessel Head Decontamination System Filter Cartridges	
Isotope	(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)	
	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum
I-132										
Te-132										
I-133										
Cs-134										
I-134										
Te-134										
I-135										
Cs-136										
Cs-137										
Cs-138										
Ba-140										
La-140										
Pr-143										
Ce-144										

Note: The activity data in this Table are based on analysis done for the original licensing of Millstone Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

TABLE 11.1-4 CURIE INVENTORY OF SOLID WASTE SHIPPED FROM MILLSTONE UNIT 2 (SEE NOTE)

Source	Normal Curies per Year	Maximum Curies per Year
Resins	4,215	42,871
Filter Cartridges	35	149
Concentrates	Less than 1	5
TOTAL	4,250	43,025

Note: The activity data in this Table are based on analysis done for the original licensing of Millstone Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11. The best source of Solid Waste data can be found in the Annual Radiological Effluent Release Report.

TABLE 11.1-5 ASSUMPTIONS FOR WASTE GAS DECAY TANK ACCIDENT

Assumption (1) Maximum Noble Gas Activity in Waste Gas Decay Tank.

Isotope	Activity, curies
Kr-85m	278
Kr-85	942
Kr-87	160
Kr-88	494
Xe-131m	486
Xe-133m	708
Xe-133	45,600
Xe-135m	26.2
Xe-135	1,280
Xe-137	28.9
Xe-138	101

Basis: Maximum activity in tanks based on 1% degraded fuel, core power = 2700 MWt and degassing one system volume.

Assumption (2) 2 Hour Ground Level Release.

Basis: Regulatory Guide 1.24

Assumption (3) Ground Level X/Q:

EAB hr. = 3.66 E -04 (sec/m^3)

LPZ 4.80 E -05 (sec/m^3)

Basis: 95% maximum X/Q's during the years 1974 -1981.

FIGURE 11.1–1 P&ID CLEAN LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–2 P&ID CLEAN LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–4 P&ID AERATED LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–5 P&ID DIAGRAM GASEOUS RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–6 DEGASIFIER PERFORMANCE CURVE

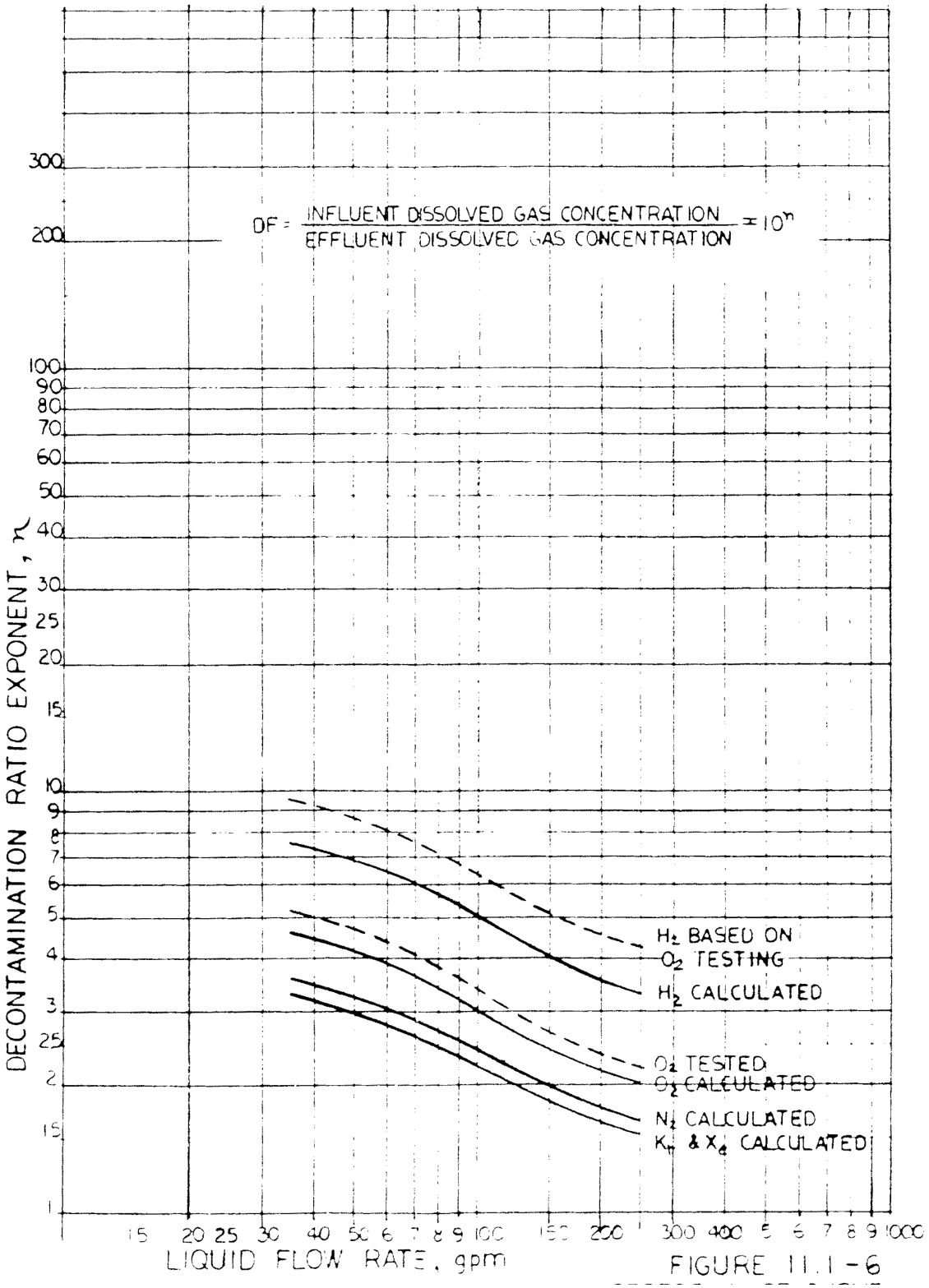


FIGURE 11.1-6

FIGURE 11.1–7 P&ID SPENT RESIN RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

11.2 RADIATION PROTECTION

11.2.1 FINAL SAFETY ANALYSIS REPORT UPDATE

It should be noted that the information in this section provided the basis and description of the shielding design prior to operation. As the station has operated, dose rates have fluctuated within the various areas as a function of time but for the most part have remained well within the zone designation limits presented below. Minor changes in equipment layout from that indicated in Figures 11.2–1 through 11.2–10 have been made, but since the values and figures given in this section provided the design basis for the major shielding structures, they are not being changed.

Many minor shielding changes (mostly through the use of portable shielding) have been incorporated in the interest of maintaining occupational exposures ALARA. The only major shielding change was the installation of a reactor cavity neutron shield.

The values and figures given in this section should not be used to estimate area dose rates. Actual health physics surveys are available and should be consulted to get a more accurate and up-to-date indication of radiological conditions.

Appendices 11.D and 11.E provide some of the design basis considerations on the control and expected levels of airborne radioactivity in the plant. Again, health physics survey data should be consulted for more realistic indications of actual airborne radioactivity levels. The occupational exposure due to airborne radioactivity has been insignificant in comparison to the direct radiation exposure.

11.2.2 DESIGN BASES

The shielding is designed to perform two primary functions: to ensure that during normal operation the radiation dose to operating personnel and to the general public is within the radiation exposure limits set forth in 10 CFR Part 20 and to ensure that operating personnel are adequately protected in the event of a reactor incident so that the incident can be terminated without undue hazard to the general public. The shielding design is based on operating at design power level of 2700 MWt with reactor coolant system activity levels corresponding to one percent failed fuel. The shielding design is governed by the limits for radiation levels as follows:

Zone Description	Maximum Dose Rate (mrem/hr)
Uncontrolled, unlimited access (Zone A)	0.5
Controlled, unlimited (40 hr/week) (Zone B)	1.0
Controlled, limited access (6-2/3 hr/week) (Zone C)	15
Controlled, limited access (1 hr/wk) (Zone D)	100
Normally inaccessible (Zone E)	> 100

The maximum external dose rates within the station are shown in the radiation zone diagrams, Figures 11.2–1 through 11.2–10, as described in this subsection and later in Section 11.2.2. These maximum external dose rates are based on one percent failed fuel. Since the expected amount of failed fuel is much less than this, the average external dose rates will be much lower. The unrestricted areas in the plant where construction workers and plant visitors may be are all radiation Zone A areas which have a maximum dose rate of 0.5 mrem/hr. Only plant workers are allowed in radiation zones with dose rates above this level, and their stay times are increasingly limited as the dose rate increases.

The anticipated average dose rates will vary within each zone and with operating conditions. In general, the dose rate normally around process equipment is expected to be at least an order of magnitude lower than indicated by the applicable zone. This is due to the extremely conservative assumptions made concerning the quantity of failed fuel. The direct dose at the site boundary is negligible both in the case of normal operation and under postulated incident conditions.

Shielding wall thicknesses, as shown in Figures 11.2–1 through 11.2–10, were found using the theory of the Reactor Shielding Design Manual by T. Rockwell, Reactor Physics Constants ANL-5800, Table of Isotopes by Perlman and Lederer, Engineering Compendium on Radiation Shielding and Nuclear Engineering Handbook by H. Etherington. The geometric considerations were determined by each physical situation, such as source configurations and distances to dose points.

Assumptions used included those of TID 14844 and the following:

1. If the source was liquid, then self-attenuation by water was used. If the source was gaseous, then no self-attenuation was assumed. If the source was solid, the volume percentages of each solid were used for self-attenuation (for example, solidified waste in drums).

2. Shields included the tank or pipe walls which were assumed to have an attenuation coefficient similar to that for iron.

3. The shielding materials were lead, iron, water, and concrete.

4. Sources were given in Mev/cm^3 sec for each of 7 standard energy bins: 0.4, 0.8, 1.3, 1.7, 2.2, 2.5, > 2.5 Mev, and 6.1 and 7.1 Mev for nitrogen 16. One percent failed fuel was assumed.

5. Flux to dose conversion factors used were those given by Henderson in XDC-59-8-179.

6. The shield thicknesses were determined by the amount of shielding required to achieve a dose rate less than the upper limit for the radiation zone in which the dose point was located.

The shielding design, therefore, is based on conservative assumptions ensuring adequate radiation protection to the general public and to operating personnel as well.

11.2.3 DESCRIPTION

Shielding throughout Millstone Unit 2 is designed in accordance with the criteria specified in Section 11.2.1. Figures 11.2–1 through 11.2–10 show the equipment placement, the quantity of shielding used and the appropriate radiation zones. All components containing radioactive fluids are shown on the figures except the refueling water storage tank (RWST), located as shown on Figure 1.2–2. The maximum direct dose rate from the content of the RWST is 5.6×10^{-3} mr/hr, based on reactor operation with one percent failed fuel. The maximum total site boundary dose due to the RWST is approximately 1.66 mrem/yr. For normal operating conditions, the site boundary doses will be approximately one-tenth of those values given above.

11.2.3.1 Containment Shielding

The containment shielding consisting of the primary shielding, the reactor cavity neutron shielding, the secondary shielding, and the containment wall, is shown on Figure 1.2–6.

Primary shielding is provided to limit radiation emanating from the reactor vessel. The primary shielding is designed to:

a. Attenuate the neutron flux to limit the activation of components and structural steel.

b. Limit the radiation level after shutdown to permit access to the reactor coolant system equipment.

c. To reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel to allow limited access to the containment during normal operation.

The primary shield consists of a minimum of five feet of reinforced concrete surrounding the reactor vessel. The cavity between the primary shield and the reactor vessel insulation is air-cooled to prevent overheating and dehydration of the concrete primary shield wall.

A reactor cavity neutron shield reduces the operating neutron and gamma dose rates in the containment. A dose reduction factor of 40 on the operating floor of the containment was the design goal for the shield. With the postulation of leak-before-break applied to the reactor coolant piping, the neutron shield design does not consider dynamic effects resulting from a design basis LOCA.

The permanent reactor cavity neutron shield consists of borated concrete blocks contained inside stainless steel containers. The concrete blocks are supported by the shielding support bars, which span between the reactor vessel flange and the embedment ring. The borated concrete blocks are 15 inches high and 23.25 inches wide, located all around the reactor vessel beneath the permanent

reactor cavity seal. The bottom and the sides of the shielding are insulated by NUKON blanket insulation to maintain an acceptable shielding temperature as well as limit heat loss from the reactor vessel (Figures 11.2–11 through 11.2–12). Thermal analysis has shown that the borated concrete blocks inside the stainless steel containers could reach a temperature of up to 455°F in some sections of the concrete blocks during normal operation.

The shielding is provided with personnel access openings, located directly above the neutron detector wells. During reactor operation these openings are plugged with borated concrete shield plugs to further minimize neutron flux. The shield plugs are removed through the access openings in the permanent cavity seal for maintenance of the neutron detector instrumentation.

The shielding in combination with the reactor cavity seal described in Section 4.3.10 will provide adequate airflow channels to maintain sufficient cavity cooling during reactor operation.

In addition to the borated concrete shielding, borated polyethylene layers are installed approximately at the reactor vessel flange elevation to reduce streaming between the borated concrete shielding and the primary shield wall. The thickness of the polyethylene shielding ranges from 1 to 3 inches contained in stainless steel liner between the ribs of the support members. The borated polyethylene layers are not insulated. This shielding is cooled by cavity cooling air passing under the shielding at temperatures less than 150°F. Together, the borated concrete blocks and the borated polyethylene layers are capable of reducing the neutron radiation by a dose factor of approximately 16 to 27.

Secondary shielding is provided to reduce the activity from the reactor coolant system to radiation levels which allow limited access to the containment during normal operation and to supplement primary shielding. Nitrogen 16 is the major source of radioactivity in the reactor coolant during operation and controls the thickness of the secondary shield. The secondary shielding consists of a minimum of 3.5 feet of reinforced concrete surrounding the reactor coolant piping, pumps, steam generators and pressurizer. The N-16 activity concentration at the reactor vessel outlet nozzles is 9.8×10^{-5} Ci/cc at full power.

The containment is a reinforced prestressed concrete structure with 3.75 feet thick cylindrical walls and a 3.25 feet thickness dome. In conjunction with the primary and secondary shield, it will limit the radiation level outside the enclosure building due to sources inside the structure to no more than 0.5 mrem/hr at full power operation. The structure is also designed to protect plant personnel from radiation sources inside the structure following a postulated incident.

11.2.3.2 Auxiliary Building Shielding

The function of the auxiliary building shielding is to protect personnel working near various system components, such as those in the CVCS, the radioactive waste processing system, sampling system, and the spent fuel pool cooling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded to reduce the radiation level in it and adjacent compartments as reflected by the zone designations. Source terms used in the design of shielding for major components throughout the

auxiliary building are listed in Table 11.2-1. Similar source terms for all components containing radioactive materials were developed and used in the shielding design.

11.2.3.3 Control Room Shielding

The layout of the control room is shown in Figure 1.2–7. In conformance with General Criterion Number 19 of 10 CFR Part 50, the control room shielding is designed to ensure that the dose will not exceed 5 rem for the duration of the incident (see Subsection 14.18.3.3). The walls of the control room are two feet thick. Under normal operating conditions, the control room is a Zone A region with expected dose rates well below the indicated limits.

11.2.3.4 Spent Fuel Pool Shielding and Fuel Handling Shielding

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the spent fuel pool. it is designed to protect personnel against the radiation emitted from the spent fuel and control rod assemblies.

The refueling cavity above the reactor vessel is flooded to Elevation 36 feet 6 inches to provide a temporary water shield above the components being withdrawn from the reactor vessel.

The water height is approximately 24 feet above the reactor vessel flange. This height assures a minimum of 108 inches of water above the active portion of a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate from the spent fuel assembly is less than 1.0 mrem/hr at the water surface.

Upon removal of the fuel assembly from the reactor vessel, it is moved to the spent fuel pool by the fuel transfer mechanism, via the fuel transfer tube. Concrete shielding is provided around reactor internals storage and the steam generator for personnel protection during refueling. The spent fuel pool in the auxiliary building is permanently flooded to provide a minimum of 108 inches of water above the active portion of a fuel assembly when being withdrawn from the fuel transfer tube and raised by the fuel pool platform crane, prior to insertion in the spent fuel storage rack. The minimum water height above stored fuel assemblies is approximately 24.5 feet during operation of the spent fuel pool cooling system, described in Section 9.5.2.1, to avoid air entrainment at the pump suction intakes. Otherwise, a minimum height of 23 feet prevails in the spent fuel pool and also, while fuel is in movement, above the reactor pressure vessel, to satisfy fission product retention assumptions for fuel handling accident calculations (see Section 14.7.4.2). The sides of the spent fuel pool are six feet thick concrete to ensure a dose rate of less than 0.035 mrem/hr on the outer surface of the spent fuel pool.

11.2.3.5 Piping Systems Shielding

All piping systems containing radioactive material are routed and/or provided with shielding in accordance with the radiation zones given in Section 11.2.1. Consideration is given to maintenance and inspection requirements for components located in shielded compartments through which these lines are routed.

Isometrics of field run piping, two inches and smaller in diameter, were reviewed and approved by Bechtel Corporation Engineering Department.

11.2.4 HEALTH PHYSICS PROGRAM

Information regarding the health physics program organization is presented in Section 12.5 of Millstone 3 Final Safety Analysis Report (Reference 11.2-1). That information is contained herein by reference.

11.2.5 REFERENCES

11.2-1 Millstone Unit 3, Final Safety Analysis Report, Section 12.5 - Health Physics Program

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGN

Isotope	Letdown Pre-Filter (Curies)	Purification Ion Exchanger (Curies)	Letdown Post-filter (Curies)	Volume Control Tank Demineralizer	Degasifier (Curies)
Br-84	-	1.23	-	840.6	4.24×10^{-3}
Kr-85m	-	-	-	34.0	0.0
Kr-85	-	-	-	87.1	0.0
Kr-87	-	-	-	18.4	0.0
Kr-88	-	-	-	58.2	0.0
Rb-88	-	38.20	-	4.63	0.23
Rb-89	-	0.82	-	0.11	4.02×10^{-3}
Sr-89	-	339.80	-	0.01	5.24×10^{-4}
Sr-90	-	85.90	-	5.0×10^{-4}	2.63×10^{-5}
Y-90	-	1.79	-	2.1×10^{-2}	1.05×10^{-3}
Sr-91	-	-	-	6.7×10^{-3}	3.32×10^{-4}
Y-91	-	-	-	5.22	0.26
Mo-99	-	-	-	40.86	2.06
Ru-103	-	222.0	-	8.4×10^{-3}	4.24×10^{-4}
Ru-106	-	59.4	-	4.9×10^{-4}	2.49x10–5
Te-129	-	1.54	-	5.0×10^{-2}	2.52×10^{-3}
I-129	-	2.4×10^{-2}	-	1.44×10^{7}	7.29×10^{-9}
I-131	-	4.26×10^{4}	-	8.01	4.04×10^{-1}
Xe-131m	-	-	-	50.8	0.0
Te-132	-	1.40×10^{3}	-	0.65	3.27×10^{-2}
I-132	-	129.0	-	2.03	1.03×10^{-1}
I-133	-	6.32×10^{2}	-	10.90	5.51×10^{-1}
Xe-133	-	-	-	5012.3	0.0
Te-134	-	0.92	-	4.7×10^{2}	2.39×10^{-3}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGN (CONTINUED)

Isotope	Letdown Pre-Filter (Curies)	Purification Ion Exchanger (Curies)	Letdown Post-filter (Curies)	Volume Control Tank Demineralizer	Degasifier (Curies)
I-134	-	27.55	-	1.13	5.72×10^{-2}
Cs-134	-	-	-	8.19	4.13×10^{-1}
I-135	-	933.4	-	5.04	2.54×10^{-1}
Xe-135	-	-	-	172.4	0.0
Cs-136	-	-	-	0.28	1.45×10^{-2}
Cs-137	-	-	-	39.78	2.0
Xe-138	-	-	-	8.16	0.0
Cs-138	-	-	-	1.24	3.27×10^{-2}
Ba-140	-	104.2	-	1.23×10^{-2}	6.23×10^{-4}
La-140	-	13.1	-	1.19×10^{-2}	6.00×10^{-4}
Pr-143	-	-	-	1.08×10^{-2}	5.45×10^{-3}
Ce-143	-	-	-	7.74×10^{-3}	3.91×10^{-3}
Co-60	263.0	-	2.63	1.98×10^{-4}	1.0×10^{-5}
Fe-59	0.45	-	4.5×10^{-3}	3.83×10^{-6}	2.09×10^{-7}
Co-58	690.0	-	6.90	1.42×10^{-3}	7.18×10^{-6}
Mn-56	3.27	-	3.27×10^{-2}	4.14×10^{-3}	2.09×10^{-5}
Mn-54	3.69	-	3.69×10^{-2}	4.95×10^{-6}	2.50×10^{-7}
Cr-51	88.10	-	0.88	4.32×10^{-4}	2.18×10^{-7}
Zr-95	1.54	-	0.15	3.42×10^{-6}	1.73×10^{-7}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS (CONTINUED)

Isotope	Waste Gas Surge Tank	Waste Gas Decay Tank	Aerated Waste Drain Tank	Aerated Waste Demineralizer	Aerated Waste Monitor Tank
Br-84	-	-	8.8×10^{-3}	2.78×10^{-5}	8.8×10^{-4}
Kr-85m	4.09×10^{2}	2.80×10^{2}	-	-	-
Kr-85	$3.13.10^{2}$	2.61×10^{3}	-	-	-
Kr-87	6.97×10^{1}	1.41×10^{1}	-	-	-
Kr-88	4.80×10^{2}	2.10×10^{2}	-	-	-
Rb-88	-	-	0.486	8.63×10^{-4}	0.243
Rb-89	-	-	1.2×10^{-2}	1.85×10^{-5}	6×10^{-3}
Sr-89	-	-	1.08×10^{-3}	7.68×10^{-3}	5.4×10^{-4}
Sr-90	-	-	5.47×10^{-5}	1.94×10^{-3}	2.74×10^{-5}
Y-90	-	-	2.20×10^{-4}	4.05×10^{-5}	2.2×10^{-4}
Sr-91	-	-	7.04×10^{-4}	-	3.52×10^{-4}
Y-91	-	-	5.49×10^{-2}	-	5.49×10^{-2}
Mo-99	-	-	0.430	-	0.430
Ru-103	-	-	8.84×10^{-4}	5.02×10^{-3}	8.84×10^{-5}
Ru-106	-	-	5.19×10^{-5}	1.34×10^{-3}	5.19×10^{-6}
Te-129	-	-	5.24×10^{-3}	3.48×10^{-5}	5.24×10^{-4}
I-129	-	-	1.52×10^{-8}	5.42×10^{-7}	1.52×10^{-9}
I-131	-	-	0.842	9.63×10^{-1}	8.42×10^{-2}
Xe-131m	1.80×10^{3}	1.37×10^{4}	-	-	-
Te-132	-	-	6.81×10^{-2}	3.16×10^{-2}	6.81×10^{-3}
I-132	-	-	0.214	2.92×10^{-3}	2.14×10^{-2}
I-133	-	-	1.15	1.43×10^{-2}	1.15×10^{-1}
Xe-133	1.71×10^{5}	1.17×10^{6}	-	-	-
Te-134	-	-	4.98×10^{-3}	2.08×10^{-5}	4.98×10^{-4}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS (CONTINUED)

Isotope	Waste Gas Surge Tank	Waste Gas Decay Tank	Aerated Waste Drain Tank	Aerated Waste Demineralizer	Aerated Waste Monitor Tank
I-134	-	-	0.119	6.23×10^{-4}	1.19×10^{-2}
Cs-134	-	-	0.861	-	0.430
I-135	-	-	0.530	2.11×10^{-2}	5.3×10^{-2}
Xe-135	3.41×10^{2}	4.86×10^{2}	-	-	-
Cs-136	-	-	3.03×10^{-2}	-	1.52×10^{-2}
Cs-137	-	-	4.18	-	2.09
Xe-138	6.75	2.99×10^{-1}	-	-	-
Cs-138	-	-	1.31×10^{-1}	-	6.6×10^{-2}
Ba-140	-	-	1.30×10^{-3}	2.35×10^{-3}	1.30×10^{-4}
La-140	-	-	1.25×10^{-3}	2.96×10^{-4}	1.25×10^{-4}
Pr-143	-	-	1.14×10^{-3}	-	1.14×10^{-4}
Ce-143	-	-	8.14×10^{-4}	-	8.14×10^{-5}
Co-60	-	-	2.08×10^{-4}	-	2.08×10^{-5}
Fe-59	-	-	4.03×10^{-6}	-	4.03×10^{-7}
Co-58	-	-	1.50×10^{-3}	-	1.50×10^{-4}
Mn-56	-	-	4.35×10^{-3}	-	4.35×10^{-4}
Mn-54	-	-	5.29×10^{-6}	-	5.20×10^{-7}
Cr-51	-	-	4.54×10^{-4}	-	4.54×10^{-3}
Zr-95	-	-	3.60×10^{-6}	-	3.6×10^{-7}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS (CONTINUED)

Isotope	Primary Demineralizer	Coolant Waste Receiver	Secondary Demineralizer	Coolant Waste Monitor Tanks	Primary Drain Tank
Br-84	$4.87x10^{-1}$	0.106	$4.87x10^{-2}$	$5.30x10^{-3}$	$2.65x10^{-1}$
Kr-85m	-	-	-	-	-
Kr-85	-	-	-	-	-
Kr-87	-	-	-	-	-
Kr-88	-	-	-	-	-
Rb-88	$4.87x10^{-1}$	5.84	$4.87x10^{-2}$	$2.92x10^{-2}$	$1.50x10^{+1}$
Rb-89	$1.04x10^{-2}$	0.145	$1.04x10^{-3}$	$7.24x10^{-3}$	$3.62x10^{-1}$
Sr-89	4.34	$1.30x10^{-2}$	$4.34x10^{-2}$	$6.48x10^{-4}$	$3.24x10^{-2}$
Sr-90	1.10	$6.56x10^{-4}$	$1.10x10^{-1}$	$3.28x10^{-5}$	$1.64x10^{-3}$
Y-90	$4.71x10^{-2}$	$2.63x10^{-2}$	$4.71x10^{-3}$	$1.32x10^{-3}$	$6.59x10^{-3}$
Sr-91	$2.28x10^{-2}$	$8.45x10^{-3}$	$2.28x10^{-3}$	$4.22x10^{-4}$	$2.11x10^{-2}$
Y-91	$2.50x10^{2}$	6.59	$2.50x10^{-1}$	$3.29x10^{-1}$	1.65
Mo-99	$9.47x10^{1}$	$5.16x10^{1}$	9.47	2.58	$1.29x10^{1}$
Ru-103	2.83	$1.06x10^{-2}$	$2.83x10^{-1}$	$5.30x10^{-4}$	$2.65x10^{-2}$
Ru-106	$7.59x10^{-1}$	$6.22x10^{-4}$	$7.59x10^{-2}$	$3.11x10^{-4}$	$1.56x10^{-3}$
Te-129	$1.96x10^{-2}$	$6.29x10^{-2}$	$1.96x10^{-3}$	$3.15x10^{-3}$	$1.57x10^{-1}$
I-129	$3.07x10^{-4}$	-	$3.04x10^{-5}$	-	-
I-131	$5.43x10^{2}$	$1.01x10^{1}$	$5.43x10^{1}$	$5.05x10^{-1}$	$2.53x10^{1}$
Xe-131m	-	-	-	-	-
Te-132	$1.77x10^{1}$	0.818	1.77	$4.09x10^{-2}$	2.04
I-132	1.64	2.566	$1.65x10^{1}$	$1.28x10^{-1}$	6.42
I-133	$8.06x10^{1}$	$1.378x10^{1}$	8.06	$6.89x10^{-1}$	$3.45x10^{1}$
Xe-133	-	-	-	-	-
Te-134	$1.16x10^{-2}$	$5.97x10^{-2}$	$1.16x10^{-3}$	$2.99x10^{-3}$	$1.49x10^{-1}$

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS (CONTINUED)

Isotope	Primary Demineralizer	Coolant Waste Receiver	Secondary Demineralizer	Coolant Waste Monitor Tanks	Primary Drain Tank
I-134	3.52×10^{-10}	1.43	3.52×10^{-2}	7.15×10^{-2}	3.58
Cs-134	1.50×10^{4}	1.03×10^{1}	1.5×10^{3}	5.17×10^{-1}	2.58×10^{1}
I-135	3.52×10^{-1}	6.36	3.52×10^{-2}	3.18×10^{-1}	1.59×10^{1}
Xe-135	-	-	-	-	-
Cs-136	3.06×10^{1}	3.63×10^{-1}	3.06	1.82×10^{-2}	9.08×10^{-1}
Cs-137	8.38×10^{4}	5.0×10^{1}	8.38×10^{3}	2.50	1.25^{2}
Xe-138	-	-	-	-	-
Cs-138	2.34×10^{-1}	8.18×10^{-1}	2.34×10^{-2}	4.09×10^{-2}	3.92
Ba-140	1.33	1.56×10^{-2}	1.33×10^{-1}	7.79×10^{-4}	3.89×10^{-2}
La-140	1.68×10^{-1}	1.5×10^{-2}	1.68×10^{-2}	7.51×10^{-4}	3.75×10^{-2}
Pr-143	1.24	1.24×10^{-1}	1.24×10^{-1}	6.81×10^{-3}	3.41×10^{-2}
Ce-143	1.09×10^{1}	9.77×10^{-2}	1.09	4.88×10^{-3}	2.44×10^{-2}
Co-60	-	-	-	-	6.25×10^{-3}
Fe-59	-	-	-	-	1.21×10^{-4}
Co-58	-	-	-	-	4.49×10^{-2}
Mn-56	-	-	-	-	1.31×10^{-1}
Mr-54	-	-	-	-	1.56×10^{-4}
Cr-51	-	-	-	-	1.36×10^{-2}
Zr-95	-	-	-	-	1.08×10^{-4}

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–1 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 45 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–2 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 29 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–3 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 5 FEET 0 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–4 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT BUILDING - ELEVATION 14 FEET 6 INCHES AND 38 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–5 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL AUXILIARY BUILDING - ELEVATION 14 FEET 6 INCHES AND 25 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–6 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL TURBINE BUILDING - ELEVATION 14 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–7 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL AUXILIARY BUILDING - ELEVATION 36 FEET 6 INCHES



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–8 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - SECTION A-A



Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–9 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - SECTION B-B

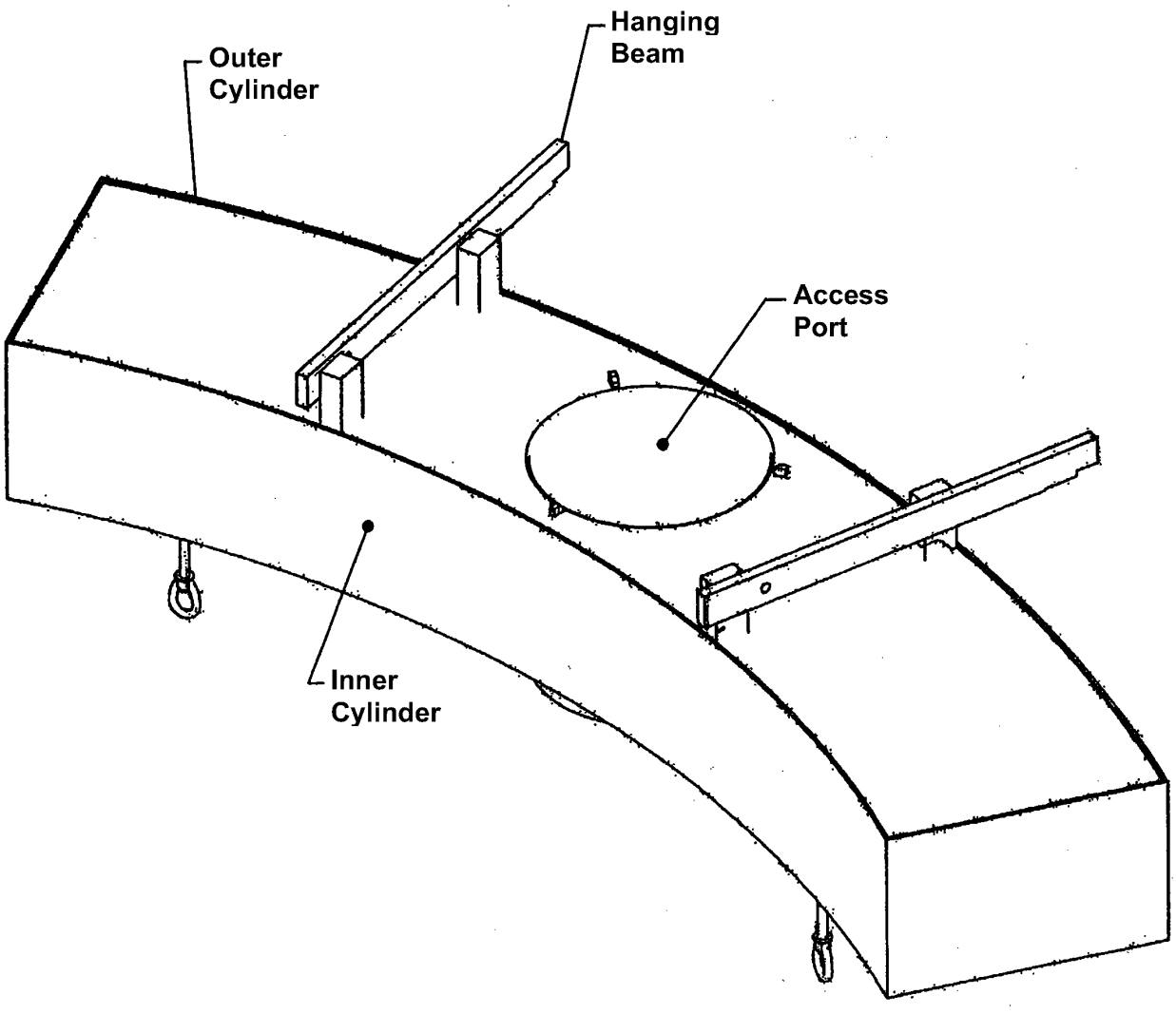


Withhold under 10 CFR 2.390 (d) (1)

FIGURE 11.2–10 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY



FIGURE 11.2–11 NEUTRON SHIELD SEGMENT



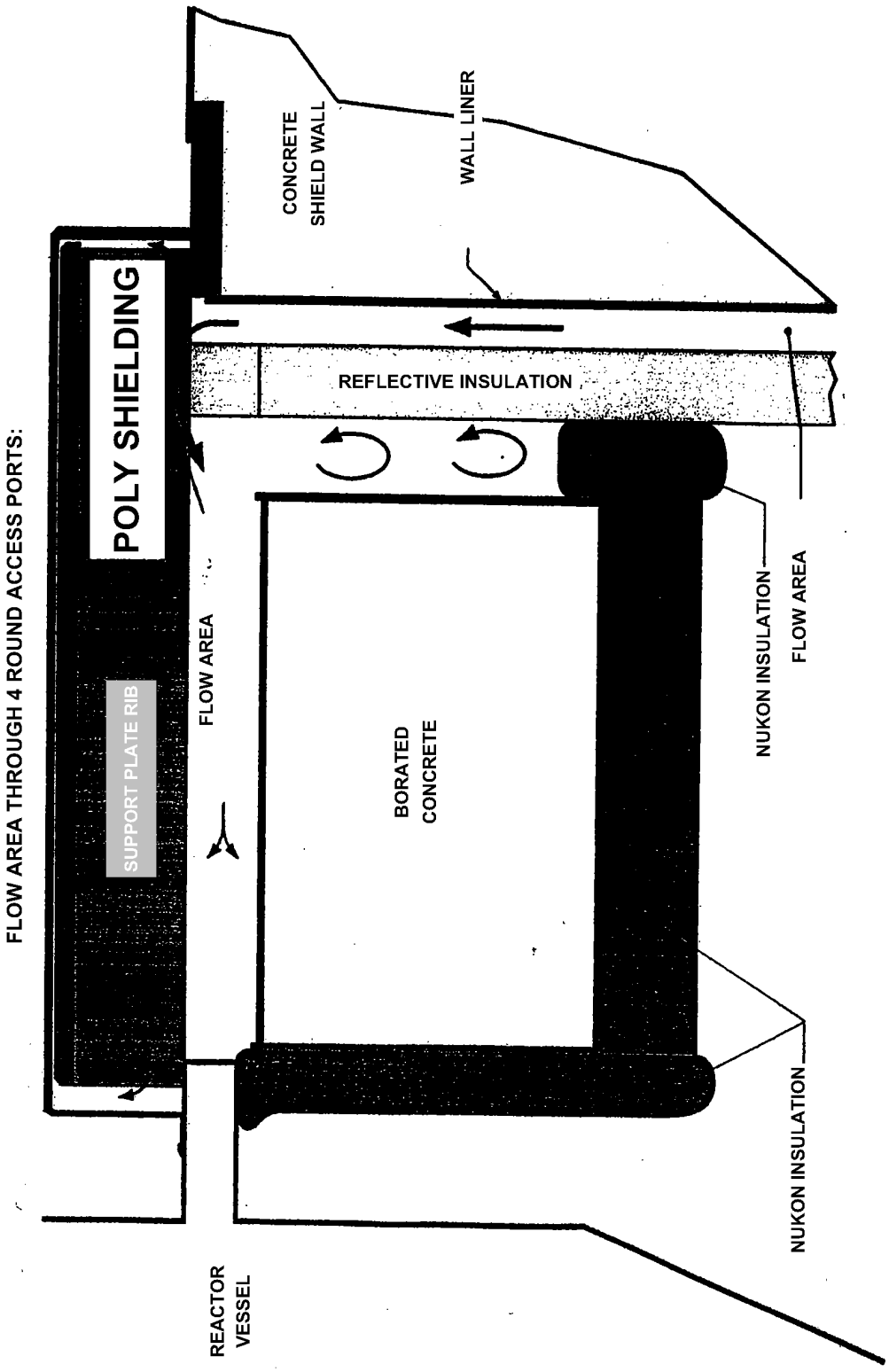


FIGURE 11.2–12 NEUTRON SHIELDING - SECTIONAL VIEW

FIGURE 11.2–13 NOT USED

FIGURE 11.2–14 NOT USED

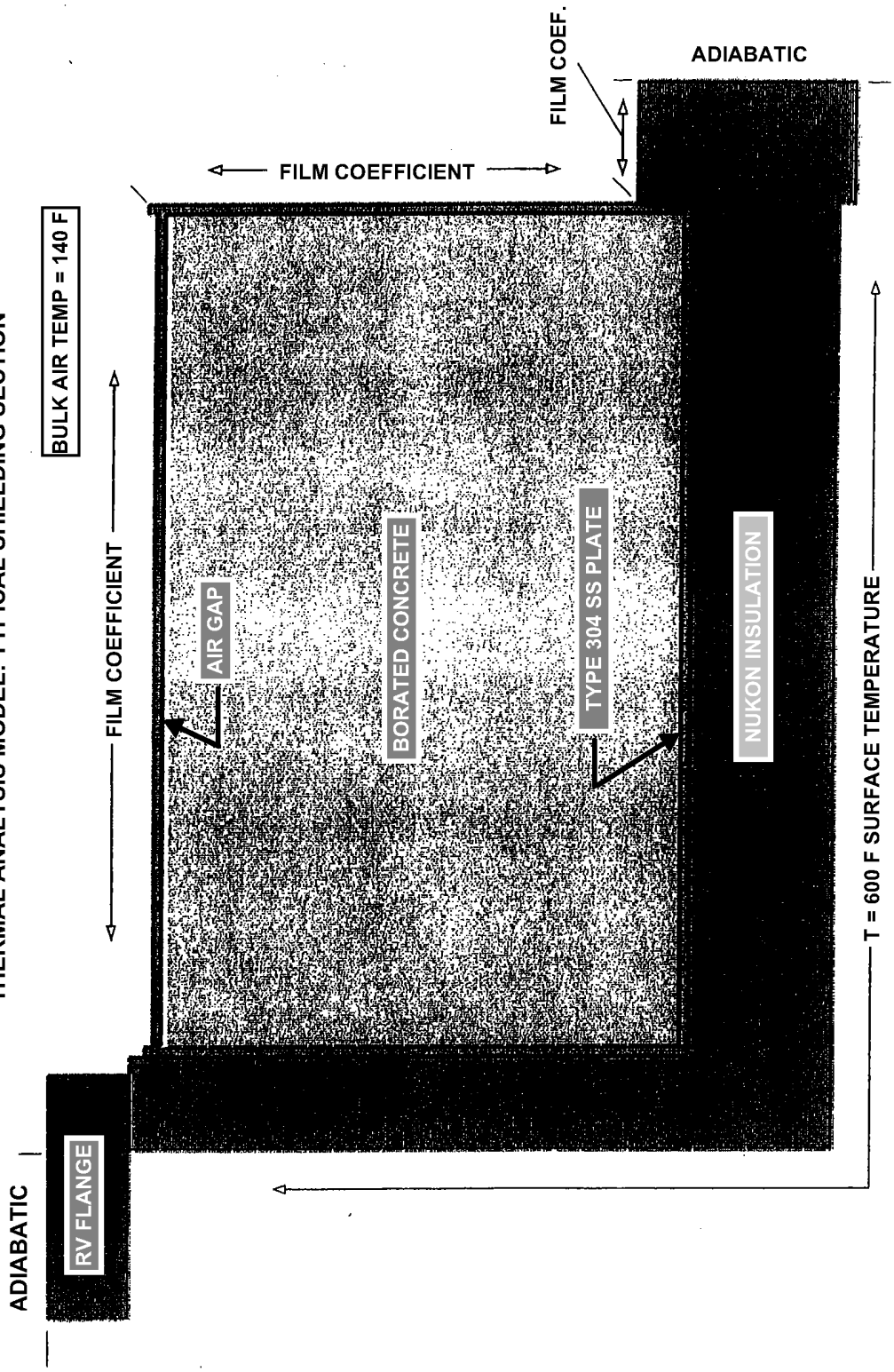


FIGURE 11.2–15 NEUTRON SHIELD - THERMAL LOADINGS (TYPICAL ANNULAR SECTION)

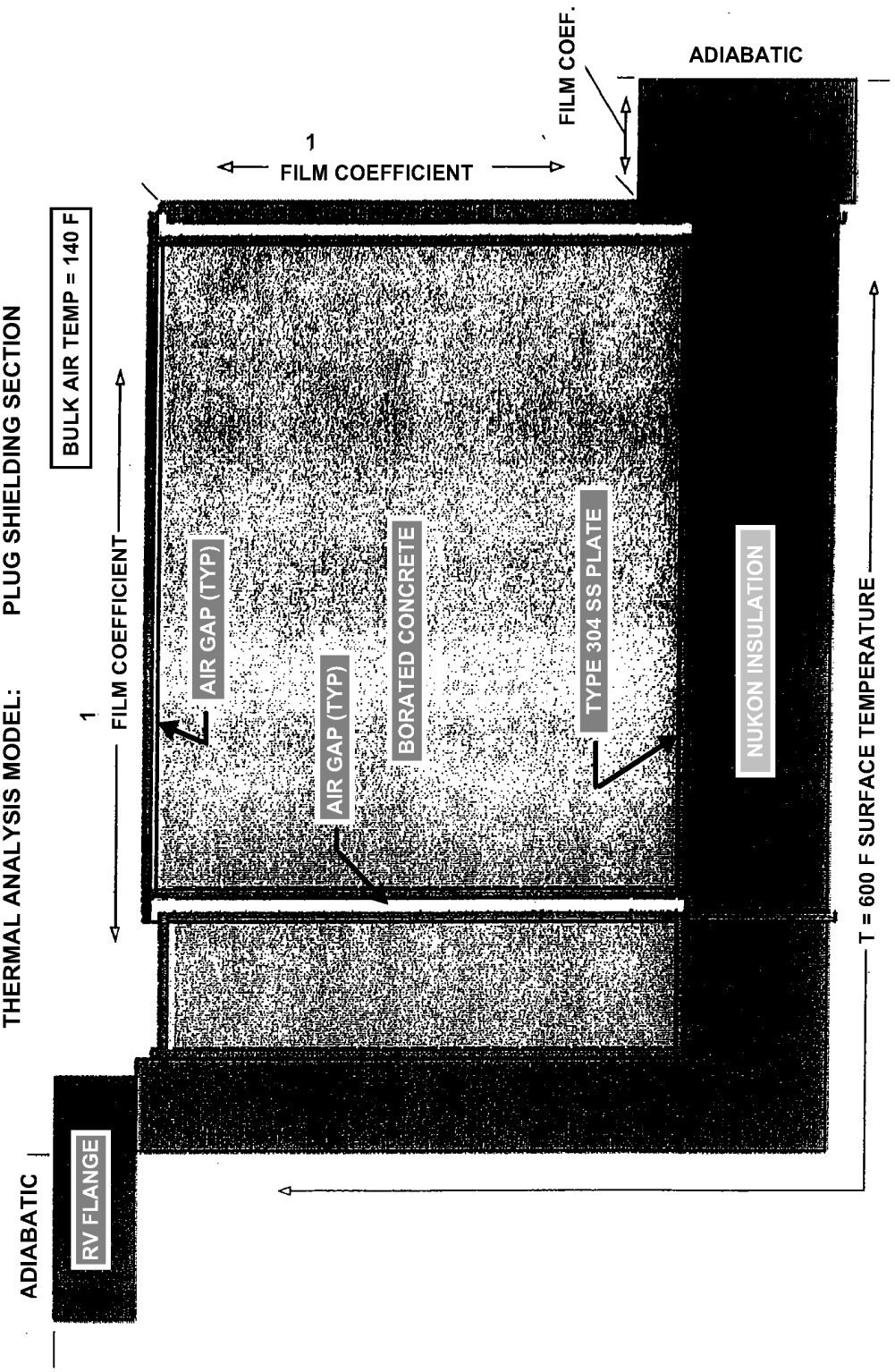


FIGURE 11.2–16 NEUTRON SHIELD - THERMAL LOADINGS (PLUG SHIELD SECTION)

11.A SOURCE TERMS FOR RADIOACTIVE WASTE PROCESSING AND RELEASES TO THE ENVIRONMENT

11.A.1 REACTOR COOLANT DESIGN BASIS RADIONUCLIDE ACTIVITIES

To calculate reactor coolant design-basis radionuclide activities, a calculation of reactor core source term (discussed in Section 11.A.1.1 below) was performed. Reactor coolant activities were calculated taking into account (a) leakage from the fuel pellets to the primary coolant due to fuel pin cladding failure, (b) removal by radioactive decay, (c) removal by the CVCS purification system, (d) removal of primary coolant for boron reduction, and (e) discharge of primary coolant to the liquid waste processing system. Included in the resultant reactor coolant design-basis radionuclide inventory are activated corrosion products (crud) and tritium (discussed in Sections 11.A.1.2 and 11.A.1.3, respectively). Table 11.A-1 provides a tabulation of the design-basis reactor coolant radionuclide activities. Bases for the calculation of these activities are presented in Table 11.A-10.

11.A.1.1 Development of Reactor Core Radionuclide Activities

For conservatism, the model used to calculate reactor core radionuclide activities employs an end-of-cycle equilibrium fuel cycle condition. Cycle characteristics are consistent with nominal two-year fuel cycles (with an 80% capacity factor included). The ORIGEN-S computer program is used to model the buildup and decay of core fission product radionuclides.

ORIGEN-S calculates the following three categories of core radionuclide activities:

1. Fission Products
2. Light Elements
3. Actinides

The light elements are neutron activation products of fuel assembly metallic structural components. The ORIGEN-S model used in the calculation of Millstone Unit 2 reactor core radionuclide activities is predicated on a standard 14-by-14 fuel pin arrangement. Actinides simulate the buildup, decay, and loss due to fissioning. Only the ^{239}Np actinide data are used.

The ORIGEN-S results are calculated in terms of gram-atoms per metric tonne of initial heavy metal. These results are converted, as necessary, to activities (in Curies) in the core using the following equation:

$$\text{Acore} = \frac{N_A \cdot \lambda}{\text{ACF}} \cdot \{M_{u1} \cdot (B1C1 + B1C2 + B1C3) + M_{u2} \cdot (B2C1 + B2C2)\}$$

where N_A is in atoms per gram-atom, λ is in seconds^{-1}, ACF is the activity conversion factor (equal to 3.7×10^{10} disintegrations/second per Curie), M_{u1} is equal to 19.208 metric tonnes of uranium in Batch 1, M_{u2} is equal to 13.72 metric tonnes of uranium in Batch 2, and where *B1C1, B1C2, B1C3, B2C1,* and *B2C2* are batch/cycle-specific values (in gram-atoms per metric tonnes of uranium) calculated by ORIGEN-S.

The reactor core radionuclide activities thus calculated and used as basic source terms are tabulated in Table 11.A-2. Also presented in Table 11.A-2 are the bases for the calculation of these reactor core radionuclide activities.

11.A.1.2 Corrosion Products

The activity concentrations of activated corrosion products (crud) in the reactor coolant have been calculated based on the model in the ANSI/ANS standard[1] dealing with the radioactive source term for normal operation of light water reactors. The starting point for this calculation was the reactor coolant radionuclide activity concentrations specified in the standard for a reference PWR with U-tube steam generators (such as those used at Millstone Unit 2). These activity concentrations were altered by adjustment factors that were prepared in accordance with the ANSI/ANS model to reflect the operating parameter differences between Millstone Unit 2 and the reference PWR, and were then further adjusted to the Technical Specification radionuclide concentration limits.

11.A.1.3 Tritium Production

Tritium may be produced in the coolant or enter the coolant from a number of sources. One source is from fissioning of uranium within the fuel, yielding tritium as a tertiary fission product. Since zircaloy fuel cladding reacts with tritium to form zircaloy hydride, no tritium diffuses through the cladding [2, 3]. Therefore, the tritium released to the coolant from the fuel originates only from defective fuel.

Tritium is also produced by the reaction of neutrons with boron in the control element assemblies (CEAs). Data from operating plants using B_4C control rods indicates that no tritium is released from the control rods. The tritium may combine with carbon to form hydrocarbons and/or with lithium to form lithium hydride, thereby preventing diffusion through the inconel cladding. Another possibility is that the low internal temperature of the B_4C control rods (relative to

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1 ANSI/ANS-18.1-1984, American National Standard — Radioactive Source Term for Normal Operation of Light Water Reactors, dated 12/31/1984

2 James M. Smith, Jr., The Significance of Tritium in Water Reactors, GE, APED, 9/19/67

3 Joseph W. Ray, et. al., Investigation of Tritium Generation and Release in PM Nuclear Power Plants, BMI-1787, 10/31/66

stainless clad fuel rods from which about 45% escape can be expected) may prohibit tritium diffusion. To account for possible control rod cladding defects, it is assumed that one percent of the tritium produced in the CEAs is released to the coolant.

Another source of tritium is the activation of boron, lithium, deuterium, and nitrogen within the reactor coolant. Boron in the form of boric acid is used in the coolant for reactivity control. Lithium is produced in the coolant as a result of neutron-boron reaction and may also be added as a pH control agent. Deuterium is a naturally occurring constituent of water. Nitrogen may be present due to aeration of the coolant during shutdown and due to aerated makeup water.

The expected tritium releases from the combined liquid and vapor pathways was assumed to be 0.4 Ci/year per MWt, based on review of the tritium release rates at a number of PWRs evaluated in NUREG-0017, Rev. 1. The quantity of tritium released in the liquid pathway is based on the calculated volume of liquid released, excluding secondary system wastes, with a primary coolant concentration of 1.0 μCi/ml. It was assumed that the remainder was released as a gas from building ventilation system exhaust.

11.A.1.4 Fuel Experience

Past operation of stainless steel-clad fuel rods in the Connecticut Yankee reactor showed fuel failure rates on the order of 0.01%.

Zircaloy-clad UO_2 fuel in the Obrigheim reactor in Germany sustained a fuel failure rate just over 0.1% in its first cycle, but this had fallen in the second cycle to essentially zero (0.001%). The fuel failure rate in the Dresden 1 reactor over a nine-year period had averaged < 0.1%, with the rate more recently being even lower. Fuel in the Mihama reactor in Japan and in the Point Beach reactor had exceeded the burn up at which failures in fuel of similar design were observed in Ginna, without exhibiting increases in coolant activity (indicative of fuel defects).

Fuel failure rates in the current generation of reactors can be controlled to very low levels. Widespread fuel defects in certain reactors have been observed, the cause having been attributed to fuel clad contamination. Appropriate corrective actions have been devised to ensure that the occurrence of such fuel defects will be greatly minimized in the future. Nevertheless, there is always the possibility, despite careful testing and manufacture, that other defects will become apparent in new fuel designs in the future that, because of statistical considerations or unrecognized or uncontrollable environmental differences, could not be foreseen. The design refinements continuously introduced in nuclear power reactors and their fuel as a natural outcome of a dynamic industry will, on rare occasions, introduce such defects. Existing licensing regulations limit coolant activity to that associated with 1% failed fuel, even during these transitory and infrequent periods.

The fuel failure rate attainable under more normal conditions has been demonstrated to be nearer 0.01%. Over the lifetime of an operating reactor, the latter rate is expected to predominate.

11.A.2 REACTOR COOLANT EXPECTED RADIONUCLIDE ACTIVITIES

Expected reactor coolant radionuclide activities are based on data generated from operating plants, field and laboratory tests, and plant-specific design considerations. These activities are built into the PWR-GALE Code (henceforth referred to as 'GALE'), which is a computerized mathematical model for calculating the expected releases of radioactive material in liquid and gaseous effluents from pressurized water reactors (PWRs). The expected reactor coolant radionuclide activities are tabulated in Table 11.A-1, juxtaposed with the design-basis reactor coolant radionuclide activities.

11.A.3 CALCULATION OF LIQUID AND GASEOUS EFFLUENT RELEASES

11.A.3.1 Expected Liquid and Gaseous Radioactive Effluent Releases

The Millstone Unit 2 expected liquid and gaseous radioactive effluent releases are calculated to determine compliance with 10 CFR Part 50 Appendix I. This calculation is performed using GALE. As previously stated, the calculation is based on data generated from operating reactors, field and laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences. The calculation performed by GALE is based on (a) American Nuclear Society (ANS) 18.1 Working Group recommendations for adjustment factors, (b) the release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (c) plant-specific design features used to reduce the quantities of radioactive materials that are ultimately released to the environment, and (d) information received on the operation of nuclear power plants. The principal mechanisms that affect the concentrations of radioactive materials in the primary coolant are the following:

- fission product leakage to the primary coolant from defects in the fuel cladding and fission product generation in tramp uranium,
- corrosion products activated in the core (i.e., crud),
- radioactivity removed in the reactor coolant treatment systems, and
- activity removed because of primary coolant leakage.

The descriptions of the Millstone Unit 2 liquid and gaseous waste processing systems, as well as the GALE input parameters unique to those systems, are presented in Appendix 11.B. The following tables provide tabulations of the GALE results:

- Table 11.A-3: Expected Annual Airborne Effluent Releases (Curies per Year), by Radionuclide, from Each Release Point
- Table 11.A-4: Expected Annual Liquid Activity Releases (Curies/Year), by Radionuclide, from Each Waste Stream

- Table 11.A-5: Expected Annual Liquid Effluent Concentrations (Diluted and Undiluted), by Radionuclide, from Each Liquid Waste Stream

- Table 11.A-8: Total Annual Design Basis And Expected Releases Of Radioactive Liquid Waste To The Environment From All Sources Combined, In Curies Per Year

- Table 11.A-9: Total Annual Design Basis And Expected Releases Of Airborne Radioactive Waste To The Environment From All Release Points Combined, In Curies Per Year

11.A.3.2 Design Basis Liquid and Gaseous Radioactive Effluent Releases

The Millstone Unit 2 design basis liquid and gaseous radioactive effluent releases are calculated to determine compliance with 10 CFR Part 20. To perform this calculation, the design basis primary coolant activity concentrations are used, in conjunction with their corresponding expected primary coolant activity concentrations, to determine isotopic scaling factors. These scaling factors are then applied to the expected liquid and gaseous effluent release inventory, as calculated by GALE, in order to determine the design basis liquid and gaseous radioactive effluent releases.

The calculated design basis liquid and gaseous radioactive effluent releases are provided in the following tables:

- Table 11.A-6: Design Basis Radionuclide Concentrations in Liquid Effluent, in Fractions of 10 CFR Part 20 Concentration Limits

- Table 11.A-7: Design Basis Radionuclide Airborne Concentrations at the Site Boundary from All Gaseous Effluent Release Points Combined, in Fractions of 10 CFR Part 20 Concentration Limits

- Table 11.A-8: Total Annual Design Basis And Expected Releases Of Radioactive Liquid Waste To The Environment From All Sources Combined, In Curies Per Year

- Table 11.A-9: Total Annual Design Basis And Expected Releases Of Airborne Radioactive Waste To The Environment From All Release Points Combined, In Curies Per Year

11.A.4 SOLID WASTE PROCESSING SYSTEM

The information in this section is based on analyses done for the original licensing of Millstone Unit 2 and is not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

11.A.4.1 Spent Resins

11.A.4.1.1 Spent Resins from CVCS Ion Exchanger

The radioactivity buildup on the CVCS ion exchangers is based on a continuous letdown flow of 40 gpm. One of the two purification ion exchangers and the deborating ion exchanger will be sluiced to the solid waste processing system at the end of each core cycle. Each of the two purification ion exchangers operates for two cycles, one cycle as a Li removal exchanger and the next cycle as the continuous purification exchanger, such that one of the two exchangers will be on the Li removal cycle while the other is for continuous purification.

The decontamination factor for ion exchangers is assumed to be 10 for all soluble isotopes except Y, Mo, and Cs. For Y, Mo, and Cs, a removal factor of 10 with a 20% usage factor is used for the Li removal cycle for each purification ion exchanger.

The buildup of activity on the CVCS ion exchangers' resins (32 ft^3 each) is given in Table 11.A-3.

11.A.4.1.2 Spent Resins from Clean Liquid Waste Processing System Demineralizers

Activity buildup on the clean liquid waste processing system demineralizers is based on the processing of 14 system volumes per year of reactor coolant wastes. An average processing rate based on annual volumes of liquid wastes is assumed, with a decontamination factor of 10^3 for soluble isotopes. Operation is assumed to be divided equally between the two primary demineralizers. The loading on the secondary demineralizer resin bed(s) will be much less than that on the primary demineralizers and has not been tabulated.

The activity buildup on the clean liquid waste processing system demineralizers' resins (42 ft^3 each) is given in Table 11.1-3.

11.A.4.1.3 Spent Resins from Aerated Liquid Waste Processing System Demineralizer

Activity buildup on the aerated liquid waste demineralizer resin is based on the processing of the volumes of aerated liquid radwaste generated annually. An average processing rate based on annual volumes of liquid waste is assumed, with a total DF of 500 taken for soluble isotopes.

The resin activity buildup on the hard piped demineralizer resin (42 ft^3) is given in Table 11.1-3.

The resin activity buildup on the portable demineralizers (15 ft^3 each) is not addressed, since this activity is bounded by the resin activity buildup on hard-piped demineralizer.

11.A.4.1.4 Spent Resins from Spent Fuel Pool Demineralizer

The activity buildup on resins from the spent fuel demineralizer is based on the processing of spent fuel pool water and refueling water with a radioactivity level corresponding to one-tenth of reactor coolant. The assumption is made that complete mixing of reactor coolant, spent fuel pool

water, and refueling water occurs during refueling operations, resulting in a dilution of the reactor coolant by a factor of approximately 10. An expected DF of 1000 is used for the demineralizer. The activity buildup on the demineralizer resin (42 ft^3) is given in Table 11.1-3.

11.A.4.1.5 Contaminated Filter Cartridges

Buildup of activity on filter cartridges is based on processing of liquid radioactive waste. A decontamination factor (DF) of 10 is taken for each filter. This DF is consistent with operating experience from nuclear power stations utilizing three micron filter cartridges.

The activity buildup on the filter cartridges is given in Table 11.1-3.

TABLE 11.A-1 DESIGN-BASIS AND EXPECTED PRIMARY COOLANT ACTIVITY CONCENTRATIONS

Noble Gases

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Kr-85m	1.33E+00	1.40E-01
Kr-85	4.50E+00	2.12E-02
Kr-87	7.64E-01	1.41E-01
Kr-88	2.36E+00	2.53E-01
Xe-131m	2.32E+00	1.17E-01
Xe-133m	3.39E+00	3.03E-02
Xe-133	2.17E+02	6.88E-01
Xe-135m	1.25E-01	1.25E-01
Xe-35	6.13E+00	6.72E-01
Xe-137	1.38E-01	3.28E-02
Xe-138	4.81E-01	1.15E-01

Halogens

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
I-131	5.57E+00	4.51E-02
I-132	1.18E+00	2.04E-01
I-133	7.29E+00	1.39E-01
I-134	7.77E-01	3.30E-01
I-135	3.65E+00	2.55E-01
Br-84	4.84E-02	1.55E-02

TABLE 11.A-1 DESIGN-BASIS AND EXPECTED PRIMARY COOLANT ACTIVITY CONCENTRATIONS (CONTINUED)

Activated Corrosion Products (Crud)

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Na-24	1.40E+00	4.63E-02
Cr-51	9.50E-02	3.09E-03
Mn-54	4.90E-02	1.60E-03
Fe-55	3.70E-02	1.20E-03
Fe-59	9.20E-03	2.99E-04
Co-58	1.40E-01	4.59E-03
Co-60	1.60E-02	5.29E-04
Zn-65	1.60E-02	5.09E-04
W-187	7.60E-02	2.47E-03
Np-239	6.70E-02	2.19E-03

TABLE 11.A-1 DESIGN-BASIS AND EXPECTED PRIMARY COOLANT ACTIVITY CONCENTRATIONS (CONTINUED)

Other Particulate Radionuclides

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Rb-88	5.04E-02	1.84E-01
Sr-89	3.74E-03	1.40E-04
Sr-90	3.83E-04	1.20E-05
Sr-91	1.83E-03	9.42E-04
Y-91m	1.43E-04	4.46E-04
Y-91	4.83E-03	5.19E-06
Y-93	1.53E-03	4.12E-03
Zr-95	1.20E-02	3.89E-04
Nb-95	6.39E-03	2.79E-04
Mo-99	8.11E-01	6.36E-03
Tc-99m	1.77E-03	4.60E-03
Ru-103	5.57E-03	7.49E-03
Ru-106	1.94E-03	8.99E-02
Ag-110m	1.28E-05	1.30E-03
Te-129m	2.20E-02	1.90E-04
Te-129	7.99E-03	2.33E-02
Te-131m	4.75E-02	1.48E-03
Te-131	8.51E-03	7.45E-03
Te-132	4.46E-01	1.69E-03
Cs-134	1.92E+00	7.56E-03
Cs-136	5.10E-01	9.21E-04
Cs-137	1.58E+00	1.00E-02
Ba-140	6.49E-03	1.30E-02
La-140	5.14E-03	2.48E-02
Ce-141	6.16E-03	1.50E-04
Ce-143	4.02E-03	2.77E-03
Ce-144	4.93E-03	3.89E-03

TABLE 11.A-2 CALCULATED REACTOR CORE ACTIVITIES

Nobel Gases

Nuclide	Reactor Core Activities (Curies) [1]
Kr-85m	1.859E+07
Kr-85	8.557E+05
Kr-87	3.736E+07
Kr-88	5.193E+07
Xe-131m	9.452E+05
Xe-133m	4.636E+06
Xe-133	1.482E+08
Xe-135m	3.021E+07
Xe-135	4.288E+07
Xe-137	1.344E+08
Xe-138	1.266E+08

Halogens

Nuclide	Reactor Core Activities (Curies) [1]
Br-84	1.658E+07
I-131	7.120E+07
I-132	1.036E+08
I-133	1.479E+08
I-134	1.644E+08
I-135	1.400E+08

TABLE 11.A-2 CALCULATED REACTOR CORE ACTIVITIES (CONTINUED)

Cesium and Rubidium

Nuclide	Reactor Core Activities (Curies) [1]
Rb-88	5.320E+07
Cs-134	1.202E+07
Cs-136	3.475E+06
Cs-137	9.888E+06

Other Nuclides

Nuclide	Reactor Core Activities (Curies) [1]
Cr-51	2.413E+06
Mn-54	9.754E+04
Fe-55	4.496E+05
Fe-59	3.382E+04
Co-58	8.739E+05
Co-60	8.888E+05
Zn-65	6.496E-02
Np-239	1.347E+09
Sr-89	7.321E+07
Sr-90	7.406E+06
Sr-91	9.077E+07
Y-91	9.435E+07
Y-91m	5.255E+07
Y-93	7.297E+07
Zr-95	1.250E+08
Nb-95	1.255E+08
Mo-99	1.343E+08
Tc-99m	1.188E+08
Ru-103	1.093E+08
Ru-106	3.752E+07

Other Nuclides

Nuclide	Reactor Core Activities (Curies) [1]
Ag-110m	2.479E+05
Te-129	2.136E+07
Te-129m	4.324E+06
Te-131	6.042E+07
Te-131m	1.373E+07
Te-132	1.025E+08
Ba-140	1.314E+08
La-140	1.359E+08
Ce-141	1.213E+08
Ce-143	1.127E+08
Ce-144	9.545E+07

1. The reactor core radionuclide activities tabulated above are based on the following: (a) a core power level of 2,700 MWt and (b) a three-region equilibrium cycle core, with an end-of-cycle core average burn-up of 36,142 MWD/MTU, the three regions having operated at a specific power of 31.74 MWt/MTU for 705, 1333 and 1550 EFPD, respectively.

TABLE 11.A-3 EXPECTED ANNUAL EFFLUENT RELEASES (CURIES PER YEAR), BY RADIONUCLIDE, FROM EACH RELEASE POINT

Release Points: Iodines

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
I-131	0.00E+00	1.77E-01	1.00E-03
I-133	1.80E-04	5.51E-01	1.00E-03
Totals	1.80E-04	7.28E-01	2.00E-03

Release Points: Noble Gases

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
Kr-85m	0.00E+00	3.00E+00	1.00E+00
K-r85	0.00E+00	3.80E+01	5.60E+02
Kr-87	0M.00E+00	3.00E+00	1.00E+00
Kr-88	0.00E+00	5.00E+00	4.00E+00
Xe-131m	0.00E+00	2.70E+01	2.30E+01
Xe-133m	0.00E+00	1.00E+00	2.00E+00
Xe-133	0.00E+00	8.10E+01	7.10E+01
Xe-135m	0.00E+00	3.00E+00	1.00E+00
Xe-135	0.00E+00	1.90E+01	1.20E+01
Xe-137	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	2.00E+00	1.00E+00
Totals	0.00E+00	1.82E+02	6.76E+02

TABLE 11.A-3 EXPECTED ANNUAL EFFLUENT RELEASES (CURIES PER YEAR), BY RADIONUCLIDE, FROM EACH RELEASE POINT (CONTINUED)

Release Points: Others (Totals Allocated to Unit 2 Vent Release for Conservatism)

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
H-3	none	1.10E+02	none
C-14	none	7.30E+00	none
Ar-41	none	3.40E+01	none
Totals	none	1.51E+02	none

Release Points: Particulates

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
Cr-51	none	9.70E-05	1.40E-05
Mn-54	none	5.68E-05	2.10E-06
Co-57	none	8.20E-06	0.00E+00
Co-58	none	4.79E-04	8.70E-06
Co-60	none	1.13E-04	1.40E-05
Fe-59	none	2.75E-05	1.80E-06
Sr-89	none	1.59E-04	4.40E-05
Sr-90	none	6.29E-05	1.70E-05
Zr-95	none	1.00E-05	4.80E-06
Nb-95	none	4.23E-05	3.70E-06
Ru-103	none	1.66E-05	3.20E-06
Ru-106	none	7.50E-07	2.70E-06
Sb-125	none	6.09E-07	0.00E+00
Cs-134	none	4.74E-05	3.30E-05
Cs-136	none	3.25E-05	5.30E-06
Cs-137	none	8.92E-05	7.70E-05
Ba-140	none	4.00E-06	2.30E-05
Ce-141	none	1.33E-05	2.20E-06
Totals	none	1.26E-03	2.57E-04

TABLE 11.A-4 EXPECTED ANNUAL LIQUID EFFLUENT ACTIVITY RELEASES (CURIES/YEAR), BY RADIONUCLIDE, FROM EACH WASTE STREAM

NUCLIDE	BORON RS (Ci/yr)	MISC. WASTES (Ci/yr)	SECONDARY (Ci/yr)	TURB. BLDG. (Ci/yr)	DETERGENT (Ci/yr)
Corrosion and Activation Products:					
Na-24	2.46E-04	9.26E-03	4.62E-01	5.94E-05	0.00E+00
Cr-51	3.39E-04	1.96E-03	3.81E-02	6.23E-06	4.70E-04
Mn-54	2.11E-04	1.04E-03	1.94E-02	3.13E-06	3.80E-04
Fe-55	1.61E-04	7.84E-04	1.47E-02	2.36E-06	7.20E-04
Fe-59	3.55E-05	1.92E-04	3.54E-03	5.76E-07	2.20E-04
Co-58	5.71E-04	2.97E-03	5.63E-02	9.13E-06	7.90E-04
Co-60	7.12E-05	3.47E-04	6.59E-03	1.06E-06	1.40E-03
Zn-65	6.71E-05	3.32E-04	6.27E-03	1.01E-06	0.00E+00
W-187	2.36E-05	7.16E-04	2.61E-02	3.72E-06	0.00E+00
Np-239	5.48E-05	9.81E-04	2.45E-02	3.85E-06	0.00E+00
Fission Products:					
Br-84	4.64E-10	1.90E-05	3.21E-02	4.26E-09	0.00E+00
Rb-88	3.25E-10	1.44E-03	2.42E-01	3.36E-11	0.00E+00
Sr-89	1.69E-05	8.99E-05	1.68E-03	2.74E-07	8.80E-06
Sr-90	1.62E-06	7.85E-06	1.47E-04	2.36E-08	1.30E-06
Sr-91	2.71E-06	1.23E-04	8.95E-03	9.88E-07	0.00E+00
Y-91m	1.75E-06	7.86E-05	1.37E-03	6.33E-07	0.00E+00
Y-91	1.45E-06	6.86E-06	6.47E-05	1.35E-08	8.40E-06
Y-93	1.29E-05	5.68E-04	3.82E-02	4.31E-06	0.00E+00
Zr-95	4.80E-05	2.51E-04	4.74E-03	7.68E-07	1.10E-04
Nb-95	3.95E-05	1.85E-04	3.32E-03	5.31E-07	1.90E-04
Mo-99	1.90E-04	3.02E-03	7.26E-02	1.15E-05	6.00E-06
Tc-99m	1.80E-04	3.02E-03	3.73E-02	8.30E-06	0.00E+00
Ru-103	8.72E-04	4.79E-03	9.12E-02	1.49E-05	2.90E-05
Ru-106	1.19E-02	5.87E-02	1.11E+00	1.78E-04	8.90E-04
Ag-110m	1.71E-04	8.47E-04	1.58E-02	2.55E-06	1.20E-04

TABLE 11.A-4 EXPECTED ANNUAL LIQUID EFFLUENT ACTIVITY RELEASES (CURIES/YEAR), BY RADIONUCLIDE, FROM EACH WASTE STREAM (CONTINUED)

NUCLIDE	BORON RS (Ci/yr)	MISC. WASTES (Ci/yr)	SECONDARY (Ci/yr)	TURB. BLDG. (Ci/yr)	DETERGENT (Ci/yr)
Te-29m	2.16E-05	1.21E-04	2.29E-03	3.74E-07	0.00E+00
Te-29	1.40E-05	2.47E-04	9.12E-02	6.49E-07	0.00E+00
Te-131m	1.85E-05	4.97E-04	1.60E-02	2.36E-06	0.00E+00
Te-131	3.38E-06	9.50E-05	1.31E-02	4.32E-07	0.00E+00
I-131	2.91E-03	2.64E-02	4.51E-01	1.43E-04	1.60E-04
Te-132	5.87E-05	8.39E-04	1.91E-02	3.06E-06	0.00E+00
I-132	8.64E-05	5.45E-03	1.07E+00	6.24E-05	0.00E+00
I-133	1.04E-03	3.66E-02	1.24E+00	3.42E-04	0.00E+00
I-134	5.82E-07	1.41E-03	9.69E-01	2.92E-06	0.00E+00
Cs-134	1.14E-01	2.47E-01	9.36E-02	1.54E-05	1.10E-03
I-135	3.76E-04	2.22E-02	1.91E+00	3.48E-04	0.00E+00
Cs-136	9.16E-03	2.81E-02	1.11E-02	1.84E-06	3.70E-05
Cs-137	1.52E-01	3.28E-01	1.25E-01	2.05E-05	1.60E-03
Ba-140	1.14E-03	7.90E-03	1.51E-01	2.48E-05	9.10E-05
La-140	1.48E-03	1.29E-02	2.76E-01	4.42E-05	0.00E+00
Ce-141	1.69E-05	9.53E-05	1.79E-03	2.92E-07	2.30E-05
Ce-143	3.86E-05	9.81E-04	2.96E-02	4.42E-06	0.00E+00
Ce-144	5.15E-04	2.54E-03	4.78E-02	7.70E-06	3.90E-04
All Others Except Tritium	1.56E-01	3.73E-01	6.69E-02	2.20E-04	2.31E-04
TOTAL	4.54E-01	1.19E+00	8.90E+00	1.56E-03	8.98E-03

TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
Corrosion and Activation Products:										
Na-24	5.45E-08	6.04E-06	1.57E-06	5.97E-09	0.00E+00	2.64E-13	9.94E-12	4.96E-10	6.38E-14	0.00E+00
Cr-51	7.50E-08	1.28E-06	1.30E-07	6.26E-10	6.30E-07	3.64E-13	2.10E-12	4.09E-11	6.69E-15	5.05E-13
Mn-54	4.67E-08	6.78E-07	6.60E-08	3.15E-10	5.09E-07	2.27E-13	1.12E-12	2.08E-11	3.36E-15	4.08E-13
Fe-55	3.56E-08	5.11E-07	5.00E-08	2.37E-10	9.65E-07	1.73E-13	8.42E-13	1.58E-11	2.53E-15	7.73E-13
Fe-59	7.86E-09	1.25E-07	1.20E-08	5.79E-11	2.95E-07	3.81E-14	2.06E-13	3.80E-12	6.18E-16	2.36E-13
Co-58	1.26E-07	1.94E-06	1.91E-07	9.18E-10	1.06E-06	6.13E-13	3.19E-12	6.04E-11	9.80E-15	8.48E-13
Co-60	1.58E-08	2.26E-07	2.24E-08	1.07E-10	1.88E-06	7.65E-14	3.73E-13	7.07E-12	1.14E-15	1.50E-12
Zn-65	1.49E-08	2.16E-07	2.13E-08	1.02E-10	0.00E+00	7.20E-14	3.56E-13	6.73E-12	1.08E-15	0.00E+00
W-187	5.22E-09	4.67E-07	8.88E-08	3.74E-10	0.00E+00	2.53E-14	7.69E-13	2.80E-11	3.99E-15	0.00E+00
Np-239	1.21E-08	6.40E-07	8.33E-08	3.87E-10	0.00E+00	5.88E-14	1.05E-12	2.63E-11	4.13E-15	0.00E+00
Fission Products:										
Br-84	1.03E-13	1.24E-08	1.09E-07	4.28E-13	0.00E+00	4.98E-19	2.04E-14	3.45E-11	4.57E-18	0.00E+00
Rb-88	7.19E-14	9.39E-07	8.23E-07	3.38E-15	0.00E+00	3.49E-19	1.55E-12	2.60E-10	3.61E-20	0.00E+00
Sr-89	3.74E-09	5.86E-08	5.71E-09	2.75E-11	1.18E-08	1.81E-14	9.65E-14	1.80E-12	2.94E-16	9.45E-15
Sr-90	3.59E-10	5.12E-09	5.00E-10	2.37E-12	1.74E-09	1.74E-15	8.43E-15	1.58E-13	2.53E-17	1.40E-15
Sr-91	6.00E-10	8.02E-08	3.04E-08	9.93E-11	0.00E+00	2.91E-15	1.32E-13	9.61E-12	1.06E-15	0.00E+00
Y-91m	3.87E-10	5.13E-08	4.66E-09	6.36E-11	0.00E+00	1.88E-15	8.44E-14	1.47E-12	6.80E-16	0.00E+00
Y-91	3.21E-10	4.47E-09	2.20E-10	1.36E-12	1.13E-08	1.56E-15	7.37E-15	6.95E-14	1.45E-17	9.02E-15
Y-93	2.86E-09	3.70E-07	1.30E-07	4.33E-10	0.00E+00	1.39E-14	6.10E-13	4.10E-11	4.63E-15	0.00E+00
Zr-95	1.06E-08	1.64E-07	1.61E-08	7.72E-11	1.47E-07	5.15E-14	2.70E-13	5.09E-12	8.25E-16	1.18E-13
Nb-95	8.74E-09	1.21E-07	1.13E-08	5.34E-11	2.55E-07	4.24E-14	1.99E-13	3.56E-12	5.70E-16	2.04E-13

TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM (CONTINUED)

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
Mo-99	4.21E-08	1.97E-06	2.47E-07	1.16E-09	8.04E-09	2.04E-13	3.24E-12	7.79E-11	1.23E-14	6.44E-15
Tc-99m	3.98E-08	1.97E-06	1.27E-07	8.34E-10	0.00E+00	1.93E-13	3.24E-12	4.00E-11	8.91E-15	0.00E+00
Ru-103	1.93E-07	3.12E-06	3.10E-07	1.50E-09	3.89E-08	9.36E-13	5.14E-12	9.79E-11	1.60E-14	3.11E-14
Ru-106	2.63E-06	3.83E-05	3.77E-06	1.79E-08	1.19E-06	1.28E-11	6.30E-11	1.19E-09	1.91E-13	9.56E-13
Ag-110m	3.79E-08	5.52E-07	5.37E-08	2.56E-10	1.61E-07	1.84E-13	9.09E-13	1.70E-11	2.74E-15	1.29E-13
Te-129m	4.78E-09	7.89E-08	7.79E-09	3.76E-11	0.00E+00	2.32E-14	1.30E-13	2.46E-12	4.02E-16	0.00E+00
Te-129	3.10E-09	1.61E-07	3.10E-07	6.52E-11	0.00E+00	1.50E-14	2.65E-13	9.79E-11	6.97E-16	0.00E+00
Te-131m	4.10E-09	3.24E-07	5.44E-08	2.37E-10	0.00E+00	1.99E-14	5.34E-13	1.72E-11	2.53E-15	0.00E+00
Te-131	7.48E-10	6.20E-08	4.45E-08	4.34E-11	0.00E+00	3.63E-15	1.02E-13	1.41E-11	4.64E-16	0.00E+00
I-131	6.44E-07	1.72E-05	1.53E-06	1.44E-08	2.14E-07	3.12E-12	2.83E-11	4.84E-10	1.54E-13	1.72E-13
Te-132	1.30E-08	5.47E-07	6.50E-08	3.08E-10	0.00E+00	6.30E-14	9.01E-13	2.05E-11	3.29E-15	0.00E+00
I-132	1.91E-08	3.55E-06	3.64E-06	6.27E-09	0.00E+00	9.28E-14	5.85E-12	1.15E-09	6.70E-14	0.00E+00
I-133	2.30E-07	2.39E-05	4.22E-06	3.44E-08	0.00E+00	1.12E-12	3.93E-11	1.33E-09	3.67E-13	0.00E+00
I-134	1.29E-10	9.19E-07	3.30E-06	2.94E-10	0.00E+00	6.25E-16	1.51E-12	1.04E-09	3.14E-15	0.00E+00
Cs-134	2.52E-05	1.61E-04	3.18E-07	1.55E-09	1.47E-06	1.22E-10	2.65E-10	1.00E-10	1.65E-14	1.18E-12
I-135	8.32E-08	1.45E-05	6.50E-06	3.50E-08	0.00E+00	4.04E-13	2.38E-11	2.05E-09	3.74E-13	0.00E+00
Cs-136	2.03E-06	1.83E-05	3.77E-08	1.85E-10	4.96E-08	9.84E-12	3.02E-11	1.19E-11	1.98E-15	3.97E-14
Cs-137	3.36E-05	2.14E-04	4.25E-07	2.06E-09	2.14E-06	1.63E-10	3.52E-10	1.34E-10	2.20E-14	1.72E-12
Ba-140	2.52E-07	5.15E-06	5.14E-07	2.49E-09	1.22E-07	1.22E-12	8.48E-12	1.62E-10	2.66E-14	9.77E-14
La-140	3.28E-07	8.41E-06	9.39E-07	4.44E-09	0.00E+00	1.59E-12	1.39E-11	2.96E-10	4.75E-14	0.00E+00
Ce-141	3.74E-09	6.21E-08	6.09E-09	2.94E-11	3.08E-08	1.81E-14	1.02E-13	1.92E-12	3.14E-16	2.47E-14
Ce-143	8.54E-09	6.40E-07	1.01E-07	4.44E-10	0.00E+00	4.14E-14	1.05E-12	3.18E-11	4.75E-15	0.00E+00
Ce-144	1.14E-07	1.66E-06	1.63E-07	7.74E-10	5.23E-07	5.53E-13	2.73E-12	5.13E-11	8.27E-15	4.19E-13

TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM (CONTINUED)

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
All Others Except Tritium	3.45E-05	2.43E-04	2.28E-07	2.21E-08	3.10E-07	1.68E-10	4.01E-10	7.18E-11	2.36E-13	2.48E-13
TOTAL	1.01E-04	7.73E-04	3.03E-05	1.57E-07	1.20E-05	4.88E-10	1.27E-09	9.56E-09	1.67E-12	9.64E-12

TABLE 11.A-6 DESIGN BASIS RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENT, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS

NUCLIDE	FRACTION OF 10 CFR PART 20 MAXIMUM PERMISSIBLE CONCENTRATIONS [1]
Na-24	7.79E-05
Cr-51	6.77E-07
Mn-54	6.90E-06
Fe-55	7.03E-07
Fe-59	2.20E-06
Co-58	2.03E-08
Co-60	5.52E-06
Zn-65	2.29E-06
W-187	1.27E-05
Np-239	8.54E-06
Br-84	2.77E-07
Rb-88	1.84E-07
Sr-89	1.73E-05
Sr-90	1.83E-05
Sr-91	2.73E-07
Y-91m	1.72E-10
Y-91	2.76E-06
Y-93	5.18E-07
Zr-95	2.87E-06
Nb-95	9.34E-07
Mo-99	5.27E-05
Tc-99m	2.82E-09
Ru-103	9.79E-07
Ru-106	2.78E-06
Ag-110m	5.99E-09
Te-129m	1.03E-05

TABLE 11.A-6 DESIGN BASIS RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENT, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS (CONTINUED)

NUCLIDE	FRACTION OF 10 CFR PART 20 MAXIMUM PERMISSIBLE CONCENTRATIONS [1]
Te-129	4.28E-08
Te-131m	9.77E-06
Te-131	1.99E-07
I-131	2.17E-01
Te-132	1.89E-04
I-132	8.52E-04
I-133	7.32E-02
I-134	1.25E-04
Cs-134	1.39E-02
I-135	7.69E-03
Cs-136	3.24E-04
Cs-137	5.26E-03
Ba-140	2.86E-06
La-140	3.23E-06
Ce-141	9.79E-07
Ce-143	1.21E-06
Ce-144	7.07E-06
H-3	3.54E-04
Sum of MPC Fractions [2]	3.19E-01

1. Based on 10 CFR Part 20, Appendix B, Table II, Column 2, prior to 1994.
2. In the course of plant operation, individual isotopic fractions of maximum permissible concentration (MPC) might vary from the values tabulated above. The purpose of this table is to demonstrate that the design of the liquid radioactive waste processing system is adequate at design conditions such that the sum of the fractions of MPCs in the pre-1994 version of 10 CFR Part 20 will not exceed the limit of 1.0.

TABLE 11.A-7 DESIGN-BASIS RADIONUCLIDE AIRBORNE CONCENTRATIONS AT THE SITE BOUNDARY FROM ALL GASEOUS EFFLUENT RELEASE POINTS COMBINED, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS

Iodines

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
I-131	4.13E-04
I-133	7.65E-04

Noble Gases

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
Kr-85m	7.03E-05
Kr-85	7.87E-02
Kr-87	2.01E-04
Kr-88	7.77E-04
Xe-131m	4.60E-04
Xe-133m	2.07E-04
Xe-133	2.93E-02
Xe-135m	1.85E-05
Xe-135	5.23E-04
Xe-137	none
Xe-138	1.16E-04

TABLE 11.A-7 DESIGN-BASIS RADIONUCLIDE AIRBORNE CONCENTRATIONS AT THE SITE BOUNDARY FROM ALL GASEOUS EFFLUENT RELEASE POINTS COMBINED, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS (CONTINUED)

Particulates

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
Cr-51	7.83E-09
Mn-54	3.35E-07
Co-57	2.53E-10
Co-58	1.38E-06
Co-60	2.43E-06
Fe-59	8.26E-08
Sr-89	9.92E-07
Sr-90	2.37E-06
Zr-95	8.57E-08
Nb-95	6.50E-08
Ru-103	9.19E-10
Ru-106	6.99E-11
Sb-125	1.61E-10
Cs-134	9.39E-06
Cs-136	6.49E-07
Cs-137	9.94E-06
Ba-140	2.49E-09
Ce-141	2.28E-08
Ar-41	1.57E-04
C-14	1.35E-05
H-3	1.02E-04

Sum of MPC Fractions (2) 1.12E-01.

1. Based on 10 CFR Part 20, Appendix B, Table II, Column 1, prior to 1994.
2. In the course of plant operation, individual isotopic fractions of maximum permissible concentration (MPC) might vary from the values tabulated above. The purpose of this table is to demonstrate that the design of the gaseous radioactive waste processing system is adequate at design conditions such that the sum of the fractions of MPCs in the pre-1994 version of 10 CFR Part 20 will not exceed the limit of 1.0.

TABLE 11.A-8 TOTAL ANNUAL DESIGN-BASIS AND EXPECTED RELEASES OF RADIOACTIVE LIQUID WASTE TO THE ENVIRONMENT FROM ALL SOURCES COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS LIQUID EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED LIQUID EFFLUENT RELEASE (Ci/yr)
Activated Corrosion Products (Crud):		
Na-24	1.45E+01	4.80E-01
Cr-51	1.26E+00	4.10E-02
Mn-54	6.43E-01	2.10E-02
Fe-55	5.24E-01	1.70E-02
Fe-59	1.23E-01	4.00E-03
Co-58	1.89E+00	6.20E-02
Co-60	2.57E-01	8.50E-03
Zn-65	2.14E-01	6.80E-03
W-187	8.31E-01	2.70E-02
Np-239	7.95E-01	2.60E-02
Fission Products:		
Br-84	1.03E-01	3.30E-02
Rb-88	6.85E-02	2.50E-01
Sr-89	4.81E-02	1.80E-03
Sr-90	5.11E-03	1.60E-04
Sr-91	1.79E-02	9.20E-03
Y-91m	4.82E-04	1.50E-03
Y-91	7.73E-02	8.30E-05
Y-93	1.45E-02	3.90E-02
Zr-95	1.60E-01	5.20E-03
Nb-95	8.70E-02	3.80E-03
Mo-99	9.81E+00	7.70E-02
Tc-99m	1.58E-02	4.10E-02
Ru-103	7.29E-02	9.80E-02
Ru-106	2.59E-02	1.20E+00
Ag-110m	1.67E-04	1.70E-02

TABLE 11.A-8 TOTAL ANNUAL DESIGN-BASIS AND EXPECTED RELEASES OF RADIOACTIVE LIQUID WASTE TO THE ENVIRONMENT FROM ALL SOURCES COMBINED, IN CURIES PER YEAR (CONTINUED)

NUCLIDE	ANNUAL DESIGN BASIS LIQUID EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED LIQUID EFFLUENT RELEASE (Ci/yr)
Te-129m	2.89E-01	2.50E-03
Te-129	3.19E-02	9.30E-02
Te-131m	5.46E-01	1.70E-02
Te-131	1.49E-02	1.30E-02
I-131	6.05E+01	4.90E-01
Te-132	5.28E+00	2.00E-02
I-132	6.35E+00	1.10E+00
I-133	6.82E+01	1.30E+00
I-134	2.33E+00	9.90E-01
Cs-134	1.17E+02	4.60E-01
I-135	2.86E+01	2.00E+00
Cs-136	2.71E+01	4.90E-02
Cs-137	9.79E+01	6.20E-01
Ba-140	7.98E-02	1.60E-01
La-140	6.00E-02	2.90E-01
Ce-141	8.22E-02	2.00E-03
Ce-143	4.50E-02	3.10E-02
Ce-144	6.59E-02	5.20E-02
H-3	9.90E+02	9.90E+02
TOTAL WITHOUT TRITIUM	4.25E+02	1.10E+01

TABLE 11.A-9 TOTAL ANNUAL DESIGN BASIS AND EXPECTED RELEASES OF AIRBORNE RADIOACTIVE WASTE TO THE ENVIRONMENT FROM ALL RELEASE POINTS COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS AIRBORNE EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED AIRBORNE EFFLUENT RELEASE (Ci/yr)
I-131	2.23E+01	1.80E-01
I-133	2.89E+01	5.50E-01
Kr-85m	3.80E+01	4.00E+00
Kr-85	1.28E+05	6.00E+02
Kr-87	2.17E+01	4.00E+00
Kr-88	8.38E+01	9.00E+00
Xe-131m	9.94E+02	5.00E+01
Xe-133m	3.36E+02	3.00E+00
Xe-133	4.74E+04	1.50E+02
Xe-135m	4.00E+00	4.00E+00
Xe-135	2.83E+02	3.10E+01
Xe-137	0.00E+00	0.00E+00
Xe-138	1.25E+01	3.00E+00
Cr-51	3.38E-03	1.10E-04
Mn-54	1.81E-03	5.90E-05
Co-57	8.20E-06	8.20E-06
Co-58	1.49E-02	4.90E-04
Co-60	3.93E-03	1.30E-04
Fe-59	8.92E-04	2.90E-05
Sr-89	5.35E-03	2.00E-04
Sr-90	2.56E-03	8.00E-05
Zr-95	4.63E-04	1.50E-05

TABLE 11.A-9 TOTAL ANNUAL DESIGN BASIS AND EXPECTED RELEASES OF AIRBORNE RADIOACTIVE WASTE TO THE ENVIRONMENT FROM ALL RELEASE POINTS COMBINED, IN CURIES PER YEAR (CONTINUED)

NUCLIDE	ANNUAL DESIGN BASIS AIRBORNE EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED AIRBORNE EFFLUENT RELEASE (Ci/yr)
Nb-95	1.05E-03	4.60E-05
Ru-103	1.49E-05	2.00E-05
Ru-106	7.55E-08	3.50E-06
Sb-125	6.10E-07	6.10E-07
Cs-134	2.03E-02	8.00E-05
Cs-136	2.10E-02	3.80E-05
Cs-137	2.69E-02	1.70E-04
Ba-140	1.35E-05	2.70E-05
Ce-141	6.16E-04	1.50E-05
Ar-41	3.40E+01	3.40E+01
C-14	7.30E+00	7.30E+00
H-3	1.10E+02	1.10E+02

TABLE 11.A-10 BASIS FOR REACTOR COOLANT SYSTEM ACTIVITY NUREG-0017 GALE CODE INPUT

EPower Level, MWt	2754
Percent Failed Fuel	1.0
CVCS Purification Ion Exchanger	
Decontamination Factors:	
Iodine	100
Cs, Rb	2
others	100
Purification Flow Rate (CVCS Purification Ion Exchanger), gpm	60
Effective Purification Flow Rate for Lithium and Cesium Removal, gpm	0
Fission Product Escape Rate Coefficients, sec^{-1}	
Noble Gases	6.5×10^{-8}
Halogens, Cs	2.3×10^{-8}
Te, Mo	1.4×10^{-9}
All others	1.4×10^{-11}
Feed and Bleed Liquid Waste, gal./day	2670

FIGURE 11.A–1 ESCAPE RATE COEFFICIENTS

11.B RADIOACTIVE WASTE PROCESSING OF RELEASES TO ENVIRONMENT

11.B.1 BASES

Discussed in this section are the bases used for determining the expected and design-basis annual releases of radionuclides from the radioactive waste processing system to the environment. The description of radioactive waste processing system design and operation provided in this appendix is a representation of how the design and operation have been modeled in the 10 CFR 50 Appendix I, NUREG-0017, GALE Code analysis. GALE, the NRC computer program, was used to calculate expected releases of radioactive material in liquid, gaseous and airborne effluents. The inputs to GALE are summarized in Table 11.B-1. The total expected annual releases are given in Appendix 11.A, Tables 11.A-3 and 11.A-4.

Expected releases, with anticipated operational occurrences included, are calculated (using GALE) for the purpose of ascertaining that the radioactive waste processing systems and building ventilation systems have sufficient capability to ensure that annual releases will be within the limits specified by 10 CFR Part 50, Appendix I. Design basis releases are calculated (as discussed in Appendix 11.A) to demonstrate that the radioactive waste processing systems and building ventilation effluents are within the limits of 10 CFR Part 20, Sections 105 and 106 and Appendix B (version prior to January 1, 1994).

Actual system operation (discussed in FSAR Section 11.1) may differ from the model input to GALE. However, meeting the applicable criteria is ensured through the use of process and effluent radiation monitoring and sampling systems used in conjunction with the Radiological Effluent Monitoring & Offsite Dose Calculation Manual (REMODCM).

11.B.2 LIQUID WASTE PROCESSING SYSTEM

11.B.2.1 Processing of Clean Liquid Waste

The expected releases during normal operation from the clean liquid waste processing system (comprised of primary coolant waste and shim bleed) are based on processing and discharging of approximately 1,200,000 gallons per year of reactor coolant wastes with quantities as shown in Figure 11.B–1.

The clean liquid waste processing system is designed to process reactor coolant waste as well as the letdown flow from the CVCS. The following three sources of liquid waste contribute to the clean liquid waste processing system:

- The primary drain tank, containing clean liquid waste from containment
- The equipment drain sump tank, containing clean liquid waste from the auxiliary building
- Shim bleed from the primary coolant letdown that has been processed by a deborating or purification ion exchanger

Primary coolant letdown is processed by either the purification or deborating ion exchangers. A portion of what leaves these ion exchangers, referred to as shim bleed, is diverted to the clean liquid waste system for further processing. The remainder of the primary coolant letdown is passed on to the volume control tank (VCT). In the volume control tank, primary coolant liquid is degassed and the gases are purged and sent to the gaseous waste processing system with the remaining liquid returned to the primary coolant system.

The three liquid sources, listed above, are sent to a mixed-bed ion exchanger. Following processing in the mixed-bed ion exchangers, the liquid is sent to the coolant waste receiver tanks. From there, the liquid is then sent to a secondary mixed-bed ion exchanger for further processing, after which this liquid is passed into the coolant waste monitor tanks. From these tanks, the liquid is released to the environment via the discharge canal.

This process stream is shown schematically in Figure 11.B–1. Shown in that figure are the tank volumes, flow rates, and decontamination factors (DFs) associated with this liquid waste process stream that are inputs to GALE.

11.B.2.2 Processing of Aerated Liquid Waste

Contributions to the aerated liquid waste processing system include the following:

- Waste water from on-site laundry
- Drainage from hand wash sinks
- Equipment and area decontamination waste water
- Spent fuel pool liner drainage
- Primary coolant sampling system drainage
- Auxiliary building floor drainage
- Primary coolant leakage from miscellaneous sources

These contributions are directed into aerated waste drain tanks. The aerated liquid waste collected in these drain tanks is then sent to a mixed-bed ion exchanger for processing, after which it is sent to an aerated waste monitor tank. The liquid content of this tank is released to the environment via the discharge canal.

11.B.2.3 Processing of Secondary Side Liquid Waste

Secondary side liquid waste is comprised of the following three contributions:

- Steam generator blowdown

- Turbine building floor drainage

- Condensate demineralizer regenerant solutions from condensate polishing facility

Steam generator blowdown is sent to a blowdown flash tank, where one-third of the liquid blowdown flashes to steam and can be released as gaseous effluent. Turbine building floor drainage flows to the turbine building sump and is released to the environment via the discharge canal when radioactivity is detected. With respect to the liquid waste from the condensate polishing facility, the condensate demineralizer regenerant solutions are sent to the waste neutralization sumps. The liquid in the waste neutralization sumps is released to the environment via the discharge canal.

These three contributions to the secondary side liquid waste (identified above) are shown schematically in Figure 11.B–1.

11.B.3 GASEOUS WASTE PROCESSING SYSTEM

Airborne releases from the gaseous radioactive waste processing system, main condenser air ejector, and containment vents are discharged via the Millstone stack. Containment purges, as well as airborne releases from buildings such as the auxiliary and fuel building, are discharged via the Unit 2 enclosure building roof vent. Turbine building releases are discharged via the Unit 2 turbine building roof vent. Steam generator blowdown is discharged via a separate blowdown vent located on the Unit 2 enclosure building roof. For demonstrating compliance with the requirements of 10 CFR Part 20, Sections 105 and 106, and Appendix B (version prior to January 1, 1994) and 10 CFR Part 50, Appendix I, maximum dispersion factors derived from annual average meteorological data were assumed in the 10 CFR Part 50 Appendix I analysis.

The sources and pathways of airborne releases are the following:

- Auxiliary Building/Fuel Building Ventilation: Airborne releases from these sources are normally processed by HEPA filter and monitored prior to release to the environment via the Unit 2 enclosure building roof vent.

- Steam Generator Blowdown Tank Vent: Approximately one-third of the steam generator blowdown flashes to steam and is released to the environment at the Unit 2 enclosure building roof.

- Containment Building: Containment purge air is processed by a HEPA filter and monitored before being released to the environment via the Unit 2 enclosure building roof vent. Releases of routine venting of containment are processed by charcoal adsorber and HEPA filter and monitored before being released to the environment via the Millstone stack.

- Gaseous Waste Gas Processing System: Gases from liquid waste processing system tanks and gases stripped in the VCT from the primary coolant letdown flow are stored in six gas

decay tanks and monitored before being released to the environment via the Millstone stack. No credit is taken for the HEPA filter because it does not satisfy Reg. Guide 1.140.

- Main Condenser/Air Ejector: Non-condensable gases from steam flow to the main condenser are monitored and released to the environment via the Millstone stack.

- Turbine Building: Steam leakage into the turbine building is released to the environment at the Unit 2 turbine building roof vent.

The sources and pathways to the gaseous radioactive waste processing system (identified above) are shown schematically in Figure 11.B–2. These contributions are included in the inputs to GALE and are summarized in Table 11.B-1.

TABLE 11.B-1 INPUTS TO PWR-GALE CODE

DESCRIPTION	UNITS	VALUE
Thermal Power Level	MWt	2,754
Mass of Primary Coolant	10^3 lbm	461.1
Primary Coolant Letdown Rate	GPM	60
Letdown Cation Demineralizer Flow Rate	GPM	0
Number of Steam Generators		2
Total Steam Flow	10^6 lbm/hr	11.8
Mass of Liquid in Each Steam Generator	10^3 lbm	132.257
Total Blowdown Rate	10^3 lbm/hr	73.7
Blowdown Treatment Method	0, 1, or 2	2
Condensate Demineralizer Regeneration Time	days	56
Condensate Demineralizer Flow Fraction (equal to average flow for all condensate demineralizers ÷ total steam flow)	fraction	0.813
CLEAN LIQUID WASTE (see notes)		
Shim Bleed Rate:		
Shim Bleed Flow Rate	GPD	2,670
DF for Iodine		10,000
DF for Cs, Rb		200
DF for Other Nuclides		5,000
Collection Time	days	16.66
Process and Discharge Time	days	0.2865
Fraction Discharged		1
Equipment Drains (Coolant Waste) Inputs:		
Flow Rate	GPD	600
PCA (Fraction of Primary Coolant Activity)		0.145
DF for Iodine		1,000
DF for Cs, Rb		20
DF for Other Nuclides		1,000
Collection Time	days	16.66

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Process and Discharge Time	days	0.2865
Fraction Discharged		1
Clean Wastes Inputs: None (see notes)		
AERATED LIQUID WASTE (see note)		
Dirty Wastes (Aerated Waste) Inputs:		
Flow Rate	GPD	1,110
PCA (Fraction of Primary Coolant Activity)		0.0427
DF for Iodine		100
DF for Cs, Rb		2.0
DF for Other Nuclides		100
Collection Time	days	2.61
Process and Discharge Time	days	0.0598
Fraction Discharged		1
MISC. other liquid waste inputs (see note)		
Steam Generator Blowdown:		
Blowdown Fraction Processed		1
DF for Iodine		1
DF for Cs, Rb		1
DF for Other Nuclides		1
Collection Time	days	0
Process and Discharge Time	days	0
Fraction Discharged		1
Regenerant Inputs:		
Regenerant Flow Rate	GPD	850
DF for Iodine		1
DF for Cs, Rb		1
DF for Other Nuclides		1
Collection Time	days	0
Process and Discharge Time	days	0
Fraction Discharged		1

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Misc. other GALE Code INPUTS:		
Continuous Stripping of Full Letdown Flow		2
Hold-up Time For Xenon	days	90
Hold-up Time for Krypton	days	90
Fill Time for Decay Tanks for Gas Stripping	days	90
AIRBORNE INPUTS:		
Waste Gas System Particulate (HEPA) Filter Removal Efficiency		0%
Fuel Handling Building Charcoal Filter Removal Efficiency		0%
Fuel Handling Building HEPA Filter Removal Efficiency		99%
Auxiliary Building Charcoal Filter Removal Efficiency		0%
Auxiliary Building HEPA Filter Removal Efficiency		99%
Containment Volume	10^6 ft^3	1.899
Containment Atmosphere Clean-up Charcoal Removal Efficiency		0%
Containment Atmosphere Clean-up HEPA Removal Efficiency		0%
Containment Atmosphere Clean-up Rate	10^3 CFM	0
Containment High Volume (Shutdown) Purge Charcoal Removal Efficiency		0%
Containment High Volume (Shutdown) Purge HEPA Filter Removal Efficiency		99%
No. of Purges per Year (2 purges internal to GALE code, see notes)		0
Containment Low Volume Purge (Normal) Charcoal Efficiency (i.e.: containment venting, see note)		70%
Containment Low Volume Purge (Normal) Filter Efficiency (i.e.: containment venting, see note)		99%
Containment Low Volume Purge (Normal) Rate (i.e.: containment venting, see note)	CFM	10
Fraction of Iodine Released from Blowdown Tank Vent		0.05

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Percent of Iodine Removed From (Main Condenser) Air Ejector Release		0
DETERGENT WASTES:		
Detergent Waste PF (Purification Factor), $(DF)^{-1}$		0.1

NOTES:

The component and system designations used in this table reflect the terminology used in NUREG-0017, GALE Code, and may differ from that used by Millstone Unit 2.

The GALE Code has internal calculations that impact the input data. The data provided as GALE Code inputs in this table and Figures 11.B–1 and 11.B–2 may differ from data listed in other tables in the FSAR. A specific example of this are the tank volumes listed on Figure 11.B–1 versus Table 11.1-1. The GALE Code applies a 0.8 or 0.4 factor to tank volumes to calculate collection, process, discharge and decay times.

FIGURE 11.B–1 NUREG-0017 GALE CODE INPUT DIAGRAM - LIQUID WASTE SYSTEM

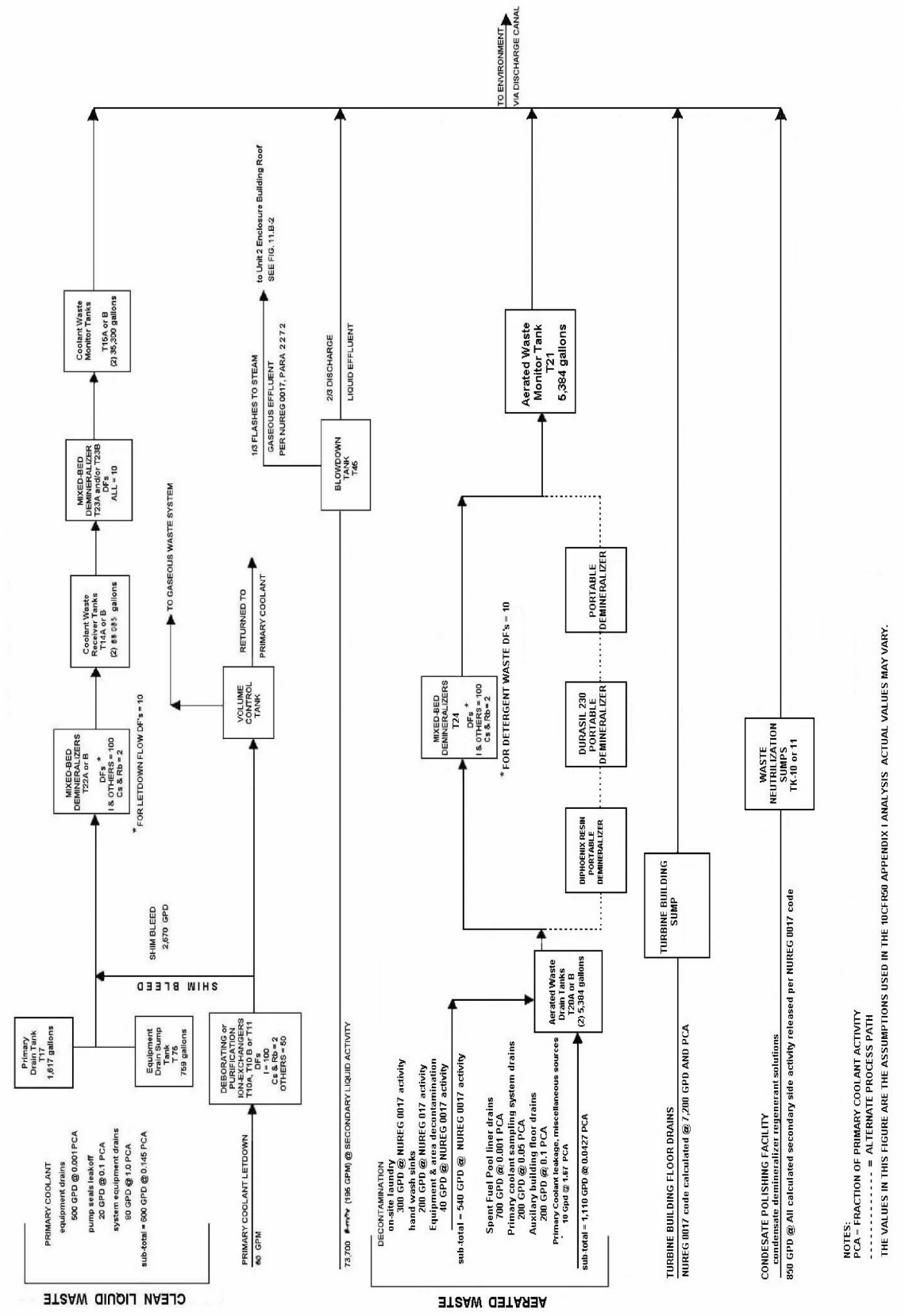
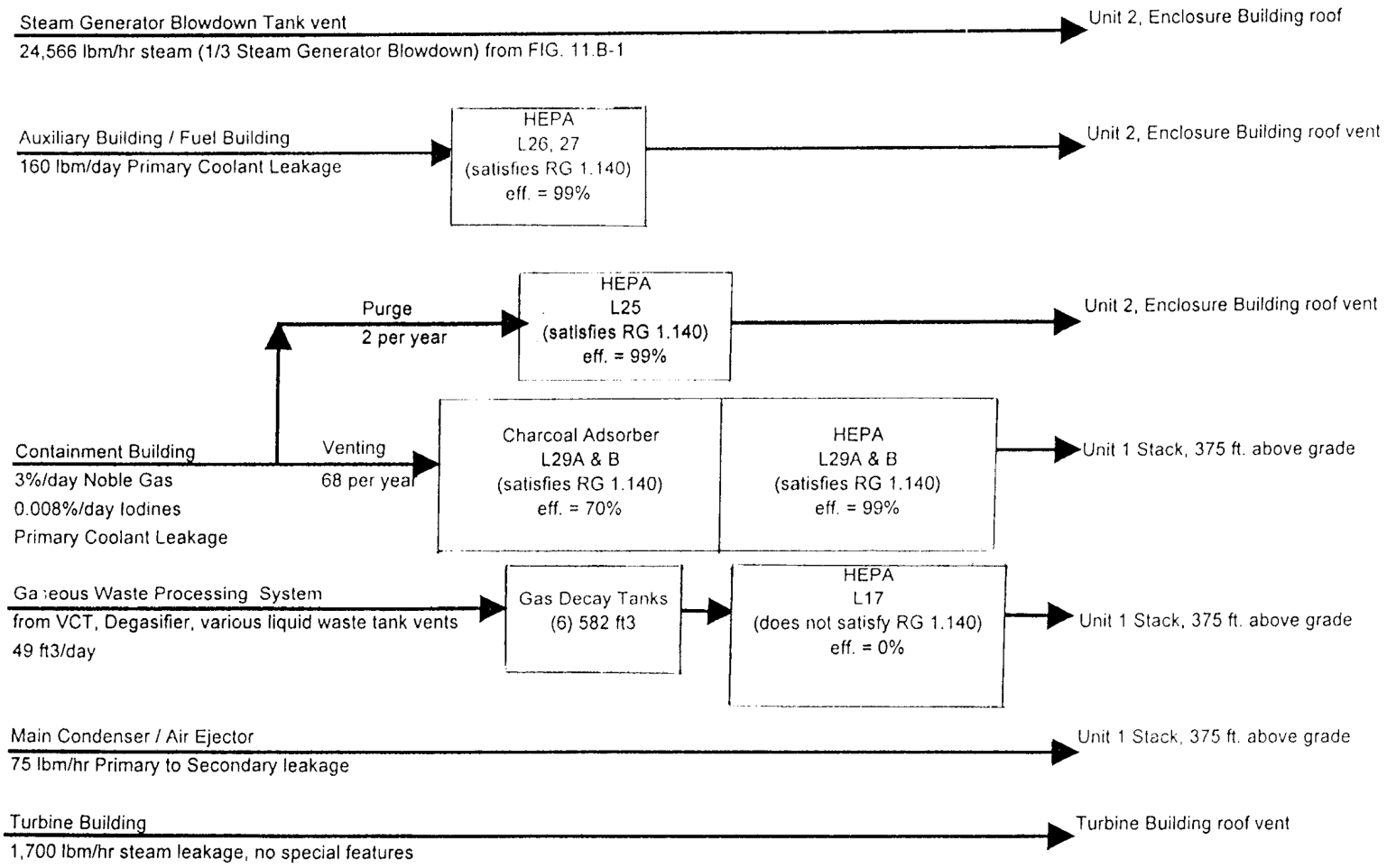


FIGURE 11.B–2 NUREG-0017 GALE CODE INPUT DIAGRAM - AIRBORNE



NOTE: THE VALUES IN THIS FIGURE ARE THE ASSUMPTIONS USED IN THE 10CFR50 APPENDIX I ANALYSIS. ACTUAL VALUES MAY VARY.

11.C DOSES FROM RADIOACTIVE RELEASES AND COST-BENEFIT ANALYSIS

11.C.1 DOSES TO HUMANS

The analysis of annual doses to the maximum individual and to the population residing within an 50 mile radius of Millstone Unit 2 are based on the methodology and equations presented in U. S. NRC Regulatory Guide 1.109 Rev. 1, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for The Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I.* To automate the Millstone Unit 2 pathways-to-man dose analyses, U.S. NRC Computer Code LADTAP II was used for the liquid effluent releases, and U.S. NRC Computer Code GASPAR was used for the gaseous effluent releases. The Regulatory Guide 1.109 equations germane to liquid and gaseous releases are presented in Sections 11.C.2 and 11.C.3, respectively. In the Millstone Unit 2 liquid and gaseous pathways-to-man dose analyses, the NRC default values provided in Regulatory Guide 1.109 were used as input to LADTAP II and GASPAR whenever site-specific data were unavailable.

The ensuing sections present the equations/methodologies employed in LADTAP II and GASPAR to calculate the dose contributions corresponding to the release/uptake pathways considered in the Millstone Unit 2 pathways-to-man dose analyses.

11.C.2 METHODS FOR CALCULATING DOSES FROM LIQUID RELEASES

LADTAP II was used to calculate the radiation exposure to man from ingestion of aquatic foods, shoreline deposits, swimming, and boating. Doses are calculated for both the maximum individual and for the population and are summarized for each pathway by age group and organ. LADTAP II also calculates the doses to certain representative biota other than man in the aquatic environment, such as fish, invertebrates, algae, muskrats, herons, and ducks, using the models presented in WASH-1258.

The equations and assumptions used in the LADTAP II analysis are presented in the ensuing subsections. Final dose results based on pathway doses calculated by LADTAP II are presented in Table 11.C-1.

11.C.2.1 Generalized Equation for Calculating Radiation Doses to Humans via Liquid Pathways

$$R_{aipj} = (C_{jp}) \cdot (U_{ap}) \cdot (D_{aipj})$$

where:

R_{aipj} = the annual dose to organ j of an individual of age group a from nuclide i via pathway p, in mrem/yr

C_{ip} = the concentration of nuclide i in the media of pathway p, in pCi/l, pCi/kg, or pCi/m^2

U_{ap} = the exposure time or intake rate (usage) associated with pathway *p* for age group *a*, in hr/yr, yr^{-1}, or kg/yr (as appropriate)

D_{aipj} = the dose factor, specific age group *a*, radionuclide *i*, pathway *p*, and organ *j*, in mrem/pCi ingested, mrem/hr per pCi/m^2 from exposure to deposited activity in sediment or on the ground, or mrem/hr per pCi/liter due to exposure from boating and swimming

11.C.2.2 Doses from Aquatic Foods

$$R_{apj} = (1{,}100)[(U_{ap})(M_p)/F] \cdot \Sigma_i Q_i l.\ B_{ip} \cdot D_{aipj} \cdot e^{-\lambda_i t_p}$$

where:

B_{ip} = the equilibrium bioaccumulation factor for nuclide *i* in pathway *p*, expressed as the ratio of the concentration in biota (in pCi/kg) to the radionuclide concentration in water (in pCi/l), in l/kg

M_p = the mixing ratio (reciprocal of the dilution factor) at the point of exposure or point of harvest of aquatic food, dimensionless

F = the flow rate of the liquid effluent in ft^3/sec

Q_i = the release rate of nuclide *i*, in Ci/yr

R_{apj} = the total annual dose to organ *j* of individuals of age group a from all of the nuclides *i* in pathway *p*, in mrem/yr

λ_i = the radioactive decay constant of nuclide *i*, in hours

t_p = the average transit time required for nuclides to reach the point of exposure. For internal dose, *t* is the total time elapsed between release of the nuclides and ingestion of food, in hours.

1,100 = the factor to convert from (Ci/yr)/(ft^3/sec) to pCi/l

All the other symbols are as previously defined.

11.C.2.3 Doses from Shoreline Deposits

$$R_{apj} = 110{,}000 \times [(U_{ap})(M_p)(W)/F] \times \Sigma_i Q_i T_i D_{aipj} \times e^{-\lambda_i t_p} \times (1 - e^{-\lambda_i t_b})$$

where:

W = the shoreline width factor that describes the geometry of the exposure, dimensionless

T_i = the radiological half-life of nuclide *i*, in days

t_b = the period of time for which sediment or soil is exposed to the contaminated water, in hours

110,000 = the factor to convert from Ci/yr per ft^3/sec to pCi/l and to account for the proportionality constant used in the sediment radioactivity model

All other symbols are as previously defined.

11.C.2.4 Doses from Swimming and Boating

The doses from swimming and boating were calculated using the methodology described in WASH 1258 (Atomic Energy Commission 1973).

The equation for calculation of the external dose to skin and the total body dose from swimming (water immersion) or boating (water surface) is:

$$R_{apj} = 1{,}000 \times [(U_{ap})(M_p)/(F)(K_p)] \times \Sigma_i Q_i D_{aipj} \times e^{-\lambda_i t_p}$$

where:

K_p = geometry correction factor equal to 1 for swimming and 2 for boating, dimensionless (no credit is taken for the shielding provided by the boat).

All other symbols are as previously defined.

11.C.3 METHOD FOR CALCULATING DOSES FROM GASEOUS RELEASES

FASPAR implements the air release dose models of the NRC Regulatory Guide 1.109 for noble gases (semi-infinite plume only) and for radioiodine and particulate emissions. GASPAR computes both population and individual doses using site data, meteorological data, and radionuclide release source terms as inputs. Location-dependent meteorological data for selected individuals are specified as input data. The site data include the population distribution and the quantities of milk, meat, and vegetation produced. The meteorological data include dispersion X/Q, X/Q decayed, X/Q decayed and depleted, and deposition D/Q. Population doses, individual doses, and cost benefit results are calculated.

There are two basic types of calculations, the individual dose calculation and the population dose calculation. Seven pathways by which the nuclides travel to man are considered. These are plume, ground, inhalation, vegetation, cows' milk, goats' milk, and meat. For the individual dose calculations, man is subdivided into the four age groups of infant (0 to 1 year), child (1 to 11 years), teenager (12 to 18 years), and adult (over 18 years). Each of these calculations takes into account eight body organs: total body, gastrointestinal (GI) tract, bone, liver, kidney, thyroid, lung, and skin.

The equations and assumptions used in the GASPAR analysis are presented in the ensuing subsections. Final dose results by GASPAR, based on activity releases tabulated in Table 11.A-3, are presented in Table 11.C-1.

11.C.3.1 Gamma and Beta Doses from Noble Gas Discharged to the Atmosphere

11.C.3.1.1 Annual Air Doses from Noble Gas Releases (Non-Elevated)

- Annual Gamma Air Dose Equation:

$$D^{\gamma}(r,\theta) = 3.17 \times 10^{4} \cdot \sum_{i} Q_i \cdot \left[X/Q\right](r,\theta) \cdot \left(DF_i^{\gamma}\right)$$

- Annual Beta Air Dose:

$$D^{\beta}(r,\theta) = 3.17 \times 10^{4} \cdot \sum_{i} Q_i \cdot \left[X/Q\right](r,\theta) \cdot \left(DF_i^{\beta}\right)$$

where:

$D^{\gamma}(r,\theta)$ = the annual gamma air doses at the distance r in the sector at angle θ from the discharge point in mrad/year

$D^{\beta}(r,\theta)$ = the annual beta air doses at the distance r in the sector at angle θ from the discharge point in mrad/year

Q_i = the release rate of the radionuclide i, in Ci/year

$X/Q(r,\theta)$ = the annual average gaseous dispersion factor at distance r in sector θ, in sec/m^3 (corrected for radioactive decay)

DF^{γ}_i = the gamma air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad·m^3/pCi·yr

DF^{β}_i = the beta air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad·m^3/pCi·yr

3.17 x 10^4 = the number of pCi per Ci divided by the number of seconds per year

11.C.3.1.2 Annual Total Body Dose from Noble Gas Releases (Non-Elevated)

$$D^{T}_{\infty}(r,\theta) = S_F \sum_i \chi_i(r,\theta) \cdot DFB_i$$

where:

$D^T\infty(r,\theta)$ = the total body dose due to immersion in a semi-infinite cloud at distance r in sector θ, in mrem/year

S_F = the attenuation factor that accounts for dose reduction due to shielding provided by residential structures, dimensionless

$\chi_i(r,\theta)$ = the annual average ground-level concentration of radionuclide i at distance r in sector θ, in pCi/ m^3

DFB_i = the total body dose factor for a semi-infinite cloud of the radionuclide i, which includes radiation attenuation through a depth of 5 cm into the body, in mrem·m^3/ pCi·yr

11.C.3.1.3 Annual Skin Dose from Noble Gas Releases (Non-Elevated)

$$D^{S}_{\infty}(r,\theta) = 1.11 \cdot S_F \cdot \sum_i \chi_i(r,\theta) \cdot DF^{\gamma}_i + \sum_i \chi_i(r,\theta) \cdot DFS_i$$

where:

$D^S\infty(r,\theta)$ = the annual skin dose due to immersion in a semi-infinite cloud at the distance r in sector θ, in mrem/yr

DFS_i = the beta skin dose factor for a semi-infinite cloud of radionuclide i, which includes the attenuation by the outer "dead" layer of the skin, in mrem·m^3/pCi·yr

1.11 = the average ratio of tissue to air energy absorption coefficients

All other parameters are as previously defined.

11.C.3.1.4 Annual Gamma Air Dose and Annual Total Body Dose Due to Noble Gas Releases from Free-Standing Stacks More Than 80 Meters Tall

Regulatory Guide 1.109 provides gamma air dose and total body dose equations that are applicable to elevated releases of noble gases from free-standing stacks that are more than 80 meters tall. However, GASPAR does not include these equations, and any attempt to use GASPAR to calculate cloud shine doses from elevated releases would lead to non-conservative results within approximately 1 kilometer from the point of release. What was done in the Millstone Unit 2 analysis was to simulate an elevated release by making a separate GASPAR run using the stack release radionuclide inventory as input, but with ground-level release X/Qs used in lieu of stack release X/Qs. This conservative approach was taken to calculate noble gas gamma shine doses (both gamma air doses and total body doses) to maximum individuals.

11.C.3.2 Doses from Radioiodines and Other Radionuclides, Exclusive of Noble Gases, Released to the Atmosphere

11.C.3.2.1 Annual Organ Dose Due to External Irradiation from Ground Deposition of Radionuclides

$$D_j^G(r,\theta) = 8{,}760 \cdot S_F \cdot \sum_i C^{G_i}(r,\theta) \cdot DFG_{ij}$$

where:

$D^G_j(r,\theta)$ = the annual dose to organ j at location (r,θ), in mrem/yr

S_F = a shielding factor that accounts for the dose reduction due to shielding provided by residential structures during occupancy (assumed to be 0.7), dimensionless

$C^G_i(r,\theta)$ = the ground plane concentration of radionuclide i at distance r in sector θ, in pCi/m^2

DFG_{ij} = the open field ground plane dose conversion factor for organ j from radionuclide i, in $mrem \cdot m^2/pCi \cdot hr$

8,760 = the number of hours in a year

11.C.3.2.2 Annual Organ Dose from Inhalation of Radionuclides in Air

$$D^A_{ja}(r,\theta) = R_a \cdot \sum_i \chi_i(r,\theta) \cdot DFA_{ija}$$

where:

D^{A}_{ja} (r,θ) = the annual dose to organ *j* of an individual in the age group *a* at location (r,θ) due to inhalation, in mrem/yr

R_a = the annual air intake for individuals in the age group *a*, in m^3/yr

χ_i (r,θ) = the annual average concentration of radionuclide *i* in air at location (r,θ), in pCi/m^3

DFA_{ija} = the inhalation dose factor for radionuclide *i*, organ *j*, and age group *a*, in mrem/pCi

11.C.3.2.3 Annual Organ Dose from Ingestion of Atmospherically Released Radionuclides in Food

$$D^{D}_{ja}(r,\theta) = \sum_i DFI_{ija} \cdot \left[U^{V}_{a} \cdot f_g \cdot C^{V}_{i}(r,\theta) + U^{m}_{a} \cdot C^{m}_{i}(r,\theta) + U^{F}_{a} \cdot C^{F}_{i}(r,\theta) + U^{L}_{a} \cdot f_l \cdot C^{L}_{i}(r,\theta) \right]$$

where:

$C^{v}_{i}(r,\theta)$, $C^{m}_{i}(r,\theta)$, $C^{L}_{i}(r,\theta)$, $C^{F}_{i}(r,\theta)$ = the concentrations of radionuclide *i* in produce (non-leafy vegetables, leafy vegetables, fruits, and grains), milk, and meat, at location (r,θ), in pCi/kg or pCi/l

$D^{D}_{ja}(r,\theta)$ = the annual dose to the organ *i* of an individual in age group *a* from ingestion of produce, milk, leafy vegetables, and meat at location (r,θ), in mrem/year

DFI_{ija} = the ingestion dose factor for radionuclide *i*, organ *j*, and age group *a* in mrem/pCi

f_g, f_l = the respective fractions of the ingestion rates of produce and leafy vegetables that are produced in the garden of interest

U^{v}_{a}, U^{m}_{a}, U^{F}_{a}, U^{La} = the annual intake (usage) of produce, milk, meat, and leafy vegetables, respectively, for individuals in the age group *a*, in kg/yr or l/yr

11.C.4 COMPARISON OF CALCULATED ANNUAL MAXIMUM INDIVIDUAL DOSES WITH APPENDIX I DESIGN OBJECTIVES

A comparison of calculated annual maximum individual doses with 10 CFR Part 50 Appendix I design objectives is provided in Table 11.C-1.

11.C.5 GENERAL EXPRESSION FOR POPULATION DOSES

The general expression for calculating the annual population-integrated dose is:

$$D_j^P = 0.001 \sum_d P_d \cdot \sum_a D_{jda} \cdot f_{da}$$

where:

D^P_j = the annual population-integrated dose to organ *j* (total body or thyroid), in man-rem or thyroid man-rem

P_d = the population associated with sub-region *d*

D_{jda} = the annual dose to organ *j* (total body or thyroid) of an average individual of age group *a* in sub-region *d*, in mrem/yr

f_{da} = the fraction of the population in sub-region *d* that is in age group *a*

0.001 = the conversion factor from mrem to rem

The equation above, used in conjunction with the preceding equations with parameters adjusted for each age group, is used to calculate the population doses. The population doses due to annual releases of expected liquid and airborne radionuclide concentrations are presented in Table 11.C-2.

11.C.6 COST-BENEFIT ANALYSIS

Presented in this section are the cost-benefit analyses and their results, performed in accordance with the requirements set forth in 10 CFR Part 50, Appendix I, Section II.D. In these analyses, potential augments to the liquid and gaseous radioactive waste processing systems are examined for cost-effectiveness using the methodology and data provided in U. S. NRC Regulatory Guide 1.110 Rev. 0 (March 1976), *Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors*. The beneficial savings of each augment is calculated by multiplying the calculated dose reduction by $1,000 per man-rem. The cost of borrowed money is conservatively assumed to be 9%[1]. Provided in Appendix 11.B is information pertinent to the model input to GALE representing the design and configuration of the Millstone Unit 2 radioactive waste processing systems.

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[1] This is predicated on Northeast Utilities' June 1998 Cost of Capital Study, in which it was concluded that the weighted cost of borrowed money for NU, CL&P, WMECO, PSNH, and combined system is 9.3%. Based on this conclusion, it is conservative in the cost-benefit analysis to round down to 9%.

11.C.6.1 Procedure Used for Performing Cost-Benefit Analysis

The procedure employed for performing the cost-benefit analyses presented in Sections 11.C.6.2 and 11.C.6.3 below is taken directly from Regulatory Guide 1.110 Rev. 0. This procedure is summarized in the following steps:

1. Obtain the direct cost of equipment and materials from Table A-1 in Regulatory Guide 1.110.

2. Multiply the direct labor cost obtained from Table A-1 in Regulatory Guide 1.110 by the appropriate labor cost correction factor, obtained from Table A-4 in Regulatory Guide 1.110, to obtain the corrected labor cost for the geographical region in which the plant is located (Millstone Unit 2 being in Region I).

3. Add the costs obtained from the previous two steps to obtain the total direct cost (*TDC*).

4. Obtain the appropriate indirect cost factor (*ICF*) from Table A-5 in Regulatory Guide 1.110.

5. Determine the total capital cost (*TCC*) using the following equation:

   $TCC = TDC \times ICF$

6. Obtain the appropriate capital recovery factor (*CRF*) from Table A-6 in Regulatory Guide 1.110.

7. Determine the annual fixed cost (*AFC*) using the following equation:

   $AFC = TCC \times CRF$

8. Obtain the annual operating cost (*AOC*) and the annual maintenance cost (*AMC*) from Regulatory Guide 1.110 Tables A-2 and A-3, respectively.

9. Determine the total annual cost (*TAC*) using the following equation:

   $TAC = AFC + AOC + AMC$

10. Determine the benefit of each augment by multiplying the dose reduction to be achieved by $1,000 per man-rem.

11. The benefit calculated in the previous step minus *TAC* provides the net benefit of adding the augment to the radwaste system. A positive net benefit means that adding the augment would be cost-beneficial; conversely, a negative net benefit means that it would not be cost-beneficial to add the augment to the radwaste system.

11.C.6.2 Augments to the Liquid Radioactive Waste Processing System

Table 11.C-2 presents the calculated base case annual total body dose (man-rem), annual maximum organ dose (man-rem), and annual thyroid dose (man-rem) associated with the operation of the Millstone Unit 2 liquid radioactive waste processing system for the population expected to live within a 50 mile radius of the plant. In the cost-benefit analysis performed for the liquid radioactive waste processing system, the augment that was chosen was a 50-gpm demineralizer on the blow-down waste stream. This demineralizer was selected because it was judged to be the least expensive augment that could provide any benefit in the reduction of population dose. Assuming that this augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum annual benefit to be realized would be approximately $4,880. However, off-setting this benefit is a total annual cost associated with this augment that is estimated to be approximately $43,800.

Conclusion: It would not be cost-beneficial to add additional liquid radwaste processing equipment to the existing system.

The salient inputs to the liquid radioactive waste processing system cost-benefit analysis (taken from the tables in Regulatory Guide 1.110) are summarized below:

- Equipment/material cost: $43,000
- Labor cost: $29,000
- Labor cost correction factor: 1.6
- Indirect cost factor (ICF): 1.58, based on the fact that Millstone is a three unit site, with each unit having its own radwaste system
- Capital recovery factor (CRF): 0.0973
- Annual operating cost (AOC): $25,000
- Annual maintenance cost (AMC): $5,000
- Dose reduction goal: 4.88 man-rem (assuming that the goal is to reduce to zero the largest population dose resulting from liquid effluent released to the environment, that population dose having been calculated to be the maximum organ dose)

11.C.6.3 Augments to the Gaseous Radioactive Waste Processing System

Table 11.C-2 presents the calculated base case annual total body dose (man-rem) and annual thyroid dose (man-rem) associated with the operation of the Unit 2 gaseous radioactive waste processing system for the population expected to live within a 50 mile radius of the plant. In the cost-benefit analysis performed for the gaseous radioactive waste processing system, the augment that was chosen was a pair of 30,000 cfm charcoal/HEPA filtration systems on the Unit 2

auxiliary building vent. This choice was made because it was judged to be the least expensive augment that could provide any benefit in the reduction of population dose. Assuming that this augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum annual benefit to be realized would be approximately $19,600. However, off-setting this benefit is a total annual cost associated with this augment that is estimated to be approximately $127,500.

Conclusion: It would not be cost-beneficial to add the two 30,000-cfm charcoal/HEPA filtration systems to the existing gaseous radioactive waste processing system. The salient inputs to the gaseous radioactive waste processing system cost-benefit analysis (taken from the tables in Regulatory Guide 1.110) are summarized below:

- Equipment/material cost: $314,000
- Labor cost: $102,000
- Labor cost correction factor: 1.6
- Indirect cost factor (ICF): 1.58, based on the fact that Millstone is a three-unit site, with each unit having its own radwaste system
- Capital recovery factor (CRF): 0.0973
- Annual operating cost (AOC): $18,000
- Annual maintenance cost (AMC): $36,000
- Dose reduction goal: 19.6 man-rem (assuming that the goal is to reduce to zero the largest population dose resulting from liquid effluent released to the environment, that population dose having been calculated to be the thyroid dose)

TABLE 11.C-1 COMPARISON OF CALCULATED ANNUAL MAXIMUM INDIVIDUAL DOSES WITH 10 CFR PART 50 APPENDIX I DESIGN OBJECTIVES

	10 CFR PART 50 APPENDIX I DESIGN OBJECTIVE DOSE [1]	CALCULATED ANNUAL DOSE
Airborne Effluent:		
• Gamma Air Dose	10 mrad	0.2 mrad
• Beta Air Dose	20 mrad	0.078 mrad
• Total Body Dose	5 mrem	0.15 mrem
• Skin Dose	15 mrem	0.25 mrem
• Maximum Organ Dose (Thyroid)	15 mrem	8.3 mrem
Liquid Effluent:		
• Total Body Dose	3 mrem	0.06 mrem
• Maximum Organ Dose (GI-LLI)	10 mrem	0.91 mrem

1 Per reactor.

TABLE 11.C-2 ANNUAL TOTAL BODY AND THYROID DOSES TO THE POPULATION WITHIN 50 MILES OF THE MILLSTONE SITE, IN MAN-REM, FROM EXPECTED LIQUID AND AIRBORNE EFFLUENT RELEASES

1. Annual Total Population Doses from Airborne Effluent Releases:
    - Total Body Dose: 0.607 man-rem
    - Thyroid Dose: 19.6 man-rem
2. Annual Total Population Doses from Liquid Effluent Releases:
    - Total Body Dose: 1.89 man-rem
    - Thyroid Dose: 2.86 man-rem
    - Maximum Organ Dose (GI-LLI): 4.88 man-rem

11.D EXPECTED ANNUAL INHALATION DOSES AND ESTIMATED AIR CONCENTRATIONS OF RADIOACTIVE ISOTOPES FOR MP2 FACILITIES

The expected annual inhalation doses to plant personnel and estimated air concentrations were computed as outlined below:

Equilibrium airborne concentrations:

Equilibrium concentration = (Production/Losses) = fA/λ

where: f = flow rate of activity source (Ci/sec)

A = activity of source (Ci/cc)

λ = loss rates = $\Sigma\lambda d = \lambda e + \lambda R$

λe = leakage or exhaust

λR = recirculation for iodines, if any

Doses:

The whole body dose is determined using a semi-infinite cloud model (Safety Guide 4). The thyroid inhalation dose is found using the inhalation model and dose conversion factors of TID 14844 and Safety Guide 4. The other organ doses are found by the fraction of allowable MPC, which yields the limiting doses.

Whole Body Doses

Lung Dose:

$$D_I Lung = \frac{C_I}{MPC_I Lung} \times MPD^{Lung}$$

where: C_I = Equilibrium concentration of the Ith isotope

$MPC_I Lung$ = Maximum permissible concentration for the Ith isotope for lung based on a 40 hour workweek taken from ICRP II for the airborne case

MPD^{Lung} = Maximum permissible yearly dose to lung

For Cs 138 and Rb 88, D_I^{Lung} is computed as follows:

$D^{Lung} = 2.81 \times 10^5 \times T \times E \times C$

where: Y = radiological half life

E = effective energy absorbed per disintegration in MeV/disintegration

C = Concentration in μCi/cc of CS^{138} or ^{88}Rb.

	Y Min.	E(MeV/dis)
CS 138	32.2	1.39
Rb 88	17.8	1.68

The total lung dose D^L is found as follows:

$$D^L = \sum_{I=1}^{n} D_I Lung$$

Thyroid Dose:

$D_I Thyroid = CI \times t \times BR \times DCF_I$

where: $D_I Thyroid$ = Thyroid dose in REMs

C_I = Concentration of the Iodine isotope I

t = Exposure time in day = 300 days

BR = Breathing rate in cc/day = 2×10^7 cc/day

DCF = Dose conversion factors for Iodine in REMs/Ci per TID 14844

The total thyroid dose is found as follows:

$$D^T = \Sigma_{I=1}^{n} D_I Thyroid$$

Bases and Results:

Source terms are based on reactor operation with 0.1 percent failed fuel.

Containment Building:

1. Containment volume - 1.92×10^6 ft^3

2. Purge containment for a minimum of five hours at 32,000 CFM prior to personnel entering the containment (see FSAR Section 9.9.2).

3. Activity source is conservatively assumed to be 40 GPD of reactor coolant leakage. The expected and calculated leakage of reactor coolant to the containment atmosphere will be a small fraction of the 40 GPD rate used in the analysis. The 40

GPD leakage rate was taken for “Twelfth AEC Air Cleaning Conference-Analysis of Power Reactor Gaseous Waste System.”

4. Assume 75 days of activity buildup in the containment.

5. A decontamination factor of 10, used for iodines and particulates, was taken from “Twelfth AEC Air Cleaning Conference-Analysis of Power Reactor Gaseous Waste System.”

The results indicate the whole body dose rate will be 0.85 mr/hr. After 10 hours, the dose rate decreases to 0.44 mrem/hr. Concentrations, given in Table 11.D–1, are below the 10 CFR Part 20 allowable airborne concentrations.

Auxiliary Building (Excludes nonradioactive areas and fuel handling area):

1. Assume 20 GPD leakage of 120°F fluid and 1 GPD leakage of 550°F fluid with reactor coolant concentrations of activity.

2. Ventilation air flow is 40,000 CFM for radioactive areas of auxiliary building. This corresponds to approximately five air changes per hour.

3. Iodine and particulate partition factors assumed as follows:

   Coolant at 120°F - 10^{-4}

   Coolant at 550°F - 10^{-1}

4. 2000 hours of exposure per year.

Doses based on 40 hours per week, 52 weeks per year exposure are as follows:

Lung Dose - 0.046 REM

Whole Body Dose - 0.468 REM

Thyroid Dose - 0.61 REM

Airborne concentrations are given in Table 11.D–2.

Auxiliary Building (Fuel handling area):

1. Sources arise due to evaporation from the spent fuel pool surface.

2. Sources assumed as one-tenth of reactor coolant concentrations (0.1% failed fuel). Only tritium is considered since separation factors for iodines and particulates in spent fuel pool water will greatly reduce airborne concentrations. Contribution from noble gases will be insignificant.

3. Evaporation rate from spent fuel pool at 120°F is 0.37 gpm.

4. There will be 20 air changes per hour at average exhaust rate of 18,000 CFM (FSAR Section 9.9).

5. Only whole body dose is considered as the critical organ for tritium.

6. Exposure calculated for 2000 hrs/year.

The airborne equilibrium concentration for tritium is 1.48×10^{-8} μCi/cc. The whole body dose is 25.2 mr/yr.

Turbine Building

1. Sources arise due to leakage from the secondary systems into the turbine building.

2. Primary coolant system operating at 0.1% failed fuel.

3. Primary to secondary leakage of 100 gallons per day.

4. 5 gpm continuous blowdown per steam generator.

5. Processed blowdown liquid is discharged.

6. Feedwater cleanup factor is 0.

7. Steam leakage from the secondary side is 1700 lb/hr.

8. Liquid leakage from the secondary side is 15 gpm.

9. Volume of the turbine building 2,882,000 ft^3.

10. Exhaust flow rate is 200,000 cfm.

Doses based on 40 hours per week, 50 weeks per year exposure are as follows:

Lung Dose - 7.8×10^{-3} REM

Whole Body Dos - 2.4×10^{-5} REM

Thyroid Dose - 0.314 REM

Airborne concentrations are given in Table 11.D–3.

TABLE 11.D–1 CONTAINMENT BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION μCi/cc
I-131	1.31×10^{-8}
I-132	9.75×10^{-10}
I-133	7.46×10^{-9}
I-134	3.99×10^{-10}
I-135	2.95×10^{-9}
Xe-131M	6.35×10^{-8}
Xe-133M	--
Xe-133	4.60×10^{-6}
Xe-135M	--
Xe-135	8.61×10^{-8}
Xe-137	--
Xe-138	1.24×10^{-9}
Kr-83M	--
Kr-85M	1.52×10^{-8}
Kr-85	1.35×10^{-7}
Kr-87	6.07×10^{-9}
Kr-88	2.43×10^{-8}
Kr-89	--

TABLE 11.D–2 AUXILIARY BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION µCi/cc
Kr-85M	8.90×10^{-9}
Kr-85	5.45×10^{-9}
Kr-87	4.52×10^{-9}
Kr-88	1.53×10^{-8}
Xe-131M	9.10×10^{-9}
Xe-133	1.11×10^{-6}
Xe-135	4.56×10^{-8}
Xe-138	1.42×10^{-9}
I-129	2.96×10^{-18}
I-131	1.63×10^{-10}
I-132	4.23×10^{-11}
I-133	2.31×10^{-10}
I-134	2.21×10^{-11}
I-135	1.09×10^{-10}
Br-84	1.53×10^{-12}
Rb-88	7.24×10^{-11}
Rb-89	1.72×10^{-12}
Sr-89	2.08×10^{-13}
Sr-90	1.07×10^{-14}
Sr-91	1.44×10^{-13}
Te-129	9.24×10^{-13}
Te-132	1.35×10^{-11}
Te-134	9.08×10^{-13}
Be-140	2.50×10^{-13}
Ru-103	1.69×10^{-13}

TABLE 11.D–2 AUXILIARY BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION µCi/cc
Ru-106	1.02×10^{-15}
La-140	2.39×10^{-13}
Ce-144	1.69×10^{-13}
Pr-143	2.39×10^{-13}
Mo-99	8.31×10^{-11}
Y-90	4.17×10^{-14}
Y-91	4.51×10^{-12}
Cs-134	4.10×10^{-12}
Cs-136	1.05×10^{-12}
Cs-137	1.31×10^{-11}
Cs-138	2.27×10^{-11}
Cr-51	1.56×10^{-14}
Mn-54	1.31×10^{-14}
Fe-59	8.73×10^{-15}
Co-58	1.91×10^{-12}
Co-60	2.13×10^{-13}
Zr-95	3.83×10^{-16}

TABLE 11.D–3 TURBINE BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION Ci/cc
H-3	5.14E-10
Cr-51	1.59E-15
Mn-54	1.21E-15
Fe-59	9.08E-16
Co-58	2.01E-13
Co-60	2.29E-14
Br-84	1.09E-14
Kr-85M	4.61E-12
Kr-85	2.88E-12
Kr-87	2.19E-12
Kr-88	7.80E-12
Rb-88	5.74E-14
Rb-89	1.17E-15
Sr-89	2.17E-14
Sr-90	1.15E-15
Sr-91	3.24E-15
Y-90	2.85E-15
Y-91	4.72E-13
Zr-95	4.04E-17
Mo-99	5.75E-12
Ru-103	1.75E-14
Ru-106	1.09E-16
Te-129	2.96E-15
Te-132	9.90E-13
Te-134	1.80E-15
I-129	1.56E-18
I-131	7.22E-11
I-132	1.29E-12
I-133	4.32E-11

TABLE 11.D–3 TURBINE BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION Ci/cc
I-134	2.60E-13
I-135	8.81E-12
Xe-131M	4.83E-12
Xe-133	5.89E-10
Xe-135	2.41E-11
Xe-138	5.88E-13
Cs-134	4.40E-13
Cs-136	1.01E-13
Cs-137	1.41E-12
Cs-138	3.36E-14
Ba-140	2.40E-14
La-140	1.34E-14
Ce-144	1.81E-14
Fr-143	2.24E-14

Where E-14 = 10^{-14}, etc.

11.E AIRBORNE ACTIVITY SAMPLING SYSTEM FOR CONTAINMENT, SPENT FUEL AND RADWASTE ATMOSPHERES

In-plant control of personnel exposure from airborne radioactivity will be effected by a continuing program of sampling for airborne activity and administrative controls through radiation work permits.

1. Containment atmosphere is continuously monitored by two sample lines inside containment. The sample is pulled through a fixed particulate filter, a charcoal cartridge and a gas chamber before being pumped back to the containment. Particulate activity is measured by a beta scintillator through a derivative (rate of change) circuit. Any rapid changes in beta activity will be alarmed in the control room. Gaseous iodine is absorbed on the charcoal cartridge. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period. Gaseous activity is measured with a beta scintillator type detector. Increased activity above pre-set levels will alarm in the control room.

   Upon indication of a significant increase in activity in the containment building, the Shift Supervisor will notify radiation protection. Radiation protection will obtain and analyze samples of containment air to determine the isotopes responsible for the increased activity. During power operation, if conditions warrant, radiation protection personnel may enter the containment and take portable air samples in selected areas in order to locate the source of the activity. Appropriate respiratory equipment will be indicated on the radiation work permits.

   During shutdown and maintenance conditions, air samples will be taken with the portable sampling equipment throughout the containment building on a scheduled basis and as required, to determine the presence of any unusual concentrations of airborne activity. Appropriate respiratory protection, if required, will be indicated on the radiation work permits.

   While work is being performed on the reactor vessel head and there is a potential for release of airborne activity, continuous radiation protection monitoring of the work area will be in effect. This monitoring will consist of particulate and iodine sampling. Workers will be instructed to leave the area if they suspect any unusual problem.

   The spent fuel pool exhaust monitor is a continuous sampling system that takes suction upstream to the HEPA filter. This sample passes through a particulate filter, charcoal cartridge and gaseous 4 pi beta, gamma detection chamber. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with

the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period. The gaseous channel is monitored and alarmed in the control room. An alarm condition will require notification of radiation protection to determine the source of the increased activity. Work area air sampling will be established consistent with spent fuel pool ventilation line up and work in progress. These samples will supplement the spent fuel pool exhaust monitor. During periods when the spent fuel pool exhaust or the containment atmosphere monitors are not available, air samples are taken with portable sampling equipment in these areas on a scheduled basis per the Millstone Effluent Control Program to determine the presence of any unusual concentrations of airborne activity.

2. The area with the highest potential for creating airborne activity is the sampling room. All other radwaste areas contain equipment that is closed and therefore of a low potential for creating airborne activity.

   The sampling room (primary sample sink area) has a separate ventilation system via the sample hood. The hood design is such that at least 100 linear feet per minute flow is maintained over the working area in the hood to ensure that there is no contamination or airborne activity spread into the sampling room. Visual indication is available to the personnel frequenting the sampling room to indicate the condition of the exhaust system. A shutdown exhaust system would require health physics evaluation of airborne conditions prior to entry.

3. The radwaste airborne radioactivity monitoring system consists of four radiation monitors, each capable of continuously monitoring airborne radioactive particulates. In addition, each monitor contains an iodine sampling assembly consisting of a replaceable charcoal cartridge mounted in series and downstream of the particulate monitor. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period.

   Since the waste gas system of the radwaste areas has the potential for the release of gaseous activity, the airborne radiation monitor sampling the ventilation exhaust duct servicing that area also contains a gaseous radioactivity monitor in series with the particulate and iodine monitor. The gaseous monitor will be used to detect and measure significant noble gas releases from the waste gas system.

   Sampling for each monitor is accomplished by extracting a representative sample from the radwaste ventilation exhaust system by using an isokinetic nozzle designed in accordance with ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.

Since the airflow at each point selected represents the exhaust of several compartments, the effect of dilution on the compartment having the least flow was analyzed to demonstrate the capability of detection and measurement of airborne contaminants in the areas being served by that section of the ventilation exhaust system.

For this analysis, a concentration of $3x10^{-9}$ microcuries per cubic centimeter was used. To determine the radioactive concentration at the sampling point, this value was multiplied by the ratio of the flow for the compartment having the lowest airflow to the total flow at the sample point.

Point A (RM-8999 P&ID 25203-26029)

The airflow at this sampling point is 15,495 scfm. The ventilation exhaust system at this point services the storage area, the offices, the heat exchanger pump area, the coolant waste tank areas, and the evaporator rooms.

The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 800 scfm. Assuming the concentration of $3x10^{-9}$ microcuries per cubic centimeter exists in this room with the least flow, the radioactive concentration C_A at this point is:

$$C_A = 3 \times 10^{-9} \ \mu Ci/cc \ \frac{800 \ scfm}{15,495 \ scfm}$$
$$= \underline{1.548 \times 10^{-10} \mu Ci/cc}$$

Manufacturer's data for the radiation monitoring equipment states that the monitoring equipment placed in a 1 mrem/hr background of 1 Mev gamma energies produces a count rate of 78 cpm. Concentrations of $1x10^{-11}\mu Ci/cc$ (limiting isotope Cs-137) produces a count rate 107 cpm. Therefore, the count rate for the concentration at point A is equal to 1656 cpm.

Point B (RM-8998 P&ID 25203-26029)

The airflow at this sampling point is 18,335 scfm. The ventilation exhaust system at this point services the sampling room, the letdown heat exchanger room, the coolant waste receiver area, and the volume control tank area.

The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 160 scfm. Assuming the value of $3x10^{-9}\mu Ci/cc$ exists in this area, the radioactive concentration C_B at this point is:

$$C_B = 3 \times 10^{-9} \ \mu Ci/cc \left(\frac{160 \ scfm}{18,335 \ scfm}\right)$$

$$= \underline{2.61 \times 10^{-11} \mu Ci/cc}$$

Assuming the 78 cpm for the background counting rate and the 107 cpm for the concentration $1x10^{-11}$μCi/cc (Cs-137), the count rate for point B is 279 cpm.

Point C (RM-8997 P&ID 25203-26029)

The airflow at this sampling point is 10,170 scfm. The ventilation exhaust system at this point services the charging pump areas, the degasifier areas, and the engineered safety features areas. The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 235 scfm. Assuming the value of $3x10^{-9}$ microcuries per cubic centimeter exists in this area, the radioactive concentration C_C, at this point is:

$$C_C = 3 \times 10^{-9} \ \mu Ci/cc \left(\frac{235 \ scfm}{10,170 \ scfm}\right)$$

$$= \underline{6.93 \times 10^{-11} \mu Ci/cc}$$

Assuming the 78 cpm for the background count rate and the 107 cpm for the concentration 1 x 10^{-11}μCi/cc (Cs-137), the count rate for the concentration at point C is 742 cpm.

Point D (RM-8434 A&B P&ID 25203-26029)

The air flow at this sampling point is 17,920 scfm. The ventilation system at this point services the equipment laydown area, the waste gas decay system areas, the aerated waste gas system area, and the RBCCW heat exchanger area. The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 800 scfm. Assuming the value of $3x10^{-9}$μCi/cc exists in this area, the radioactive concentration, C_D, at this point D is:

$$C_D = 3 \times 10^{-9} \ \mu Ci/cc \ \frac{800 \ scfm}{17,920 \ scfm}$$
$$= \underline{1.34 \times 10^{-10} \mu Ci/cc}$$

Assuming the 78 cpm for the background count rate and 107 cpm for the concentration 1 x 10^{-11}μCi/cc (Cs-137), the count rate for the concentration at point D is 1434 cpm.

The calculations and analysis for the four sampling points shown above indicate a count rate ranging from 279 counts per minute to 1656 counts per minute. Since these count rates are greater than three times the background rate, as specified by the manufacturer, the concentration shown in the above calculations can be detected and alarmed.

The above analysis is conservative in that the value used is based on an exposure to the concentrations specified for forty hours in any period of seven consecutive days. However, since the normal occupancy for these areas is significantly less than this 40-hour period, the exposure of personnel to airborne radioactivity will be considerably less than that stated in the above calculations.

In conclusion, the above analysis clearly demonstrates the capability of the airborne radiation monitoring system to detect and measure appropriate concentrations in the radwaste ventilation exhaust system and thus to comply with the requirements for personnel safety stated in 10 CFR 20.103 and 10 CFR 20.203(d).

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 12: Conduct of Operations

CHAPTER 12—CONDUCT OF OPERATIONS

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CHAPTER 12 – CONDUCT OF OPERATIONS

12.1 ORGANIZATIONAL STRUCTURE

Information regarding the organizational structure is presented in Section 1.0, Organization, of the Quality Assurance Program Description (QAPD) Topical Report (Reference 12.1-1). With the exception given below, that information is incorporated herein by reference.

The owner, holding 100 percent of the Millstone 2 nuclear plant, is Dominion Nuclear Connecticut, Inc.

12.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

Information regarding the management and technical support organization is presented in Section 1.0, Organization, of the QAPD Topical Report (Reference 12.1-1). That information is incorporated herein by reference.

12.1.1.1 Technical Support for Operations

The organization providing technical support for operations is described in Section 1.0 of the QAPD Topical Report (Reference 12.1-1). That information is incorporated herein by reference.

12.1.1.2 Organizational Arrangement

The organizational arrangement is as described in Section 1.0 of the QAPD Topical Report (Reference 12.1-1). That information is incorporated herein by reference.

12.1.2 OPERATING ORGANIZATION

12.1.2.1 Plant Organization

The plant organization is as shown in Reference 12.1-1.

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.

12.1.3 QUALIFICATION OF NUCLEAR PLANT PERSONNEL

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

12.1.4 REFERENCES

12.1-1 Quality Assurance Program Description Topical Report.

12.2 TRAINING PROGRAM

Formal training programs have been established to train and qualify the personnel who operate and maintain the Millstone nuclear units. These programs are structured to fulfill the requirements for training set forth in ACAD 91-015 (Reference 12.2-1). The programs are based on a Systems Approach to Training and are accredited by the National Academy for Nuclear Training. Initial accreditation of these programs was awarded on August 21, 1986, for operator training and on December 15, 1987, for Maintenance and Technical training. These programs are implemented for the following categories of nuclear power plant personnel:

- Nonlicensed Operator
- Reactor Operator
- Senior Reactor Operator
- Shift Manager
- Continuing (Requalification) Training for Licensed Personnel
- Shift Technical Advisor
- Instrument and Control Technician
- Electrical Maintenance Personnel
- Mechanical Maintenance Personnel
- Chemistry Technician
- Radiological Protection Technician
- Engineering Support Personnel

12.2.1 REFERENCES

12.2-1 ACAD 91-015, National Academy for Nuclear Training, “The Objectives and Criteria for Accreditation of Training in the Nuclear Power Industry.”

12.3 EMERGENCY PLANNING

The Staff approved Millstone Nuclear Power Station Emergency Plan (Reference 12.3-1) addresses the criteria set forth in NUREG-0654, FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Revision 1, November 1980 and NUREG-0737, Supplement 1. As such, the NRC approved Emergency Plan provides for an acceptable state of emergency preparedness and meets the requirements of 10 CFR Part 50 and Appendix E thereto.

12.3.1 REFERENCES

12.3-1 J. F. Opeka letter to U. S. Nuclear Regulatory Commission Document Control Desk, transmitting Revision 6 to the Millstone Nuclear Power Station, Unit Numbers 1, 2, and 3 Emergency Plan, dated 11/4/91, and subsequent revisions thereto submitted on an annual basis.

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 Quality Assurance Program Description (QAPD) Topical Report (Reference 12.4-1).

12.4.2 INDEPENDENT REVIEW

Independent review of activities affecting the unit's safety is performed by the Management Safety Review Committee as described in the QAPD Topical Report.

12.4.3 AUDITS

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

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, nuclear industry management assessments, insurance inspections, etc.

12.4.5 REFERENCES

12.4-1 Quality Assurance Program Description Topical Report.

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, 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 Quality Assurance Program Description (QAPD) Topical Report.

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.

Equipment Control Procedures

Instructions are written to specify proper methods of obtaining clearances on plant equipment for maintenance or construction and to specify procedures for control of jumpers, inhibits, and wire removal. The clearance procedure assigns responsibility for clearance issue to the shift supervisor. A licensed operator, after ensuring he is aware of the effect of the activity on the system, is required to authorize all maintenance, tests, and surveillances performed on plant systems. Upon completion of the item, the document is returned to the operator for acceptance or for the purpose of returning the system to service. The administrative procedures which control these evolutions provide the required explicit notification of operational personnel whenever a safety related system is removed from and returned to service. The clearance procedure also contains certain restrictions on the issuance of a clearance. The instructions for control of jumpers, inhibits, and wire removal allow temporary alterations to critical structures, systems, or components to facilitate tests, maintenance, or operations. They specify administrative procedures to be followed in performing such alterations.

Control of Maintenance and Modifications

Administrative control procedures implement the review and approval requirements for maintenance and modifications. These procedures include the control of plant modifications and maintenance on safety-related equipment. These procedures establish a framework of special process and maintenance procedures.

Master Surveillance Testing Schedule

An administrative control procedure establishes a master test control list, implements the surveillance test program, and assigns responsibility for review and approval of surveillance procedures in accordance with Technical Specifications. Written surveillance procedures are controlled as station or unit procedures.

Procedures for Logbook Usage and Control

An administrative control procedure establishes the requirements for logbook usage and control.

Maintenance and Testing of Safety-Related Systems

Procedures for maintenance and testing of safety-related systems specify that prior to the removal of a safety-related system from service, the redundant system is verified operable. For equipment that requires specific surveillance in accordance with Technical Specifications, the surveillance testing is completed prior to removing the system from service.

These procedures are reviewed and improved, if necessary, to ensure operability of safety systems prior to taking credit for the system(s) to satisfy Technical Specification requirements.

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 QAPD Topical Report.

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 abnormal condition, automatic actions that may occur, and the appropriate immediate and subsequent operator actions.

12.5.2.1.4 Emergency Operating Procedures

Emergency operating procedures are prepared for conditions which might possibly lead to injury of plant personnel or the public if the release of radioactivity in excess of established limits occurs. These procedures include symptoms of the emergency conditions, automatic actions that may or should occur, and immediate and subsequent operator actions. All immediate actions are required to be memorized by the operator since the primary responsibility for detection of an emergency and initiation of corrective action rests upon the operator.

12.5.2.2 Station Procedures

Station procedures are written by the chemistry, health physics, security, generation test, building services, stores, nuclear records, computer operations, station services engineering and any other station group. These procedures control the specific activities of these departments in support of unit or station operation (may be common site or unit specific). Station calibration procedures written by the maintenance or instrument departments are also station procedures.

12.5.2.2.1 Radiation Protection Procedures

Radiation Protection procedures support Section 13.5 and 10 CFR 20 requirements.

12.5.2.2.2 Instrument Maintenance Instructions

Instrument maintenance instructions are prepared for the performance of periodic calibration, testing, and channel checking of safety-related plant instrumentation and all instruments used to satisfy Technical Specification requirements. These instructions ensure measurement accuracies adequate to maintain plant safety parameters within operational and safety limits. In addition, instrument maintenance instructions outline the periodic calibration and accuracy requirements of test equipment necessary to support the calibration of safety-related instrumentation.

12.5.2.2.3 Chemistry Procedures

Chemistry procedures are prepared covering the routine analysis and sampling methods to ensure compliance with plant chemistry and discharge limits.

12.5.2.2.4 Radioactive Waste System Procedures

Procedures for operation of radwaste systems are included in system operating procedures.

12.5.2.2.5 Material Control Procedures

This topic is covered by administrative procedures in Section 12.5.1.3.

12.5.2.2.6 Maintenance and Modification Procedures

Maintenance procedures are prepared to cover safety-related work which requires a specific technique or sequence not normally part of an individual's routine skill.

The procedures support the requirements and programs of Section 12.5.1.33 which covers administrative control of maintenance and modification.

12.5.2.2.7 Fire Protection Procedures

The Fire Protection Program is described in Section 9.10.

12.5.2.2.8 Special Procedures

This topic is covered by administrative procedures.

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 as described in the Quality Assurance Program Description Topical Report. These controls include all quality related records including plant and as-built drawings.

12.7 PHYSICAL SECURITY PLANS

The security plan (Reference 12.7-1) states the security measures to be employed by the licensee for the protection of Millstone Units 1, 2, and 3 at the Millstone Nuclear Power Station, Waterford, Connecticut, against radiological sabotage. The plans have been submitted in accordance with 10 CFR Part 73, Section 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage."

These plans include measures to deter or prevent malicious actions that could result in the release of radioactive materials into the environment through sabotage. This protection is provided through the use of armed guards, physical barriers, monitors, personnel access controls, alarms, communications, response to security contingencies, and liaison with appropriate law enforcement agencies.

12.7.1 REFERENCES

12.7-1 J. F. Opeka letter to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station Unit Numbers 1, 2, and 3 Physical Security Plan, Revision 15," dated December 16, 1991, and subsequent revisions thereto.

12.8 QUALITY ASSURANCE PROGRAM

12.8.1 QUALITY ASSURANCE PROGRAM DESCRIPTION (QAPD) TOPICAL REPORT

A comprehensive Quality Assurance Program (QAP) has been developed and implemented to assure conformance with established regulatory requirements, set forth by the Nuclear Regulatory Commission, and accepted industry standards. The participants in the QAP assure that the design, procurement, construction, testing, operation, maintenance, repair, and modification of nuclear power plants are performed in a safe and effective manner.

The QAPD Topical Report complies with the requirements set forth in Appendix B of 10 CFR Part 50, along with applicable sections of the Safety Analysis Report (SAR) for each license application, and is responsive to NUREG-0800, which describes the information presented in the Quality Assurance Section of the SARs for nuclear power plants. The QAPD Topical Report is incorporated herein by reference.

The QAP is also established, maintained, and executed with regard to Radioactive Material Transport Packages as allowed by 10 CFR 71.101(f).

The QAPD Topical Report is submitted periodically to the NRC in accordance with 10 CFR 50.54(a).

12.9 DELETED BY PKG FSC MP2-UCR-2011-014

12.10 RISK INFORMED CATEGORIZATION AND TREATMENT

12.10.1 INTRODUCTION

Millstone has implemented 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-3, and RISC-4 structures, systems, and components (SSCs) using:

- Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; and the shutdown safety assessment process to assess shutdown risk;

- The Arkansas Nuclear One, Unit 2 (AN0-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports;

- The results of non-PRA evaluations that are based on the IPEEE [Individual Plant Examination of External Events] Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ ANS PRA Standard RA-Sa-2009.

12.10.2 SSC CATEGORIZATION

As outlined in the Safety Evaluation Report (SER) for License Amendment 337 to Renewed Facility Operating License No. DPR-65 (Reference 12.10-3), Millstone 2 will use the methodology described in Section 12.10.

It should be noted that 10 CFR 50.69 does not replace the existing “safety-related” and “nonsafety-related” classification. Instead 10 CFR 50.69 divides these classifications into two subcategories based on high or low safety significance, such that there are four categories of risk-informed safety class (RISC), as shown below:

RISC-1: safety-related SSCs that perform (high) safety significant functions.

RISC-2: nonsafety-related SSCs that perform (high) safety significant functions.

RISC-3: safety-related SSCs that perform low safety-significant functions.

RISC-4: nonsafety-related SSCs that perform low safety-significant functions

10 CRF 50.69(f)(2) requires updating the UFSAR to reflect which systems have been categorized. The list below is revised as part of the periodic UFSAR update to reflect systems that have been categorized.

System	Name
Radiation Monitoring (2404)	

12.10.3 SSC TREATMENT

12.10.3.1 Treatment of Component Categories

The programs or processes that implement the special treatment requirements are revised to recognize that the special treatments no longer apply to RISC-3 and RISC-4 SSCs. The programs or processes either allow continued application of the special treatments or acceptable alternative treatments, as applicable, to provide reasonable confidence that these SSCs would perform their safety-related function under design basis conditions.

For those components that are categorized as Low Safety Significant, 10 CFR_50.69(b)(1) allows compliance with alternative requirements in lieu of the following special treatment requirements:

1. 10 CFR Part 21

2. The portion of the 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR Part 50

3. 10 CFR 50.49

4. 10 CFR 50.55(e)

5. The in-service testing requirements in 10 CFR 50.55a(f): the in-service inspection and repair, and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279 and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h)

6. 10 CFR 50.65, except for paragraph (a)(4)

7. 10 CFR 50.72

8. 10 CFR 50.73

9. Appendix B to 10 CFR Part 50

10. The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, for penetrations and valves meeting the following criteria:

a. Containment penetrations that are either 1-in. nominal size or less, or continuously pressurized.

b. Containment isolation valves that meet one or more of the following criteria:

1. The valve is required to be open under accident conditions to prevent or mitigate core damage events;

2. The valve is normally closed and in a physically closed, water-filled system;

3. The valves is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and is not connected to the reactor coolant pressure boundary; or

4. The valve is 1-in. nominal size or less.

11. Appendix A to Part 100, Sections VI(a)(1) and VI(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake.

When applying alternative treatment, 10 CFR 59.69 requires that the licensee "shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life."

Performance monitoring is being performed on all RISC-1, RISC-2, RISC-3, and RISC-4 SSCs and the station adjusts, as necessary, to either the categorization or treatment process so that the categorization process and results are maintained valid.

12.10.3.2 Enhanced Treatment of RISC-2 SSCs

The 10 CFR 50.69 procedures and 10 CFR 50.69(d)(1) require that RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

12.10.4 REFERENCES

12.10-1 NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1, dated June 1999.

12.10-2 Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," dated September 1999.

12.10-3 Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 337 to Renewed Facility Operating License No. DPR-65 Dominion Energy Nuclear Connecticut, Inc. Millstone Power Station, Unit No. 2, Docket No. 50-336.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 13: Initial Tests and Operation

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CHAPTER 13 – INITIAL TESTS AND OPERATION

13.1 DESCRIPTION OF THE STARTUP TEST PROGRAM

13.1.1 SCOPE

The Millstone Unit 2 Startup Test Program consisted of all testing activities commencing at the completion of the construction phase and ending with the completion and acceptance of all test data. The Startup Test Program was divided into the following test phases.

Phase I: Initial Inspection and Component Testing

Phase II: Preoperational and Acceptance Testing

Phase III: Pre-Core Hot Functional Testing

Phase IV: Initial Fuel Loading

Phase V: Post-Core Hot Functional Testing

Phase VI: Reactor Initial Criticality

Phase VII: Low Power Physics Testing

Phase VIII: Power Ascension Testing

Each of the above testing phases will be discussed in detail in succeeding sections. The Initial Inspection and Component Testing phase began as the various system construction was completed. This phase was followed by the preoperational, acceptance and pre-core hot functional test phases. After fuel load, the final 4 phases of testing were completed by May 1976.

13.1.2 OBJECTIVES

The Startup Test Program demonstrated that components and systems operate and function in accordance with the FSAR and system descriptions. The test results confirmed that the plant can be safely operated and that performance levels can be maintained in accordance with the safety requirements established by the Final Safety Analysis Report. The program results established baseline performance data, and served as verification that normal plant operating and plant emergency procedures accomplish the purposes for which they were prepared.

13.2 STARTUP TEST PROGRAM ORGANIZATION AND RESPONSIBILITIES

13.2.1 PLANT ORGANIZATION

The Millstone Point Company was responsible for supervising, directing, and ensuring that all phases of plant testing are accomplished in accordance with established criteria. The Plant Superintendent had the responsibility for ensuring that all phases of the Startup Test Program were properly accomplished and accepted. The Startup Group and Plant Staff was responsible to the Plant Superintendent for planning and accomplishment of the Startup Test Program. The organization of the Plant Staff including Millstone Startup personnel is shown in Figure 13.2–1. A detailed description of the Startup Group is provided in the following section.

13.2.2 STARTUP TEST GROUP

13.2.2.1 Organization

The Startup Group was organized as shown on Figure 13.2–2. The Joint Test Group, composed of the Assistant Plant Superintendent, Combustion Engineering Site Manager, and the Bechtel Supervising Startup Engineer, administratively controlled the conduct of the Startup Test Program. The plant operating staff conducted all testing operations under the direction of the Millstone Startup Engineers. The Millstone Lead Startup Engineer coordinated the entire Startup Test Program, working with the Joint Test Group and directing the activities of the Millstone Startup Engineers.

During the equipment checkout and preoperational tests phases of the Startup Test Program, Millstone Startup Engineers were on shift only if required for the performance of a particular pre-operational/acceptance test. The Millstone Startup Engineer normally acted as the Test Supervisor. Bechtel Startup Engineers were on shift during this time if a need for their particular expertise exists as determined by the Lead Startup Engineer. Combustion Engineering Startup Engineers were on site whenever CE supplied equipment was being tested. This includes a commitment for full shift coverage if necessary.

During hot functional testing, initial fuel loading, initial criticality, low power physics testing and power ascension testing, Millstone Startup Engineers normally provided full shift coverage directing the test program. During long periods of normal operation using fully tested systems, the JTG elected to temporarily suspend full coverage by Millstone Startup Engineers. Bechtel Startup Engineers were assigned to full shift coverage as necessary by the Millstone Lead Startup Engineer.

Combustion Engineering provided full coverage as requested by the Lead Startup Engineer during periods of changing core reactivity or testing NSSS components.

13.2.2.2 Duties and Responsibilities

a. Joint Test Group

1. Coordinate the Startup Test Program including formulating, reviewing and approving the schedule of startup testing events.

2. Shall meet as required to ensure the safety and proper sequencing of evolutions of the Startup Test Program.

3. Shall ensure that the Startup Test Program is conducted in accordance with approved procedures and the Millstone Startup Manual.

4. Shall ensure that each organization's engineering requirements are satisfied.

5. Shall approve and release for execution and authorize changes to startup test procedures except for Phase I.

6. Shall review and approve the results of tests (Phase II and subsequent) performed under the Startup Test Program.

7. Shall recommend the disposition of deficiencies.

8. Shall direct the return of systems or portions thereof to Bechtel Construction for correction of deficiencies.

9. Shall recommend modifications to system design, equipment, or operating procedures to the Plant Superintendent, based on the evaluation of the Startup Test Program.

10. Shall recommend acceptance of the satisfactorily tested systems by the Plant Superintendent.

b. Assistant Plant Superintendent

1. Shall serve as Chairman of the Joint Test Group, or may designate a member of the Plant Staff to act as Chairman in his absence.

2. Shall be responsible to the Plant Superintendent for the execution of the Startup Program in accordance with the Startup Manual.

c. Combustion Engineering Startup Test Organization

The C-E Startup Test Group shall consist of a Site Manager, a Chief Test Engineer, and three Startup Engineers. In addition, design engineers will be temporarily assigned to the site to supplement the C-E Startup Group.

The group shall provide technical advice and consultation on all matters relating to the design, operation, and testing of C-E supplied systems and equipment.

Accordingly, the C-E Startup personnel will:

1. Monitor the conduct of preoperational, hot functional, fuel loading, criticality, low power physics testing and power testing of C-E supplied equipment.

2. Shall provide site evaluation of test results.

3. Shall coordinate resolution of problem areas acting as a liaison between the site and home office design groups.

4. Will arrange for onsite vendor representation as required.

5. The C-E Site Manager or his authorized representative shall be a member of the Joint Test Group and as such shall review and approve all startup Test procedures and changes thereto.

6. Minimum qualifications for this group are included in Section 13.2.2.3.

d. Bechtel Startup Test Organization

The Bechtel Startup Test Group shall consist of a Supervising Startup Engineer and three Startup Engineers (i.e., Mechanical, Electrical, and Instrumentation and Control).

The group shall provide technical advice and consultation on matters relating to the design, operations, and testing of systems and equipment.

Accordingly, the Bechtel Startup personnel will:

1. Monitor the conduct of preoperational, hot functional and power testing of systems and equipment.

2. Shall provide site evaluation of test results.

3. Shall coordinate resolutions of problem area acting as a liaison between the site and home office design groups.

4. Will arrange for onsite vendor representation as required.

5. The Lead Supervising Startup Engineer or his authorized representative shall be a member of the Joint Test Group and as such shall review and approve all startup test procedures and changes thereto.

6. Minimum qualifications for this group are included in Section 13.2.2.3.

e. Millstone Lead Startup Engineer

1. Shall attend meetings of the Joint Test Group and may, if so designated, act as Chairman of the JTG in the absence of the Assistant Plant Superintendent.

2. Shall be responsible to the JTG for implementation of the Startup Test program.

3. Shall coordinate the preparation and review of all procedures required for the Startup Test Program.

4. Shall, as directed by the JTG, coordinate the transfer of systems, or portions thereof, from Bechtel Construction to Millstone Point Company for testing.

5. Shall coordinate, as appropriate, the flushing and hydrostatic testing of systems.

6. Shall coordinate the testing functions of Millstone Maintenance, Instrument, and Operations Departments.

7. Shall ensure timely completion of test prerequisites and notification to the JTG of system readiness to test.

8. Shall ensure that sufficient personnel and equipment are available to execute each test of the Startup Test Program.

9. Shall submit test results to the JTG for evaluation.

10. Shall coordinate resolution of design, construction, or testing deficiencies.

11. Shall, when directed by the JTG, supervise the transfer of systems or portion thereof to Bechtel Construction for further work or correction or deficiencies.

12. Shall approve for use Phase I Test Procedures.

f. Millstone Startup Engineers

1. Shall assist the Lead Startup Engineer in the performance of his duties.

2. Shall prepare test procedures.

3. Shall supervise the performance of test, ensuring that the tests are properly conducted in accordance with the test procedure, applicable operating procedures, and the Startup Manual.

13.2.2.3 Qualifications

Minimum qualifications of the key personnel comprising the Millstone Startup Group are as follows:

a. Millstone Lead Startup Engineer

1. At least two years of formal college level academic training or equivalent.

2. At least six years of power plant-related work experience including not less than two years directly involved in the operation, engineering, and/or maintenance disciplines.

3. At least three years of training or experience with nuclear power plants or equivalent.

4. Senior Operator’s License.

b. Millstone Startup Engineers

1. At least two years of formal college level academic training or equivalent.

2. At least six years of power plant related work experience including not less than two years directly involved in the operation, engineering, and/or maintenance disciplines.

3. At least two years of training or experience, with nuclear power plants or equivalent.

4. Senior Operator’s License or equivalent knowledge and background.

c. Combustion Engineering Site Manager

1. B.S. Engineering Nuclear Power Training (Military, ORSORT or equivalent).

2. Ten years’ nuclear reactor operations, testing and training.

3. SRO or equivalent on power reactor (military, commercial, or test reactor).

d. Combustion Engineering Chief Test Engineer

1. M.S. or B.S. Nuclear Engineering with equivalent experience.

2. Ten years' reactor analysis, design, operation and testing. Previous experience in installation and startup testing, power and research reactor.

3. SRO or equivalent on power reactor (military, commercial or test reactor).

e. Combustion Engineering Startup Engineer

1. B.S. Engineering Nuclear Power Training (Military, ORSORT or equivalent).

2. Five years' nuclear reactor operation testing or training.

3. SRO or equivalent on power reactor (military, commercial or test reactor).

f. Bechtel Supervising Startup Engineer

1. Will have previously held a similar capacity of Supervising Startup Engineer on another nuclear plant, or been a group leader (Electrical, Mechanical, I&C) on another nuclear plant.

2. Will be familiar with AEC requirements for the pre-operational test program, including documentation, quality control and quality assurance.

3. Will be capable of planning and coordinating the entire plant startup program.

g. Bechtel Startup Engineers

1. Will have held a similar position of Startup Engineer on another nuclear plant.

2. Will be familiar with AEC requirements for pre-operational test program, including documentation, quality control and quality assurance.

3. Will be capable of planning and coordinating the startup activities within his discipline (Electrical, Mechanical, and I&C).

FIGURE 13.2–1 MILLSTONE NUCLEAR POWER STATION ORGANIZATION DIAGRAM DURING INITIAL STARTUP

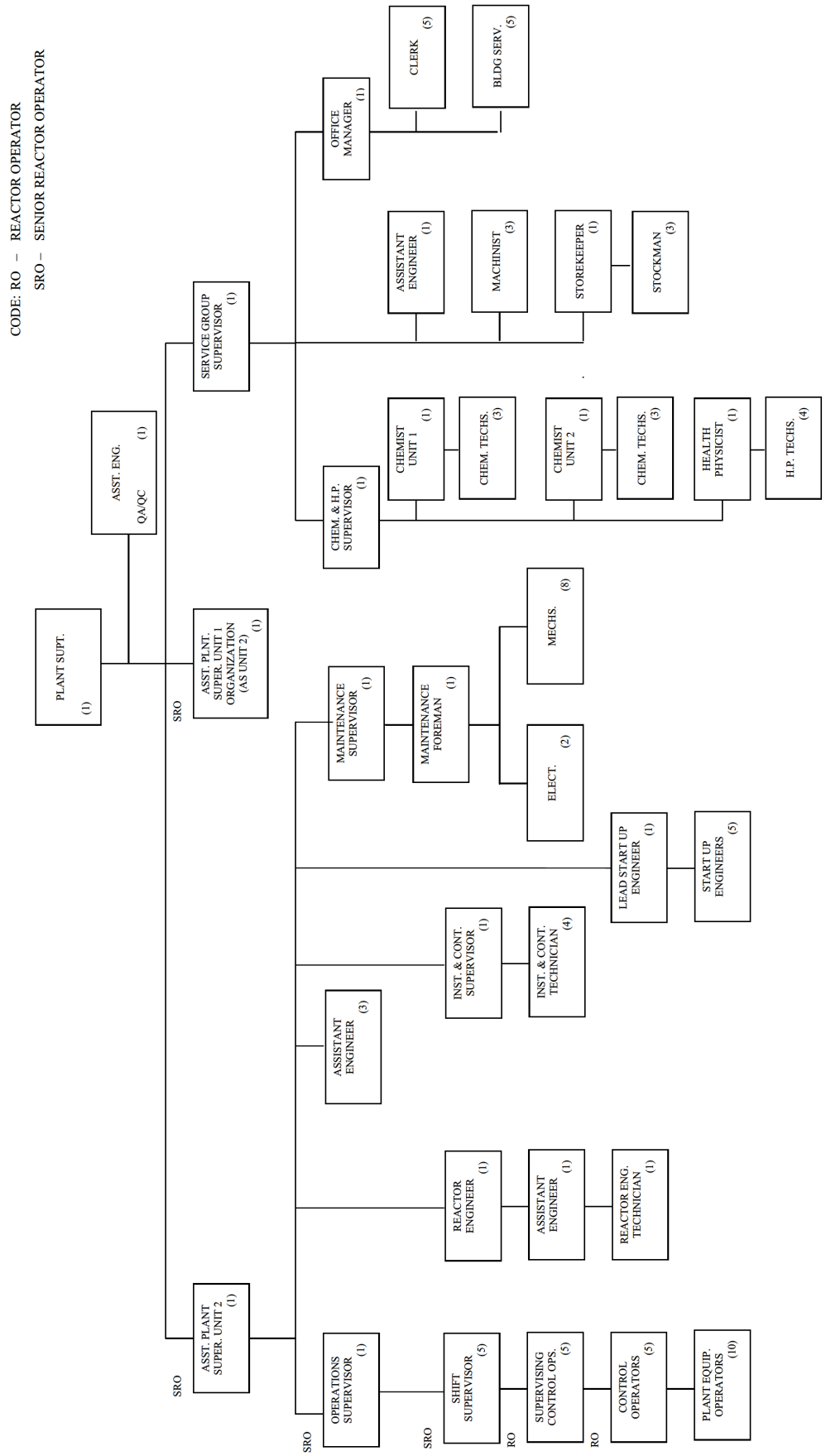
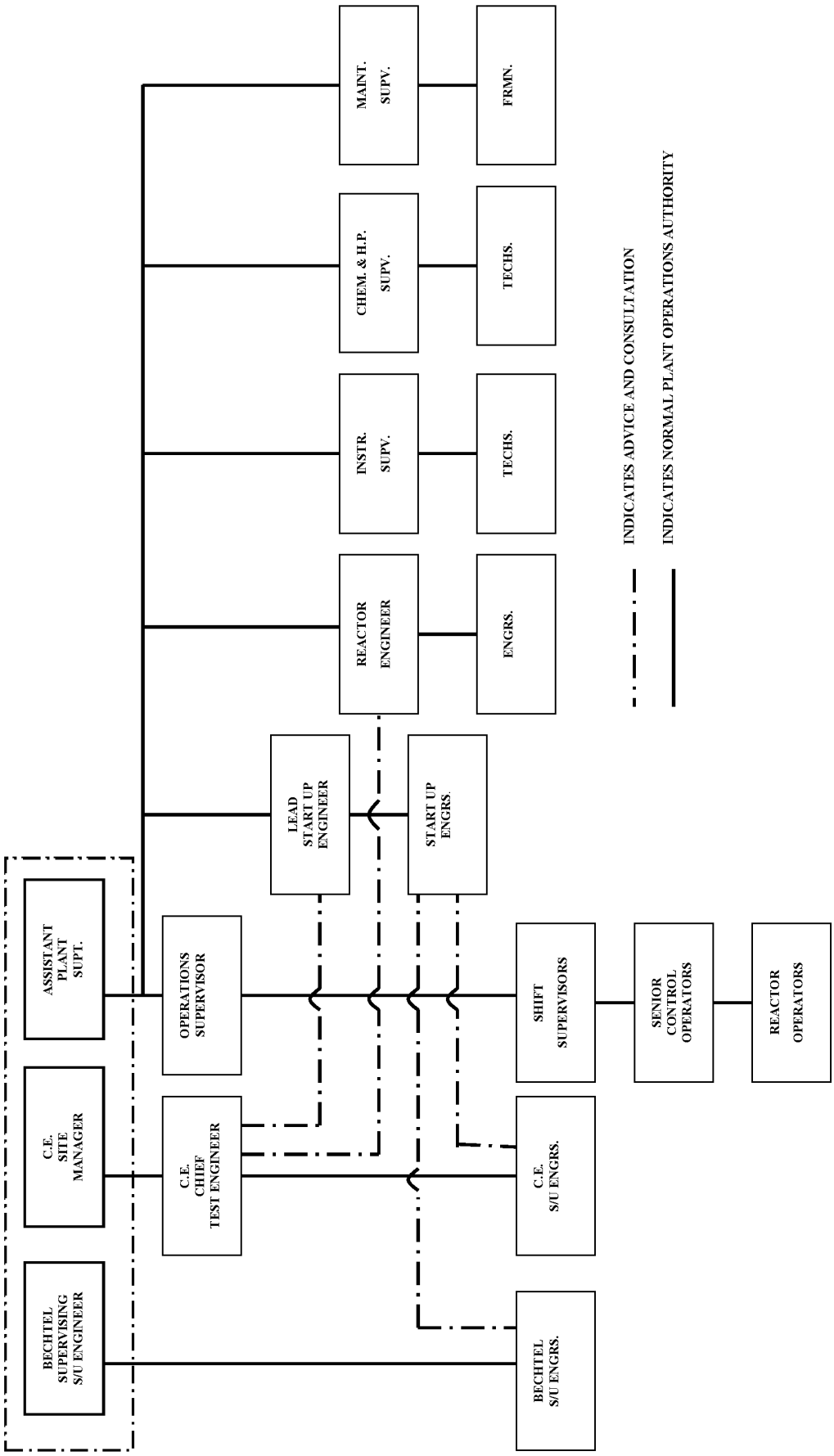


FIGURE 13.2–2 STARTUP GROUP ORGANIZATION



13.3 ADMINISTRATIVE CONDUCT OF THE STARTUP TEST PROGRAM

13.3.1 TEST PROCEDURES - DISCUSSION

Test procedures were prepared by the plant staff using test guidelines provided by Bechtel or C-E. These tests were reviewed by NUSCO, Bechtel and C-E. Additionally, other members of the plant staff were assigned to review test procedures. With the exception of Phase I test procedures, all procedures were reviewed and approved by the JTG. The tests were performed by Millstone personnel, under the direction of a Millstone Startup Engineer. All procedures were adhered to with deviations permitted only through administrative contracts.

Test data was recorded and evaluated. Test deficiencies were resolved and the test accepted before the test was considered complete by the JTG (for Phase II and subsequent tests).

All tests and associated documents will be retained for the life of the plant.

13.3.2 TEST PROCEDURE PREPARATION

Detailed test procedures were prepared by the plant staff using test guidelines provided by Bechtel or Combustion Engineering, for each specific test performed during the various phases of the Startup Test Program. Each test procedure contained the following major sections:

Test Objective

Acceptance Criteria

References

Prerequisites

Initial Conditions

Special Precautions

Procedure Steps

System Restoration

Upon completion of preparation of a test procedure, the procedure was processed through a review circuit. The Millstone Lead Startup Engineer coordinated the review of all procedures.

As appropriate for a particular procedure, the document was reviewed by some or all of the following:

Combustion Engineering

Bechtel Startup Group

Northeast Utilities Service Company

Millstone Instrument Department

Millstone Chemistry/Health Physics Department

Millstone Reactor Engineering Department

Upon completion of these reviews, any necessary changes were incorporated by the originator. Test procedures for Phase I of the Startup Test Program were approved for use by the Millstone Lead Startup Engineer. All other test procedures were forwarded to the JTG for approval.

13.3.3 TEST PROCEDURE EXECUTION AND CHANGES THERETO

Once the test procedure had been approved, the Millstone Lead Startup Engineer coordinated the completion of prerequisites. When all test prerequisites were complete and any allowable exceptions noted, the test was returned to the Millstone Lead Startup Engineer who discussed any prerequisite exceptions with the JTG and arranged for the test to be scheduled.

When authorized for performance, each test was performed in conformance with the test procedure and authorized changes. All test data was recorded on data blanks within the procedure or on specially prepared data sheets. When the test was completed, the Millstone Startup Engineer reviewed the test data against the stated acceptance criteria and resolved discrepancies in accordance with the Startup Manual.

At all times the shift supervisor was responsible for the safe and proper operation of plant equipment and systems and for the safety of plant personnel. In the fulfillment of this responsibility he took whatever action was necessary including but not limited to stopping any test and placing plant equipment in a safe condition.

Once a procedure had been approved, all procedure changes were made using the Test Procedure Change Form. This form provided a means for making on-the-spot changes providing the change did not alter the intent of the procedure or endanger plant personnel or equipment. In the case of a Phase I test procedure only, the shift supervisor and test Supervisor provided joint change authorization. Changes for all other phases were jointly authorized by the Shift Supervisor and the Millstone Startup Engineer. Concurrence of the C-E Startup Engineer and/or Bechtel Startup Engineer as appropriate was obtained prior to making the change. The change was then confirmed by the JTG by the end of one working week. JTG approval was not required for changes to Phase I test procedures.

13.3.4 EVALUATING TEST PROCEDURE DATA

All tests were evaluated by a minimum of one individual not a member of the department performing the test. This would normally be done by NUSCO, Bechtel, or C-E. Additionally, other members of the Millstone Plant Staff were also requested to evaluate the completed test. This evaluation included determining that all deficiencies had been identified. The evaluation of all test procedures was coordinated by the Millstone Lead Startup Engineer.

Once the test had been evaluated, test deficiencies resolved, and no additional retesting required, the Millstone Lead Startup Engineer signed for test acceptance on all Phase I tests.

For all other tests, the JTG reviewed the test results and, when satisfied, signed for test acceptance.

13.3.5 TEST RESULT DOCUMENTATION

As a test procedure was being performed, test data was recorded on data blanks within the body of the procedure or on provided data sheets. Included were the expected or desired value of the parameters.

The execution copy of all test procedures, records, startup forms, recorder tracings, and photographs are the sole property of the Millstone Point Company and will be retained for the life of the plant.

13.4 MILLSTONE STARTUP GROUP TRAINING PROGRAM

An intensive training program was conducted to prepare the Millstone Unit 2 Startup Engineers, Staff Supervisors, Supervising Control Operators, Reactor Operators, Plant Equipment Operators and certain members of the Millstone Plant Staff for their responsibilities in conducting the Unit 2 Startup Test Program. The details of the training program are given in Chapter 12.

13.5 INITIAL INSPECTION AND COMPONENT TESTING PHASE

During this phase of the Startup Test Program, each system component was individually tested for conformance with acceptance criteria. Typical items to be performed during this phase by the maintenance or instrumentation departments were phase rotation checks, megger checks, alignment checks, etc. These checks were performed in accordance with prepared test procedures. Additionally, the Operations Department directed the performance of flushes and hydrostatic tests. These operations were performed in accordance with prepared test procedures and served as some of the prerequisites for the preoperational/acceptance test phase of the Startup Test Program.

13.6 PREOPERATIONAL AND ACCEPTANCE TEST PROGRAM

13.6.1 DESCRIPTION OF PREOPERATIONAL AND ACCEPTANCE TEST PROGRAM

The comprehensive preoperational and acceptance testing phase were performed to ensure that all equipment and systems were capable of performance in accordance with their acceptance criteria, and to establish initial baseline performance data for the equipment and systems. The preoperational and acceptance test phase in general included adjustments, calibrations, determination of pump head characteristics and overall system capability under ambient conditions. The tests involved actual operations of the systems and equipment where possible. All tests were performed in accordance with written procedure. Administrative control of the preoperational test procedures is described in Section 13.3 of this chapter.

Testing operations were conducted under the direction of the personnel described in Section 13.2.2.

13.6.2 INDEX OF PREOPERATIONAL TESTS AND GENERAL TESTING SEQUENCE

Listed below, in an appropriate sequence of performance, are the systems which were preoperationally tested.

Communications System

125 Volt DC System

Reserve Station Service Transformer

Primary Water Treatment System

4,160 Volt Electrical System

480 Volt Load Centers

Service Water System

Clean Liquid Radwaste System

Aerated Liquid Radwaste System

480 Volt Motor Control Centers

Turbine Building Closed Cooling Water System

Fire Protection System

Recovered Boric Acid System

Instrument Air System

Primary Water Storage System

120 Volt Nonvital Instrument AC System

120 Volt Vital Instrument System

Gaseous Radwaste System

Condensate Storage and Transfer System

Auxiliary and Emergency Feedwater System

Steam Generator Feedwater System

Condensate System

Feedwater Heater Drain and Vents

Main Steam (Bypass and Steam Dump Valves)

Sampling System

Spent Fuel Cooling and Purification

Reactor Building Closed Cooling Water System

Fuel Handling Systems

Safety Injection Systems (High-Pressure, Low-Pressure, and Safety Injection Tanks)

Diesel Fuel Oil System

Diesel Generators

Containment Spray System

Shutdown Cooling System

Engineered Safeguards Actuation System

Reactor Coolant System

Reactor Regulating System

Reactor Protection System

Chemical Volume Control and Purification System

Nuclear Instrumentation Systems

In-Core Instrumentation System

Process Radiation Monitoring Systems

Area Radiation Monitoring Systems

Boronometer and Process Radiation Monitor

Main and Normal Station Service Transformer

Enclosure Building Filtration System

Control Element Assemblies - CEDS BOX SIMULATOR

Solid Radwaste

Containment Building Integrated Leak Test

Containment Building Structural Integrity Test

Containment Penetration Cooling System

Containment and Enclosure Building Purge System

Control Element Drive Mechanism Cooling System

Containment Air Recirculation System

Control Room Air Conditioning System

Nonradioactive Ventilation System

Diesel Generator Ventilation System

Main Exhaust System

Fuel Handling Area Ventilation System

Radwaste Ventilation System

Primary Chemical Addition System

Access Control Area Air Conditioning System

Engineered Safety Features Room (Air Recirculation and Cooling) Ventilation System

Vital Electrical Switchgear (Emergency Cooling) Ventilation System

Containment Post-Incident Hydrogen Control

Preoperational test descriptions are given in Appendix 13.A.

13.6.3 INDEX OF ACCEPTANCE TESTS AND GENERAL TESTING SEQUENCE

Listed below, in an approximate sequence of performance, are the systems which were acceptance tested:

Domestic Water System

345 kV Switchyard

Plant Computer

Screen Wash System

Chlorine System

Auxiliary Boiler Fuel Oil System

Plant Heating and Condensate Recovery System

Condenser Air Removal System

Water Box Primary System

Condenser Air Removal Exhaust System

Intake Structure Ventilation System

Cathodic Protection System

Main Circulating Water System

Station Air System

Extraction Steam System

Secondary Chemical Addition System

6,900 Volt Electrical System

Auxiliary Steam Reboiler and Deaerating Feedwater System

Reheat Steam and Moisture Separator System

Warehouse Ventilation System

Containment Auxiliary Circulation System

Turbine Building Heating and Ventilation System

Turbine Seal Oil System

Turbine Lube Oil System

Exciter Air Cooler System

Main Turbine and Controls

Hydrogen System

Nitrogen System

Generator Hydrogen Cooler System

Carbon Dioxide and Hydrogen System

Main Generator

Turbine Gland Seal and Drain System

Stator Liquid Cooling System

Station Lighting and Grounding

13.7 HOT FUNCTIONAL TEST PROGRAM

13.7.1 DESCRIPTION OF HOT FUNCTIONAL TEST PROGRAM

The integrated pre-core hot functional test phase of the Startup Test Program was performed on major plant systems prior to initial fuel loading with the reactor coolant system at operating temperature and pressure. The pre-core hot functional test heated the primary plant from ambient to normal operating temperature and pressure, stopping at specific temperature plateaus and performing specified evaluations (i.e., pressurizer bubble, coolant flow and vibration checks, primary plant chemistry analysis, bypass and dump valve operation, etc.).

The post-core hot functional test phase was conducted upon the completion of the initial core loading operations. This test essentially consisted of CEDM coupling, CEA position checks, CEA drop tests and loss of flow and flow coast down tests.

13.7.2 REACTOR COOLANT SYSTEM HOT FUNCTIONAL TEST (PRE-CORE) DESCRIPTION

a. Test Objective

1. To demonstrate the proper performance of the NSSS under normal operating conditions of temperature, pressure, and flow prior to performing the initial core loading.

2. To provide chemical passivity of the RCS piping prior to reactor operation.

b. General Prerequisites

1. All pre-operational test procedures have been completed and discrepancies corrected or waived.

2. The reactor vessel internals have been installed with the exception of the fuel.

3. Temporary instrumentation to measure reactor coolant pump and reactor vessel differential pressures during flow testing is installed and has been calibrated.

4. Prestartup system checkoff lists have been completed.

5. The RCS, SIS, CVCS, RBCCWS, and all other associated systems have been lined up for normal operation in accordance with the applicable P&IDs.

6. A suitable source for heat rejection from the RCS is available.

7. All systems including the RCS, the SIS, and the CVCS have been filled with demineralized water in accordance with the water chemistry

specifications for cold shutdown with the exception that the boron concentration is zero ppm.

c. Test Methods

1. Using normal operating procedures bring the RCS to a temperature of 260°F and a pressure of 415 psia with a bubble in the pressurizer, normal zero-power pressurizer level, and two reactor coolant pumps operating.

2. With the RCS at 260°F and 415 psia, perform a test data record, measure RCS flow under various pump combinations, perform a calibration check of the process instrumentation, measure the RCS component expansion, and measure the CEDM temperatures.

3. Perform RCS heatup and pressurization using the same pressure and temperature steps that will be used during the Post-Core Hot Functional Tests and Low Power Physics Testing. The steps will involve heatup from 260°F, 415 psia to 360°F, 415 psia; pressurization from 415 psia to 1100 psia to 2250 psia, 460°F, and heatup from 460°F to 532°F, 2250 psia.

4. During the heatup and pressurization the heatup rate will be determined, the pressurizer heatup rate will be measured, calibration checks will be conducted on temperature detectors, CEDM temperatures will be monitored, RCS components expansion will be measured, test data records will be completed, and pressure detector calibrations will be checked.

5. At 532°F, 2250 psia test data records will be taken, reactor coolant flow measurements will be performed, reactor coolant heat loss measurements will be made, CEDM temperatures will be monitored, RCS component expansion will be measured, process instrumentation will have their calibrations checked.

6. The reactor will be cooled down in accordance with normal plant operating procedures to 260°F, 415 psia.

7. The RCS will be cooled down in accordance with normal operating procedures to approximately 130°F and all systems will be prepared for initial fuel loading.

d. Acceptance Criteria

The NSSS performs in accordance with the FSAR description, the various system descriptions, appropriate vendor manuals, and Technical Specifications.

13.7.3 REACTOR COOLANT SYSTEM HOT FUNCTIONAL TEST (POST-CORE) - DESCRIPTION

a. Test Objective

1. To demonstrate operation of the NSSS with fuel prior to performing the initial criticality.

2. To measure the RCS heatup rates, heat losses, and system flows.

3. To demonstrate the proper functioning of the CEDMs and the CEAs.

b. General Prerequisites

1. Initial core loading has been completed.

2. Reactor vessel internals and head have been installed with all appropriate wiring connections made.

3. Prestartup system checkoff lists have been completed.

4. All systems are ready for normal operation.

c. Test Methods

1. Using standard operating procedures, heatup the plant to 260°F, 415 psia with a bubble in the pressurizer, normal zero-power pressurizer level, and two or three reactor coolant pumps operating.

2. With the RCS at 260°F, 415 psia conduct a test data record, and perform the cold CEA performance test. Cold rod drop times will be performed once on each CEA with no reactor coolant pumps operating.

3. Increase RCS temperature and pressure using the same steps as used in NH-RC-2. During these heatups and pressurizations, remeasure the RCS heatup rate and the pressurizer heatup rate.

4. With the RCS at 532°F, 2250 psia, conduct a test data record, measure the RCS flows under various two-, three- and four-pump combinations, perform flow coast down measurements, determine the RCS heat loss, perform the hot CEA performance test, and conduct boron dilution and boration tests. The performance tests include determining CEA drop times once on each CEA with full RCS flow plus additional measurement of the fastest and slowest CEA’s ten times each at full flow.

5. Prepare for performing initial criticality.

d. Acceptance Criteria

The RCS flows, the CEA performance, the RCS heat losses, etc., are in agreement with the FSAR description, the appropriate system descriptions, the appropriate vendor manuals, and the Technical Specifications.

13.8 INITIAL FUEL LOADING

The Millstone Startup Group directed the initial fuel loading operations with the responsibility for the operation resting with the Plant Superintendent. The Combustion Engineering Site Manager and Chief Test Engineer or their representatives provided technical advice as required during the fuel loading operations.

13.8.1 THE PURPOSES OF THE INITIAL FUEL LOADING PROCEDURE ARE:

a. To provide a safe, organized method of accomplishing the initial fuel loading for the Millstone Unit 2 reactors. General procedures and supplemental detailed procedures will be provided to perform the following:

   1. Loading of 217 fuel assemblies.

   2. Insertion of 81 CEA's.

   3. Insertion of neutron source assemblies.

b. To provide a fuel loading sequence which will result in early coupling of the loaded portion of the core and the reactor plant out-of-core nuclear instrumentation system.

   Fuel loading shall be performed in accordance with written procedure. After completion of each step of the procedure, approval must be obtained from the Millstone Startup Engineer before the next procedural step will be initiated. Administrative control of the fuel loading procedures shall be as described in Section 13.3 of this chapter.

13.8.2 INITIAL CONDITIONS WHICH WILL BE ESTABLISHED PRIOR TO THE INITIAL FUEL LOADING ARE:

a. Pre-core loading briefing for operating and supervisory personnel has been conducted.

b. The reactor protective system calibration tests, preoperational tests, and hot functional tests have been satisfactorily completed as far as possible with all known discrepancies documented and corrected or waived for the purpose of proceeding with the initial core loading;

c. The reactor vessel head and upper guide structure are removed. The reactor coolant system is filled to a specified level with refueling concentration borated water. Valve check lists are completed to minimize the possibility of an inadvertent change in boron concentration. Representative samples of system coolant will be withdrawn periodically and analyzed for boron concentration during the loading operation;

d. Temporary neutron detectors will be installed in the core and final calibration and alignment completed. Readouts will be available in the loading area. A communications link will be established between the control room, the refueling pool area, and the spent fuel pool area;

e. A start-up neutron source will be installed in the core with the first fuel assembly and verification of the neutron level monitoring instrumentation made. Equipment will be provided to conduct inverse count rate monitoring of the core during loading;

f. A tag board will be provided in the control room for identifying the location of all movable core components. These include sources, neutron detectors, fuel assemblies and control element assemblies;

g. The reactor coolant system, chemical and volume control system, and auxiliary systems are lined up for fueling operations in accordance with plant operating procedures; and

h. Continuous area radiation monitoring in the spent fuel pool handling and refueling pool areas is maintained.

13.8.3 PRECAUTIONS WHICH SHALL BE OBSERVED DURING THE INITIAL FUEL LOADING ARE:

a. Continuous monitoring of the temporary neutron instrumentation will be conducted and operations halted if any unusual increase in count rate is observed, or if the addition of two consecutive fuel assemblies causes the inverse count rate plot to vary significantly from the expected behavior.

b. A fueling monitor will be assigned to the refueling pool area to observe and verify all fuel handling operations. His sole responsibility will be the verification of serial numbers and core locations of fuel assemblies and CEA's. He shall not perform any other task during the fueling operation. He will verify the fuel assembly serial number prior to its removal from the reactor side transfer machine. Core location and assembly serial number orientation will be verified and recorded once the bundle is placed in the core. CEA serial numbers will be verified upon insertion into the proper fuel assembly and their installed location will be recorded;

c. Reactor coolant boron concentration shall be maintained at 1770 ± 50 ppm. Boron concentration of the reactor coolant in the reactor vessel will be verified periodically throughout the fuel loading process, and the results reviewed by the Millstone Startup Engineer. Water level in the reactor coolant system will be monitored and recorded periodically.

d. Whenever makeup water is added to the reactor coolant system the refueling water boron concentration must be verified at the refueling concentration of 1770 ± 50 ppm boron;

e. Water level in the reactor vessel during loading will be maintained at a specified distance from the reactor vessel flange. If any unexplained increase or decrease in water level occurs the fueling operation will be stopped until the cause for the change in water level has been determined and the boron concentration verified;

f. Extreme care will be taken throughout the fuel loading operation to maintain accurate records of fuel assembly and CEA locations;

g. An overall fuel and CEA location map, including the exact status of each fuel assembly and CEA, will be maintained in the refueling pool area and the control room. A step-by-step record of the fueling procedure will be maintained in each area;

h. The fueling sequence will be strictly followed; and

i. Core loading operations will be stopped if communications are lost between the control room and the refueling pool area.

13.8.4 THE FUEL LOADING PROCEDURES SHALL INCLUDE THE FOLLOWING CONSIDERATIONS:

a. The refueling machine will be operated in accordance with plant operating procedures;

b. A record of the time and date as a given step is completed will be written on the appropriate procedure sheet;

c. As each step is completed the Shift Supervisor must initial for the step. His signature must appear at the bottom of each page of this procedure. The Shift Supervisor must be informed prior to the performance of each step of the procedure and at the completion of each step;

d. Count rate of each wide range channel and temporary channel will be recorded on an appropriate data sheet;

e. 1/m (Inverse count rate multiplication) values will be calculated, plotted, and evaluated after each loading step;

f. The count rate of the temporary neutron detectors must be normalized after each move of the detector. Each detector move must be documented on the plotting sheet to account for possible discontinuities of the plot;

g. When fuel loading has been completed, a thorough visual inspection will be conducted, recording serial numbers on a core map, and noting any obvious misalignment of the fuel assemblies; and

h. The upper guide structure, instrumentation and reactor vessel head will be installed in accordance with applicable procedures.

13.8.5

The plant reactor engineer has the responsibility and authority, as delegated by the Plant Superintendent, necessary to assure the safe and efficient conduct of the core loading operation and the safeguarding and accountability of Special Nuclear Material. In this regard, he has the authority to initiate alterations to the fuel loading procedure, and stop fuel loading operations when deemed necessary in the interest of safeguarding personnel, the plant and SNM. Approved written procedures will be strictly adhered to. As a minimum, these procedures will contain the following: criteria for minimum operable instrumentation; scope of communications between the control room and the fuel handling areas; a containment evacuation plan and its associated safety measures; actions to be followed in the event of fuel damage; sequencing instructions for core loading, including positioning, orienting and seating of fuel and components; independent verification of fuel and component locations; documentation instructions for SNM accountability; methods for the gathering, reduction and evaluation of statistical count-rate data; and criteria for stopping the fuel loading evolution.

13.9 INITIAL CRITICALITY

The purpose of the Initial Approach to Criticality Procedure was to provide a safe and controlled method of achieving initial reactor criticality. The initial approach was performed by primary coolant boron dilution with control rods in the nearly fully withdrawn position at a temperature and pressure specified in the procedure.

Administrative control of the initial criticality procedure is described in Section 13.3 of this chapter. After completion of each step of the procedure, approval was obtained from the Millstone Startup Engineer before the next procedural step was initiated. Initial criticality approach was conducted under the direction of the organization described in Section 13.2 of this chapter.

Initial conditions which will be established prior to the initial approach to criticality are:

a. Preoperational and hot functional testing will have been completed;

b. Reactor coolant system boron concentration is at the refueling concentration;

c. Reactor coolant system is maintained at specified temperature and pressure;

d. Pressurizer and steam generators are maintained at specified water levels;

e. Reactor protective system calibration tests have been performed;

f. Reactor coolant pumps are in the status specified in the procedure; and

g. Containment integrity established in accordance with the containment integrity check procedures.

Precaution which must be observed during the initial approach to criticality are:

a. Operating personnel will be vigilant in seeking out and evaluating unexpected behavior of the reactor plant throughout the approach to criticality;

b. Criticality will be anticipated whenever CEAs are being withdrawn or whenever boron dilution operations are being performed;

c. The maximum startup rate will not be exceeded;

d. Any changes in plant operating conditions that may produce a sudden change in primary loop water temperature or boron concentration will be avoided if possible;

e. The initial approach to criticality will be guided by plots of the inverse count rate multiplication versus time and versus boron concentration (chemical analysis) and of boron concentration (chemical analysis and boronometer) versus dilution time will be maintained.

f. Reactor coolant system boron concentration will be determined periodically. Results of boron analysis will be recorded in the control room and compared with the boronometer reading.

The procedure for the initial approach to criticality will include the following considerations:

a. Reactor coolant system will be sampled to establish initial boron concentration and general reactor coolant system chemistry;

b. Periodic sampling of the reactor coolant system for boron will be initiated and results recorded in the control room;

c. A boronometer check will be made periodically for proper operation;

d. Actuation of the low steam generator pressure and the thermal margin/low-pressure trip will be bypassed;

e. Withdrawal of each group of CEAs by the increment specified will be performed. Test personnel shall maintain plots of the inverse count rate multiplication versus CEA group sequence position for the wide-range instrumentation;

f. Reactor coolant system boron concentration will be decreased by dilution. Plots of the inverse count rate multiplication versus time, versus actual boron concentration, and versus calculated boron concentration will be maintained. Dilution will be halted when criticality is achieved;

g. Slightly above the level at which criticality was achieved, power level will be stabilized using regulating CEA group specified; and

h. After plant conditions have stabilized, the following data for just critical condition will be recorded:

1. CEA positions
2. System pressure
3. System temperature
4. Pressurizer level
5. Boron concentration
6. Indicated power level
7. Reactor coolant system flow
8. Routine chemistry analysis for RCS

13.10 LOW POWER PHYSICS TESTING

The purpose of the Low Power Physics Testing program was to obtain as-built reactor characteristics and to verify Technical Specification, safety limits, and physics design parameters. Administrative control of the Low Power Physics Test procedures was as described in Section 13.3 of this chapter. After completion of each procedural step, approval was obtained from the Millstone Startup Engineer before the next procedural step was initiated. Low Power Physics Testing was conducted under the direction of the organization described in Section 13.3 of this chapter.

Initial conditions which will be established prior to the start of the low-power tests are:

a. Initial Approach to Criticality has been completed;

b. The reactor is critical with excess reactivity as close to zero as practical and a reactor startup rate of zero;

c. Boron concentration and CEA position is as remaining from initial criticality;

d. The reactor coolant system temperature is being maintained by pump heat and by dumping steam to the atmosphere and/or to the condenser;

e. System pressure is being maintained by control of the backup and proportional pressurizer heaters and spray valves;

f. Pressurizer level is being maintained by operating the charging pump and proper positioning the letdown stop valves and the letdown control valves; and

g. Reactor protective system calibration tests have been performed subsequent to the most recent alteration to any part of that system.

Precautions which shall be observed during the low-power tests are:

a. The specified low-power power level as indicated by the wide-range logarithmic channels will not be exceeded;

b. If any abnormal conditions develop during this procedure which could or do significantly affect core reactivity or increase reactor power above the low-power power level, the reactor operator will trip the reactor, borate if necessary, and restore the plant to a safe normal condition;

c. When increasing neutron level, particular note of reactor plant temperature, pressure, pressurizer level, and charging rate changes will be taken that might indicate reactor operation in the sensible heat region;

d. While maintaining criticality during primary coolant system borations and/or dilutions, adjustment will be made of the boration or dilution rate as required when approaching the desired end point to prevent overshoot. Pressurizer spray will be provided during boron concentration changes to equalize pressurizer boron concentration with loop boron concentration;

e. Prior to commencing a specific test, it will be required that all CEA's in a group are within a specified band of each other unless a test condition specifies that a CEA is at other than bank height;

f. All plant parameters other than those to be changed during a given operation will be maintained as constant as possible; and

g. Tests involving abnormal CEA configurations will be performed in accordance with Technical Specifications, and the appropriate special operating precautions will be observed.

Tests to be performed as part of the Low Power Physics Testing program are:

a. Neutron and gamma radiation surveys;

b. Verification of radiation monitor response to known source;

c. All rods out, clean core, critical boron concentration;

d. Temperature coefficient of reactivity with various CEA group configurations;

e. Nonoverlapped regulating and shutdown CEA group worths;

f. Composite regulating CEA group worth with normal group sequencing;

g. Worth of the most reactive CEA for a dropped rod situation to verify safety analysis;

h. Worth of the most reactive CEA for an ejected rod situation occurring with a full power and a zero power configuration to verify safety analysis;

i. Worth of the most reactive CEA for a stuck rod situation to verify safety analysis;

j. Part-length CEA group worth measurement;

k. Chemistry and radiochemistry tests to verify control of water quality;

l. Boron reactivity worth measurement; and

m. Shutdown margin determination.

13.11 POWER ASCENSION TESTING

The power test procedures describe the detailed steps required for the initial plant startup from completion of the Low Power Physics Tests phase to full-rated power level, including those tests necessary to demonstrate safe plant operation within design specifications. Administrative control of this phase of the program was as described in Section 13.3 of this chapter. After completion of each step of the procedure, approval was obtained from the Millstone Startup Engineer before the next procedural step will be initiated.

Power Ascension testing was conducted under the direction of the organization described in Section 13.2 of this chapter.

Initial conditions which will be established prior to the ascension to power are:

a. All preoperational, hot functional, and low power physics testing have been completed;

b. System check lists completed for the Initial Approach to Criticality Procedure need not be repeated on any system unless that system was removed from service or abnormal valving was performed on that system;

c. The reactor coolant system temperature is being maintained by pump heat and by bypassing steam to the main condenser;

d. Pressurizer pressure control is in automatic maintaining system pressure;

e. The chemical and volume control system is automatically maintaining programmed pressurizer water level;

f. Steam generator water level is being maintained within the normal operating range;

g. The required reactor coolant pumps are operable;

h. The gain sensitivity switches for the linear nuclear instrument power channels are in the X10 position;

i. Main condenser vacuum is being maintained;

j. The turbine-generator unit is on turning gear; and

k. The applicable requirements of the Technical Specifications for plant startup are satisfied.

During the ascension to power, the following precautions will be observed:

a. If an unplanned reactor trip should occur during the conduct of this procedure, the plant will be returned, after necessary corrective measures have been taken, to the conditions which existed at the beginning of the procedure step during which the trip occurred following approved operating procedures; and

b. All operations where certain plant conditions are being established for the first time shall be performed slowly and deliberately to avoid sudden changes in parameter affecting reactor behavior.

The types of tests that are expected to be performed are listed below. The individual test procedural steps will include instructions and precautions for establishing special conditions necessary for conducting tests.

TEST	POWER LEVELS (%)
NSS Test Data Record	10, 20, 30, 40, 50, 60, 70, 80, 90, 100
Core Power Distribution Data Record	10, 20, 30, 40, 50, 60, 70, 80, 90, 100
Transient Test Data Record	20, 50, 80, 100
Xe Equilibrium ARO	20, 50, 80, 100
Xe Equilibrium Rods Inserted	20, 50, 80, 100
NSSS Calorimetric	20, 50, 80, 100
Nuclear and ΔT Power Calibration	20, 50, 80, 100
Chemistry & Radiochemistry Tests	20, 50, 80, 100
Radiation Surveys and Shielding Effectiveness	20, 50, 80, 100
Process Radiation Monitor Calibration Check	20, 50, 80, 100
Variable T_{avg} Test	20, 50, 80, 100
MCB Reactor Trip	20
RCS ΔT Power Determination	20
Static “Ejected” CEA	20
Power Range Subchannel Calibration	20
Generator Trip	100
Shutdown Outside Control Room	25-40
Reactor Regulating System	50
Partial Loss of RC Flow	80
Total Loss of RC Flow	40

TEST	POWER LEVELS (%)
Natural Circulation	0
Loss of Offsite Power	40
Dropped CEA	50
Turbine Runback	80, 90
Part Loop Operation	≤ 47, ≤ 78.5
Induced Oscillation	80
Minimum DNBR	50, 80
Turbine Trip	50 (a)
Warranty	100

(a). The Generator Trip at 100% power will result in a Turbine Trip, thus, verifying trip response.

13.12 PIPING OPERATIONAL TESTING

The seismic Category I piping systems designed to ASME Nuclear Classes l and 2 are listed in Table 13.12-1.

The following discusses the stress analysis techniques used to evaluate the dynamic effects in piping systems. The discussion covers the dynamic effect produced by valve operations and pump trips.

Valve Operations

When dynamic effects are expected in a piping system, excluding the pressurizer safety/relief and shutdown cooling piping systems, due to valve operation, the stress analysis is done using maximum forces created by valve closures by means of a static analysis with a dynamic loading factor of 2 applied for the conservative dynamic loads. For such closed discharging piping systems, supports are installed to take the dynamic transient force from rapid valve closure and/or relief valve opening on each straight leg of piping and to cause the piping to have its fundamental frequency in the axial direction above the minimum acceptable natural frequency to make the method of static analysis valid.

The magnitude of the forces created by valve openings in closed piping system is estimated by determining the change in momentum in each section of straight piping with linear increased valve flow rate. For valve closures, a pressure wave is generated in the system which causes a dynamic response of the system. The maximum magnitude of the pressure wave is calculated from $Ap = e\Delta V\ C$, where e is fluid density, ΔV is flow velocity change and C is velocity of sound. The stresses caused by these dynamic transient forces are appropriately combined with those caused by seismic, deadweight, and pressure. The total stress is limited to less than those allowed by the applicable codes.

The pressurizer and shutdown cooling piping systems, which are subject to rapid relief valve openings, are analyzed on a time-history basis. The loading (forcing function) as a result of valve operation is calculated on a time-dependent basis and applied to a dynamic model of the system. The load is applied at specific locations in the system with respect to the time it takes the pressure wave to reach that location. It is this time-dependent application of the load which results in the excitation of the piping system.

The stress evaluation due to the loads developed by rapid relief opening is performed by using the rules for Class I piping according to Section III of the ASME BPVC, NB-3650, as modified by Code Case 1569. The value of Mi in equations 9 through 14 includes the reaction force moment as a result of relief valve opening.

Pump Trip

The stress analysis includes the dynamic effects of pump trips if estimated to be significant. However, coastdown time for a centrifugal pump is usually of sufficient duration that water hammer effects in the piping systems are normally negligible.

The following discusses the dynamic acceptability of the piping systems covered in Table 13.12-1.

Main Steam System

The analysis of the main steam system included the transient conditions of rapid turbine stop valve closure and main steam relief valve lifting. As a result of a turbine stop valve closure, a pressure wave is generated within the steam line which causes a dynamic response of the system. An instantaneous stop valve closure is assumed, and the magnitude of t@e pressure wave is calculated from $\Delta p = e\ \Delta V\ C$.

The main steam relief valve system is designed so that blowdown force is transmitted directly to the structure and not through the piping by use of mechanical/structural devices. These relief valves are provided with discharge stacks to direct the steam flow to atmosphere. The stacks are designed so that back pressure does not cause a valve reaction force.

The stresses caused by the dynamic transient forces when combined with those caused by seismic, dead weight and pressure are less than 1.2 S_h, which is allowed by ANSI B31.7.

From past experience, flow induced steady state vibration in the main steam line has been shown to be insignificant by actual measurements with remote instrumentation. This was verified for Millstone 2 by the startup test program.

Main Feedwater

An analysis of the feedwater system has been done to determine any potential for damage due to water hammer. The area of most concern is the pressure waves caused by rapid changes in flow velocity. The feedwater control valve closure time of 20 seconds or pump coastdown time is much too slow to cause water hammer.

On the basis of the above discussion, water hammer effects in the feedwater system are negligible.

Pressurizer Relief/Safety Valves and Shutdown Cooling Systems

These systems are being analyzed by the force-time history approach with respect to relief valve opening. The piping will be sufficiently restrained to carry the imposed loading.

Charging Line

The reciprocating type charging pumps will produce vibratory response in the charging system. To minimize these effects, each charging pump will be provided with a pulsation dampener at the pump discharge connection to limit the pressure pulsations to less than 4 percent of the working pressure. These dampeners are expected to limit the dynamic effects in the piping system to an acceptable level. The startup test program was used to verify the above.

Other Systems

The dynamic effects experienced by the letdown, safety injection, containment penetration, boric acid, and reactor coolant sampling piping systems are expected to be insignificant. Therefore, the stress analyses for these systems did not include any transient conditions. Verification of the above assumption will be made by the test program.

Acceptance Criteria

Steady state vibration of piping systems, excluding the pressurizer relief/safety and shutdown cooling piping systems, was measured in the field. The actual displacements were compared with allowable displacements. Allowables are defined as those displacements which produce stresses whose amplitudes are less than one-half the endurance limits as defined in ASME Section III. A start-up test program was established to perform the above measurements.

Since the force-time history approach was utilized for the pressurizer relief/safety valve and shutdown cooling piping systems, these systems will be excluded from the startup test program.

Conclusions

The stress analysis performed for the piping system subjected to dynamic effects is believed adequate to insure the integrity of seismic Category I piping systems which are in accordance with ASME Section III, Classes l and 2. A startup test program was established to verify that the analysis and predictions concerning dynamic effects in piping systems are correct.

For the systems listed in Table 13.12-1, either the transient effect were proven to be minimal or a test program was conducted. A location for measurement of deflection was selected for each system to be tested. The location selected was determined by reviewing the existing analyses and determining which portion of the system will respond to imposed transient conditions.

The instrumentation was located at the specific points defined in Item l above and was monitored during plant warm-up to indicate that actual pipe movements are following anticipated movements. (This indicated that supports intended to deflect during normal heat-up are free.) During imposed plant transients, visual observation of the specific lines was made and the deflections observed and recorded.

The measured deflections were compared to the calculated deflections and determination made as to acceptability.

The criteria for acceptability will be as follows:

a. Class I Systems: The allowable variation of the resultant deflection shall be determined for each system.

$$\frac{\delta \text{ Measured}}{\delta \text{ Calculated}} \leq \frac{3S_M}{C_2 M_i D_o^*}$$

* Note: Calculated thermal expansion stress for operating condition investigated

b. Class 2 Systems:

$$\frac{\delta \text{ Measured}}{\delta \text{ Calculated}} \qquad \frac{S_A}{S_E^*}$$

In order to conservatively estimate these dynamic effects, valve opening times, pump coastdown times, etc., have been conservatively estimated.

TABLE 13.12-1 SEISMIC CATEGORY I-ASME SECTION III CLASSES 1 AND 2 PIPING SYSTEMS

Main steam

Main feedwater

Charging

Letdown

Steam generator blowdown

Shutdown cooling

Safety Injection

Containment penetration piping

Pressurizer relief / safety valve

Boric acid

Reactor Coolant sampling

Pressurizer spray

Auxiliary Pressurizer spray

Reactor coolant pump controlled bleedoff

Reactor coolant

13.13 ACTIVE VALVES OPERATIONAL TESTING

Active valves are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the reactor coolant pressure boundary. These valves are located within the engineered safety features systems, vital support systems, and containment isolation system and are automatically actuated during a post-incident condition to perform their intended function.

Pneumatically actuated valves without air accumulators are excluded from the above definition of active valves since the compressed air system is a nonessential system and is not relied upon to function during post-incident conditions. Pneumatically actuated valves, whether receiving an automatic signal or not, are assumed to be operated by spring action to their respective safe position. Pneumatic actuated valves which are not required to remain operational following a seismic disturbance are classified as passive components. A listing of all safety-related pneumatically actuated valves, including those with air accumulator, is tabulated in the response to AEC Question 9.32 in Amendment 15.

Pneumatically actuated valves provided with air accumulator are considered active valves even though their operation is required during the long term post-incident condition. The air accumulator is provided to assure an adequate supply of compressed air for valve operation during the post-incident case assuming the normal compressed air system is operable. In addition, valves with air accumulator that are located in areas which are accessible during LOCI conditions are provided with handwheels for manual operation as backup.

The air accumulator package consists of tanks, tubing and check valve which are connected to the normal compressed air supply arrangement. The check valve prevents loss of accumulator air following a postulated loss of the normal compressed air system. The package is completely separate from the valve pneumatic operator except for the air tubing connection to the operator. The accumulator package is designed as a vital system with all components through the check valve analyzed for operation during and after a seismic event.

The electric motor-operated valves which are required for post-incident operation for vital processes or for containment isolation are considered active valves. A tabulation of active process motor-operated valves is given in the response to AEC Question 4.4 in Amendment 15. The electric motor operators for these valves are all of the Limitorque SMB series and have a history of qualification testing to verify their reliability and operability. Extensive testing has been carried out by the valve operator manufacturer and by an independent institute. The testing has been done over a number of years, and the most recent testing in the summer of 1972 bears out the operability confirmed initially.

1. MANUFACTURER’S TESTS

a. Electric Motor Operator Qualification Testing Shock and Vibration Testing

In August, 1970, a Limitorque SMB Series operator was mounted on a test stand having a threaded valve stem driven by the operator simulating

opening and closing a valve. The operator was electrically connected so as to stop at the full close position by means of a torque switch and stop at the full open position by means of a geared limit switch. The operator had a 4-train geared limit switch installed and all contacts not being used for motor control were wired to electric indicating lights at a remote panel.

The unit successfully completed a 5.3 g shock level at 32 Hz with no discrepancies noted. An exploratory scan of 5 Hz to 35 Hz was made and no critical resonant frequencies were noted on the operator. The unit was shocked and vibrated in each of three different axes a total of 2 minutes on, 1 minute off, 3 times per axis. The unit was operated electrically to both the full-open and full-close position, and all torque switches and limit switches functioned properly. None of the auxiliary limit switches wired to indicating light ever flickered or indicated they were opening or flickering. All electrical and mechanical devices on the operator worked successfully.

b. Heat Testing

In January 1969, a completely assembled and operational SMB series operator was placed in an oven where the temperature was maintained at approximately 325°F for a duration of 12 hours. The unit was electrically operated every thirty minutes for a period of approximately two minutes per cycle, and the geared limit switches were used to stop the actuator at the full-open and full-closed position of travel. Indicating light circuits were also wired to the geared limit switches.

The test was successful in every respect. There were no malfunctions of the operator and upon inspection of the component parts used, there was no noticeable deterioration or wear.

c. Live Steam Testing

In January 1969, a complete SMB series operator was set up for electrical operation, and live steam was piped into he conduit taps on the top of the limit switch compartment. One of the bottom conduit taps was left open to drain off any condensate. The operator was set up on a timer basis for operation over a period of approximately nine hours and operating every thirty minutes for two minutes per cycle. During this test, the live steam in the switch compartment seemed to have no effect whatever on the function of the torque and limit switches in their control of the operator at the full-open and full closed position of travel. In addition, the limit switches were wired up to indicating lights which operated satisfactorily.

The test was successful and there was no noticeable effect on the function of any of the parts in the limit switch compartment.

d. Life Cycle Testing

In January 1969, the operator was mounted on a stand inside a test chamber and a 150 cycle (anticipated equivalent to 40 year life of valve) load test was made on the unit. This test cycle consisted of stroking a 2-3/8 inch diameter valve stem at a speed of 6 inches per minute for a total of approximately 12 inches in two minutes. The valve stem in the full closed position produced a thrust of 16,500 pounds on a rigid plate securely bolted to the test chamber. The unit was wired so that the closing direction and the open position geared limit switch stopped the unit in the full open position.

After the life cycle testing was completed, the unit was inspected and found to be in excellent condition. There was no noticeable wear on any of the parts.

e. Simulated Accident Environment Testing

An electric motor operator was tested under conditions which simulated the temperature humidity and chemical environments that could be expected to exist in the containment following some postulated accident such as the rupture of a major reactor coolant pipe.

The operator was placed in an Autoclave type chamber and subjected to 90 psig saturated steam. At specified intervals, the operator was cycled to assure proper operation. Forty minutes after the introduction of steam, a 1.5% boric acid solution was sprayed on the operator assembly. The operator continued to operate satisfactorily. Later, the steam pressure was periodically reduced to simulate post-accident conditions. The boric acid spray was allowed to continue for four hours. The steam pressure was eventually reduced to 15 psig. The test continued for 7 days.

During this time, the operation of the operator became erratic. The corrosive effects of the steam and boric acid spray caused electrical contact malfunctions which were bypassed by the use of an appropriate jumper. The valve continued to cycle during the seven day period.

A design change was made to the limit switch in order to correct the erratic operation, and it was tested under similar accident conditions and found to operate satisfactorily. This design change has been incorporated into all subsequent applicable models of this operator.

2. INDEPENDENT TESTING

a. Franklin Institute Tests

More recent tests on a Limitorque SMB series operator were conducted during the summer of 1972 by the Franklin Institute Research Laboratories.*

The following testing was conducted:

1. The previous Simulated Accident Environment Test (Item 1-E above) was reconducted with the Limitorque operators exposed to gamma radiation (200 megarads) and increased steam/chemical exposure time from 7 days to 12 days.

2. The previous Live Steam Testing (Item 1-C above) was reconducted in a steam environment as high as 340°F during the first day of tests and increased exposure time from 9 hours to 30 days.

3. The previous shock and Vibration Test (Item 1-A above) was reconducted.

During all of these tests, as appropriate, the operator was periodically cycled and was found to operate satisfactorily.

*1. Qualification Test of Limitorque Valve Operator Motor Brake, and other Units in a Simulated Reactor Containment Post-Accident Environment, Final Report F-C3327, July 1972.

2. Qualification Test of Limitorque Valve Operators in a Simulated Reaction Containment Post-Accident Steam Environment, Final Report F-C3441, September 1972.

3. VALVE PURCHASE SPECIFICATION

All safety-related valves, including pneumatically actuated valves with and without air accumulators, receive extensive preoperational testing prior to initial fuel loading and periodic surveillance testing after startup. These valves must comply with the purchase specification requirements for testing and seismic analysis.

The purchase specifications for these valves require that they be hydrostatically tested, leak tested and cycled from the extremes of allowed movement of opened or closed.

The valve vendor is also required to submit calculations which show that when the valve assembly is subjected to a 3g inertial load in any direction that stresses incurred are within the code allowable stresses. It is also verified that the first natural frequency is above 20 cps for all valves.

4. PREOPERATIONAL TESTING

Testing procedures for valves that require operation to meet engineered safety feature requirements were quite extensive during the preoperational testing program. These tests demonstrated proper installation, strength and functional performance of valves. Subsequent to satisfactory preoperational testing, surveillance testing requirements have been established to assure continued satisfactory operation of these valves. A planned surveillance program of valve motor operator testing through the whole operating cycle will be undertaken to prove the operability and setting of the torque and limit switches.

a. System Electrical Tests

The purpose of these tests was to verify electrical characteristics of valve operators in performing their function. Preliminary checkout of the operator valve assembly requires that the valve be free to move and that if the motor-operated valve travels in the wrong direction from its midtravel position, its breaker must be tripped immediately, as there would be no torque limit protection. The valve can be operated manually with a handwheel to ascertain its freedom of movement.

The phase rotation of the operator was checked. During valve operation, verification was obtained that the valve travel and motor are stopped by opening the torque limit switch. Similarly, the opening of the valve is terminated by the opening of the limit switch.

In checking the valve for engineered safety feature actuation, the valve was placed in the position opposite to its engineered safety feature position, and then the appropriate signal is simulated. The valve moves to its engineered safety feature position.

Acceptance criteria for these electrical tests are:

1. Valves must open, close, and travel in the proper direction in response to control and engineered safety features signals.

2. The valve open and closed indicating lights must indicate correctly.

3. Valve motor resistance-to-ground reading must be within specification.

4. The specified valve travel time is within specification requirements.

b. System Engineered Safety Features Tests

The performance of these tests was to demonstrate actual valve performance for its intended Engineered safety feature use. Initially, all valves were placed in their nonengineered safety feature position prior to simulating an engineered safety feature signal. Upon initiation of an engineered safety feature signal, the tests for the subject valve demonstrated that the valve travels to its post-incident position.

c. System Functional Testing

The purpose of this testing was to verify that the valves perform as intended for normal operation. Cycling the valves under conditions of specified differential pressure and/or flow that may be encountered during plant operation verified that the valve motor did not exceed maximum operating current and cycle time.

d. Integrated Actuation Test

The purpose of this test, for which these valves are used to perform, was to demonstrate the full operational sequence that would bring the emergency core cooling systems and the containment cooling systems into action, including the transfer to alternate power sources.

General acceptance criteria for this test are:

1. Upon actuation of the appropriate safety feature, the systems operate as described in the FSAR.

2. Upon loss of normal station power, the safety feature systems continue to perform their designed functions without interruption.

5. THE TECHNICAL SPECIFICATIONS

The Technical Specification testing requires that these valves be operated periodically to assure their continued availability and reliability. During the life of the facility, these valves will require maintenance, inspection, repair, and possibly replacement or modification.

In summary, the entire scope of testing verified the valves operability from conditions of extreme duress to normal operation, and the results of the earliest environmental, vibratory, and load testing have been repeatedly verified in later testing by independent research.

13.A PREOPERATIONAL TEST DESCRIPTIONS

13.A.1 SYSTEM: 125 VOLT DC SYSTEM

13.A.1.1 Objective

To demonstrate that the 125V DC System is capable of providing power during normal and abnormal conditions.

13.A.1.2 Prerequisites

a. 480V, 3ϕ, 60 cycle supply is available.

b. Battery room ventilation is operational.

c. Batteries and DC distribution system, including protective devices, are checked for operation and are in service.

d. Ammeters and voltmeters are calibrated and are in place.

e. Battery chargers are installed and are operational.

13.A.1.3 Method

a. Energize the battery chargers.

b. Adjust alarms and interlocks.

c. Discharge the batteries at a controlled rate to determine ampere-hour capacity.

d. Adjust chargers to supply DC load and charge batteries.

13.A.1.4 Acceptance Criteria

a. The DC System is capable of supplying the loads connected on a loss of AC power.

b. The chargers are capable of recharging the batteries and carrying the normal DC loads.

c. Alarms and interlocks function as designed.

13.A.2 SYSTEM: RESERVE STATION SERVICE TRANSFORMER

13.A.2.1 Objective

To verify that the Reserve Station Service Transformer 15G-22S(U3) can be energized from the 345 kV switchyard, also that all interlocks and protective devices function properly.

13.A.2.2 Prerequisites

a. Feeder breakers from reserve station service transformer racked out and tagged.

b. All meters, relays and protective devices calibrated and tested.

c. 125V DC and 480V AC available.

d. All construction work complete on transformer and switchgear.

13.A.2.3 Method

a. Install phase rotation indicators and voltmeters on 6.9-kV and 4,160V secondary winding power transformers.

b. Energize RSS transformer 15G-22S(U3).

c. Record voltage and verify phase relationship.

13.A.2.4 Acceptance Criteria

a. RSS transformer is capable of being energized from the 345-kV switchyard and proper phase relationship is exhibited.

b. The RSS transformer responds correctly to a loss of station power and a reactor safeguards signal.

13.A.3 SYSTEM: PRIMARY WATER TREATMENT SYSTEM

13.A.3.1 Objective

To demonstrate that the water treatment system is capable of meeting the requirement of high purity water for makeup to the primary loop and for various other services throughout the plant, the greatest demand occurring during refueling.

13.A.3.2 Prerequisites

a. 480V, 3ϕ, 60 cycle supply is available.

b. 120V, 60 cycle AC is available.

c. 125V DC is available.

d. Domestic water system is functional.

e. Instrument and station air systems are functional.

f. The water treatment system shall be operable.

g. Laboratory equipment for making various chemical analysis are available.

13.A.3.3 Method

a. Line up the system for operation.

b. Adjust alarms and interlocks.

c. Domestic water booster pumps are started to feed the system with water of known quality.

13.A.3.4 Acceptance Criteria

a. Pump characteristics curves are in accordance with manufacturer's specification.

b. Alarms and interlocks operate as designed.

c. The quality of treated water meets the design requirement.

d. The system is capable of supplying maximum operating requirement.

13.A.4 SYSTEM: 4,160 VOLT ELECTRICAL SYSTEM

13.A.4.1 Objective

To verify that 4160V buses A1 through A5 can be energized from their respective normal and alternate source and to verify that all interlocks and protective devices function properly.

13.A.4.2 Prerequisites

a. All feeder breakers to the 4160V buses are racked out and tagged.

b. Meters, relays, and protective devices calibrated and tested.

c. 125V DC available.

13.A.4.3 Method

a. Install phase rotation indicators and voltmeters on 4160V bus power transformers.

b. Rack in and close 4160V breakers to energize associated 4160V buses.

c. Record voltage and verify phase relationship.

13.A.4.4 Acceptance Criteria

a. The 4160V buses are capable to being energized from their normal and alternate source and that proper phase relationship is exhibited.

b. The 4160V System responds correctly to a loss of station power and a reactor safeguards signal.

13.A.5 SYSTEM: 480-VOLT ELECTRICAL LOAD CENTERS

13.A.5.1 Objective

To verify that the 480V safeguard and non safeguard buses can be energized from their normal and alternate sources and also that all interlocks and protective devices function properly.

13.A.5.2 Prerequisites

a. All breakers on 480V load center buses B1 through B6 racked out and tagged.

b. Meters, relays, and protective devices calibrated and tested.

c. 125V DC and 4.16 kV buses A1 through A4 energized.

13.A.5.3 Method

a. Install phase rotation indicators and voltmeters on Bus PT’s.

b. Close 4.16 kV breakers to energize load center transformers and buses.

c. Measure voltage and verify phase relationship.

13.A.5.4 Acceptance Criteria

a. 480V safeguard and nonsafeguard buses are capable of being energized from their normal and alternate sources and that proper phase relationship is exhibited.

b. The 480V load centers respond correctly to a loss of station power and a reactor safeguards signal.

13.A.6 SYSTEM: SERVICE WATER SYSTEM

13.A.6.1 Objectives

To demonstrate the capability of the service water system to respond to a SIAS signal; also to provide adequate cooling water to the turbine building and reactor building.

13.A.6.2 Prerequisites

a. Instrument air system operational.

b. Required electrical systems operational.

13.A.6.3 Method

a. With system in normal operation, insert a safety injection actuation signal.

13.A.6.4 Acceptance Criteria

a. Pump characteristic curves are in accordance to manufacturer specifications.

b. System must respond as designed to SIAS signal.

c. Flow capability of the system is in accordance with design.

13.A.7 SYSTEM: CLEAN LIQUID RADWASTE SYSTEM

13.A.7.1 Objective

Demonstrate the capability of the clean liquid radwaste system to operate as designed during normal operation or high radiation conditions.

13.A.7.2 Prerequisites

a. Clean liquid radwaste system operable.

b. Instrument air system operable.

c. Required electrical systems operational.

d. Safety injection actuation system operational.

13.A.7.3 Method

a. Line up system for normal operation.

b. Simulate low- and/or high-level signal to appropriate valves and pumps.

c. Simulate low- and/or high-temperature signal to appropriate valves.

d. Simulate high radiation signal to appropriate valves.

e. Initiate containment isolation actuation signal.

13.A.7.4 Acceptance Criteria

a. Upon appropriate level signal, pumps and valves must respond as designed.

b. Upon appropriate temperature signal, appropriate valves must respond as designed.

c. On high radiation signal, radiation isolation valves will close.

d. On initiation of a containment isolation actuation signal, system will respond as designed.

13.A.8 SYSTEM: AERATED LIQUID RADWASTE SYSTEM

13.A.8.1 Objective

Demonstrate the capability of the aerated liquid radwaste system to operate as designed during normal or high radiation conditions.

13.A.8.2 Prerequisites

a. Aerated liquid radwaste system operable.

b. Instrument air system operable.

c. Required electrical systems operable.

13.A.8.3 Method

a. Line up system for normal operation.

b. Simulate low level signal to appropriate pumps.

c. Simulate a high radiation signal to appropriate valves.

13.A.8.4 Acceptance Criteria

a. Pump characteristic curves must be in accordance with manufacturer's specifications.

b. Pumps must respond as designed to low level signal.

c. Appropriate valves must respond as designed to a high-radiation signal.

13.A.9 SYSTEM: 480-VOLT MOTOR CONTROL CENTERS

13.A.9.1 Objective

To verify that the 480V safeguard and non-safeguard motor control centers can be energized from their normal source. Also that all interlocks and protective devices function properly.

13.A.9.2 Prerequisites

a. All breakers on motor control centers to be energized off and tagged.

b. Meters, relays and protective devices calibrated and tested.

c. 125V DC available.

d. 480V buses B1 through B6 energized.

13.A.9.3 Method

a. Install phase rotation indicator in MCC starter.

b. Close associated MCC supply breaker and verify phase relationship.

13.A.9.4 Acceptance Criteria

a. Safeguard and non-safeguard motor control centers are capable of being energized from their normal source and that proper phase relationship is exhibited.

13.A.10 SYSTEM: TURBINE BUILDING CLOSED COOLING WATER SYSTEM

13.A.10.1 Objective

To demonstrate the capability of turbine building closed cooling water system to provide adequate cooling water to the various equipment in the turbine building.

13.A.10.2 Prerequisites

a. 480V, 120V AC and 125V DC electrical systems are available.

b. Service water system is functional.

c. Instrument air system is functional.

d. Chemical addition tank is available.

e. Turbine building closed cooling water system is operable.

13.A.10.3 Method

a. Line up the system for operation.

b. Adjust alarms and interlocks.

c. Surge tank level should be carefully watched to note loss of water which would be indicative of leakage.

d. Establish required cooling water flow to various equipment.

13.A.10.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer's specifications.

b. Alarms and interlocks function as designed.

c. Surge tank level remains steady.

d. Flow capability of the system meets the design requirement.

13.A.11 SYSTEM: FIRE PROTECTION SYSTEM

13.A.11.1 Objective

To demonstrate that the fire protection system is capable of providing a firm supply of water for firefighting under all conditions, including loss of electrical feed.

13.A.11.2 Prerequisites

a. 480V AC and 125V DC electrical supplies are available.

b. Adequate water is available in fire water storage tank.

c. Fire protection system is operable.

13.A.11.3 Test Method

a. Line up the system for operation.

b. Adjust alarms and interlocks.

c. Check sprinkler systems.

13.A.11.4 Acceptance Criteria

a. Pump characteristic curve is in accordance with manufacturer’s specification.

b. Alarms and interlocks function as designed.

c. Fire water is available from Unit 3 loop in case of failure of Unit 2 electric driven fire pump.

d. The system is capable of providing required flow to the sprinkler systems.

13.A.12 SYSTEM: RECOVERED BORIC ACID SYSTEM

The testing requirements for this system are covered under the Clean Liquid Radwaste System.

13.A.13 SYSTEM: INSTRUMENT AIR SYSTEM

13.A.13.1 Objectives

To demonstrate the capability of the instrument air system to provide adequate low-moisture air for the plant.

13.A.13.2 Prerequisites

a. Turbine building closed cooling water system operational.

b. Require electrical systems available.

13.A.13.3 Method

a. Line up system for normal operation.

b. Simulate a low pressure signal to the normally operating instrument air compressor.

c. Simulate a low-low-pressure signal to the standby instrument air compressors.

d. Simulate a low-pressure signal to instrument air cross-connect valve.

13.A.13.4 Acceptance Criteria

a. All three instrument air compressors must respond as designed to a low pressure signal.

b. The instrument air cross-connect valve must respond to a low-pressure signal as designed.

c. Instrument air flow and quality must be in accordance to system design.

13.A.14 SYSTEM: PRIMARY WATER STORAGE SYSTEM

13.A.14.1 Objective

Demonstrate that the primary water storage system is capable of providing an adequate supply of deaerated water.

13.A.14.2 Prerequisites

a. Primary water storage tank operable.

b. Domestic water system operational.

c. Water treatment system operational.

d. Instrument air operational.

e. Required electrical systems functional.

13.A.14.3 Method

a. Line up system and place in normal operation.

b. Simulate low- and/or high-pressure signal to the primary water transfer pumps.

c. Simulate a low- and/or high-level signal to the appropriate level control valves.

d. Simulate a low temperature signal to the primary water circulation pumps.

13.A.14.4 Acceptance Criteria

a. All pumps and level control valves must operate in accordance with design when receiving the appropriate simulated signal.

13.A.15 SYSTEM: 120-VOLT INSTRUMENT AC SYSTEM

13.A.15.1 Objective

To demonstrate that the 120V Instrument AC System is capable of supplying non-essential instrument and control load requiring voltage regulation.

13.A.15.2 Prerequisites

a. 480V AC supply is available.

b. 120V Instrument AC System is operable.

13.A.15.3 Method

a. Line up the system for operation.

b. Adjust regulator transformers to supply non-essential instrument and control load.

13.A.15.4 Acceptance Criteria

a. Voltage regulation meets the design requirement.

b. The system is capable of carrying non-essential instrument and control requiring voltage regulation.

13.A.16 SYSTEM: 120-VOLT VITAL AC SYSTEM

13.A.16.1 Objective

To demonstrate that the system is capable of supplying power to essential instrumentation and control under all operating conditions.

13.A.16.2 Prerequisites

a. 125V DC power is available.

b. 120V instrument AC is available.

c. Inverters are installed and operational.

d. 120V vital AC system is operable.

13.A.16.3 Method

a. Line up the system for operation.

b. Adjust inverters to supply vital AC load.

c. Adjust alarms and interlocks.

d. Test the transfer capability of the static switch.

13.A.16.4 Acceptance Criteria

a. Alarms and interlocks function properly.

b. Inverters are capable of carrying the vital AC loads.

c. The static switch will transfer to the backup electrical source upon manual initiation.

13.A.17 SYSTEM: GASEOUS RADWASTE SYSTEM

13.A.17.1 Objectives

Determine the capability of the gaseous radwaste system to respond to CIAS and to a high-discharge radiation level.

13.A.17.2 Prerequisites

a. Instrument air system operational.

b. Required electrical systems operational.

c. Reactor building closed cooling water system operational.

13.A.17.3 Method

a. Line up system and place in normal operation.

b. Simulate a containment isolation actuation signal.

c. Simulate a high discharge radiation signal.

d. Through use of compressor controller, simulate a high- and/or low-pressure signal to each waste gas compressor.

13.A.17.4 Acceptance Criteria

a. Appropriate valves must respond to containment isolation actuation signal (CIAS) according to design.

b. Appropriate valves must respond to a high discharge radiation signal in accordance with design.

c. Each waste gas compressor must operate in accordance with design.

d. Systems must meet all design criteria.

13.A.18 SYSTEM: CONDENSATE STORAGE SYSTEM

13.A.18.1 Objective

To demonstrate the capability of the condensate storage system to transfer water in accordance with design.

13.A.18.2 Prerequisites

a. Condensate storage tank operational.

b. Instrument air and appropriate electrical systems operational.

13.A.18.3 Method

a. Line up system for normal operation.

b. Simulate low-temperature signal to actuate recirculation pump.

c. Simulate low level to actuate proper level control valves.

13.A.18.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer specifications.

b. Recirculation pump and level control valves respond to level and temperature signals as designed.

13.A.19 SYSTEM: AUXILIARY AND EMERGENCY FEEDWATER SYSTEM

13.A.19.1 Objective

To demonstrate that the auxiliary and emergency feedwater system is capable of providing feedwater for the removal of decay heat of the primary system. Also to supply feedwater to the steam generators when the feed pumps are inoperative.

13.A.19.2 Prerequisites

a. 4.16 kV, 480V AC and 125V DC power sources are available.

b. (Deleted.)

c. Condensate storage system is functional.

d. Steam generator system is functional.

e. Instrument air system is functional.

f. Auxiliary and emergency feedwater system is operable.

13.A.19.3 Method

a. Line up the system for operation.

b. Adjust interlocks.

c. (Deleted.)

13.A.19.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer's specification.

b. Control valves operate smoothly and can be actuated from the control room and emergency shutdown cooling panel.

c. All interlocks function properly.

d. (Deleted.)

e. Feedwater flow capability of the system meets the design requirements.

13.A.20 SYSTEM: STEAM GENERATOR FEEDWATER SYSTEM

13.A.20.1 Objective

To prepare the steam generator feedwater system for operation. The feedwater pump turbines will be tested uncoupled during hot functional tests. The feedwater system will be functionally tested during power ascension tests.

13.A.20.2 Prerequisites

a. 480V, 120 VAC and 125V DC power sources are available.

b. Feedwater turbine lube oil system is operational.

c. Instrument air system is operational.

13.A.20.3 Method

a. Using simulated signals, verify alarms, interlocks and operation of turbine controls, lube oil system and valves.

13.A.20.4 Acceptance Criteria

a. Turbine lube oil system is in accordance with manufacturer's specifications.

b. Turbine controls are in accordance with manufacturer's specifications.

c. System valves with postioners operate in accordance with design.

13.A.21 SYSTEM: CONDENSATE SYSTEM

13.A.21.1 Objective

To demonstrate that the condensate system is capable of transferring adequate quantity of condensate water from condensate hotwell to steam generator feedwater system.

13.A.21.2 Prerequisites

a. 6.9 kV, 120V AC and 125V DC power sources are available.

b. (Deleted.)

c. Instrument air system is functional.

d. (Deleted.)

e. Turbine building closed cooling water is functional.

f. Condensate system is operable.

13.A.21.3 Method

a. Line up the system for operation.

b. Adjust alarms and interlocks.

13.A.21.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer's specification.

b. Alarms and interlocks function properly.

c. Condensate water flow capability of the system meets the design requirement.

13.A.22 SYSTEM: FEEDWATER HEATER DRAINS AND VENTS

13.A.22.1 Objective

To demonstrate that the heater drains and vents system will function in accordance with the design intentions and to check the functions of pumps, valves, instrumentation and interlocks.

13.A.22.2 Prerequisites

a. 4.16 kV, 120 VAC, 125 VDC power sources are available.

b. Compressed air system available.

c. Floor drain system functional.

d. (Deleted.)

e. (Deleted.)

f. (Deleted.)

g. (Deleted.)

13.A.22.3 Method

a. Using simulated signals, verify alarms, interlocks and operation of feedwater heater level control valves.

13.A.22.4 Acceptance Criteria

a. Alarms and interlocks must function properly.

b. (Deleted.)

c. Control valves operate smoothly at designed set points.

d. Pump performance curves must meet manufacturer’s specifications.

13.A.23 SYSTEM: MAIN STEAM

13.A.23.1 Objective

Demonstrate the capability of the main steam system components to operate in accordance with design intentions and to check the functions of valves, instrumentation and interlocks.

13.A.23.2 Prerequisites

a. 480 VAC, 120 VAC, 125 VDC power sources are available.

b. Instrument air system functional.

c. (Deleted.)

d. Main steam system operable.

e. Engineered Safety Features Actuation Systems functional.

f. Electro-hydraulic control system functional.

g. Steam dump/bypass control system functional.

13.A.23.3 Method

a. (Deleted.)

b. Simulate CIAS and MSI signals to appropriate equipment.

c. Initiate a steam dump signal.

d. Simulate a low steam generator pressure signal.

e. Initiate a high-radiation signal.

13.A.23.4 Acceptance Criteria

a. On initiation of a Containment Isolation Actuation signal appropriate valves must close.

b. On a high-radiation signal appropriate valves must close.

c. On initiation of a Main Steam Isolation Signal appropriate valve must close.

13.A.24 SYSTEM: SAMPLING SYSTEM

13.A.24.1 Objectives

Demonstrate the capability of the sampling system to operate under normal conditions and to respond to a high radiation signal or a containment isolation actuation signal.

13.A.24.2 Prerequisites

a. Sampling system operable.

b. Required electrical systems functional.

c. Instrument air system functional.

d. Containment isolation actuation system functional.

13.A.24.3 Method

a. Line up system for normal operation.

b. Initiate a containment isolation actuation signal (CIAS).

c. Simulate a high-radiation signal.

13.A.24.4 Acceptance Criteria

a. On initiation of a containment isolation actuation signal, the appropriate valves must close.

b. On initiation of a high-radiation signal, the appropriate valves must close.

c. System must meet design criteria.

13.A.25 SYSTEM: SPENT FUEL COOLING AND PURIFICATION

13.A.25.1 Objective

To demonstrate that the spent fuel pool and refueling pool cooling and cleanup system flow paths are as designed and to verify the capability of supplying makeup and cooling water from the emergency cooling and makeup systems.

13.A.25.2 Prerequisites

a. 480V, 120V AC and 125V DC electrical systems are available.

b. Service water system is functional.

c. Water treatment system is functional.

d. Reactor building closed cooling water system is functional.

e. Instrument air system is functional.

f. Spent fuel pool and refueling pool cooling and cleanup system is operable.

13.A.25.3 Method

a. Line up the system for operation.

b. Adjust alarms, interlocks and recorders.

13.A.25.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer's specifications.

b. Alarms and interlocks function properly.

c. Ensure cooling, skimming and purification flow paths are as designed.

d. Ensure emergency cooling and makeup flow paths are as designed. These flow paths consist of the following:

1. Makeup from refueling water storage tank.

2. Makeup from condensate storage tank.

3. Emergency cooling by way of the low-pressure safety injection pumps and shutdown heat exchangers.

13.A.26 SYSTEM: REACTOR BUILDING CLOSED COOLING WATER SYSTEM

13.A.26.1 Objective

To demonstrate that the reactor building closed cooling water system is capable of providing adequate cooling water to systems and components containing radioactive or potentially radioactive fluids during reactor operation and after shutdown.

13.A.26.2 Prerequisites

a. 4.16 kV, 480V AC and 125V DC power supplies are available.

b. Water treatment system is functional.

c. Instrument air system is functional.

d. Chemical addition tank is available.

e. Reactor building closed cooling water system is operable.

13.A.26.3 Method

a. Line up the system for normal operation.

b. Adjust alarms, interlocks and recorders.

c. Operate the system on all possible modes of operation.

d. Simulate safety injection actuation and sump recirculation signals.

e. Surge tank level should be carefully watched.

13.A.26.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer's specification.

b. Alarms and interlocks function properly.

c. The system responds to SIAS and SRAS signals.

d. Flow capability of the system is according to design.

e. Surge tank level remains steady during any particular mode of operation.

13.A.27 FUEL HANDLING SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.27.1 Purpose

To conduct a complete checkout of the fuel handling equipment including operation of the system over its full range of capability.

13.A.27.2 General Prerequisites

a. Refueling equipment installation and installation checks have been completed.

b. The reactor vessel internals with the exception of the fuel and the upper guide structure are installed in the reactor.

c. The reactor vessel head is removed and the refueling cavity is capable of being filled.

d. Refueling equipment has been checked to insure that all components are captured to prevent inadvertent introduction of any part of the equipment to the reactor vessel.

e. The refueling machine alignment has been completed.

13.A.27.3 Test Method

a. Operate the refueling machine using manual drive and normal power drive while checking all interlocks and controls on the bridge, trolley, and hoist.

b. Operate the transfer system in manual drive and normal power drive while checking all interlocks and controls.

c. Use the refueling equipment to handle a dummy fuel bundle or equivalent from the spent fuel storage area into the reactor and verify selected positions and orientation of the dummy when installed in the core support barrel.

13.A.27.4 Acceptance Criteria

The fuel handling equipment performs in accordance with the Fuel Handling Technical Manual.

13.A.28 SAFETY INJECTION SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.28.1 Purpose

To demonstrate the proper functioning of SIS components prior to performing the integrated Pre-Core Hot Functional Test.

13.A.28.2 General Prerequisites

a. Construction, flushing and hydrostatic testing of the SIS has been completed.

b. The SIS has been released for operation.

c. Installation and calibration checks of SIS instrumentation have been completed.

d. Cycling of manual valves has been performed to demonstrate freedom of movement and proper indication where appropriate.

e. The SIS has been lined up for normal operation in accordance with the SIS Piping and Instrumentation Diagram.

f. The RCS and the reactor cavity are ready to receive water.

g. Initial run-in and operation of SIS equipment has been performed where possible.

h. The refueling water storage tank contains at least 450,000 gallons of Grade A water.

i. The plant computer and those input sections associated with the SIS are operational.

13.A.28.3 Test Method

a. Operate all actuator operated valves in the SIS from the appropriate control locations and verify proper operations, indications, and fail positions.

b. Operate low- and high-pressure safety injection pumps under both recirculation flow and full-flow conditions.

c. Fill, pressurize and discharge the safety injection tanks.

d. Operate the shutdown cooling system.

e. Verify satisfactory operation of any alarm and interlock circuitry not verified during the previous operations.

13.A.28.4 Acceptance Criteria

The SIS system components operate in accordance with the FSAR description, the SIS system description, and the appropriate vendor manual.

13.A.29 SYSTEM: DIESEL FUEL OIL TRANSFER SYSTEM

13.A.29.1 Objective

To demonstrate that the diesel fuel oil system is capable of transferring adequate quantity of fuel oil from underground diesel oil storage tank to diesel oil day-supply tank.

13.A.29.2 Prerequisites

a. 480V AC and 125V DC electrical systems are available.

b. Fire protection water system is functional.

13.A.29.3 Method

a. Line up the system for operation.

b. Adjust alarms and interlocks.

13.A.29.4 Acceptance Criteria

a. Alarms and interlocks function properly.

b. Fuel transfer capability of the system is according to design.

13.A.30 SYSTEM: DIESEL GENERATORS

13.A.30.1 Objective

To demonstrate the capability of the Emergency Diesel Generators to start, accept load, and perform according to design. This will include a test of the engine controls as well as the generator controls.

13.A.30.2 Prerequisites

a. 480V, 120 VAC and 125 VDC power available.

b. Ventilation in Diesel Room operational.

c. Mechanical checks of engine completed.

d. Engine auxiliaries and support systems operational.

e. Generator and electrical tests of regulator and exciter completed.

f. Relays calibrated and all normal bus protection checked for operation and in service.

13.A.30.3 Method

a. Engine controls are operated to verify capability to control engine in all modes of operation.

b. Generator controls are operated to verify capability of responding to under voltage and engineering safeguard system signals.

c. Circuit breakers and protective relaying respond to distribution system signals.

13.A.30.4 Acceptance Criteria

a. Exciter and Field Control

1. Regulator functions to regulate and maintain voltage in all modes of operation.

2. Generator local and remote controls function according to design.

3. Engine and support systems function according to design.

4. Emergency diesel generators respond to engineering safeguard signal, bus undervoltage signal, and maintains safeguard buses on loss of offsite power.

13.A.31 SYSTEM: CONTAINMENT SPRAY SYSTEM

13.A.31.1 Objective

Demonstrate that the containment spray system will respond to a containment spray actuation signal (CSAS) as designed. Demonstrate spray nozzles are open.

13.A.31.2 Prerequisites

a. Refueling water storage tank operable.

b. Required electrical systems operable.

13.A.31.3 Method

a. Line up system to recirculate to the refueling water storage tank.

b. Initiate a simulated containment spray actuation signal.

c. Force air or smoke through each spray header.

13.A.31.4 Acceptance Criteria

a. Pump characteristic curves are in accordance with manufacturer’s specifications.

b. System must respond as designed to simulated containment spray actuation signal.

c. Spray nozzle flow paths are open.

13.A.32 SYSTEM: SHUTDOWN COOLING SYSTEM

13.A.32.1 Test Objectives

To demonstrate that the shutdown cooling system and components have the capacity to maintain the design flow.

13.A.32.2 Prerequisites

a. 4.16-kV, 480V AC, 125V DC power supplies available.

b. Instrument air system functional.

c. RBCCW system operational.

d. RX coolant system operational.

13.A.32.3 Method

a. Line up the system for normal operation.

b. Adjust alarms, interlocks and recorders.

c. Measure flow through heat exchangers.

d. Check operation of auto flow controller.

13.A.32.4 Acceptance Criteria

a. Pump/characteristic curves are in accordance with manufacturer's specification.

b. Alarms and interlocks function properly.

c. Flow of the system according to design.

13.A.33 SYSTEM: ENGINEERED SAFEGUARDS ACTUATION SYSTEM

13.A.33.1 Objectives

Demonstrate that the engineered safeguards actuating system will, when initiated, function as designed to protect against the accidental release of radioactivity to the atmosphere.

13.A.33.2 Prerequisites

a. Required electrical systems operational.

b. All systems with engineered safeguard actuation signals as control must be operable.

13.A.33.3 Method

a. Place all safety-related systems in normal (depending on plant conditions at the time the test is conducted) operating mode.

b. Initiate signal from engineered safeguards actuating system.

13.A.33.4 Acceptance Criteria

a. Upon initiation of an engineered safeguards actuation signal, the following response is required:

1. All system components (pump, fan, etc.) must respond to the signal as designed.

2. All valves must respond to the signal as designed.

13.A.34 REACTOR COOLANT SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.34.1 Test Objective

To demonstrate that the RCS components are operating properly prior to performing the integrated Pre-Core Hot Functional Test.

13.A.34.2 General Prerequisites

a. The RCS construction, flushing and cleaning is completed.

b. The RCS has been released for operations.

c. The reactor internals with the exception of the fuel and the upper guide structure are installed.

d. The reactor vessel head is removed and the refueling cavity is ready to be filled.

e. Preoperational tests on all necessary auxiliary systems have been completed.

f. Installation and calibration checks of RCS instrumentation have been completed.

g. Pressurizer safety valves have been set in accordance with Technical Specifications.

h. Cycling of manual operated valves has been performed to demonstrate freedom of movement and proper indication where applicable.

i. RCS valves are lined up for normal operation in accordance with the RCS Piping and Instrumentation Diagram.

j. Pressurizer heater breakers and reactor coolant pump breakers are in the Test position.

k. Initial run-in and operation of RCS auxiliary equipment has been performed where possible.

l. The plant computer and the input section concerned with the RCS are operational.

13.A.34.3 Test Method

a. Operate all actuator operated valves in the RCS from applicable control locations and check the fail positions of these valves.

b. Operate the reactor coolant pump auxiliary systems to verify proper operation and to demonstrate that the associated process instrumentation interlocks and controls are operating.

c. Operate the quench tank to demonstrate operation of level, pressure, and temperature instrumentation.

d. Test process instrumentation to verify setpoints and indications.

e. Operate pressurizer pressure and level control systems to demonstrate proper operation.

13.A.34.4 Acceptance Criteria

All RCS valves, instrumentation, and auxiliary systems perform in accordance with the FSAR description, the RCS system description, and appropriate vendor manuals.

13.A.35 REACTOR REGULATING SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.35.1 Purpose

To insure that the Reactor Regulating System has been properly installed and functions as designed prior to actual reactor operation.

13.A.35.2 General Prerequisites

a. The Reactor Regulating System has been installed and has been thoroughly inspected for damaged or loose components and other discrepancies.

b. All associated interconnecting wiring has been completed and has been checked for continuity, shorts, grounds, insulation resistance, etc.

c. All process instrumentation channels providing inputs to the Reactor Regulating System have been checked and calibrated.

d. The CEDS and power range control channels have been installed and calibrated.

e. The results of all factory testing on the Reactor Regulating System are available.

13.A.35.3 Test Method

a. Energize equipment.

b. Perform a static test of the Reactor Regulating System.

c. Perform the dynamic test of the Reactor Regulating System.

13.A.35.4 Acceptance Criteria

The Reactor Regulating System performs in accordance with the FSAR description, the Reactor Regulating System description, and the appropriate vendor manuals.

13.A.36 REACTOR PROTECTIVE SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.36.1 Purpose

a. To demonstrate that the RPS has been properly installed and functions as designed in all modes of operation.

b. To demonstrate that the RPS is properly interfaced with the nuclear instrumentation, the protective system process instrumentation, and the control element drive mechanism power supply.

13.A.36.2 General Prerequisites

a. The RPS has been installed in the control room, all external connections have been made, and the complete RPS assembly has been thoroughly inspected for damage or loose components and other discrepancies.

b. All external wiring to the RPS has been checked for continuity, short circuits, grounds, insulation resistance, etc.

c. All NSSS parameter measurement channels which provide input signals to the RPS have been installed and are operable.

d. Both CEDM motor generators (MG) have been installed and are ready for operation.

e. Main control board reactor trip and annunciator buttons have been installed and are operable.

f. The control circuitry for the following interlocks and actuations has been installed and is operable: CEA withdrawal prohibits, automatic withdrawal prohibits, diesel generator start, power-operated relief valve actuation, turbine runback actuation, and turbine trip.

13.A.36.3 Test Method

a. Energize equipment and verify operation of indicating lights, alarms, and proper voltages.

b. Perform system conditioning on the nuclear instrumentation by checking the indication of the systems when operating the function switches.

c. Perform system conditioning on the bistable trip units utilizing the trip tester.

d. Perform measurement channel functional tests on all parameters providing trip singles to the RPS.

e. Perform an overall system functional check verifying proper operational trip setpoints, pretrip setpoints, bistable calibrations, output of the Reactor Protective System, and the operation of all BYPASS switches.

13.A.36.4 Acceptance Criteria

The RPS and all associated equipment perform in accordance with the FSAR description, the RPS system description, and appropriate vendor manuals.

13.A.37 CHEMICAL AND VOLUME CONTROL SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.37.1 Test Objective

To demonstrate that the CVCS components are operating properly prior to performing the integrated Pre-Core Hot Functional Test.

13.A.37.2 General Prerequisites

a. CVCS construction, flushing and hydrostatic tests have been completed.

b. The CVCS has been released for operation.

c. Installation and calibration checks of process instrumentation have been completed.

d. Cycling of manual valves has been performed to demonstrate freedom of movement and proper indication where applicable.

e. CVCS valve lineup for normal operation has been performed in accordance with the CVCS P&ID.

f. The RCS or an alternate flow patch is ready to receive water from the CVCS.

g. Initiate run-in and operation of CVCS components has been performed where possible.

h. The plant computer and those input sections concerned with the CVCS are operational.

13.A.37.3 Test Method

a. Operate all actuator-operated valves in the CVCS from the appropriate control locations and check the fail positions of these valves.

b. Check the performance of the primary makeup system including the boric acid batching tank, concentrated boric acid pumps, and the concentrated boric acid storage tanks. This section will also include gravity feed checks of the system.

c. Check the performance of the volume control tank including process instrumentation, level indication, alarms and control systems. Check the performance of the volume control tank nitrogen and hydrogen pressurization systems.

d. Check the performance of the charging pumps and auxiliary systems.

e. Check the performance of all other CVCS temperature, pressure and flow controls and interlocks.

13.A.37.4 Acceptance Criteria

The CVCS performs in accordance with the FSAR description, the CVCS system description, and appropriate vendor manuals.

13.A.38 NUCLEAR INSTRUMENTATION SYSTEM PREOPERATIONAL TEST - DESCRIPTION

13.A.38.1 Purpose

To demonstrate that the nuclear instrumentation system has been properly installed and functions as designed.

13.A.38.2 General Prerequisites

a. The nuclear instrumentation system has been installed.

b. Installation of preamplifiers, cables, connectors, and detectors has been completed and all wiring has been checked for discontinuities, shorts, grounds, and insulation resistance.

13.A.38.3 General Test Methods

a. Perform resistance measurements on all detectors and associated cabling including preamplifiers.

b. Perform alignment checks for the wide-range nuclear instrumentation.

c. Check response of the wide-range nuclear instrumentation to the startup source.

d. Perform an alignment check of the power- range nuclear instrumentation.

e. Using simulated input signals, verify the proper performance of both wide-range and power-range nuclear instrumentation including output signals to the reactor regulating system, reactor protective system, and remote meters.

13.A.38.4 Acceptance Criteria

The nuclear instrumentation system performs in accordance with the FSAR description, the nuclear instrumentation system description and appropriate vendor manuals.

13.A.39 IN-CORE INSTRUMENTATION PREOPERATIONAL TEST - DESCRIPTION

13.A.39.1 Purpose

To verify the proper wiring of the signal cabling for the in-core instrumentation and to verify the detector assembly resistance values.

13.A.39.2 General Prerequisites

a. Cabling between the refueling disconnect panel and the plant computer has been run and has been checked for continuity, shorts, grounds, and insulation resistance.

b. The plant computer has been checked out and programmed for the in-core program with the appropriate constants entered.

c. All in-core detector assemblies have been received and are available for checkout.

13.A.39.3 Test Method

a. Measure the insulation resistance of each cable between the refueling disconnect panel and the plant computer.

b. Obtain background readings for all cables by use of the plant computer.

c. Using a simulated input check the computer readout for each detector assembly.

d. Measure the insulation resistance of each detector assembly.

13.A.39.4 Acceptance Criteria

Measurements made during this test agree with the specifications for the in-core instrumentation and in-core instrumentation wiring.

13.A.40 SYSTEM: PROCESS RADIATION MONITORING SYSTEMS

13.A.40.1 Objective

To check and adjust the system, fluid flows, alarms, self checking features and interlock functions. To check calibration and calibrate the system to primary radioactivity calibration data.

13.A.40.2 Prerequisites

a. Radiation monitoring system operable.

b. Process systems being monitored must be in operation before final adjustments can be made.

c. Required electrical systems operable.

13.A.40.3 Method

a. Using secondary standard radioactive sources, calibrate output to primary radioactivity calibration data.

b. Adjust alarms, interlocks and recorders.

c. Verify sampling valve sequences and self checking functions.

13.A.40.4 Acceptance Criteria

a. Outputs must be in agreement with primary calibration data.

b. Proper sampling, alarm, interlock and self checking functions must operate as designed.

13.A.41 SYSTEM: AREA RADIATION MONITORING SYSTEM

13.A.41.1 Objective

To determine the system alarms, self checking features and interlocks are functional. Also to verify manufacturer’s initial calibration.

13.A.41.2 Prerequisites

a. Radiation monitoring system operable.

b. Required electrical systems operable.

13.A.41.3 Method

a. Using standard radiation source check detectors calibration.

b. Check alarms, self-checking features, interlocks and recorders.

13.A.41.4 Acceptance Criteria

a. Detector outputs must be in agreement with standard source radiation levels.

b. System must demonstrate proper alarm, interlock and self checking functions.

13.A.42 BORONOMETER AND PROCESS RADIATION MONITOR PREOPERATIONAL TEST - DESCRIPTION

13.A.42.1 Purpose

To verify the proper installation, calibration, and operation of the boronometer and process radiation monitor.

13.A.42.2 General Prerequisites

a. Installation of the boronometer and process radiation monitor have been completed.

b. All wiring to the boronometer and process radiation monitor have been checked for continuity, short circuits, grounds, and insulation resistance.

c. All associated remote meters and wiring the control room have been installed and are operational.

d. The RCS has been filled.

13.A.42.3 Test Methods

a. Verify proper operation of the process radiation monitor using the electrical test switches and the check source button.

b. Adjust the alarm setpoints of the process radiation monitor to the proper setpoints.

c. Verify calibration of the boronometer using the electrical test switches.

d. Verify proper operation of the boronometer using a standardized solution of borated reactor coolant-grade water.

13.A.42.4 Acceptance Criteria

The boronometer and process radiation monitor perform in accordance with the system description and the technical manuals.

13.A.43 SYSTEM: MAIN AND NORMAL STATION SERVICE TRANSFORMERS

13.A.43.1 Test Objectives

To demonstrate the capability of the main and normal station service transformers to supply electrical power to the 345 kV switchyard and plant buses. To verify operation of protective devices, calibration and functional operation of controls and interlocks.

13.A.43.2 Prerequisites

a. All meters, relays and protective devices calibrated and tested.

b. 125V DC and 480V AC available.

c. All erection work on transformers and switchgear complete.

d. Transformer oil and gas systems tested and in service.

e. Isolated phase bus tested and ready for service.

f. Breaker controls and transfer scheme verified.

g. PT & CT circuits checked for polarity and continuity.

13.A.43.3 Method

a. Simulate signals to temperature controls and verify operation of transformer oil pumps and fans.

b. Simulate signals to verify annunciators for transformer protective devices.

c. Energize transformers from the 345 kV switchyard.

d. Verify transfer to reserve station service transformer when normal station service transformer is deenergized.

13.A.43.4 Acceptance Criteria

a. Transformers are capable of supplying power to plant loads and switchyard.

b. Normal station service transformer loads transfer to the reserve station service transformer on a turbine trip.

13.A.44 SYSTEM: ENCLOSURE BUILDING FILTRATION AND HYDROGEN REMOVAL SYSTEM

13.A.44.1 Objective

To demonstrate that the enclosure building filtration and H_2 removal system is capable of minimizing environmental activity levels resulting from all sources of containment leakage and maintaining H_2 concentration in the containment below the lower flammability limit following any postulated incident.

13.A.44.2 Prerequisites

a. 480V, 120VAC and 125VDC power sources are available.

b. Instrument and service air systems are functional.

c. Enclosure building filtration and H_2 removal system is operable.

13.A.44.3 Method

a. Line up the system for operation.

b. Adjust alarms, recorders and interlocks.

c. Simulate enclosure building filtration actuation signal and auxiliary exhaust actuation signal.

d. Simulate high H_2 concentration in the containment.

e. Simulate containment isolation actuation signal.

13.A.44.4 Acceptance Criteria

a. Alarms, recorders and interlocks function properly.

b. The system responds to different actuating signals as designed.

c. The system performance meets design requirement.

d. The fans and dampers must respond according to the containment isolation activation signal.

13.A.45 CEDS SIMULATOR BOX OPERATION - DESCRIPTION

13.A.45.1 Purpose

To verify the proper installation and operation of the CEDS prior to performing the Post-Core Hot Functional Test.

13.A.45.2 General Prerequisites

a. The CEDS and the RRS have been installed and are operational.

b. All interconnecting cabling between the CEDS and the refueling panel disconnect has been installed and has been checked for continuity, shorts, grounds, and insulation resistance.

c. The CEDS test box and the reed switch test box are available.

d. A high-speed optical recorder has been properly set up for use in reading out the coil power arrangements.

13.A.45.3 Test Methods

a. Verify the proper operation for each CEA of the secondary position indication and limit switch actuation using the reed switch test box.

b. Using the CEDS test box, verify proper operation for each CEA of the CEDS and the primary position indication system.

c. Verify proper group operation for each group of CEAs by observing CEA digital displays and pulse counters at CEDS panels.

d. Verify proper sequential operation of the regulating groups of CEAs by observing CEA digital displays and pulse counters at CEDS panels.

13.A.45.4 Acceptance Criteria

The CEDS performs in accordance with the FSAR description, the CEDS system description, and the appropriate technical manuals.

13.A.46 SYSTEM: SOLID RADWASTE SYSTEMS

13.A.46.1 Test Objectives

To demonstrate the capability of the solid radwaste drumming system to operate as per design criteria.

13.A.46.2 Prerequisites

a. Solid radwaste drumming equipment operable.

b. 480 volt power supply available to solid radwaste area.

c. Actual shielding in phase.

13.A.46.3 Method

a. Line up equipment for normal operation.

b. Fill a 55 gallon drum with simulated waste material.

c. Mix and pour concrete shield plug.

d. Cap waste barrel.

13.A.46.4 Acceptance Criteria

a. All equipment operates as per design specifications.

b. Operation can be performed in a timely manner.

13.A.47 SYSTEM: CONTAINMENT BUILDING INTEGRATED LEAK RATE TEST

13.A.47.1 Objective

Demonstrate that the potential leakage from the containment building is within acceptable limits, under the conditions stated below.

13.A.47.2 Prerequisites

a. Penetrations are installed.

b. Penetration local leak tests must be completed before the integrated leak rate test.

c. Reactor building closed cooling water system functional.

13.A.47.3 Method

a. Penetration local leak tests by pressurizing between the appropriate boundaries and observing the pressure decay.

b. The containment building leak rate will be determined at calculated peak accident pressure and at one-half calculated peak accident pressure.

13.A.47.4 Acceptance Criteria

The leak rates must not exceed allowable limits.

13.A.48 SYSTEM: CONTAINMENT BUILDING STRUCTURAL INTEGRITY TEST

13.A.48.1 Objective

Demonstrate the structural integrity of the containment building at 115% of design pressure.

13.A.48.2 Prerequisites

Completion of containment penetrations local leak tests.

13.A.48.3 Method

The containment building will be pressurized in steps up to a maximum of 115% of design pressure. At selected pressure levels, data will be recorded and inspections will be made to verify the structural integrity. After remaining at 115% of design pressure for approximately one hour, depressurization will begin. Data test, visually inspect containment building, hatched, penetrations and gaskets.

13.A.48.4 Acceptance Criteria

Structure must be capable of withstanding an internal pressure of 1.15 times the design pressure.

13.A.49 SYSTEM: CONTAINMENT PENETRATION VENTILATION SYSTEM

13.A.49.1 Objective

To demonstrate the containment penetration ventilation system is capable of supplying the required amount of air as designed.

13.A.49.2 Prerequisites

a. Required electrical systems operational.

13.A.49.3 Method

a. Line up system and put into normal operation.

13.A.49.4 Acceptance Criteria

a. Fans must meet the design criteria.

13.A.50 SYSTEM: CONTAINMENT PURGE SYSTEM

13.A.50.1 Objectives

To demonstrate the capability of the containment purge system to operate under normal conditions and to respond to a CIAS.

13.A.50.2 Prerequisites

a. Instrument air system operational.

b. Required electrical systems operational.

13.A.50.3 Method

a. Line up system and place in normal operation.

b. Initiate a simulated containment isolation actuation signal (CIAS).

c. Initiate a simulated high-radiation signal.

13.A.50.4 Acceptance Criteria

a. The fans and dampers must respond to the CIAS as designed.

b. The dampers must respond to the high-radiation signal as designed.

13.A.51 SYSTEM: CONTROL ELEMENT DRIVE ASSEMBLY VENTILATION

13.A.51.1 Objective

Demonstrate that the control element drive assembly ventilation system is capable of providing adequate ventilation to the control element drive assembly.

13.A.51.2 Prerequisites

a. Required electrical systems functional.

13.A.51.3 Method

a. Line up system and place in normal operation.

13.A.51.4 Acceptance Criteria

a. The fans and dampers must respond as designed.

b. The fans must meet design criteria.

13.A.52 SYSTEM: CONTAINMENT AIR RECIRCULATION SYSTEM

13.A.52.1 Objectives

To demonstrate the capability of the containment air recirculation system to operate in accordance to design and to recirculate the required amount of air.

13.A.52.2 Prerequisites

a. Required electrical systems available.

13.A.52.3 Method

a. Line up system for normal operation.

13.A.52.4 Acceptance Criteria

a. Fans and dampers must operate in accordance with design.

13.A.53 SYSTEM: CONTROL ROOM AIR CONDITIONING SYSTEM

13.A.53.1 Objectives

To demonstrate that the control room air conditioning system is capable of providing a controlled environment during normal and abnormal conditions.

13.A.53.2 Prerequisites

a. Instrument air system functional.

b. Heating and cooling coils functional.

c. Verify that particulate filters are installed.

d. Charcoal and absolute filters are not to be installed during testing.

13.A.53.3 Method

a. Line up system for normal operation.

b. Simulate smoke detection signal and recirculation signal.

13.A.53.4 Acceptance Criteria

a. The dampers and fans must respond to smoke detection and recirculation signals as designed.

b. Heating and cooling coils must meet design specifications.

13.A.54 SYSTEM: NON-RADIOACTIVE VENTILATION SYSTEM

13.A.54.1 Objective

To demonstrate that the ventilation system to the 480V unit substation, controlled access area, cable vault, battery rooms and switchgear rooms is capable of providing adequate ventilation to each room.

13.A.54.2 Prerequisites

a. Instrument air system available.

b. Required electrical systems functional.

13.A.54.3 Method

a. Line up system and place in normal operation.

13.A.54.4 Acceptance Criteria

a. The fans and dampers must respond as designed.

13.A.55 SYSTEM: DIESEL GENERATOR VENTILATION SYSTEM

13.A.55.1 Objective

To demonstrate that the diesel generator ventilation system will operate during normal or abnormal conditions.

13.A.55.2 Prerequisites

a. Required electrical systems functional.

b. Instrument air system available.

13.A.55.3 Method

a. Line up system for normal operation.

b. Simulate diesel generator start signal.

13.A.55.4 Acceptance Criteria

a. Fans and dampers must respond as designed in response to diesel generator start.

13.A.56 SYSTEM: MAIN EXHAUST SYSTEM

13.A.56.1 Objectives

To demonstrate the capability of the main exhaust system to discharge the proper amount of air from the plant in accordance with design.

13.A.56.2 Prerequisites

a. Required electrical systems operational.

13.A.56.3 Method

a. Line up system and place in normal operation.

13.A.56.4 Acceptance Criteria

a. All fans must operate to design criteria.

13.A.57 SYSTEM: FUEL HANDLING AREA VENTILATION SYSTEM

13.A.57.1 Objective

To demonstrate that the fuel handling area ventilation system is capable of providing adequate ventilation during normal and abnormal conditions.

13.A.57.2 Prerequisites

a. Instrument air system available.

b. Required electrical systems functional.

13.A.57.3 Method

a. Line up system and place in normal operation.

b. Initiate simulated auxiliary exhaust actuation signal.

13.A.57.4 Acceptance Criteria

a. Fans and dampers must respond as designed.

b. In normal operation, fans must meet design criteria.

13.A.58 SYSTEM: RADWASTE VENTILATION SYSTEM

13.A.58.1 Objective

To demonstrate that the radwaste ventilation system is capable of providing adequate ventilation to the radwaste area during normal or abnormal operating conditions.

13.A.58.2 Prerequisites

a. Instrument air system available.

b. Required electrical systems functional.

13.A.58.3 Method

a. Line up system and place in normal operation.

b. Simulate an enclosure building filtration actuation signal.

13.A.58.4 Acceptance Criteria

a. The fans and dampers must respond as designed.

13.A.59 SYSTEM: PRIMARY CHEMICAL ADDITION SYSTEM

13.A.59.1 Objective

To demonstrate that the system will function in accordance with design specifications.

13.A.59.2 Prerequisites

a. Required electrical systems functional.

b. Water treatment system functional.

13.A.59.3 Method

a. Line up system and place in normal operation.

13.A.59.4 Acceptance Criteria

a. Verify that the chemical addition metering pumps start and stop manually.

b. Ensure valves operate properly.

13.A.60 SYSTEM: POST-INCIDENT RECIRCULATION SYSTEM

13.A.60.1 Objective

To demonstrate the capability of the post-incident recirculation system to provide containment air recirculation as designed.

13.A.60.2 Prerequisites

a. Required electrical systems operational.

13.A.60.3 Method

a. Line up system and place in normal operation.

13.A.60.4 Acceptance Criteria

a. Fans must operate in accordance to design criteria.

13.A.61 REACTOR COMPONENTS HANDLING SYSTEM

13.A.61.1 Purpose

To conduct a checkout of the reactor component handling system in accordance with the guidelines provided in Section 4 of the Millstone Unit 2 Reactor Internals Instruction Manual and Section 5 of the Millstone Unit 2 Reactor Vessel Assembly Instruction Manual Volume 1.

13.A.61.2 General Prerequisites

a. The reactor vessel and the vessel internals are installed as necessary to complete the various phases of testing.

b. The polar crane will be fully tested before any reactor components are fitted.

c. The reactor vessel head is removed and the refueling cavity is capable of being filled.

13.A.61.3 Test Method

a. Operate the polar crane in all modes and load test the main and auxiliary hoist and hooks.

b. Operate the polar crane in conjunction with various tools to accomplish the following:

1. Coupling and uncoupling of CEA extension shafts.

2. Insertion and withdrawal of in-core detector and installation and removal of the Upper Guide Structure.

3. Installation and removal of core support barrel assembly.

4. Removal of surveillance samples.

5. Installation and removal of neutron source and instrument thimbles.

6. Installation and removal of the reactor vessel head.

13.A.61.4 Acceptance Criteria

The reactor component handling equipment performs in accordance with the Reactor Internals and Reactor Vessel Assembly Instruction Manuals.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 14: Safety Analysis

CHAPTER 14—SAFETY ANALYSIS

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14.8.2–9	Main Steam Line Break Analysis - 102% Power with Loss of Offsite Power and Failure of Vital Bus Cabinet VA-10 or VA-20 - Affected Steam Generator Liquid Mass vs. Time
14.8.3–1	Deleted by FSARCR 04-MP2-018
14.8.3–2	Deleted by FSARCR 04-MP2-018
14.8.3–3	Deleted by FSARCR 04-MP2-018
14.8.3–4	Deleted by FSARCR 04-MP2-018

NOTE: REFER TO THE CONTROLLED PLANT DRAWING FOR THE LATEST REVISION.

CHAPTER 14-SAFETY ANALYSIS

List of Figures (Continued)

Number	Title
14.8.3–5	Deleted by FSARCR 04-MP2-018
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CHAPTER 14 – SAFETY ANALYSIS

14.0 GENERAL

Preceding sections of this report describe and evaluate the reliability of major systems and components of the plant. The purpose of this section is to assume that certain accidents occur notwithstanding the precautions taken to prevent their occurrence and to evaluate the capability of the installed safety equipment to mitigate the potential consequences of such accidents. The analyses show that the health and safety of the public are assured in the event of even the most severe of the hypothetical accidents analyzed.

All events described in the Standard Review Plan (SRP) (Reference 14.0-1) have been reviewed and placed (dispositioned) into one of the following four categories:

1. The event needs to be analyzed.

2. The event is bounded by another event which is analyzed.

3. The event is not in the licensing basis for the plant.

4. The event is not applicable to Millstone Unit 2.

In the event disposition, all of the reactor operating conditions allowed by the plant Technical Specifications (Reference 14.0-2) are examined to ensure that the bounding subevents are identified for each SRP event category. This ensures that the safety analysis will support the complete range of allowable operating conditions. Events which are not bounded by other events or by existing accepted analyses, and are in the plant licensing basis, are dispositioned to be analyzed. In the event disposition process, the event initiator is identified for each event. The magnitude of the initiator for each event is calculated and compared to the magnitude of the initiator for other events. The comparison basis includes all the plant operating conditions. This allows, in several cases, a ranking of the event initiators as to severity, allowing the lesser events to be dispositioned as bounded by the greater event. Similar logic is applied in determination of the applicability and bounding nature for existing accepted analyses.

The reactor operating modes allowed for Millstone Unit 2 by the plant Technical Specifications are listed in Table 14.0-1. Table 14.0-2 presents a summary of results of the event disposition. This chapter presents the basis and justification for the disposition of events, and analysis of those events dispositioned as requiring analysis.

14.0.1 CLASSIFICATION OF PLANT CONDITIONS

Plant operations are placed in one of four categories. The categories are:

1. Normal Operations and Operational Transients - Events which are expected to occur frequently in the course of power operation, refueling, maintenance, or plant maneuvering.

2. Faults of Moderate Frequency - Events which may occur once per year during plant operation.

3. Infrequent Faults - Events which may occur once during the lifetime of the plant.

4. Limiting Faults - Events which are not expected to occur but which are evaluated to demonstrate the adequacy of the design.

14.0.1.1 Acceptance Criteria

The acceptance criteria for the four categories of events are as given below:

1. Operational Events

   This condition describes the normal operational modes of the reactor. As such, occurrences in this category must maintain margin between operating conditions and the plant setpoints. The setpoints are established to assure maintenance of margin to design limits. The set of operating conditions, together with conservative allowances for uncertainties, establish the set of initial conditions for the other event categories.

2. Moderate Frequency Events

   a. The pressures in reactor coolant and main steam systems should be less than 110% of design values.

   b. The fuel cladding integrity should be maintained by ensuring that fuel design limits are not exceeded. That is, the minimum calculated departure from nucleate boiling ratio (DNBR) is not less than the applicable limits of the DNBR correlation being used.

   c. The radiological consequences should be less than 10 CFR 20, Sections 105 and 106 and Appendix B (version prior to January 1, 1994).

   d. The event should not generate a more serious plant condition without other faults occurring independently.

3. Infrequent Faults

   a. The pressures in reactor coolant and main steam systems should be less than 110% of design values.

   b. A small fraction of fuel failures may occur, but these failures should not hinder the capability of the core to be cooled.

c. The radiological consequences should be within the guidelines of 10 CFR 50.67 and Regulatory Guide 1.183.

d. The event should not generate a limiting fault or result in the consequential loss of the reactor coolant or containment barriers.

4. Limiting Fault Events

a. Radiological consequences should be within the guidelines of 10 CFR 50.67 and Regulatory Guide 1.183.

b. The event should not cause a consequential loss of the required functions of systems needed to cope with the reactor coolant and containment systems transients.

c. Additional criteria to be satisfied by specific events are:

1. Loss-of-Coolant Accident (LOCA) - 10 CFR 50.46 and Appendix K.

2. Rod Ejection - Radially averaged fuel enthalpy deposition < 280 cal/gm.

14.0.1.2 Classification of Accident Events by Category

Table 14.0.1-1 lists the accident category used for each event analyzed in this report. This classification is used in evaluating the acceptability of the results obtained from the analysis.

14.0.2 PLANT CHARACTERISTICS AND INITIAL CONDITIONS

Six operational modes have been considered in the analysis and are shown in Table 14.0-1. These operational modes have been considered in establishing the subevents associated with each event initiator. A set of initial conditions is established for the events analyzed that is consistent with the conditions for each mode of operation.

The nominal plant rated operating conditions are presented in Table 14.0.2-1 and principal fuel design characteristics in Table 14.0.2-2. The uncertainties used in the accident analysis applicable to the operating conditions are:

(1) Core Power, HFP Calorimetric	± 2%
(2) Primary Coolant Cold Leg Temperature	± 2.25°F
(3) Primary Coolant Pressure	+25/-14 psi
(4) Primary Coolant Flow	± 4%

14.0.3 POWER DISTRIBUTION

The Technical Specification (Reference 14.0-2) power peaking factors are used in the accident analysis. The Technical Specification Limiting Conditions of Operation (LCO) assure that the power distribution is maintained within these limits during normal operation.

14.0.4 RANGE OF PLANT OPERATING PARAMETERS AND STATES

Table 14.0.4-1 presents the range of key plant operating parameters considered in the analysis. A broader range of power, core inlet temperature, and primary pressure is considered in establishing the trip setpoints verified by the analysis results presented in this document.

The range of operating states of the reactor is also considered in the analysis. The effect of exposure on fuel thermal performance and neutronics parameters is considered. State values are selected for the event analyzed to provide the greatest challenge to the acceptance criteria for that event. Several calculations may be required to bound the range of the state variable. For example, a range of neutronic parameters is used in the analysis of rod withdrawal events in order to verify the range of protection of the challenged trip setpoints.

The range of initiating events is also considered in formulating the analysis conditions for an event. The initiating conditions are examined to identify a set which conservatively challenges the acceptance criteria. Where not obvious, sensitivity studies are performed. For example, analyses are performed for uncontrolled rod withdrawal events throughout the range of reactivity insertion rate possible from boron dilution to maximum withdrawal rate of the highest worth control banks.

14.0.5 REACTIVITY COEFFICIENTS USED IN THE SAFETY ANALYSIS

The reactivity coefficients used in the analysis are consistent with the AREVA approved methodology and the Technical Specification limits. The set of parameters used in each analysis is listed in the appropriate section for that event.

14.0.6 SCRAM INSERTION CHARACTERISTICS

Scram reactivity insertion as a function of axial shape index (ASI) was used in the analysis for reactor trip. The insertion worth includes the most reactive rod stuck out. The shutdown margin of 3.6% delta rho and a control rod drop time of 2.75 seconds (to 90% insertion) have been supported by the transient analysis.

14.0.7 TRIP SETPOINT VERIFICATION

Operating limits for the Millstone Unit 2 nuclear plant are summarized below. Methods of analysis for determining or verifying the operating limits are detailed in Section 14.0.7.5 and Reference 14.0-4. Axial power distributions and other core neutronics related parameters used in the setpoint verification analyses were generated with AREVA approved core simulator code PRISM (Reference 14.0-5). This data was generated on a three-dimensional core basis, as described in Reference 14.0-4. With this methodology, the values of F_Q used in the setpoint

verification are calculated directly with a three-dimensional model and since operation within the Technical Specification on F_r^T limits F^Q, the need for an F_{xy}^T limit is eliminated.

Results of the analyses indicate that operating limits established for Millstone Unit 2 are acceptable.

14.0.7.1 Reactor Protection System

The reactor protection system (RPS) is designed to assure that the reactor is operated in a safe and conservative manner. The input parameters for the RPS are denoted as limiting safety system settings (LSSS). The values or functional representation of the LSSSs are calculated to ensure adherence to the specified acceptable fuel design limits (SAFDL) during steady state and anticipated operational occurrences (AOO). The safe operation of the reactor is also maintained by restricting reactor operation to conform with the LCOs, which are administratively applied at the reactor plant. The LSSS and LCO parametric values are presented in the following sections.

14.0.7.2 Specified Acceptable Fuel Design Limits

The SAFDLs are limits on the fuel and cladding established in order to preclude fuel failure. These limits may not be exceeded during steady-state operation or during AOOs. The SAFDLs are used to establish the reactor setpoints to ensure safe operation of the reactor. The specific SAFDLs used to establish the setpoints are:

1. The local power density (LPD) which coincides with fuel centerline melt.

2. The minimum departure from nucleate boiling ratio (MDNBR) corresponding to the accepted criterion which protects against the occurrence of departure from nucleate boiling (DNB).

The minimum power level required to produce centerline melt in zirconium alloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in KW/ft. This FCMLHR is determined using the methodologies of Reference 14.0-3 and Reference 14.0-4. The LPD limit for Millstone Unit 2 is the FCMLHR limit. It is noted that reload fuel may contain gadolinia-bearing fuel rods which, for a given LPD, will operate with a higher fuel temperature and will consequently have a lower LPD limit. The methodology used to determine the limit considers both uranium fuel rods and gadolinia-bearing fuel rods in establishing the FCMLHR limit.

The High Thermal Performance (HTP) critical heat flux correlation (Reference 14.0-8) is used in the thermal margin analysis with statistical parameters to support the upper 95/95 limit. Reference 14.0-4 allows for the mixed use of deterministic and statistical treatment of any, or all, of the uncertainty parameters. Observance of the LCO will protect against DNB with 95% probability at a 95% confidence level during an AOO.

14.0.7.3 Limiting Safety System Settings

14.0.7.3.1 Local Power Density

The LPD trip limit is the locus of the limiting values of core power level versus ASI that will produce a reactor trip to prevent exceeding the FCMLHR limit. The correlation between allowed core power level and peripheral ASI was determined using methods which take into account the total calculated nuclear peaking and the measurement and calculational uncertainties associated with power peaking. The LPD barn for operation at 2700 MWt is shown in Figure 14.0.7–1 as a locus of power and ASI pairs which conservatively bounds the calculated power and ASI pairs. ASI is defined as the difference between the core power in the bottom half of the core and the top half divided by the sum of the top and bottom halves.

14.0.7.3.2 Thermal Margin/Low Pressure

The thermal margin/low pressure (TM/LP) trip protects against the occurrence of DNB during steady state operations and for many, but not all, AOOs. This reactor trip system monitors primary system pressure, core inlet temperature, core power and ASI. A reactor trip occurs when primary system pressure falls below the computed limiting core pressure, P_{var}. A statistical setpoint methodology (Reference 14.0-4) is used to verify the adequacy of the existing TM/LP trip. The methodology for the TM/LP trip accounts for uncertainties in core operating conditions, HTP DNB correlation uncertainties, and uncertainties in power peaking. The existing TM/LP trip function is given by:

$$P_{var} = 2215 \times A1\ (ASI) \times QR1\ (Q) + 14.28 \times T_{in} - 8240\ [psia],$$

where Q is the higher of the thermal power and the nuclear flux power, T_{in} is the inlet temperature in °F and A1 and QR1 are shown in Figures 14.0.7–2 and 14.0.7–3, respectively.

14.0.7.3.3 Additional Trip Functions

In addition to the LPD and TM/LP trip functions, other reactor system trips have been determined to provide adherence to reactor system design criteria. The analytical setpoints for these trips are shown in Table 14.0.7-1.

14.0.7.4 Limiting Conditions for Operation

14.0.7.4.1 Departure From Nucleate Boiling

The validity of the existing LCO for allowable core power as a function of ASI was verified to ensure adherence to the SAFDL on DNB during a postulated loss-of-flow operational occurrence. The statistical analysis accounted for the effects of uncertainties associated with core operating parameters, the HTP critical heat flux correlation, and power peaking. The allowed core power as a function of ASI for the existing LCO is shown to conservatively bound the present analysis in Figure 14.0.7–5.

14.0.7.4.2 Linear Heat Rate

In the event that the in-core detector system is not in operation, the linear heat rate (LHR) will be limited through the use of an LPD LCO. The verification of this LCO was performed in a fashion similar to that used in verifying the LPD LSSS (Section 14.0.7.3.1). The verification plot is shown in Figure 14.0.7–6. The LPD LCO limits core power so that the LHR LCO based on LOCA considerations is not exceeded. The LHR LCO protected by the LPD LCO is depicted in Figure 14.0.7–7.

14.0.7.5 Setpoint Analysis

14.0.7.5.1 Limiting Safety System Settings

14.0.7.5.1.1 Local Power Density

The LPD trip monitors core power and ASI in order to initiate a reactor scram which precludes exceeding fuel centerline melt conditions. In the analysis for this trip function a large number of axial power distribution cases typical of the cycle were examined to establish bounding values of total power peaking, F_Q, versus ASI. These cases were generated in a manner consistent with that discussed in Reference 14.0-4. Statistical methods were then employed to account for the uncertainties in the parameters that are given in Table 14.0.7-2.

The peak LHR in the core occurs at the position of the maximum total peaking factor, F_Q, which is the ratio of the maximum linear heat generation rate (LHGR) in the core to the average LHGR in the core.

The allowed power for each ASI was calculated statistically by incorporating the uncertainties listed in Table 14.0.7-2 as described in Reference 14.0-4. The results in Figure 14.0.7–1 are bounded by the existing Millstone Unit 2 LPD trip and thus verify the adequacy of the existing trip function.

14.0.7.5.1.2 Thermal Margin/Low Pressure Limiting Safety System Settings

The TM/LP trip is designed to shut the reactor down should the reactor conditions (ASI, inlet temperature, core power and pressure) approach the point where DNB might occur during either normal operation or an AOO. This analysis uses the HTP critical heat flux correlation and the core thermal-hydraulic methodology described in References 14.0-8 and 14.0-11. The analysis methodology is consistent with the NRC's SRP in requiring DNB to be avoided with 95% probability at a 95% confidence level.

The uncertainties shown in Tables 14.0.7-2 and 14.0.7-3 are included in the verification of the TM/LP trip as described in Reference 14.0-4. An excess margin of protection is provided by the trip.

14.0.7.5.2 Limiting Conditions for Operation

14.0.7.5.2.1 Departure from Nucleate Boiling

The TM/LP trip system does not directly monitor reactor coolant flow. Thus, the TM/LP trip generally does not provide DNB protection for the four pump coastdown AOO. The analysis of this transient is given in Section 14.3.1. The LCO presented here administratively protects the DNB SAFDL for this transient.

The method used to establish the DNB LCO involved simulations of the loss-of-flow transient using the core thermal hydraulic code XCOBRA-IIIC (Reference 14.0-11) to determine the initial power, as a function of ASI, which provides protection from DNB with 95% probability. The uncertainties listed in Tables 14.0.7-4 and 14.0.7-5 are applied using the methodology described in Reference 14.0-4. The results of the statistical analysis for the loss-of-flow transient are summarized by the points in Figure 14.0.7–5. The points are bounded by the existing DNB LCO and, thus, verify the adequacy of the existing DNB LCO for Millstone Unit 2, which is shown in this same figure by the straight line segments.

14.0.7.5.2.2 Local Power Density

The plant Technical Specifications allow plant operations for limited periods of time with the in-core detectors out of service. In this situation, the LPD barn provides protection in steady-state operation against penetration of the LPD limit established by LOCA considerations. The statistical methodology for the LPD LCO is essentially the same as that for LPD LSSS except:

1. The peak LPD limit is reduced, and

2. The uncertainties listed in Table 14.0.7-4 are used, as opposed to the values in Table 14.0.7-2.

The allowed power versus ASI was statistically analyzed to account for the appropriate uncertainties. The points in Figure 14.0.7–6 represent the statistical calculation of the 15.1 kw/ft LHR curve depicted in Figure 14.0.7–7. The LCO curve is shown by the straight line segments in Figure 14.0.7–6, and conservatively bounds the calculated verification points.

14.0.8 COMPONENT CAPACITIES AND SETPOINTS

Table 14.0.8-1 presents the component setpoints and capacities used in the analysis.

14.0.9 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

Table 14.0.9-1 is a summary of trip functions, engineered safety features (ESF), and other equipment available for mitigation of accident effects. These are listed for all SRP Chapter 15 events. A more detailed listing of available reactor protection for each event in each operating mode is given in the individual event descriptions.

14.0.10 EFFECTS OF MIXED ASSEMBLY TYPES AND FUEL ROD BOWING

In accordance with AREVA rod bow methodology (Reference 14.0-12), the magnitude of rod bow for the AREVA assemblies has been estimated. The calculations indicate that 50% closure of the rod-to-rod gap occurs at an assembly exposure in excess of the licensed burnup limit for the AREVA 14 x 14 design. Significant impact to MDNBR due to rod bow does not occur until the gap closures exceed 50%. Since the maximum design exposure for AREVA reload fuel in Millstone Unit 2 is significantly less than that at which 50% closure occurs, rod bow does not significantly impact the MDNBR for AREVA fuel. Also, total peaking is not significantly impacted.

14.0.11 PLANT LICENSING BASIS AND SINGLE FAILURE CRITERIA

All event scenarios considered in the safety analysis depend on the following single failure criteria in the RPS:

The RPS is designed with redundancy and independence to assure that no single failure or removal from service of any component or channel of a system will result in the loss of the protection function. For each event, the reactor trips occur at the specified setpoint within the specified delay time assuming a worst single active failure.

Except for the steam generator tube rupture, design basis accident (limiting fault event) scenarios considered in the Millstone 2 safety analysis depend on one of the following additional single failure criteria:

1. Each ESF is designed to perform its intended safety function assuming a failure of a single active component. For these events, the ESFs required to function in an event are assumed to suffer a worst single failure of an active component.

2. The onsite power system and the offsite power system are designed such that each shall independently be capable of providing power for the ESF assuming a failure of a single active component in either power system.

The assumptions for concurrent loss of offsite power are as follows:

1. The following postulated accidents are considered assuming a concurrent loss of offsite power: main steam line break, control rod ejection, steam generator tube rupture, and LOCA.

2. The loss of normal feedwater, an anticipated operational occurrence, is analyzed assuming a concurrent loss of offsite power.

The requirements of 10 CFR 50, Appendix A, Criteria 10, 20, 25 and 29 require that the design and operation of the plant and the RPS assure that the SAFDLs not be exceeded during AOOs. As per the definition of AOO in 10 CFR 50, Appendix A, “Anticipated Operational Occurrences mean those conditions of normal operation which are expected to occur one or more times during

the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The SAFDLs are that: 1) the fuel shall not experience centerline melt; and 2) the DNBR shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that DNB has not occurred.

14.0.12 PLOT VARIABLE NOMENCLATURE

Some of the plotted results presented in Sections 14.1 through 14.6, use PTSPWR2 (Reference 14.0-13) output variable nomenclature. Specific variables plotted are listed and defined in Table 14.0.12-1.

14.0.13 REFERENCES

14.0-1 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

14.0-2 Technical Specifications for Millstone Unit 2, Docket Number 50-336.

14.0-3 "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A), Revision 1 and Supplements 2, 4, and 5, Exxon Nuclear Company, October 1986.

14.0-4 "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000.

14.0-5 "Reactor Analysis Systems for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," EMF-96-029-(P)(A), Siemens Power Corporation, January 1997.

14.0-6 Deleted.

14.0-7 Deleted.

14.0-8 "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," EMF-92-153(P)(A), Revision 1, Siemens Power Corporation, January 2005.

14.0-9 Deleted.

14.0-10 Deleted.

14.0-11 "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, January 1986.

14.0-12 "Computational Procedure for Evaluating Fuel Rod Bowing," XN-NF-75-32(A), Supps. 1, 2, 3 & 4, Exxon Nuclear Company, Richland, WA 99352, October 1983.

14.0-13 “Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR),” XN-NF-74-5(A), Rev. 2 and Supplements 3-6, Exxon Nuclear Company, Richland, WA 99352, October 1986.

TABLE 14.0-1 REACTOR OPERATING MODES FOR MILLSTONE UNIT 2

Mode	Reactivity Condition, K_{eff}	% Rated Thermal Power *	Average Coolant Temperature
1. Power Operation	≥ 0.99	> 5%	≥ 300°F
2. Startup	≥ 0.99	≤ 5%	≥ 300°F
3. Hot Standby	< 0.99	0	≥ 300°F
4. Hot Shutdown	< 0.99	0	300°F > T_{avg} > 200°F
5. Cold Shutdown	< 0.98	0	≤ 200°F
6. Refueling **	≤ 0.95	0	≤ 140°F

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 14.0-2 DISPOSITION OF EVENTS SUMMARY

SRP Event Designation	Name	Disposition	Bounding Event
15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM			
15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3
15.1.2	Increase in Feedwater Flow		
	1) Power	Bounded	15.1.3
	2) Startup	Bounded	15.1.3
15.1.3	Increase in Steam Flow	Analyze	
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Bounded	15.1.3
15.1.5	Steam System Piping Failures Inside and Outside of Containment	Analyze	
15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM			
15.2.1	Loss of External Load	Analyze	
15.2.2	Turbine Trip	Bounded	15.2.1
15.2.3	Loss of Condenser Vacuum	Not in Licensing Basis	
15.2.4	Closure of the Main Steam Isolation Valves	Analyze	
15.2.5	Steam Pressure Regulator Failure	Not applicable; BWR Event	
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Not in Licensing Basis	
15.2.7	Loss of Normal Feedwater Flow	Analyze	
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Not in Licensing Basis	
15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW			
15.3.1	Loss of Forced Reactor Coolant Flow	Analyze	
15.3.2	Flow Controller Malfunction	Not Applicable	
15.3.3	Reactor Coolant Pump Rotor Seizure	Analyze	
15.3.4	Reactor Coolant Pump Shaft Break	Not in Licensing Basis	
15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES			

TABLE 14.0-2 DISPOSITION OF EVENTS SUMMARY (CONTINUED)

SRP Event Designation	Name	Disposition	Bounding Event
15.4.1	Uncontrolled Control Rod/Bank Withdrawal from a Subcritical or Low-Power Condition	Analyze	
15.4.2	Uncontrolled Control Rod/Bank Withdrawal at Power	Analyze	
15.4.3	Control Rod Misoperation		
	1) Dropped Control Rod/Bank	Analyze	
	2) Dropped Part-Length Control Rod	Not Applicable	
	3) Malpositioning of the Part-Length Control Rod Group	Not Applicable	
	4) Statically Misaligned Control Rod/Bank	Not in Licensing Basis	
	5) Single Control Rod Withdrawal	Analyze	
	6) Reactivity Control Device Removal Error During Refueling	Not Applicable	
	7) Variations in Reactivity Load to be Compensated by Burnup or On Line Refueling	Not Applicable	
15.4.4	Startup of an Inactive Loop	Not Applicable (Tech Specs Preclude Significant Consequences)	
15.4.5	Flow Controller Malfunction	Not applicable; No Flow Controller	
15.4.6	Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Analyze, Modes 1-6	
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Not in Licensing Basis	
15.4.8	Spectrum of Control Rod Ejection Accidents	Analyze	
15.4.9	Spectrum of Rod Drop Accidents (BWR)	Not applicable; BWR Event	
15.5 INCREASES IN REACTOR COOLANT INVENTORY			
15.5.1	Inadvertent Operation of the Emergency Core Cooling System that Increases Reactor Coolant Inventory	Not in Licensing Basis	

TABLE 14.0-2 DISPOSITION OF EVENTS SUMMARY (CONTINUED)

SRP Event Designation	Name	Disposition	Bounding Event
15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Not in Licensing Basis	
15.6 DECREASES IN REACTOR COOLANT INVENTORY			
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	Analyze	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment	Not Applicable	
15.6.3	Radiological Consequences of Steam Generator Tube Failure	Analyze	
15.6.4	Radiological Consequences of a Main Steamline Failure Outside Containment	Not applicable; BWR Event	
15.6.5	Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Analyze	
15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT			
15.7.1	Waste Gas System Failure	Analyze	
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Not in Licensing Basis	
15.7.3	Postulated Radioactive Releases due to Liquid Containing Tank Failures	Not in Licensing Basis	
15.7.4	Radiological Consequences of Fuel Handling Accidents	Analyze	
15.7.5	Spent Fuel Cask Drop Accidents	Analyze	
FSAR EVENTS NOT CONTAINED IN THE STANDARD REVIEW PLAN			
	(1) Failures of Equipment Which Provide Joint Control/Safety Functions	Not Applicable	
	(2) Containment Pressure Analysis	Analyze	
	(3) Deleted		
	(4) Radiological Consequences of the Design Basis Accident	Analyze	

TABLE 14.0.1-1 ACCIDENT CATEGORY USED FOR EACH ANALYZED EVENT

Event	Accident Category
14.1.3 Increase in Steam Flow	Moderate
14.1.5 Steam System Piping Failures Inside and Outside of Containment	Limiting Fault
14.2.1 Loss of External Load	Moderate
14.2.4 Single Main Steam Isolation Valve Closure	Moderate
14.2.7 Loss of Normal Feedwater Flow	Moderate
14.3.1 Loss of Forced Reactor Coolant Flow	Moderate
14.3.3 Reactor Coolant Pump Rotor Seizure	Limiting Fault
14.4.1 Uncontrolled Bank Withdrawal at Subcritical or Low Power	Moderate
14.4.2 Uncontrolled Bank Withdrawal at Power	Moderate
14.4.3 Control Rod Misoperation	
1) Dropped Control Rod/Bank	Moderate
5) Single Control Rod Withdrawal	Infrequent
14.4.6 Chemical and Volume Control System Malfunction Resulting in Decreased Boron Concentration	Moderate
14.4.8 Control Rod Ejection	Limiting Fault
14.6.1 Inadvertent Opening of a Pressurizer Pressure Relief Valve	Moderate
14.6.3 Radiological Consequences of Steam Generator Tube Failure	Limiting Fault
14.6.5 Loss-of-Coolant Accidents	Limiting Fault
14.7.1 Waste Gas System Failure	
14.7.4 Radiological Consequences of Fuel Handling Accidents	
14.7.5 Spent Fuel Cask Drop Accidents	
14.8.2 Containment Analysis	
14.8.4 Radiological Consequences of the Design Basis Accident	

TABLE 14.0.2-1 PLANT OPERATING CONDITIONS

Core Thermal Power	2700 MWt
Pump Thermal Power (Total)	17.1 MWt
System Pressure	2225-2280 psia
Reactor Coolant System Flow Rate (Minimum)	360,000 gpm *
Core Inlet Coolant Temperature at full power	541.0-549.0°F **

* Flow reductions to 349,200 gpm are compensated for by reductions in the F_r^T and linear heat rate limits.

** A full power coastdown to an indicated RCS cold leg temperature of 537°F at EOC is supported.

TABLE 14.0.2-2 NOMINAL FUEL DESIGN PARAMETERS

Total Number of Fuel Assemblies	217
Fuel Assembly Design Type	14 x 14
Assembly Pitch	8.180 inches
Fuel Rods per Assembly	176
Guide Tubes per Assembly	4
Instrument Tubes per Assembly	1
Rod Pitch	0.580 inches
Clad Outside Diameter	0.440 inches
Guide and Instrument Tube OD	1.115 inches
Active Fuel Length	136.70 inches
Fuel Rod Length	146.25 inches
Number of Spacers	9

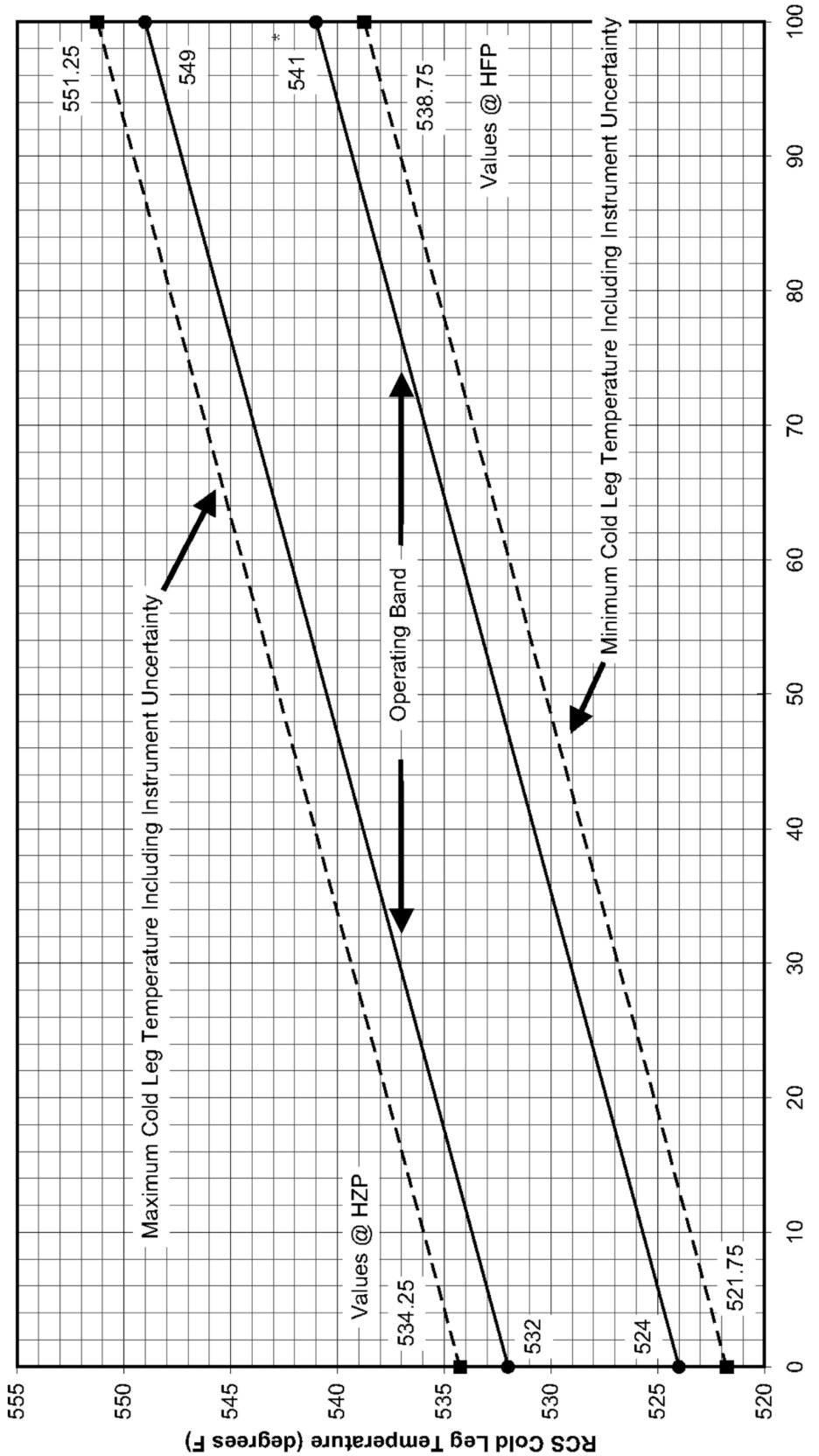
TABLE 14.0.3-1 CORE POWER DISTRIBUTION (TABLE DELETED)

TABLE 14.0.4-1 RANGE OF KEY INITIAL CONDITION OPERATING PARAMETERS

Core thermal power	Subcritical to 2754 MWt (1)
Core inlet temperature (power operation)	Figure 14.0.4-1
Reactor coolant system pressure	2225-2280 psia +14/-25 psi
Pressurizer Water Level	Programmed ± 7.5 inches
Feedwater flow and temperature	Range consistent with power level

(1) 102% of 2700 MWt

FIGURE 14.0.4-1 RCS COLD LEG TEMPERATURE AS A FUNCTION OF POWER



* A full power coastdown to an indicated RCS cold leg temperature of 537 °F at End of Cycle is supported.

FIGURE 14.0.4-2 NOT USED

FIGURE 14.0.4-3 NOT USED

FIGURE 14.0.4-4 NOT USED

FIGURE 14.0.4-5 NOT USED

FIGURE 14.0.4-6 NOT USED

FIGURE 14.0.4-7 LINEAR HEAT RATE LIMITING CONDITION FOR OPERATION USED IN LOCAL POWER DENSITY LIMITING CONDITION FOR OPERATION VERIFICATION

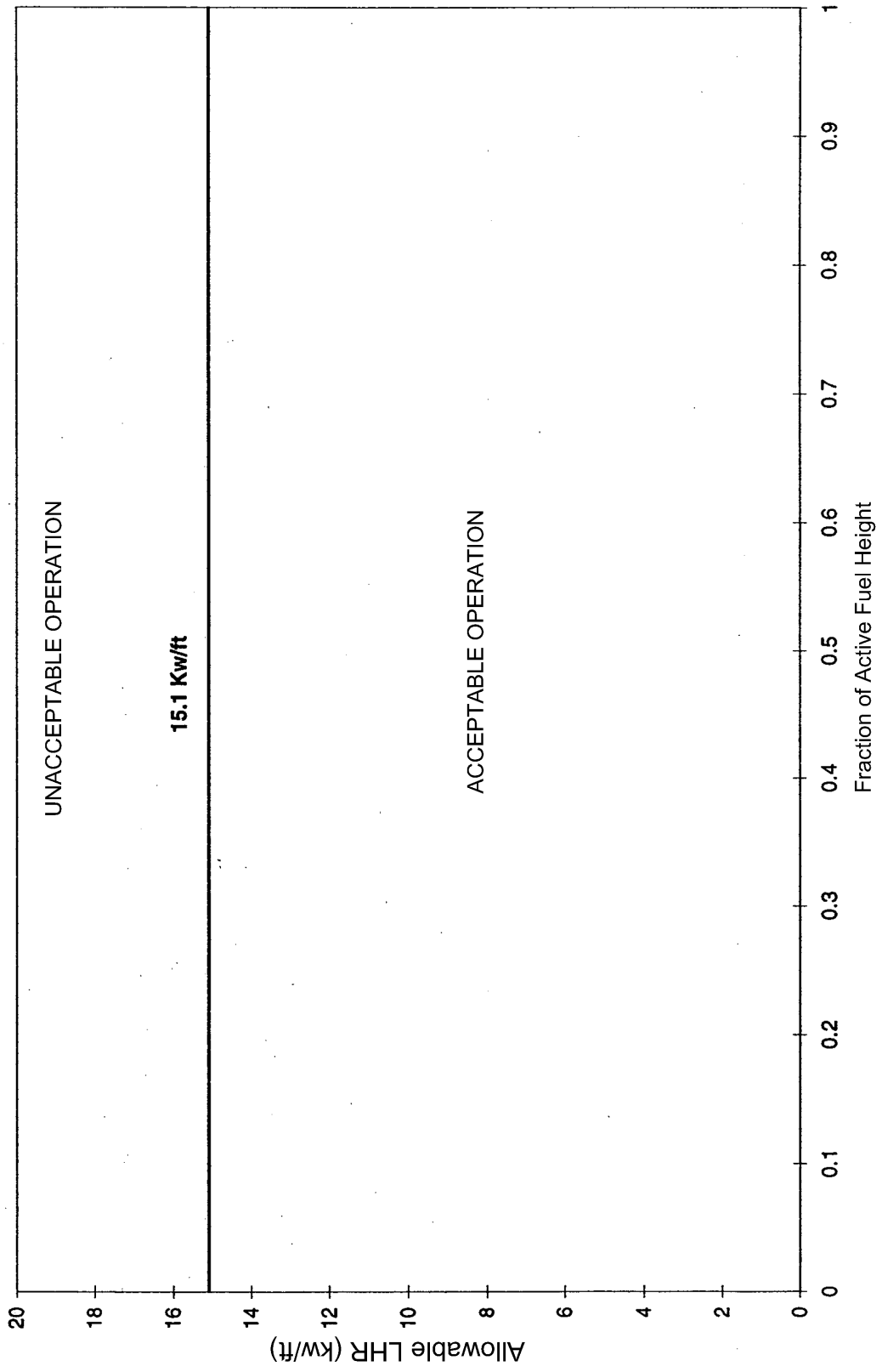


TABLE 14.0.5-1 REACTIVITY PARAMETERS (TABLE DELETED)

TABLE 14.0.7-1 ANALYTICAL TRIP SETPOINTS

Parameter	Setpoint
Low steam generator pressure	658 psia
Low steam generator water level	43%
Variable high power	111.6% ceiling on nuclear indicated power; 114% ceiling on thermal power 27.22% - Floor
Low reactor coolant flow	89.7% of Tech Spec minimum
High pressurizer pressure	2422 psia
High containment pressure	5.83 psig

TABLE 14.0.7-2 UNCERTAINTIES APPLIED AT HFP CONDITION IN LOCAL POWER DENSITY LIMITING SAFETY SYSTEM SETTINGS CALCULATIONS

Parameter	Value
Engineering tolerance	± 3% [a]
Nuclear flux power measurement uncertainty at full power	± 2.34% [a, b]
Thermal Power measurement uncertainty at full power	±4.19% [a, c]
Peaking uncertainty	≤ 7% [d]
Local power density trip transient offset	[e]
ASI uncertainty	± 0.039 [a, f]

a Two-sided 95% tolerance.

b The nuclear power measurement uncertainty used in the setpoints analyses account for calorimetric and flux signal uncertainties.

c The thermal power measurement uncertainty used in the setpoints analyses account for calorimetric and thermal power signal uncertainties. Included in this power measurement uncertainty are 2σ uncertainties on the RCS hot and cold leg temperature signals of 2.2°F and 1.725°F.

d One-sided 95% tolerance.

e Not treated statistically, treated as an event specific bias. Events where this trip is credited include the inadvertent opening of a pressurized water reactor pressurizer pressure relief valve, the uncontrolled rod/bank withdrawal at power, and the increase in steam flow transients.

f. An additional ASI bias for the INPAX-II / shape annealing factor ASI measurement uncertainty was applied.

TABLE 14.0.7-3 UNCERTAINTIES APPLIED AT HFP CONDITION IN THE THERMAL MARGIN/LOW PRESSURE LIMITING SAFETY SYSTEM SETTINGS CALCULATIONS

Parameter	Value
TM/LP trip uncertainty	± 90.30 psi [a, b]
TM/LP trip bias	[c]
Inlet coolant temperature	± 2.25°F [a]
Flow measurement uncertainty	+ 4% [d]
Fr uncertainty	± 6% [e]

a Two-sided 95% tolerance.

b Includes both pressure measurement and trip processing uncertainties.

c A 7.5 psi pressure measurement bias is applied along with an event specific set of transient biases on power, pressurizer pressure and RCS hot and cold leg temperatures. The events where this trip is credited include the inadvertent opening of a pressurized water reactor pressurizer pressure relief valve, the uncontrolled rod/bank withdrawal at power, and the increase in steam flow transients.

d One-sided 95% tolerance. Stated tolerance is percentage of design volumetric flow of 324,800 gpm.

e One-sided 95% tolerance.

TABLE 14.0.7-4 UNCERTAINTIES APPLIED AT HFP CONDITION IN THE LOCAL POWER DENSITY LIMITING CONDITION FOR OPERATION CALCULATIONS

Parameter	Value
Engineering tolerance	± 3% [a]
Power measurement uncertainty at full power	± 2.34% [a, b]
Peaking uncertainty	≤ 7% [c]
ASI uncertainty	± 0.043 [a, d, e]

a. Two-sided 95% tolerance.

b. The nuclear power measurement uncertainty used in the setpoints analyses account for calorimetric and flux signal uncertainties.

c. One-sided 95% tolerance.

d. An additional ASI bias for the INPAX-II / shape annealing factor ASI measurement uncertainty was applied.

e. Uncertainty is ASI-dependent; this is the maximum value applied over the range of ASI values considered.

TABLE 14.0.7-5 UNCERTAINTIES APPLIED IN DEPARTURE FROM NUCLEATE BOILING LIMITING CONDITION FOR OPERATION CALCULATIONS

Parameter	Value
Pressure measurement uncertainty	± 17.9 psi [a]
Inlet coolant temperature	± 2.25 °F [a]
Flow measurement uncertainty	± 4% [b]
ASI uncertainty	± 0.053 [a, d]
Power measurement uncertainty (at full power)	± 2.34% [a], [c]

a. Two-sided 95% tolerance.

b. One-sided 95% tolerance. Stated value is percentage of design volumetric flow of 324,800 gpm.

c. The nuclear power measurement uncertainty used in the setpoints analyses account for calorimetric and flux signal uncertainties.

d. An additional ASI bias for the INPAX-II / shape annealing factor ASI measurement uncertainty was applied.

FIGURE 14.0.7–1 REPRESENTATIVE VERIFICATION OF LOCAL POWER DENSITY LIMITING SAFETY SYSTEM SETTING

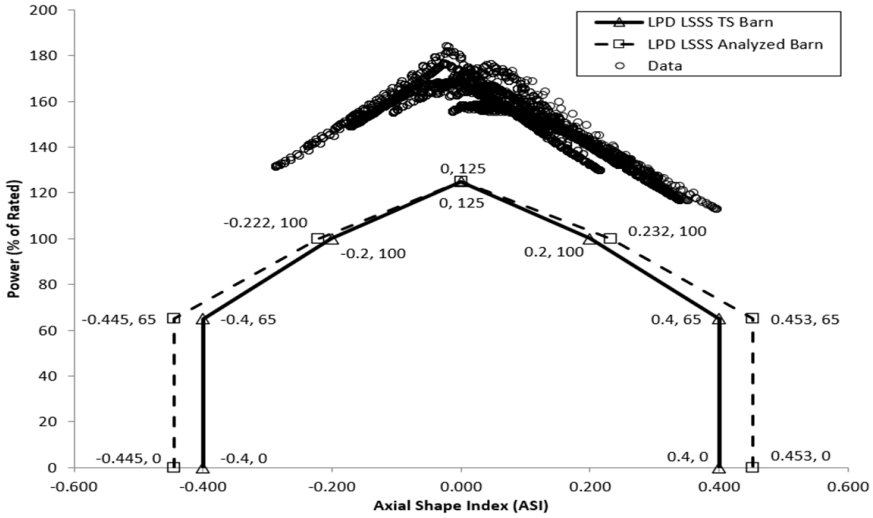
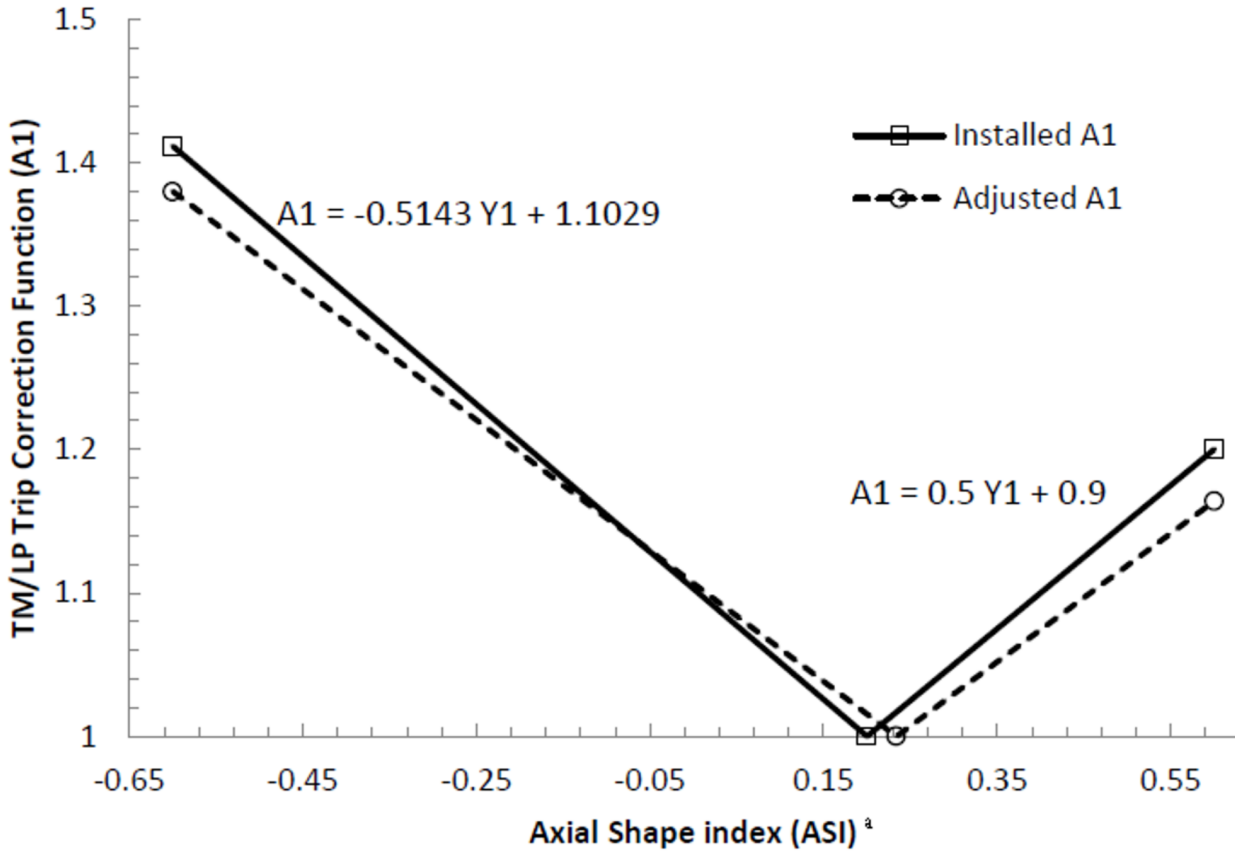


FIGURE 14.0.7–2 THERMAL MARGIN/LOW PRESSURE TRIP FUNCTION A1



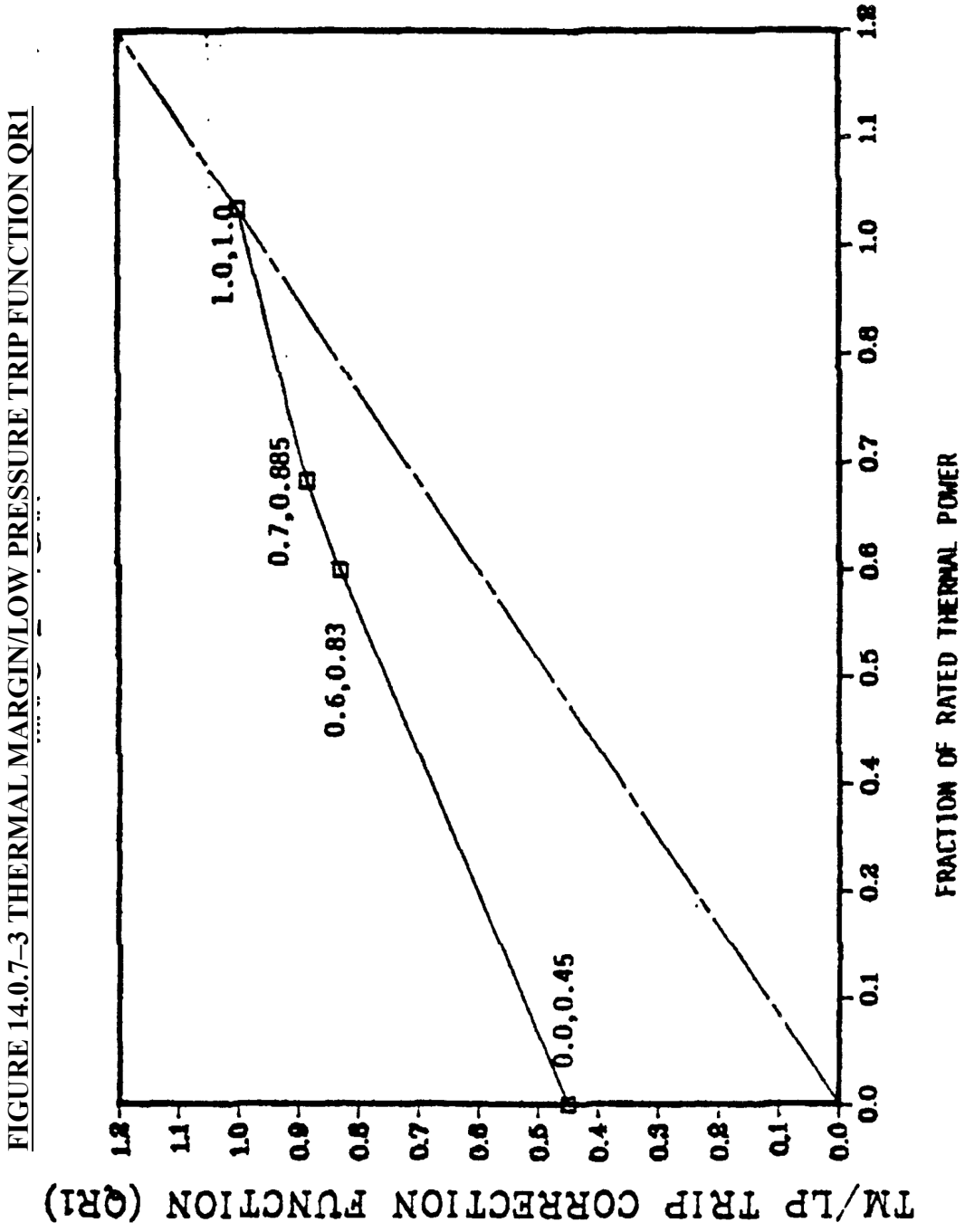


FIGURE 14.0.7–3 THERMAL MARGIN/LOW PRESSURE TRIP FUNCTION QR1

FIGURE 14.0.7–4 NOT USED

FIGURE 14.0.7–5 REPRESENTATIVE VERIFICATION OF THE DEPARTURE FROM NUCLEATE BOILING LIMITING CONDITION FOR OPERATION

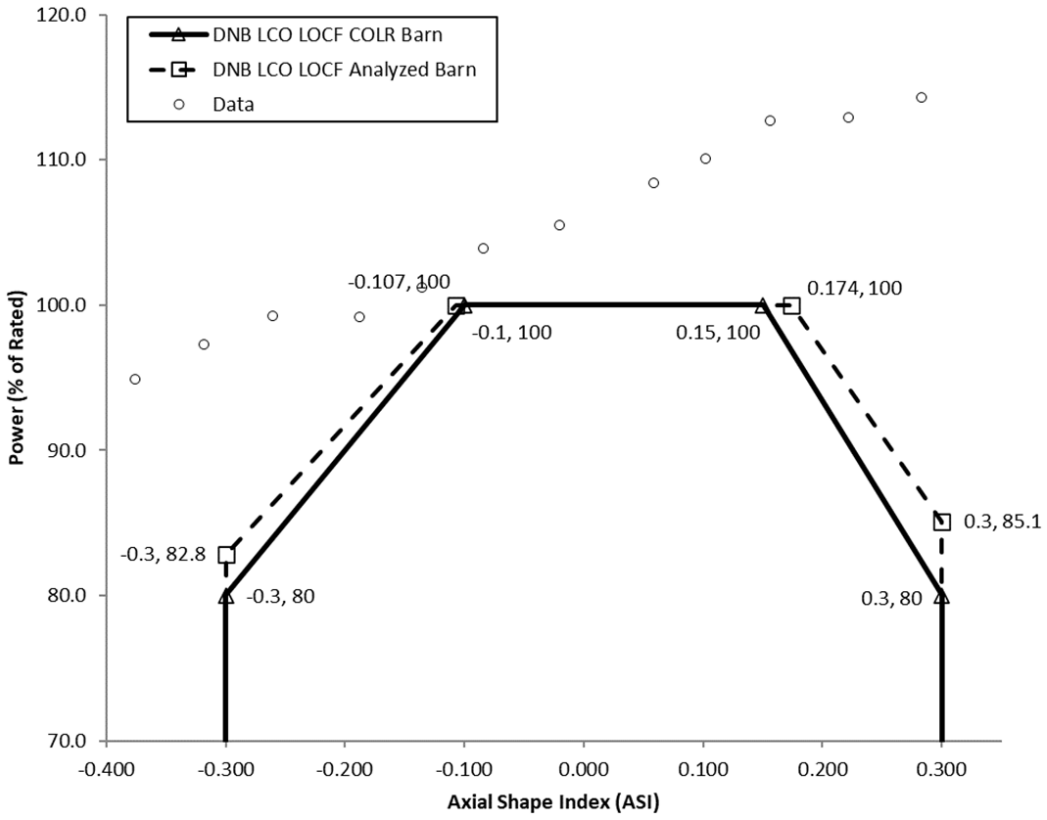
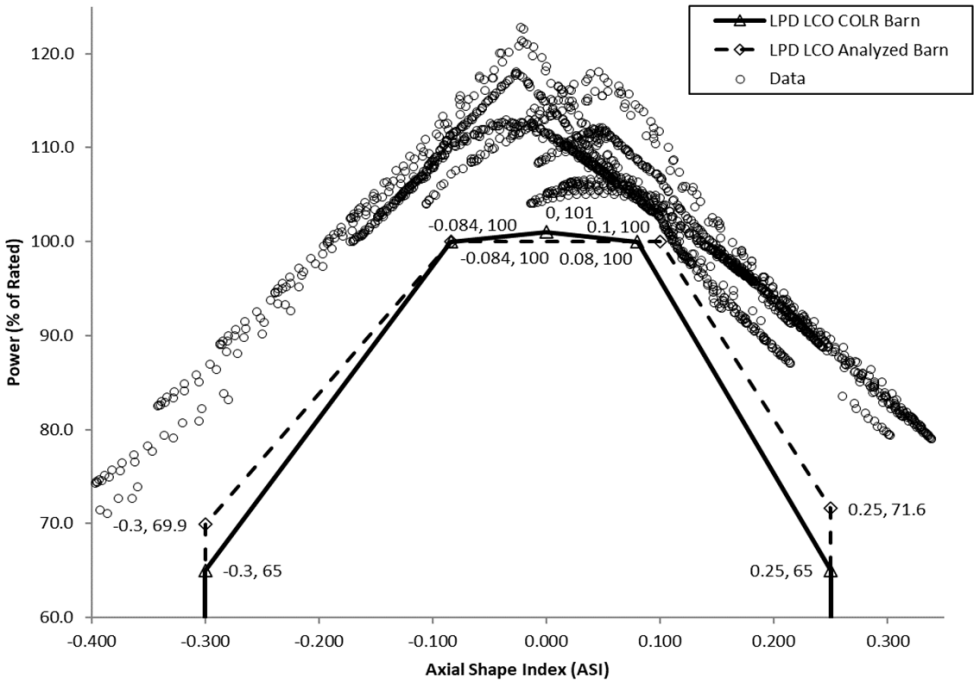


FIGURE 14.0.7–6 REPRESENTATIVE VERIFICATION OF LOCAL POWER DENSITY LIMITING CONDITION FOR OPERATION



1) The center point on the LPD LCO barn is shown at (0.0, 101.0); however, that point is used only for setting plant instrumentation. Operation is not permitted above 100% of rated power. The analysis was actually performed with that point at (0.0, 100.0).

2) The analyzed barn, which includes additional bias for the INPAX-II/shape annealing factor ASI measurement uncertainty, is represented by the dashed line. The analyzed barn is not meant for direct use in the plant.

FIGURE 14.0.7–7 LINEAR HEAT RATE LIMITING CONDITION OF OPERATION USED IN LOCAL POWER DENSITY LIMITING CONDITION OF OPERATION VERIFICATION

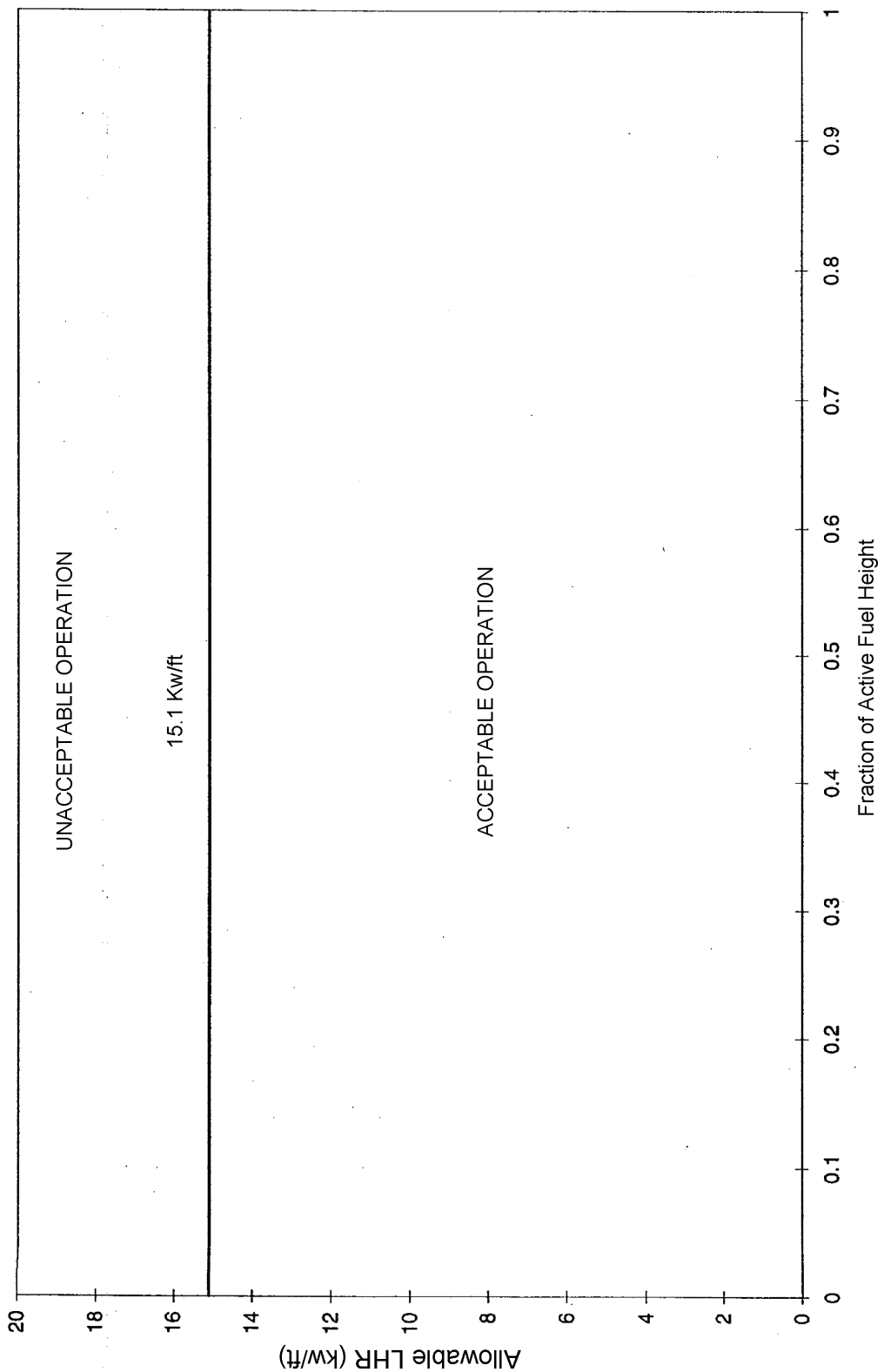


TABLE 14.0.8-1 COMPONENT CAPACITIES AND SETPOINTS

Component	Setpoint	Response Time	Capacity
Turbine main throttle valve	NA	NA	110% of flow at rated conditions
Turbine stop valve	NA	0.020 sec	NA
Main steam line isolation valves	NA	6.0 sec	NA
Feedwater flow regulating valves	NA	14 sec	120% of flow at rated conditions
Pressurizer safety valves	2500 psia ± 3%	NA	294,000 lbm/hr/valve
Steam line safety valves	2 at 1000 psia ± 3%	NA	794,060 lbm/hr/valve
	2 at 1005 psia ± 3%		
	2 at 1015 psia ± 3%		
	2 at 1025 psia ± 3%		
	2 at 1035 psia ± 3%		
	2 at 1045 psia ± 3%		
	4 at 1050 psia ± 3%		
Auxiliary feedwater pumps	NA	240 sec for Events 14.2.7, 14.6.3 and 14.6.5.2	300 gpm/MDAFP
		180 sec for Events 14.8.2 and 14.1.5	600 gpm for TDAFP
Pressurizer relief valves	2397 psia (+)14 (-)25 psi	2.0 sec	153,000 lbm/hr/ valve

TABLE 14.0.8-1 COMPONENT CAPACITIES AND SETPOINTS (CONTINUED)

Component	Setpoint	Response Time	Capacity
Pressurizer sprays	Off - 2300 psia (+)14 (-)25 psi	NA	375 gpm
	Full On - 2350 psia (+)14 (-)25 psi		
Pressurizer proportional heaters	Off - 2275 psia (+)14 (-)25 psi	NA	320 kW (may be less due to heater unavailability)
	Full On - 2225 psia (+)14 (-)25 psi		
Pressurizer backup heaters	Off - 2225 psia (+)14 (-)25 psi	NA	1280 kW (may be less due to heater unavailability)
	On - 2200 psia (+)14 (-)25 psi		

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Event	Reactor Trip Functions	Other Signals and Equipment *
14.1, Increase in Heat Removal By the Secondary System		
Feedwater System Malfunctions	High Power Trip	Steam Generator Water Level Signals
	Thermal Margin / Low Pressure Trip	Feedwater Isolation Valves
	Low Steam Generator Pressure Trip	Main Steamline Isolation Valves
	Safety Injection Actuation Signal	Turbine Trip on Reactor Trip
Increase in Steam Flow	Low Steam Generator Pressure Trip	Steam Generator Water Level Signals
	Thermal Margin / Low Pressure Trip	Main Steamline Isolation Valves
	High Power Trip	Turbine Trip on Reactor Trip
	Safety Injection Actuation Signal	Atmospheric Steam Dump Controller
		Steam Bypass to Condenser Controller
		Auxiliary Feedwater System
Inadvertent Opening of a Steam Generator Relief or Safety Valve	Low Steam Generator Pressure Trip	Steam Generator Water Level Signals
	Thermal Margin / Low Pressure Trip	Main Steamline Isolation Valves
	High Power Trip	Turbine Trip on Reactor Trip
	Safety Injection Actuation Signal	Atmospheric Steam Dump Controller
		Steam Bypass to Condenser Controller
		Auxiliary Feedwater System

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Steam System Piping Failure	Low Steam Generator Pressure Trip	Steam Generator Water Level Signals
	Thermal Margin / Low Pressure Trip	Main Steamline Isolation Valves
	High Power Trip	Turbine Trip on Reactor Trip
	Safety Injection Actuation Signal	Atmospheric Steam Dump Controller
	High Containment Pressure Trip	Steam Bypass to Condenser Controller
	Low Reactor Coolant Flow Trip	Auxiliary Feedwater System
		Containment Spray
		Containment Isolation
		Containment Air Coolers
		Feedwater Isolation
14.2, Decrease in Heat Removal by the Secondary System		
Loss of External Load / Turbine Trip / Loss of Condenser Vacuum	High Pressurizer Pressure Trip	Steam Generator Water Level Signals
	High Power Trip	Turbine Trip on Reactor Trip
	Thermal Margin / Low-Pressure Trip	Atmospheric Steam Dump Controller
	Low Steam Generator Water Level Trip	Steam Bypass to Condenser Controller
		Steam Generator Safety Valves
		Pressurizer Safety Valves
		Pressurizer Sprays

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Closure of the Main Steam Isolation Valves	Low Steam Generator Pressure Trip	Atmospheric Steam Dump Controller
	Low Steam Generator Water Level Trip	Steam Bypass to Condenser Controller
	High Power Trip	Steam Generator Safety Valves
	Thermal Margin / Low Pressure Trip	Pressurizer Safety Valves
	High Pressurizer Pressure Trip	Pressurizer Sprays
Loss of Nonemergency AC Power to the Station Auxiliaries	Low Reactor Coolant Flow Trip	Steam Generator Water Level Signals
	High Pressurizer Pressure Trip	Steam Generator Safety Valves
	Thermal Margin / Low Pressure Trip	Pressurizer Safety Valves
	Low Steam Generator Water Level Trip	Auxiliary Feedwater System
Loss of Normal Feedwater Flow	Low Steam Generator Water Level Trip	Steam Generator Water Level Signals
	High Pressurizer Pressure Trip	Steam Generator Safety Valves
	Thermal Margin/Low-Pressure Trip	Pressurizer Safety Valves
		Auxiliary Feedwater System
		Pressurizer Sprays and Level Control
Feedwater System Pipe Break	High Pressurizer Pressure Trip	Steam Generator Water Level Signals
	Thermal Margin / Low Pressure Trip	Steam Generator Safety Valves
	Low Steam Generator Water Level Trip	Pressurizer Safety Valves
	Low Steam Generator Pressure Trip	Auxiliary Feedwater System
		Pressurizer Sprays and Level Control
14.3, Decrease in Reactor Coolant System Flow		

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Loss of Forced Reactor Coolant Flow	Low Reactor Coolant Flow Trip	Atmospheric Steam Dump Controller
	Thermal Margin / Low Pressure Trip	Steam Bypass to Condenser Controller
	High Pressurizer Pressure Trip	Steam Generator Safety Valves
		Pressurizer Safety Valves
Reactor Coolant Pump Rotor Seizure/ Shaft Break	Low Reactor Coolant Flow Trip	Atmospheric Steam Dump Controller
	High Pressurizer Pressure Trip	Steam Bypass to Condenser Controller
	Thermal Margin / Low Pressure Trip	Steam Generator Safety Valves
		Pressurizer Safety Valves
14.4, Reactivity and Power Distribution Anomalies		
Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low-Power Startup Condition	Thermal Margin / Low Pressure Trip	Rod Withdrawal Prohibit Action on Pretrip Alarms
	High-Power Trip	Shutdown Margin is Maintained as Required by Technical Specifications for Modes 2-6
	High Pressurizer Pressure Trip	
Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions	High-Power Trip	Pressurizer Safety Valves
	Thermal Margin / Low Pressure Trip	Steam Generator Safety Valves
	High Pressurizer Pressure Trip	Pressurizer Spray and Level Control
		Control Rod and Bank Deviation Alarms which Initiate Rod Withdrawal Prohibit
		Rod Withdrawal Prohibit on Pretrip Alarms

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Control Rod Misoperation	High Power Trip	Pressurizer Safety Valves
	Thermal Margin / Low Pressure Trip	Steam Generator Safety Valves
	Low Steam Generator Water Level Trip	Pressurizer Spray and Level Control
	Safety Injection Actuation Signal	Control Rod and Bank Deviation Alarms
Start up of an Inactive Loop	High Power Trip	Administrative Procedures for Start up of an Idle Pump
	Thermal Margin / Low-Pressure Trip	Plant Operation with Less than All Four Primary Coolant Pumps is Controlled by Technical Specifications
Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	High Power Trip	Administrative Procedures
	Thermal Margin / Low Pressure Trip	Sufficient Operator Response Time
	High Pressurizer Pressure Trip	
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	(Technical Specification Measurement Requirement and Administrative Procedures Preclude Occurrence)	
Spectrum of Control Rod Ejection Accidents	High Power Trip	
	High Pressurizer Pressure Trip	
	Thermal Margin / Low Pressure Trip	
	Long Term, Safety Injection Actuation Signal	
14.5, Increases in Reactor Coolant System Inventory		

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Inadvertent Operation of the ECCS / CVCS Malfunction that Increases Reactor Coolant Inventory	High Power Trip	Pressurizer Safety Valves
	Thermal Margin / Low Pressure Trip	Overpressurization Mitigation System
	High Pressurizer Pressure Trip	
14.6, Decreases in Reactor Coolant Inventory		
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	High Power Trip	High Pressure Safety Injection System
	Thermal Margin / Low Pressure Trip	Pressurizer Heaters
Steam Generator Tube Failure	Thermal Margin/Low Pressure Trip	Steam Generator Safety Valves
	Safety Injection Actuation Signal	Main Steamline Isolation Valves
		Atmospheric Steam Dump Controller
		Steam Bypass to Condenser Controller
		Auxiliary Feedwater System
Small-Break Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Thermal Margin/Low Pressure Trip	Emergency Core Cooling System
	Safety Injection Actuation Signal	Auxiliary Feedwater System
	Low Reactor Coolant Flow Trip	Containment Isolation
		Containment Spray and Air Cooler

TABLE 14.0.9-1 OVERVIEW OF PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS (CONTINUED)

Event	Reactor Trip Functions	Other Signals and Equipment *
Large-Break Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	No Credit taken for a Reactor Trip by the Reactor Protection System due to the Rapid Depletion of the Moderator which Shuts Down the Reactor Core Almost Immediately, Followed by ECCS Injection Which Contains Sufficient Boron to Maintain the Reactor Core in a Subcritical Configuration	Emergency Core Cooling System
		Auxiliary Feedwater System
		Containment Isolation
		Containment Spray and Air Cooler

* All of the systems and equipment listed for an event would normally be available to mitigate the event consequences. However, the event analysis only takes credit for those systems and equipment which are safety-grade (such as safety valves). In addition, the event analysis assumes that any non safety-grade systems or equipment which would worsen the event consequences (such as condenser steam dump, for certain events) are available.

TABLE 14.0.12-1 NOMENCLATURE USED IN PLOTTED RESULTS

Variable Number	Definition
DCLEVA1	Steam Generator Downcomer Level, Loop 1
DK	Total Reactivity
DKDOP	Doppler Reactivity
DKMOD	Moderator Temperature Reactivity
PD01	Steam Generator Dome Pressure, Loop 1
PL	Core Power Level
PPR	Pressurizer Pressure
PSGSA1	Steam Generator Pressure, Loop 1
QOA	Core Average Heat Flux
TAVEC	Core Average Coolant Temperature
TAVG1	Average Coolant Temperature, Loop 1
TCIO	Core Inlet Coolant Temperature
TCL1	Cold Leg Temperature, Loop 1
THL1	Hot Leg Temperature, Loop 1
VWPR	Pressurizer Liquid Volume
WDOSLT	Total Steamline Steam Flow Rate
WFWT	Total Feedwater Flow Rate
WLPCR	Vessel Flow Rate

14.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

14.1.1 DECREASE IN FEEDWATER TEMPERATURE

14.1.1.1 Event Initiator

A decrease in feedwater temperature may be caused by loss of one or more feedwater heaters. The loss could be due to the interruption of steam extraction flow or to an accidental opening of a feedwater heater bypass line. The worst loss of feedwater heaters incident would occur if all of the low pressure heaters were bypassed. The effects of any decrease in the feedwater temperature due to flow increases (Main or Auxiliary Feedwater (AF)) are discussed in Section 14.1.2.

14.1.1.2 Event Description

Due to a malfunction in the feedwater heater system, the enthalpy of the feedwater being injected into the steam generators is reduced. The increased subcooling of the feedwater reduces the secondary system average fluid enthalpy and increases the energy removal rate from the primary system. The increase in primary to secondary heat transfer causes the reactor coolant temperature at the outlet of the steam generator to decrease. This causes a corresponding decrease in the core inlet coolant temperature. With a negative moderator temperature coefficient (MTC), the reactor core power will begin to increase as the cooler moderator fluid reaches the core.

14.1.1.3 Reactor Protection

Reactor protection is provided by the variable overpower, thermal margin/low-pressure (TM/LP), local power density (LPD), and low steam generator pressure trips. Reactor protection for the decrease in feedwater temperature event is summarized in Table 14.1.1-1.

14.1.1.4 Disposition and Justification

For operating Modes 1-3, the response of the nuclear steam supply system (NSSS) is governed by the magnitude of the overcooling introduced by the initiating event. There is no extraction to the feedwater heaters for operating Modes 4-6. As such, there is not a credible event for these reactor operating conditions.

The most limiting case for Mode 1 is from rated power conditions because the feedwater flow rate and heater duty decrease with load. Also, at rated power conditions, the initial Departure from Nucleate Boiling Ratio (DNBR) margin is minimized. The consequences of the event in Modes 2 and 3 are bounded by those of Mode 1 because the magnitude of the initiating event in Modes 2 or 3 is much smaller than in Mode 1.

This cooldown rate due to bypassing the feedwater heaters is bounded by that of the maximum cooldown event postulated in Section 14.1.3. As such, the consequences of the Increase in Steam Flow (Event 14.1.3) bound the consequences for the Decrease in Feedwater Temperature event discussed in this section. The disposition of events for the Decrease in Feedwater Temperature event is summarized in Table 14.1.1-2.

14.1.2 INCREASE IN FEEDWATER FLOW

14.1.2.1 Event Initiator

This event is initiated by a failure in the feedwater system which causes an increase in the feedwater flow to the steam generators. The initiators considered are complete opening of the feedwater control valves, overspeed of the feedwater pumps, inadvertent start of a second feedwater pump at low power, startup of the auxiliary feedwater system (AFWS), and inadvertent opening of the feedwater control valve bypass lines.

14.1.2.2 Event Description

The increased flow to the steam generators causes an increase in the energy removal capability of the steam generators by reducing the average fluid enthalpy in the steam generators. The increased energy removal from the primary system causes the reactor coolant temperature at the outlet of the steam generator to decrease. The core inlet temperature will correspondingly be reduced, which will cause the core power to increase if the moderator temperature coefficient (MTC) is negative.

Because this event is characterized as a primary system overcooling event, the primary system pressure initially decreases along with the core inlet temperature. There is also a possibility for a core power increase in the presence of a negative moderator reactivity feedback coefficient. Increased reactor power reduces the core Departure from Nucleate Boiling (DNB) margin. A potential exists that the net effect of these three factors will represent a challenge to the core DNB margin.

14.1.2.3 Reactor Protection

Reactor protection for the rated power and power operation conditions (Mode 1) is provided by the variable overpower trip, LPD trip, TM/LP trip, low steam generator pressure trip and by the safety injection actuation signal (SIAS) on low pressurizer pressure. Additional protection is provided by the control grade reactor trip on turbine trip due to high steam generator water level.

For Modes 2 and 3, protection is provided by the low steam generator pressure trip, safety injection actuation signal (SIAS), and the variable overpower trip. Reactor protection for the Increase in Feedwater Flow event is summarized in Table 14.1.2-1.

14.1.2.4 Disposition and Justification

The event consequences at rated power operating conditions will bound the consequences from all other power operating conditions. At rated power operating conditions, the initial thermal margin (DNBR) is minimized. Maximizing the increase in feedwater flow maximizes the load demand. This results in the maximum rate of moderator cooldown which, in the presence of a negative MTC, results in the maximum challenge to the specified acceptable fuel design limits (SAFDLs). Therefore, the limiting consequences of the increase in feedwater flow will occur at the full load rated power conditions and will bound all other power operating conditions due to

the initial steam generator inventory and initial margin to DNB. The greatest cooldown which can be postulated due to feedwater addition at full power is the overspeed of the feedwater pumps. This cooldown is larger than that due to inadvertent startup of the three AFW pumps but less than that calculated for Event 14.1.3, Increased Steam Flow.

The main feedwater system is off-line in Modes 4-6 but may be on-line in Mode 3. For Mode 3 operating conditions, the potential cooldown in conjunction with a negative MTC may result in a return to power at reduced primary pressure, elevated all-rods-in peaking, and less than four reactor coolant pump (RCP) conditions. This case may pose a greater challenge to the SAFDLs than the full power case, and would bound zero power operation in Mode 2 if the cooldown provides sufficient reactivity insertion to overcome the shutdown margin. This is due to the potential for prompt criticality in Mode 3. The greatest increase in feed flow would result from the startup of an idle pump with both control valves full open. The cooldown rate is less than the rate computed for Event 14.1.3 in Mode 3, and consequently Event 14.1.2 in Modes 2 and 3 is bounded by Event 14.1.3 initiated from Mode 3.

In Modes 4-6, the only increased feed flow event initiator is inadvertent startup of one or more AF pumps since the main feedwater system is off-line. The startup of all three AF pumps results in an increased energy removal rate, less than the maximum possible for the Increase in Steam Flow (Event 14.1.3) for Modes 4-6.

The disposition of events for the Increase in Feedwater Flow event is summarized in Table 14.1.2-2.

14.1.3 INCREASE IN STEAM FLOW

14.1.3.1 Event Initiator

This event is initiated by a failure or misoperation in the main steam system which results in an increase in steam flow from the steam generators. This event could be caused by the rapid opening of the turbine control valves (TCVs), the atmospheric steam dump valves (ADVs), the turbine bypass valves (TBVs), the steam dump to condenser valves (SDVs), a safety relief valve (SRV), or the turbine feed pump control valves.

14.1.3.2 Event Description

The increased steam flow resulting from the failure creates a mismatch between the heat being generated in the core and that being extracted by the steam generators. As a result of this power mismatch, the primary-to-secondary heat transfer increases and the primary system cools down. If the MTC is negative, the cooldown of the primary system coolant would cause an insertion of positive reactivity and the potential erosion of thermal margin.

14.1.3.3 Reactor Protection

The main steam system is designed to accommodate a 10% increase in load (step increase). Reactor protection against a main steam flow increase greater than a 10% step is provided by the

following trip signals: variable overpower trip, TM/LP trip, LPD trip, low steam generator water level trip, and low secondary pressure trip. In Modes 1, 2, and 3, protection is also provided by the Safety Injection Actuation Signal (SIAS) on low pressurizer pressure. Reactor protection for the Increase in Steam Flow event is summarized in Table 14.1.3-1.

14.1.3.4 Disposition and Justification

This event is predominantly a cooldown event characterized by a primary pressure and temperature drop with a power rise. Therefore, the most limiting event for power operation is one which results in the highest power, the highest core inlet temperature, and the lowest primary pressure. Core flow remains essentially unchanged. The magnitude of the pressure drop for a given increase in steam flow is about the same regardless of the initial power level. The core inlet temperature will be maximized at HFP. The margin to DNB is the smallest at full power since the product of reactor power and peaking factor decreases as power drops. Additionally, the Variable Overpower trip setpoint will decrease as power decreases, thus providing greater margin to the SAFDLs at lower powers. Therefore, for Mode 1 and for nonzero power operation in Mode 2, the bounding event is one initiated from HFP.

The maximum possible steam release results from the simultaneous opening of the steam dump to condenser valves, the atmospheric dump valves and the turbine bypass valves. Furthermore, simulating the turbine control valves as operating in the “automatic” mode, rather than the “manual” mode, is limiting. Therefore, a spectrum of HFP cases, with steam releases ranging up to that for the steam dump to condenser valves and turbine bypass valves fully open with the turbine control valves operating in the “automatic” mode, were analyze. For Cycle 18, the additional opening of the atmospheric dump valves was evaluated and determined to be bounded by the cases previously analyzed and discussed in Section 14.1.3.5. The effects of power decalibration were also included in the analysis.

The ADVs are sized to accommodate 15% of steam flow at 2700 MWt. The SDVs and the TBV are sized to accommodate 40.5% of steam flow at 2700 MWt. Each SRV will pass 6.75% of steam flow at 2700 MWt. The TCVs are sized to accommodate 111.3% of steam flow at 2700 MWt. The capacities of the control valves for the main feedwater and AF pump turbines are significantly less.

To bound the allowable plant operation with the TCVs in automatic control mode, the TCVs were opened fully, simultaneous with the SDVs and the TBV opening. This energy removal rate bounds those of the rated power operating conditions for Events 14.1.1 to 14.1.2, and 14.1.4. Therefore, this event is analyzed as part of the plant transient analysis for Millstone Unit 2. The consequences of this event for all other operating conditions are bounded by the rated power operating condition due to the increased margin to DNB at the other power operating conditions.

The disposition of events for the Increase in Steam Flow event is summarized in Table 14.1.3-2.

14.1.3.5 Definition of Events Analyzed

A spectrum of HFP cases were analyzed, with steam flows ranging from approximately 11% excess steam flow (turbine control valves fully open) to the excess steam flow associated with the steam dump to condenser valves, turbine bypass valve, and turbine control valves fully opening. The effects of power decalibration were also included in the analysis.

The end-of-cycle (EOC) Doppler feedback coefficient was selected to maximize the challenge to the specified fuel design limits. The consequences of this event are bounded at EOC conditions when the MTC is at its maximum negative value. Therefore, the core operating limits report most negative MTC limit (-32 pcm/°F) was used.

Only full power cooldown and low power events which credit power-dependent reactor trips have the potential to be adversely affected by power decalibration. Power decalibration is caused by density induced changes in the reactor vessel downcomer shadowing the power-range ex-core detectors during heatup or cooldown transients. The nuclear power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full power operation. The Variable Overpower trip, the TM/LP trip function, and the LPD trip all depend on the indicated nuclear power level. The power decalibration effect was included in the modeling of any power-dependent reactor trips credited in this analysis.

The initial conditions for the Increase in Steam Flow event is summarized in Table 14.1.3-3.

14.1.3.6 Analysis Results

The transient for the limiting case (approximately 11% excess steam flow) is initiated by a failure which causes the turbine control valves to open fully at initiation. The responses of key system variables are given in Figures 14.1.3–1 to 14.1.3–9. The sequence of events is given in Table 14.1.3-4. The peak reactor power level calculated for each of the Increase in Steam Flow cases analyzed are listed in Table 14.1.3-5.The increased steam flow (see Figure 14.1.3–7) creates a mismatch between the core heat generation rate and the steam generator heat removal rate. This power mismatch causes the primary-to-secondary heat transfer rate to increase, which in turn causes the primary system to cool down (see Figure 14.1.3–3). With a negative MTC (see Figure 14.1.3–2), the primary system cooldown causes the reactor power level to increase (see Figure 14.1.3–1. However, due to power decalibration, the indicated nuclear power level does not increase along with the reactor power level. Eventually, the indicated thermal power level reaches the Variable Overpower reactor trip ceiling, and the reactor is tripped. This terminates the power excursion.

The minimum DNBR for the limiting Increase in Steam Flow case (with approximately 11% excess steam flow) is bounded by the Section 14.3.1 four RCP loss of flow event, and is above the 95/95 DNBR safety limit. Moreover, the TM/LP trip is designed to protect DNB limits for this event. The LPD LSSS trip ensures that the maximum linear heat rate is below the FCMLHR limit. These results demonstrate that fuel failures do not occur for the Increase in Steam Flow event and that the event acceptance criteria are satisfied.

14.1.3.7 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted for the full power case is greater than the safety limit. The correlation limit assures that with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The FCMLHR limit is not violated during this event.

14.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

14.1.4.1 Event Initiator

This event is initiated by an increase in steam flow caused by the inadvertent opening of a secondary side safety or relief valve.

14.1.4.2 Event Description

The resulting mismatch in energy generation and removal rates results in an overcooling of the primary system. If the MTC is negative, the reactor power will increase.

14.1.4.3 Reactor Protection

Reactor protection is provided by the variable overpower trip, LPD trip, TM/LP trip, low secondary pressure trip, and low steam generator water level trip. In Modes 1, 2, and 3, protection is also provided by the safety injection actuation signal (SIAS) on low pressurizer pressure. Reactor protection for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event is summarized in Table 14.1.4-1.

14.1.4.4 Disposition and Justification

The inadvertent opening of a steam generator safety valve would result in an increased steam flow of approximately 6.75% of full rated steam flow. Each dump (relief) valve is sized for approximately 7.50% steam flow with the reactor at full rated power. As such, the consequences of any of these occurrences will be bounded by the events in Section 14.1.3. The disposition of events for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event is summarized in Table 14.1.4-2.

14.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT

Two separate analyses have been performed for the Steam Line Break (SLB) event.
Section 14.1.5.1 describes the pre-scram analysis performed to determine Departure from Nucleate Boiling Ratio (DNBR) and Linear Heat Generation Rate (LHGR) up to and including reactor trip. This time period represents the highest reactor power condition and the assumptions have been selected to minimize DNBR and maximize LHGR during this time frame.
Section 14.1.5.2 describes the post-scram analyses performed to determine MDNBR and LHGR during the return to power caused by the overcooling. A different set of assumptions and single

failure were determined to minimize MDNBR and maximize LHGR for the return to power time frame.

14.1.5.1 Pre-Scram Analysis

14.1.5.1.1 Event Initiator

The pre-scram SLB analysis is initiated by a rupture in the main steam piping which results in an uncontrolled steam release from the secondary system.

14.1.5.1.2 Event Description

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. With a negative MTC, the primary system cooldown causes the reactor power level to increase. If the break is not large enough to trip the reactor on a Low Steam Generator Pressure signal, the cooldown will continue until the reactor is tripped on a Variable Overpower or TM/LP signal (for breaks outside containment) or a High Containment Pressure signal (for breaks inside containment) or until the reactor reaches a new steady-state condition at an elevated power level.

Although the SLB calculation is typically a cooldown event, for the pre-scram analysis the cooldown event is not significant for the limiting pre-scram case. The case with a loss of off site power, also known as a “pumps off” case, credits the low reactor coolant flow trip for harsh conditions. In this case, the Reactor Coolant Pumps (RCPs) are tripped shortly after the initiation of the transient. The sharp reduction in reactor coolant flow causes the pre-scram pumps off calculation to become a heat up transient very similar to a Loss of Coolant Flow (LOCF). Therefore, the conditions for this case are biased as if it were a LOCF (i.e. BOC neutronics). This case becomes a combination of an MSLB and an LOCF event.

14.1.5.1.3 Reactor Protection

Reactor protection is provided by the low steam generator pressure and water level trips, variable overpower trip, LPD trip, TM/LP trip, high containment pressure trip, low reactor coolant flow, and SIAS. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.1-1.

14.1.5.1.4 Disposition and Justification

HFP initial conditions are limiting for the pre-scram SLB cases since this is the highest power condition.

The outside containment breaks do not cause harsh conditions inside containment, and therefore, do not cause the Low Reactor Coolant Flow trip to be degraded. If a loss of off site power were concurrent with an outside containment break, the primary coolant flow rate would coastdown similar to an LOCF event, without the Low Reactor Coolant Flow trip being degraded. The

outside containment break case with loss of off site power is therefore bounded by the LOCF event.

The inside containment breaks do cause harsh conditions inside containment, and therefore, an increased allowance for instrument uncertainty was applied for the Low Reactor Coolant Flow trip. Therefore, only the inside containment breaks will be analyzed with a loss of off site power.

The following pre-scram HFP Steam Line Break cases were analyzed for a range of MTCs (up to the most negative core operating limits report limit) and for break sizes ranging up to a double-ended guillotine break in a main steam line. The effects of power decalibration and harsh containment conditions (where applicable) were included in the analysis:

1. Breaks outside containment and downstream of the check valves (symmetric cases)

2. Breaks outside containment and upstream of a check valve (asymmetric cases)

3. Breaks inside containment with RCPs on (asymmetric cases)

4. Breaks inside containment with RCPs off (asymmetric cases)

The event is analyzed to support a range of MTCs up to the most negative core operating limits report limit. This event must be analyzed both with and without a coincident loss of off site power.

The single failure assumed in this analysis is the loss of one channel of Nuclear Instrumentation (NI) which provides power indication to the RPS. If one channel is out of service, the three remaining NI safety channels will be in a 2-out-of-3 coincidence mode. With the assumption of a failure in one of these channels, both of the remaining channels are required for a trip, relying on the lowest power indication for the safety function.

The disposition of events for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.1-2.

14.1.5.1.5 Definition of Events Analyzed

The pre-scram SLB event is initiated by a rupture in the main steam piping. The break location is downstream of the steam generator integral flow restrictor and either:

1. outside containment and upstream of the main steam line check valves (asymmetric break), or

2. outside containment and downstream of the main steam line check valves (symmetric break), or

3. inside containment and upstream of the main steam check valves (asymmetric break).

Steam released through a break located downstream of the main steam line check valves flows to the break from both steam generators and, therefore, results in a symmetric transient. However, steam released through a break located upstream of one of the check valves flows to the break from the upstream steam generator only (because the check valve precludes backflow to the break from the other steam generator) and, therefore, results in an asymmetric transient.

Power decalibration is caused by density induced changes in the reactor vessel downcomer shadowing of the power range ex-core detectors during heatup or cooldown transients. The nuclear power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full power operation (and higher when the vessel inlet coolant is warmer than the normal full power temperature). This effect is included in the modeling of any power-dependent reactor trips credited in the analysis of full power cooldown events and low power events. The Variable Overpower trip, the Thermal Margin/Low Pressure (TM/LP) trip function, and the Local Power Density (LPD) trip all depend on the indicated nuclear power level.

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

As outlined in Reference 14.1-1, three computerized calculations are required prior to the final calculation of the Minimum Departure From Nucleate Boiling Ratio (MDNBR) values and the maximum Linear Heat Generation Rate (LHGR) values utilized in the determination of fuel failure. The NSSS response of the pre-scram SLB with AC power available is computed using the AREVA S-RELAP5 code (Reference 14.1-1), the detailed core and hot assembly power distributions are calculated using the PRISM code (Reference 14.1-3), and the detailed core and hot assembly flow and enthalpy distributions are calculated using the XCOBRA-IIIC code (Reference 14.1-4). The HTP correlation (Reference 14.1-5) was utilized to calculate MDNBR. The previous pre-scram SLB case with the concurrent loss of off site power analyzed using the ANF RELAP (Reference 14.1-2) code was determined to remain bounding. The results of this case are described in Section 14.1.5.1.6.2.

14.1.5.1.5.1 Analysis of Results

The S-RELAP5 analysis provides the NSSS boundary conditions for the PRISM and the XCOBRA-IIIC calculations. This section presents a description of the treatment of factors which can have a significant impact on NSSS response and resultant MDNBR and LHGR values. The plant specific parameters used in this analysis are listed in Tables 14.1.5.1-3 to 14.1.5.1-5. Conservatisms are included in parameters or factors known to have significant effects on the NSSS performance and resulting MDNBR and LHGR values.

14.1.5.1.5.1.1 Break Location, Size, and Flow Model

The pre-scram SLB event analyzes breaks outside containment both downstream (symmetric cases) and upstream (asymmetric cases) of the main steam line check valves and breaks inside containment (asymmetric cases). A full range of break sizes, up to the double-ended guillotine break of a main steam line, were considered.

The S-RELAP5 break mass flow rate is computed using the Moody critical flow model modified such that only steam flows out the break.

14.1.5.1.5.1.2 Power Decalibration

Power decalibration is caused by density-induced changes in the reactor vessel downcomer shadowing of the power-range ex-core detectors during heatup or cooldown transients. The nuclear power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full power operation (and higher when the vessel inlet coolant is warmer than the normal full power temperature). This effect is included in the modeling of any power-dependent reactor trips credited in the analysis of full power cooldown events and low power events. The Variable Overpower trip, the Thermal Margin/Low Pressure (TM/LP) trip function, and the Local Power Density (LPD) trip all depend on the indicated nuclear power level.

14.1.5.1.5.1.3 Harsh Containment Conditions

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

14.1.5.1.5.1.4 Boron Injection

Boron injection into the primary system acts to mitigate the return to power. Injection of boron is modeled from the HPSI system. The HPSI system is conservatively modeled to take suction from the Refueling Water Storage Tank (RWST) at 35°F with a boron concentration of 1720 ppm. Initially, the line volume between the check valves isolating the system pumps and the cold leg injection location is assumed to be filled with unborated water. The time required to flush this unborated water from the safety injection lines is included as an integral part of the S-RELAP5 NSSS calculation. In the pre-scram SLB event, the analysis is terminated shortly after reactor trip, therefore injection of borated water is not a factor in the analysis.

14.1.5.1.5.1.5 Single Failure Assumption

In order to simulate the asymmetric thermal-hydraulic and reactivity feedback effects that occur during the pre-scram SLB event, the core is divided into an affected sector (one-half of the core) and an unaffected sector (one-half of the core). The single failure assumed in this analysis is the loss of one channel of Nuclear Instrumentation (NI) which provides power indication to the

Reactor Protection System (RPS). If one channel is out of service, the three remaining NI safety channels will be in a 2-out-of-3 coincidence mode to cause a reactor trip. The excore detectors are placed around the reactor vessel in positions that result in one detector seeing the flux only from the affected region, one seeing the flux only from the unaffected region, and two detectors seeing nearly equal flux from both regions. If one of these latter two is out of service, and the other is assumed to be a single failure, the remaining two channels will be required to cause an RPS trip (high power or TM/LP). Since the power in the affected region will always be higher than in the unaffected region, it is sufficient to model the NI channel reading the unaffected region only.

14.1.5.1.5.1.6 Feedwater

For breaks smaller than 0.4 square feet, the Main Feed Water (MFW) control valves are assumed to modulate to match MFW flow to steam flow. For larger breaks, the MFW valves are conservatively assumed to open fully at the initiation of the event and remain open. The MFW pumps are assumed to be operating at rated speed. The MFW flow increases as the secondary pressure decreases at the lowest possible fluid temperature until the feedwater regulator valve closes. Fluid temperature is determined by assuming heating of the feedwater ceases at the same time the break is initiated. The MFW flow is terminated 14 seconds after receiving the isolation signal.

14.1.5.1.5.1.7 Trips and Delays

Actuation signals and delays are given in Table 14.1.5.1-4. Biases to account for uncertainties are included in the trip setpoints as shown. In the pre-scram SLB event, the analysis is terminated shortly after reactor trip, therefore injection of borated water is not a factor in the analysis.

14.1.5.1.5.1.8 Neutronics

The core kinetics input for this calculation consisted of the minimum required control rod shutdown worth at EOC, and EOC values associated with the reactivity feedback curves, delayed neutron fraction, delayed neutron fraction distribution and related time constants, and prompt neutron generation time. The S-RELAP5 default fission product and actinide decay constants were utilized for this calculation.

The core reactivity is derived from input of several functions. These include effects from control rod worth, moderator density changes, boron concentration, and Doppler effects. The reactivity is weighted between the core sectors. The S-RELAP5 analyses for cases with off site power available were performed with an MTC ranging from -8 pcm/°F to -32 pcm/°F. In all cases, the most limiting scenarios were those performed with an MTC of -32 pcm/°F. The analyses for cases with a loss of off site power were performed with an MTC of +4.0 pcm/°F. A summary of the nuclear input and assumptions is given in Table 14.1.5.1-5.

14.1.5.1.5.1.9 Decay Heat

The presence of radioisotope decay heat at the initiation of the SLB event will reduce the rate and the extent of cooldown of the primary system. The initial decay heat is calculated on the basis of

infinite irradiation time at a power of 2754 MW prior to transient initiation. This treatment of decay heat serves to maximize the stored energy and provide limiting stored energy conditions for the SLB cases.

14.1.5.1.5.1.10 Nodalization

The NSSS transient calculations presented in this report utilized the nodalization model described in Reference 14.1-1. The nodalization treats all major NSSS components and subcomponents as discrete elements, with the exception of the secondary side of the steam generators. In addition, all components with long axial dimensions are divided into subcells adequate to minimize numerical diffusion and smearing of gradients.

In order to simulate the asymmetric thermal-hydraulic and reactivity feedback effects that occur during the pre-scram SLB event, the core is divided into an affected sector (one-half of the core) and an unaffected sector (one-half of the core).

14.1.5.1.5.1.11 Interloop Mixing

During an actual SLB transient, some mixing between the parallel channels within the reactor pressure vessel will occur in the downcomer, the lower plenum, the core, and the upper plenum due to lateral momentum imbalances, and turbulence or eddy mixing. The mixing will act to reduce the positive reactivity feedback effects due to a reduced rate and magnitude of cooldown of the affected loop and associated core sector.

In this analysis, no credit is taken for turbulent or eddy mixing of coolant between loops or the parallel flow channels within the reactor pressure vessel. However, interloop mixing is calculated to occur due to flow in interloop junctions in the upper and lower plenums. Mixing in the lower plenum was effectively reduced to zero by using an extremely high loss coefficient between the affected and intact sectors.

14.1.5.1.5.2 Minimum Departure From Nucleate Boiling and Linear Heat Generation Rate Analysis

The PRISM (Reference 14.1-3) core neutronics code is used to calculate the core radial power distributions for XCOBRA-IIIC (Reference 14.1-4) during the asymmetric transients with off site power available only. The PRISM model is a three-dimensional representation of the entire core, with four radial nodes and 24 axial nodes for each fuel assembly.

Based on the overall core conditions calculated by S-RELAP5 for the symmetric cases (or S-RELAP5 and PRISM for the asymmetric cases with off site power available) at the peak heat flux time-point, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single “channel”) linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single “channel”). The limiting assembly DNBR calculations are performed using the HTP DNB correlation (Reference 14.1-5).

The DNBR calculation is performed with the methodology described in Reference 14.1-6

For the asymmetric transients, the radial power peaking is augmented above the Technical Specification limit to account for the increase in radial power peaking which occurs during the transient. The increase in peaking is determined by PRISM.

14.1.5.1.6 Analysis Results

A summary of the calculated peak reactor power is presented in Table 14.1.5.1-6 for the limiting MDNBR and LHGR cases. The MDNBR values for the limiting cases are also in this Table. For cases where off site power was available for operation of the primary coolant system pumps, the MDNBR and the maximum LHGR occurred at the time of the maximum power condition. For cases where off site power is lost and the primary system pumps coast down, the maximum LHGR and the MDNBR occur when the worst combination of core power, flow, inlet temperature, and pressure are present.

The scenario which resulted in the highest power level and the largest LHGR is the HFP 0.20 ft^2 asymmetric break inside containment with off site power available for operation of the primary coolant pumps. This case is presented in detail.

The scenario which resulted in the limiting MDNBR is the HFP case with a loss of off site power and is also presented in detail.

14.1.5.1.6.1 Hot Full Power 0.20 ft^2 Break Inside Containment Upstream of a Check Valve with Off site Power Available

The S-RELAP5 simulation of the NSSS during the HFP asymmetric break transient with off site power available is illustrated in Figures 14.1.5.1–1 through 14.1.5.1–6. A tabulation of the sequence of events is presented in Table 14.1.5.1-7. The S-RELAP5 computation was terminated well beyond the time of MDNBR or peak LHGR. The general response of the reactor was the same for all the asymmetric break sizes but the sequence of events was quicker as the break size increased.

14.1.5.1.6.1.1 Secondary System Parameters

Upon break initiation the break flow increased sharply and then began to decline in response to falling secondary side pressure. When the turbine trip occurred, the break flow increased due to a local pressure increase. The main steam line flow rate from each generator initially increased (see Figure 14.1.5.1–6) in response to the break and the assumed instantaneous full opening of the turbine control valves. The increased steam flow creates a mismatch between the core heat generation rate and the steam generator heat removal rate. This power mismatch causes the primary-to-secondary heat transfer rate to increase, which in turn causes the primary system to cool down (see Figure 14.1.5.1–2). When the reactor scram occurred, the turbine valves closed and steam flow declined sharply.

14.1.5.1.6.1.2 Primary System Parameters

Approximately five seconds after the break occurred, the core inlet temperature began to decline. With a negative MTC (see Figure 14.1.5.1–3), the primary system cooldown caused the reactor power level to increase. The core power continues to increase until reactor scram on high containment pressure occurs. This terminated the power excursion. The pressurizer pressure and level began to decline as the volume of water in the primary system shrank. The core inlet mass flow rate increased due to the increasing density of the primary system fluid while the reactor coolant pumps' speed remained constant.

14.1.5.1.6.1.3 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for this scenario was calculated and compared to the 95/95 HTP correlation limit. The amount of failed fuel is determined by comparing the core power distribution to the power at which DNB occurs. The scenario results in no more than a predicted failure of 3.7% of the fuel rods in the core.

The peak LHR for the LHR-limiting case (0.20 ft^2 break inside containment and upstream of a check valve) is calculated to be less than the FCMLHR limit. Therefore, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.1.6.2 Hot Full Power 3.51 ft^2 Inside Containment Asymmetric Break Concurrent with a Loss of Off site Power

The ANF-RELAP NSSS simulation of the most limiting pre-scram SLB scenario from an MDNBR standpoint (i.e., HFP 3.51 ft^2 inside containment asymmetric break concurrent with a loss of off site power) is illustrated in Figures 14.1.5.1–7 through 14.1.5.1–11. A tabulation of the sequence of events is presented in Table 14.1.5.1-8. The ANF-RELAP computation was terminated 60 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR.

The transient is initiated by the opening of the break. The RCPs tripped shortly after transient initiation. The sharp reduction in the reactor coolant flow causes this pre-trip pumps off calculation to become a heat up transient very similar to a Loss of Coolant Flow event. Typically, the Steam Line Break calculation is a cooldown event. Because this case is a heat up event the most positive BOC neutronics conditions are used, and the maximum inside containment asymmetric break size is used. The maximum break size causes the biggest decrease in primary pressure. Maximizing the primary system pressure decrease causes the maximum decrease in moderator density and the maximum positive moderator feedback. The RCP trip causes the RCS flow to decrease rapidly throughout this transient. The decreasing RCS flow causes the transient time of the fluid in the core to increase and the fluid temperature begins to rise. The increasing fluid temperature causes positive moderator feedback, which in turn causes an increase in core power. However, the decreasing RCS flow causes the heat transfer to the fluid to decrease. The increase in core power is offset by the decrease in heat transfer from the fuel rods, such that, the

fuel rod heat flux decreases slightly until reactor scram. The reactor scrams on the low reactor coolant flow trip signal.

14.1.5.1.6.2.1 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for the pre-scram 3.51 ft^2 asymmetric break inside containment with a loss of off site power was calculated to be below the 95/95 HTP correlation limit. The number of failed assemblies is determined by comparing the core power distribution to the assembly power where DNB occurs. This results in a predicted failure of 3.7% of the fuel rods in the core.

The peak LHR for this case is bounded by the 0.20 ft^2 inside containment asymmetric break. Therefore, the LHGR for this case is below the FCMLHR limit and no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.1.7 Conclusions

The HFP 0.20 ft^2 break inside containment and upstream of a check valve (asymmetric break) with off site power available was determined to be the most limiting in this analysis from an LHGR standpoint. In no scenario evaluated, however, was fuel failure calculated to occur as a result of violating the FCMLHR limit.

Both the HFP 0.20 ft^2 asymmetric break inside containment with off site power available and the HFP 3.51 ft^2 asymmetric break inside containment coincident with a loss of off site power were determined to be the most limiting in this analysis from the standpoint of MDNBR. Predicted fuel failures due to MDNBR are limited to no more than 3.7% of the fuel rods in the core. However, for any outside containment main steam line break scenario, no fuel failure due to violation of the DNBR limit was predicted to occur.

14.1.5.2 Post-Scram Analysis

14.1.5.2.1 Event Initiator

This event is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the MSIVs which results in an uncontrolled steam release from the secondary system.

14.1.5.2.2 Event Description

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. In the presence of a negative MTC, this cooldown causes a decrease in the shutdown margin (following reactor scram) such that a return to power might be possible following a steam line rupture. This is a potential problem because of the high power peaking factors which exist, assuming the most reactive control rod to be stuck in its fully withdrawn position.

14.1.5.2.3 Reactor Protection

Reactor protection is provided by the low steam generator pressure and water level trips, LPD trip, TM/LP trip, high containment pressure trip, and SIAS. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.2-1.

14.1.5.2.4 Disposition and Justification

At rated power conditions, the stored energy in the primary coolant is maximized, the available thermal margin is minimized, and the pre-trip power level is maximized. These conditions result in the greatest potential for cooldown and provide the greatest challenge to the SAFDLS. Initiating this event from rated power also results in the highest post-trip power since it maximizes the concentration of delayed neutrons providing for the greatest power rise for a given positive reactivity insertion. Thus, this event initiated from rated power conditions will bound all other cases initiated from at power operation modes.

For the zero power and subcritical plant states (Modes 2-6), there is a potential for a return to power at reduced pressure conditions. The most limiting steam line break (SLB) event at zero power is one which is initiated at the highest temperature, thereby providing the greatest capacity for cooldown. This occurs in Modes 2 and 3. Thus, the event initiated from Modes 2 and 3 will bound those initiated from Modes 4-6. Further, the limiting initial conditions will occur when the core is just critical. These conditions will maximize the available positive reactivity and produce the quickest and largest return to power. Thus, the SLB initiated from critical conditions in Mode 2 will bound the results of the event initiated form subcritical Mode 3 conditions.

The Technical Specifications only require a minimum of one RCP to be operating in Mode 3. One pump operation provides the limiting minimum initial core flow case. Minimizing core flow minimizes the clad to coolant heat transfer coefficient and degrades the ability to remove heat generated within the fuel pins. Conversely, however, a maximum loop flow will maximize the primary to secondary heat transfer coefficient, thus providing for the greatest cooldown. Higher loop flow will sweep the cooler fluid into the core faster, maximizing the rate of positive reactivity addition and the peak power level.

The worst combination of conditions is achieved for the four-pump loss of off site power case. In this situation, the initial loop flow is maximized resulting in the greatest initial cooldown, while the final loop flow is minimized providing the greatest challenge to the DNB SAFDL. Since the natural circulation flow which is established at the end of the transient will be the same regardless of whether one or four pumps were initially operating the results of the four-pump loss of off site power case will bound those of the one-pump case. Thus, only four-pump operation need be analyzed for the Mode 2 case.

The event is analyzed to support the Technical Specification EOC MTC limit. This event must be analyzed both with and without a coincident loss of off site power. Typically there are two single failures which are considered for the off site power available case. The first is failure of a High Pressure Safety Injection (HPSI) pump to start. The second is failure of an MSIV to close,

resulting in a continued uncontrolled cooldown. However, Millstone 2 has combination MSIV/ swing disc check valves. A double valve failure would thus be required for steam from the intact steam generator to reach the break. This is not deemed credible. Thus, the single failure to be considered with off site power available is failure of a HPSI pump to start. For the loss-of-off site power case, the limiting single failure is the failure of a diesel generator to start. This is assumed to result in the loss of one HPSI pump. The disposition of events for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.2-2.

14.1.5.2.5 Definition of Events Analyzed

The post-scram SLB is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the MSIVs which results in an uncontrolled steam release from the secondary system. The effects of harsh containment conditions (where applicable) are included in the following analyses:

1. HFP and HZP breaks outside containment with off site power available

2. HFP and HZP breaks outside containment with a loss of off site power

3. HFP and HZP breaks inside containment with off site power available

4. HFP and HZP breaks inside containment with a loss of off site power

The event is analyzed to support the Technical Specification EOC MTC limit. This event must be analyzed both with and without a coincident loss of off site power.

The single failure assumed in this analysis results in the disabling of one of the two HPSI pumps required to be in service during normal operation. In addition to the single failure, there is no credit taken for the charging pump system. This assumption results in an additional delay in the time required for boron to reach the core. The delay is amplified when combined with the assumption of a stagnant upper head which serves to maintain the primary system pressure due to flashing of the hot fluid in the upper head.

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. In the presence of a negative MTC, this cooldown results in a large decrease in the shutdown margin and a return to power. This return to power is exacerbated because of the high power peaking factors which exist, with the most reactive control rod stuck in its full withdrawn position.

As outlined in Reference 14.1-1, three computerized calculations are required prior to the final calculation of the Minimum Departure From Nucleate Boiling Ratio (MDNBR) values and the maximum Linear Heat Generation Rate (LHGR) values utilized in the determination of fuel failure. The NSSS response is computed using the AREVA S-RELAP5 code (Reference 14.1-1), the detailed core and hot assembly power distributions and the reactivity at the time of peak post-scram power are calculated using the PRISM code (Reference 14.1-3), and the detailed core and hot assembly flow and enthalpy distributions are calculated using the XCOBRA-IIIC code

(Reference 14.1-4). The Biasi correlation (Reference 14.1-7) was utilized to calculate MDNBR due to the reduced pressures occurring during the SLB event.

14.1.5.2.5.1 Analysis of Results

The S-RELAP5 analysis provides the NSSS boundary conditions for the PRISM and the XCOBRA-IIIC calculations. This section presents a description of the treatment of factors which can have a significant impact on NSSS response and resultant MDNBR and LHGR values. The plant specific parameters used in this analysis are listed in Tables 14.1.5.2-3 to 14.1.5.2-5. Conservatisms are included in parameters or factors known to have significant effects on the NSSS performance and resulting MDNBR and LHGR values.

14.1.5.2.5.1.1 Break Location, Size, and Flow Model

The post-scram SLB event is initiated by a double-ended guillotine break of a main steam line downstream of the integral steam generator flow restrictors and upstream of the MSIVs. The flow is choked at the integral steam generator flow restrictor, which has an area of 3.51 ft^2. On the steam generator side of the break, steam flows out of the break throughout the entire transient. On the MSIV side of the break, break flow terminates after the MSIVs are fully closed. As an added conservatism, the main steam check valves are not credited in the analysis. The event occurs concurrent with the most reactive control rod stuck out of the core. The break flow areas for the affected and intact steam generators are listed in Table 14.1.5.2-3.

The S-RELAP5 break mass flow rate is computed using the Moody critical flow model modified such that only steam flows out the break.

14.1.5.2.5.1.2 Boron Injection

Boron injection into the primary system acts to mitigate the return to power. Injection of boron is modeled from the HPSI system. The HPSI system is conservatively modeled to take suction from the Refueling Water Storage Tank (RWST) at 35°F with a boron concentration of 1720 ppm. Initially, the line volume between the check valves isolating the system pumps and the cold leg injection location is assumed to be filled with unborated water. The time required to flush this unborated water from the safety injection lines is included as an integral part of the S-RELAP5 NSSS calculation. The characteristics of the HPSI system are listed in Table 14.1.5.2-3. The delivery curve for the HPSI system used in this analysis is given in Figure 14.1.5.2–1.

14.1.5.2.5.1.3 Single Failure Assumption

The single failure assumed in the engineered safeguards system results in the disabling of one of the two HPSI pumps required to be in service during normal operation. In addition to the single failure, there is no credit taken for the charging pump system. This assumption results in an additional delay in the time required for boron to reach the reactor core. The delay is further amplified when combined with the assumption of a stagnant upper head which serves to maintain the primary system pressure due to flashing of the hot fluid in the upper head.

14.1.5.2.5.1.4 Feedwater

For the HFP scenarios, normal MFW flow is assumed to be delivered to both Steam Generators. The MFW flow increases as the secondary pressure decreases at the lowest possible fluid temperature until the feedwater regulating valve closes. Fluid temperature is determined by assuming heating of the feedwater ceases at the same time the break is initiated. The MFW flow is terminated 14 seconds after receiving the isolation signal. In addition, a sensitivity study was performed on the effect of MFW on the post-scram SLB results. The study concluded that the post-scram SLB results are insensitive to the MFW flow.

For the HFP scenarios, the AFW flow is assumed to be zero at break initiation. After 180 seconds, AFW is delivered at the maximum capacity of the AFW system with flow restrictors installed on the AFW delivery lines. For the HZP scenarios, the AFW flow is increased to the maximum capacity immediately at break initiation. For all HZP and HFP cases, the maximum AFW flow deliverable due to presence of cavitating venturis (660 gpm) is delivered to both the affected and unaffected steam generator. AFW flow to the steam generators is continued until S-RELAP5 computation was terminated 600 seconds after break initiation. By 600 seconds, core power is decreasing. After this point in time, the operators will terminate AFW flow to the affected steam generator, which will cause the primary system temperatures to increase, further decreasing the core power. The timing of the isolation of AFW to the affected steam generator is not a critical parameter.

14.1.5.2.5.1.5 Trips and Delays

Trips for the HPSI, main feedwater valves, and MSIVs are given in Table 14.1.5.2-4. Biases to account for uncertainties are included in the trip setpoints as shown. For the steam and feedwater valves, the delay times given are between the time the trip setpoint is reached and the time full valve closure is reached. For the HPSI system, the delay time given is from the time the setpoint is reached until the pumps have accelerated to rated speed. Additional delay time required to sweep the lines of unborated water is accounted for by setting the boron concentration of the injected flow to zero until the volume of the injection lines has been cleared.

14.1.5.2.5.1.6 Neutronics

The core kinetics input for this calculation consisted of the minimum required control rod shutdown worth at the EOC, and EOC values associated with the reactivity feedback curves, delayed neutron fraction, delayed neutron fraction distribution and related time constants, and prompt neutron generation time. The S-RELAP5 default fission product and actinide decay constants were utilized for this calculation.

The core reactivity is derived from input of several functions. These include effects from control rod worth, moderator density changes, boron concentration, and Doppler effects. The reactivity is weighted between the core sectors. Different reactivity functions were utilized where necessary for the HZP and the HFP cases. The HZP and HFP cases were performed with an MTC of -30 pcm/°F. A summary of the nuclear input and assumptions is given in Table 14.1.5.2-5.

14.1.5.2.5.1.7 Decay Heat

The presence of radioisotope decay heat at the initiation of the SLB event will reduce the rate and the extent of cooldown of the primary system. For the HFP scenarios, the initial decay heat is calculated on the basis of infinite irradiation time at a power of 2700 MW prior to transient initiation. For the HZP scenarios, the initial decay heat is calculated on the basis of infinite irradiation time at a power of 1 W prior to transient initiation. For both scenarios, decay heat generated from return to power is calculated. This treatment of decay heat serves to maximize the stored energy in the HFP cases and to minimize it in the HZP cases. This treatment provides limiting stored energy conditions for the SLB cases.

14.1.5.2.5.1.8 Nodalization

The NSSS transient calculations utilized the nodalization model described in Reference 14.1-1. The nodalization treats all major NSSS components and subcomponents as discrete elements, with the exception of the secondary side of the steam generators. In addition, all components with long axial dimensions are divided into subcells adequate to minimize numerical diffusion and smearing of gradients.

In order to simulate the asymmetric thermal hydraulic and reactivity feedback effects that occur during an SLB transient, the core is nodalized into three radial sectors. One sector corresponds to the region immediately surrounding the assembly where the most reactive control rod is assumed stuck out of the core. This sector is termed the 'stuck rod' sector. The remainder of the region of the core which is directly affected by the loop containing the break is the second sector and is termed the 'affected' sector. The remainder of the core and the other loop is termed either the 'unaffected' or the 'intact' sector or loop.

14.1.5.2.5.1.9 Interloop Mixing

During an actual SLB transient, some mixing between the parallel channels within the reactor pressure vessel will occur in the downcomer, the lower plenum, the core, and the upper plenum due to lateral momentum imbalances, and turbulence or eddy mixing. The mixing will act to reduce the positive reactivity feedback effects due to a reduced rate and magnitude of cooldown of the affected loop and associated core sector.

In this analysis, no credit is taken for turbulent or eddy mixing of coolant between loops or the parallel flow channels within the reactor pressure vessel (RPV). However, interloop mixing is calculated to occur due to flow in interloop junctions in the upper and lower plenums. Mixing in the lower plenum was reduced to a minimum by using an extremely high loss coefficient between the affected and intact sectors.

14.1.5.2.5.1.10 Harsh Containment Conditions

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh

environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

14.1.5.2.5.2 Minimum Departure From Nucleate Boiling and Linear Heat Generation Rate Analysis

MDNBR calculations require determination of the power, enthalpy, and flow distributions within the highest power assembly of the stuck rod core sector. Similarly, determination of the maximum LHGR also requires characterization of the power distribution. The power distribution within the core, including the highest powered assembly within the stuck rod core sector, is calculated with PRISM (Reference 14.1-3). Flow and enthalpy distributions within the core, including the highest powered assembly within the stuck rod core sector, are calculated with XCOBRA-IIIC (Reference 14.1-4). In order to obtain compatible flows, moderator densities, and powers within the high power assemblies, iteration between PRISM and XCOBRA-IIIC is conducted.

The Biasi correlation is for the MDNBR calculation (Reference 14.1-7). The Biasi correlation is originally based on academic experimental burnout data. Reference 14.1-7 provides a design limit for the Biasi correlation suitable for application to HTP fuel.

14.1.5.2.6 Analysis Results

A summary of calculated results important to this analysis is presented in Table 14.1.5.2-6 for the limiting MDNBR and LHGR scenarios. The MDNBR values are listed together with the corresponding core power values at the time of MDNBR which corresponds to the maximum post-scram power level. The outside containment cases, regardless of whether or not off site power was or was not available, were found to be the most limiting. For cases where off site power was available for operation of the primary coolant system pumps, the MDNBR and the maximum LHGR occurred at the time of the maximum power condition. For cases where off site power is lost and the primary system pumps coast down, the maximum LHGR and the MDNBR occur when the worst combination of core power, flow, inlet temperature, and pressure are present. These conditions occurred at the time of peak power in this analysis.

The scenario which resulted in the highest post-scram power level and the largest LHGR is that initiated from HZP with the break occurring outside containment and with off site power available for operation of the primary coolant pumps. This case is presented in detail.

The NSSS responses for the scenarios with loss of off site power for operation of the primary system coolant pumps are different from those scenarios where off site power is available throughout the transient due to the pump coastdown and subsequent natural circulation of the primary coolant. Post-scram maximum power levels attained during the transient are significantly lower. Lower power levels result from lower positive moderator feedback. The positive moderator feedback is reduced due to the coolant density reductions that occur axially upwards in the core at low core flow rates, even for low core power levels. Lower power levels cause MDNBR values to increase, but lowering flow rates cause MDNBR values to decrease. Overall, the combination of factors results in lower MDNBR values for the reduced flow condition than for the full flow condition.

Of the two loss-of-off site-power scenarios analyzed, the HZP break occurring outside containment case resulted in lower MDNBR values. The general response of the HFP and HZP cases with loss of off site power is comparable. Because the two scenarios are quite similar in terms of their general response, only the limiting MDNBR case (i.e., HZP break outside containment and without off site power) is presented in detail.

14.1.5.2.6.1 Hot Zero Power Outside Containment with Off site Power Available

The S-RELAP5 simulation of the NSSS during the HZP transient with off site power available is illustrated in Figures 14.1.5.2–2 through 14.1.5.2–9. A tabulation of the sequence of events is presented in Table 14.1.5.2-7. The S-RELAP5 computation was terminated 600 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR.

14.1.5.2.6.1.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Break flow for the steam generators is plotted in Figure 14.1.5.2–2. Secondary pressure for the steam generators is plotted in Figure 14.1.5.2–3. After break initiation, the pressure in the affected steam generator decreased immediately and then stabilized around 180 seconds. The mass inventory in the affected steam generator decreased rapidly for the first 100 seconds of the transient and slowly thereafter. The unaffected steam generator mass inventory increased following the initial drop once the MSIV closed due to the addition of AFW.

The intact steam generator blows down for a short period until the MSIVs completely close approximately 15 seconds after the break is initiated. The pressure recovers following MSIV closure. The unaffected steam generator pressure begins to decrease again as cold AFW fills the steam generator.

14.1.5.2.6.1.2 Primary System Thermal Hydraulic Parameters

The primary system coolant temperature and pressurizer pressure and level responses resulting from the break flow are illustrated in Figures 14.1.5.2–4 through 14.1.5.2–6. The primary system pressure decays rapidly as the coolant contracts due to cooldown and the pressurizer empties. The MSIVs close at 15 seconds, ending the blowdown of the intact steam generators and reducing the rate of energy removal from the primary fluid. The pressurizer emptied at approximately 40 seconds. After the rapid decrease in system pressure caused by the overcooling resulting from the break, the system pressure stabilized and followed the pressure established by the saturation temperature of the primary coolant in the upper head of the reactor vessel.

14.1.5.2.6.1.3 Reactivity and Core Power

The reactivity transient calculated by S-RELAP5 is illustrated in Figure 14.1.5.2–8. Initially, the core is assumed to be at zero power. All control rods, except the most reactive one, are assumed to be inserted into the core following the reactor trip signal. The reactivity transient then proceeds. The total core reactivity, initially at 0.00$ decreased initially due to reactor scram worth, then steadily increased due to moderator and Doppler feedback associated with the primary system

cooldown. The reactor was approaching a quasi steady-state with the Doppler and the moderator reactivities balancing the scram reactivity, when boron began entering the core, causing the power to decrease.

HPSI flow to the RCS began 41 seconds after break initiation and 25 seconds after the HPSI actuation signal. Twenty-five seconds was the assumed time for the HPSI pumps to reach rated speed.

Figure 14.1.5.2–9 shows the transient reactor power. The reactor power initially declined due to insertion of the control rods. The severe cooldown resulted in power increasing after 80 seconds. A maximum power level of 281 MW or ~10.5% of rated power occurred at 333 seconds.

14.1.5.2.6.1.4 PRISM and XCOBRA-IIIC Results

The PRISM calculation is made initially on the basis of S-RELAP5 input. Each assembly within the three channels is assumed to have a uniform flow corresponding to the sector flows calculated with S-RELAP5. Due to high power peaking in the region of the stuck control rod, large moderator density reductions are calculated to occur in the top portions of several assemblies in this region of the core in the PRISM calculation and are responsible for the significant reduction in reactivity observed when PRISM is compared to S-RELAP5. An XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly.

A comparison of the overall change in reactivity from the event initiation to the time of maximum LHGR between S-RELAP5 and PRISM shows the S-RELAP5 power calculation is conservative.

14.1.5.2.6.1.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

For the MDNBR portion of the calculation, the radial power distribution was modified to conservatively account for local rod power distribution effects within the hot assembly. This was done by raising the power of the hot assembly to bound the peak rod power.

On the bases of these conservative assumptions, the MDNBR value was calculated to be greater than the Biasi 95/95 DNBR correlation limit. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The analysis of the peak LHGR also comes from the PRISM and XCOBRA-IIIC analysis. The peak LHGR is calculated from the S-RELAP5 total core power and the PRISM radial and axial peaking. The peak LHGR was calculated to be less than the FCMLHR limit for the HZP outside containment break with off site power available event. As such, no fuel pins are predicted to fail due to the violation of the FCMLHR limit.

14.1.5.2.6.2 Hot Zero Power Outside Containment with Loss of Off Site Power

The S-RELAP5 NSSS simulation of the most limiting SLB scenario from an MDNBR standpoint (i.e., HZP outside containment break with a loss of off site power) is illustrated in Figures 14.1.5.2–10 through 14.1.5.2–16. A tabulation of the sequence of events is presented in

Table 14.1.5.2-8. Termination of the AFW by manual operator action was assumed to occur 600 seconds after initiation of the break. The S-RELAP5 computation was terminated 600 seconds after break initiation. This is well beyond the time of MDNBR and maximum LHGR.

14.1.5.2.6.2.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Steam flow for the affected steam generator is plotted in Figure 14.1.5.2–10. Secondary pressure for the steam generators is plotted in Figure 14.1.5.2–11. The affected steam generator blows down through the break throughout the transient. The pressure and mass flow rate dropped rapidly at first and then proceeded downward at a slower decay rate until natural circulation flow was established by approximately 250 seconds.

The intact steam generators blow down for a short period until the MSIVs completely close approximately 14 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system. Subsequently, the intact steam generator pressure steadily decreases as cold AFW fills the steam generator.

14.1.5.2.6.2.2 Primary System Thermal Hydraulic Parameters

The primary system core coolant temperature and pressurizer pressure and level responses resulting from the break flow are illustrated in Figures 14.1.5.2–12 through 14.1.5.2–14. The primary system pressure decays rapidly as the coolant contracts due to the cooldown and the pressurizer empties. Continued pressure reduction in the primary system causes the relatively hot stagnant liquid in the head of the RPV vessel to flash. The flashing in the upper head, coupled with near equilibration of other NSSS parameters, retards the pressure decay from that point forward.

A comparison of intact and affected core sector inlet temperatures throughout the transient indicates significant differences due to the limited cross flow allowed between loops. The core sector flows all show the same trend due to the coastdown of the primary coolant pumps. That is, all flows decrease rapidly until natural circulation conditions are achieved in the two flow loops.

14.1.5.2.6.2.3 Reactivity and Core Power

The reactivity transient calculated by S-RELAP5 is illustrated in Figure 14.1.5.2–15. Initially, the core is assumed to be at zero power. The total core reactivity, initially at 0.00$ decreased initially due to reactor scram worth, then steadily increased due to moderator and Doppler feedback associated with the primary system cooldown. The rise in reactor power was arrested when boron began entering the core at 332 seconds. Power then declined slowly due to an increasing boron concentration in the primary system.

The HPSI actuation signal was received at 21 seconds. After a 25 second delay, during which the HPSI pumps reached rated speed, HPSI flow to the RCS began, at 46 seconds.

The transient experienced by the core power is illustrated in Figure 14.1.5.2–16. The core power, initially at 1 Watt, increased rapidly at 120 seconds and reached a peak power level of 5.9% of rated power (159 MW) at 352 seconds.

14.1.5.2.6.2.4 PRISM and XCOBRA-IIIC Results

The PRISM calculation is initially made on the basis of S-RELAP5 predicted core power, flow, pressure, and inlet temperatures. The PRISM calculations provide the radial and axial power distributions for use in the XCOBRA-IIIC code. Due to the high power peaking in the region of the stuck control rod, and the low core average natural circulation flow rates, large moderator density decreases are calculated in several assemblies in this region in the PRISM calculation and are responsible for the significant reduction in reactivity observed when PRISM is compared to S-RELAP5. An XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly.

A comparison of the overall change in reactivity from the event initiation to the time of minimum DNBR between S-RELAP5 and PRISM shows the S-RELAP5 power calculation is conservative.

14.1.5.2.6.2.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR of the hot fuel assembly is calculated to be greater than the Biasi 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

As before, the analysis of the peak LHGR comes from the PRISM and the XCOBRA-IIIC analysis. The peak LHGR was less than the FCMLHR limit. Therefore, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.2.7 Conclusions

The HFP and HZP scenarios, with off site power maintained for operation of the primary coolant pumps, resulted in a return to higher power levels than the scenarios where off site power is lost. However, these scenarios provide substantially greater margin to the MDNBR limit because of the higher coolant flow rate. In no scenario evaluated, however, was fuel failure calculated to occur as a result of penetration of the MDNBR safety limit. Even though the scenarios with off site power available have substantially greater margin to the MDNBR limit because of a higher coolant flow rate, the higher power levels in combination with the highly skewed power distribution due to the assumed stuck rod cluster resulted in them having the least margin to the fuel centerline melt limit.

The HZP outside containment break scenario concurrent with a loss of off site power was determined to be the most limiting in this analysis from an MDNBR standpoint. The MDNBR of the hot fuel assembly is calculated to be above the Biasi 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

The HZP outside containment break scenario with off site power available was determined to be the most limiting in this analysis from the standpoint of centerline melt. This scenario results in the highest return to power and highest calculated LHGR. No fuel pins are calculated to fail due to the violation of the FCMLHR limit.

14.1.5.3 Radiological Consequences of a Main Steam Line Break

The main steam line break is postulated to occur in a main steam line outside the containment. The radiological consequences of a main steam line break inside containment is bounded by the main steam line break outside containment. The plant is assumed to be operating with Technical Specification coolant concentrations. A 150 gpd primary to secondary leak is assumed to occur in each steam generator.

The main steam line break (MSLB) accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine coincident with a loss of off site power. As a result, the condenser is unavailable and cool down of the primary system is through the release of steam to the environment from the intact generator. In order to maximize doses, break scenarios are assessed inside and outside containment. The analysis for the MSLB in Containment assumes 3.7% fuel failure. The analysis for the MSLB outside containment assumes no fuel failure but addresses the pre-accident (30μCi/gm Dose Equivalent (DE) I-131) and concurrent iodine (500 times the iodine appearance rates at 0.5μCi/gm DE I-131) spike scenarios as well as Technical Specification levels of activity consistent with DE I-131 and Xe-133 limits.

An operator action is credited in the event of a MSLB outside containment. There is insufficient radioactivity in the release to alarm control room radiation monitors with subsequent isolation of the control room. This action requires an operator to isolate the control room within 4 hours of MSLB and align Control Room Emergency Ventilation for filtered recirculation within an additional 1 hour.

The noble gases and iodines in the primary coolant that leak into the faulted steam generator during the transient are released directly to the environment without holdup or decontamination except for MSLB in containment. For the MSLB in containment, releases from the faulted steam generator are released to the environment at Technical Specification containment leak rates. An iodine partition factor of 1 is used for the releases from the unaffected steam generator for the first 20 seconds. From 20 seconds until the conclusion of the release, an iodine partition factor of 0.01 is used. Off-site power is assumed to be lost, thus making the condenser unavailable. The steam releases from the main steam line break are from the turbine building blowout panels as the atmospheric dispersion factor is greater for this release point than the enclosure building blowout panels. The steam releases from the intact steam generator are from the MSSVs/ADVs.

The radiological consequences of a main steam line break to the EAB, LPZ and Millstone 2 Control Room are reported in Table 14.1.5.3-2. The assumptions used to perform this evaluation are summarized in Table 14.1.5.3-1.

The resulting doses to the EAB, LPZ and Control Room do not exceed the guidelines provided in 10 CFR 50.67 and Regulatory Guide 1.183.

14.1.6 REFERENCES

14.1-1 “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” EMF-2310(P)(A), Revision 1, Framatome-ANP, May 2004.

14.1-2 “ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events,” ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, May 1992.

14.1-3 “Reactor Analysis Systems for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,” EMF-96-029(P)(A), Siemens Power Corporation, January 1997.

14.1-4 “XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation,” XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, January 1986.

14.1-5 “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” EMF-92-153(P)(A), Revision 1, Siemens Power Corporation, January 2005.

14.1-6 “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors,” EMF-1961(P)(A), Revision 0 Siemens Power Corporation, July 2000.

14.1-7 “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” EMF-2310(P)(A), Revision 1, Supplement 2P-A, Revision 0, Framatome Inc., March 2023.

TABLE 14.1.1-1 AVAILABLE REACTOR PROTECTION FOR THE DECREASE IN FEEDWATER TEMPERATURE EVENT

Reactor Operational Modes	Reactor Protection
1	Variable Overpower Trip Thermal Margin/Low Pressure Trip Local Power Density Trip Low Steam Generator Pressure Trip
2	Variable Overpower Trip Low Steam Generator Pressure Trip
3	Variable Overpower Trip
4-6	Not a credible event for these reactor operating conditions since there is no extraction steam to the feedwater heaters

TABLE 14.1.1-2 DISPOSITION OF EVENTS FOR THE DECREASE IN FEEDWATER TEMPERATURE EVENT

Reactor Operational Modes	Disposition
1	Bounded by Event 14.1.3, Increase in Steam Flow Event
2, 3	Bounded by the above
4-6	No analysis required; not a credible event

TABLE 14.1.2-1 AVAILABLE REACTOR PROTECTION FOR THE INCREASE IN FEEDWATER FLOW EVENT

Reactor Operational Modes	Reactor Protection
1	Variable Overpower Trip Local Power Density Trip Thermal Margin/Low Pressure Trip Low Steam Generator Pressure Trip Safety Injection Actuation Signal Reactor Trip on Turbine Trip due to High Steam Generator Water Level *
2	Low Steam Generator Pressure Trip Variable Overpower Trip Safety Injection Actuation Signal
3	Variable Overpower Trip Safety Injection Actuation Signal
4	Technical Specification Requirements on Shutdown Margin Inherent Negative Doppler Feedback **
5, 6	No analysis required; no significant consequences

* Additional protection is provided by this control-grade reactor trip.

** Provides defense in depth.

TABLE 14.1.2-2 DISPOSITION OF EVENTS FOR THE INCREASE IN FEEDWATER FLOW EVENT

Reactor Operational Modes	Disposition
1	Bounded by Event 14.1.3 (Increase in Steam Flow)
2	Bounded by the Mode 3 Case
3-6	Bounded by Event 14.1.3

TABLE 14.1.3-1 AVAILABLE REACTOR PROTECTION FOR THE INCREASE IN STEAM FLOW EVENT

Reactor Operational Modes	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Thermal Margin/Low Pressure Trip Local Power Density Trip Variable Overpower Trip Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Variable Overpower Trip Safety Injection Actuation Signal
3	Variable Overpower Trip Safety Injection Actuation Signal
4	Technical Specification Requirements on Shutdown Margin Inherent Negative Doppler Feedback
5, 6	No Analysis Required; No Significant Consequences

* Provides defense in depth.

TABLE 14.1.3-2 DISPOSITION OF EVENTS FOR THE INCREASE IN STEAM FLOW EVENT

Reactor Operational Modes	Disposition
1	Analyze
2, 3, 4	Bounded by the above
5, 6	No analysis required; no significant consequences

TABLE 14.1.3-3 INITIAL CONDITIONS FOR THE INCREASE IN STEAM FLOW EVENT

Parameter	Initial Value
Reactor power	2754 MWt (102% of rated)
Cold leg temperature	549°F
Total RCS flow rate	360,000 gpm
Pressurizer pressure	2250 psia
Pressurizer level	65% of span
Steam generator pressure	875 psia
Steam generator fluid mass (feedwater and steam)	167,237 lbm per steam generator

TABLE 14.1.3-4 EVENT SUMMARY FOR THE INCREASE IN STEAM FLOW EVENT

Event	Time (seconds)
Steam dump to condenser valves and turbine bypass valves open fully	0.0
Turbine control valves open fully	0.0
Indicated thermal power reaches Variable Overpower trip ceiling (111.7%)	22.1
Turbine trips on reactor scram signal	23.0
Reactor power reaches maximum value	23.5
Scram CEA insertion begins	23.5
MDNBR occurs	23.7

TABLE 14.1.3-5 PEAK REACTOR POWER LEVELS FOR INCREASE IN STEAM FLOW EVENT

Case	Amount of Steam Release	Peak Reactor Power (% of Rated)
Turbine control valves fully open	~11% excess	117.8%
Intermediate case	~20% excess	117.8%
Steam dump to condenser valves, turbine bypass valves, and turbine control valves fully open	~41% excess	117.7%

FIGURE 14.1.3–1 NORMALIZED POWER AND HEAT FLUX FOR THE INCREASE IN STEAM FLOW EVENT

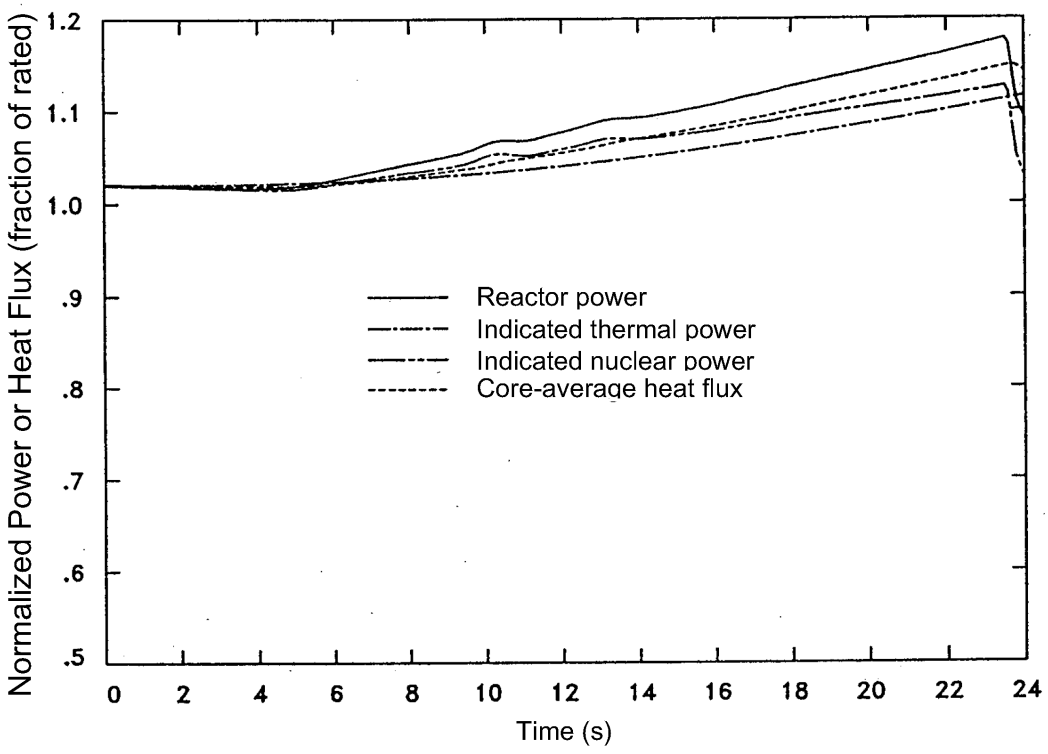


FIGURE 14.1.3–2 REACTIVITY FEEDBACK FOR THE INCREASE IN STEAM FLOW EVENT

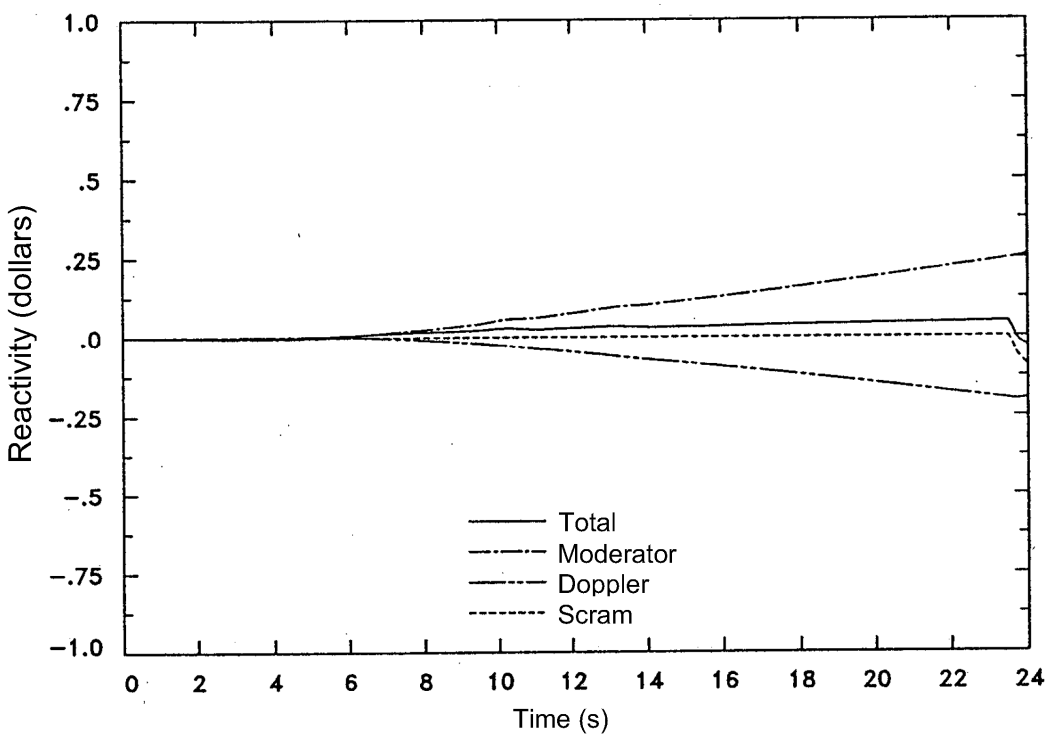


FIGURE 14.1.3–3 REACTOR COOLANT TEMPERATURES FOR INCREASE IN STEAM FLOW EVENT

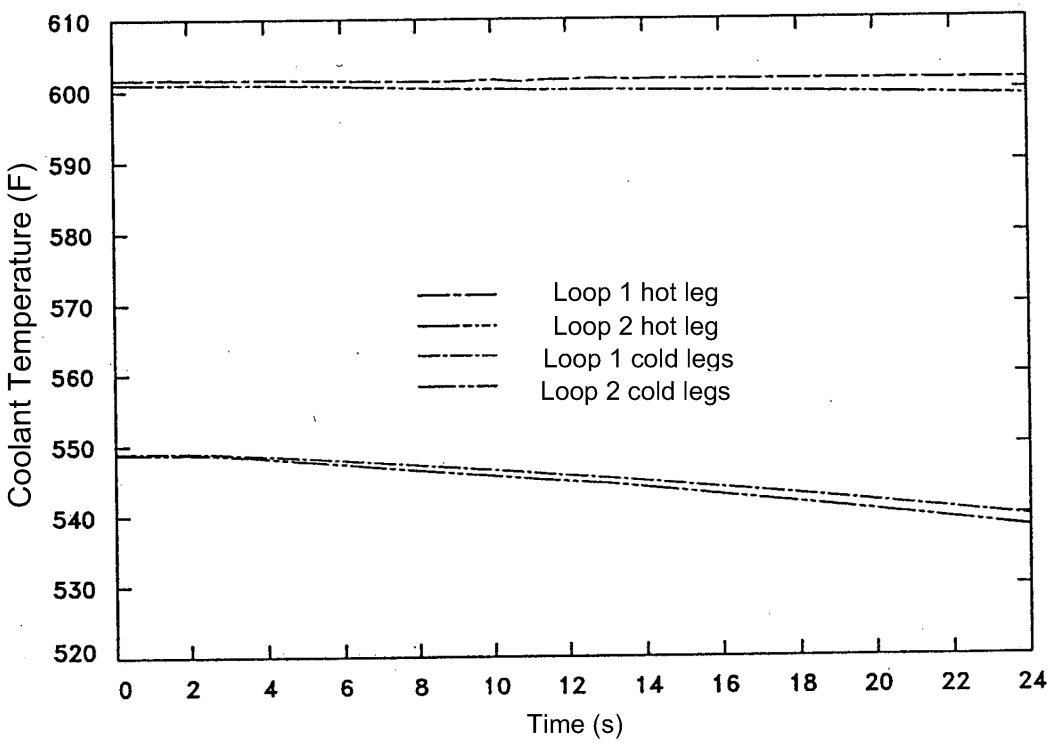


FIGURE 14.1.3–4 CORE INLET MASS FLOW RATE FOR THE INCREASE IN STEAM FLOW EVENT

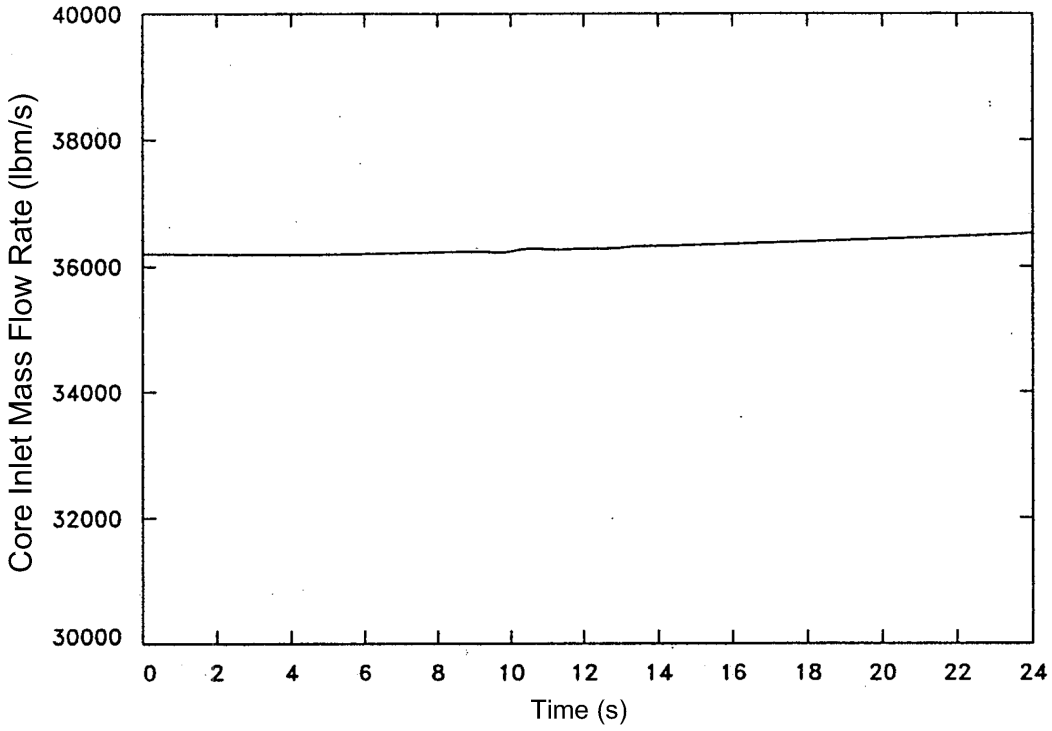


FIGURE 14.1.3–5 PRESSURIZER PRESSURE FOR THE INCREASE IN STEAM FLOW EVENT

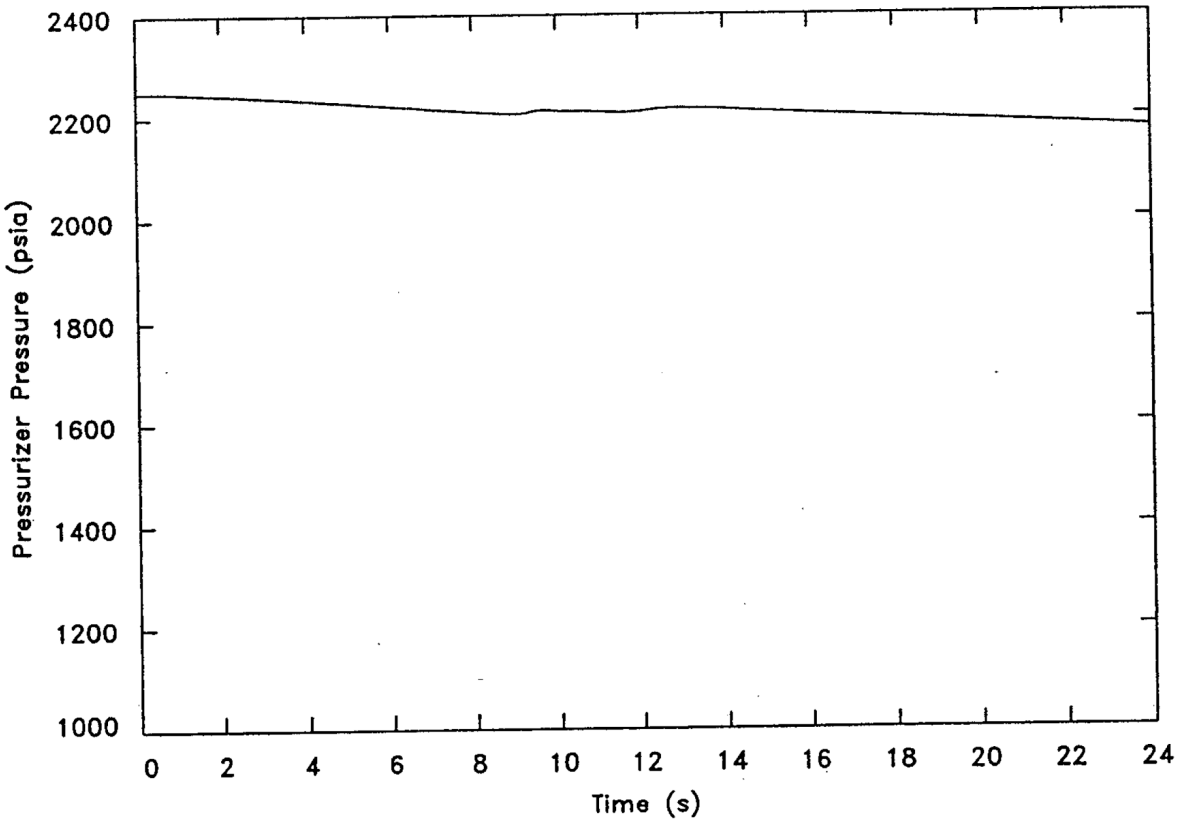


FIGURE 14.1.3–6 STEAM GENERATOR PRESSURES FOR THE INCREASE IN STEAM FLOW EVENT

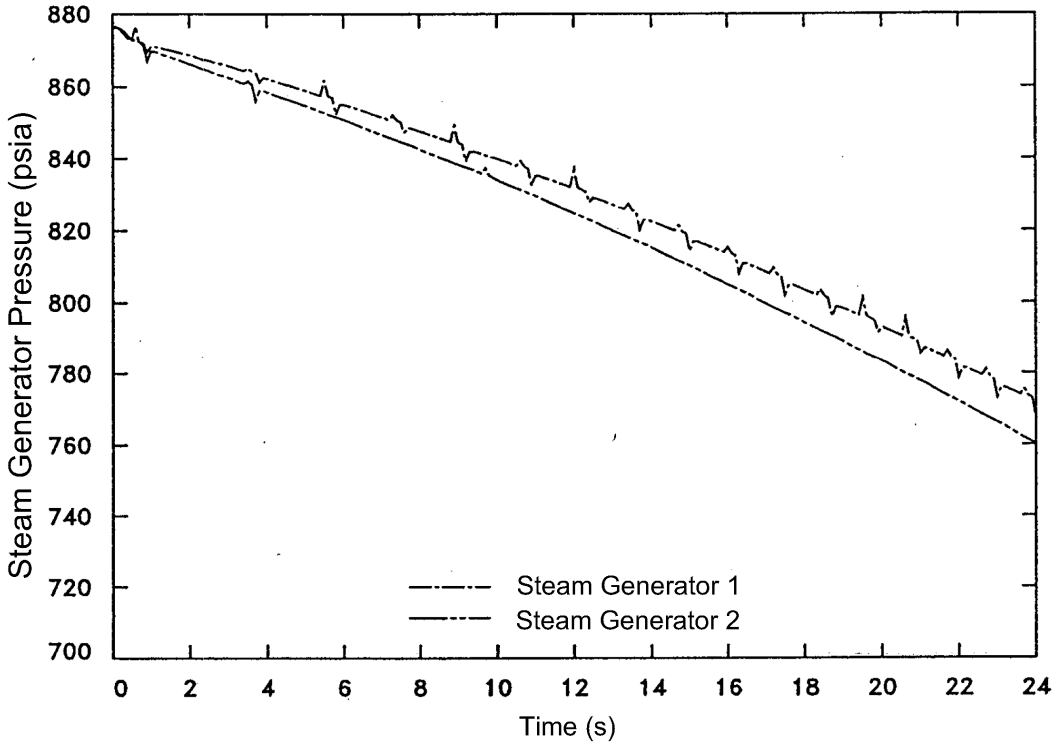


FIGURE 14.1.3–7 STEAM MASS FLOW RATES FOR THE INCREASE IN STEAM FLOW EVENT

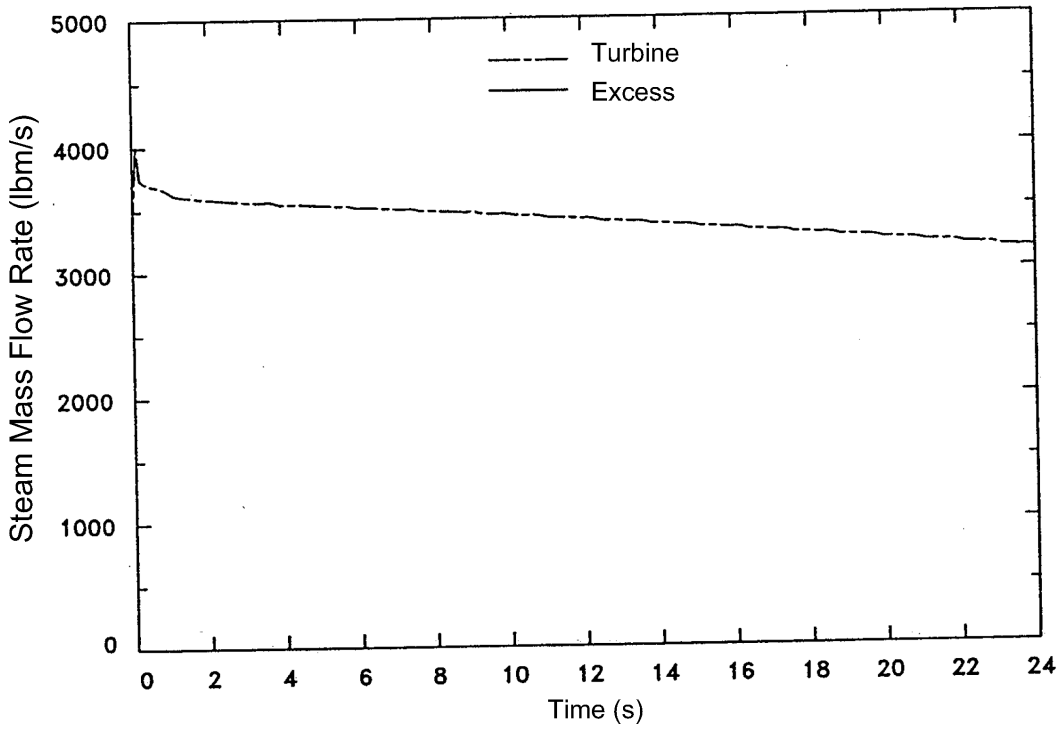


FIGURE 14.1.3–8 MAIN FEEDWATER FLOW FOR THE INCREASE IN STEAM FLOW EVENT

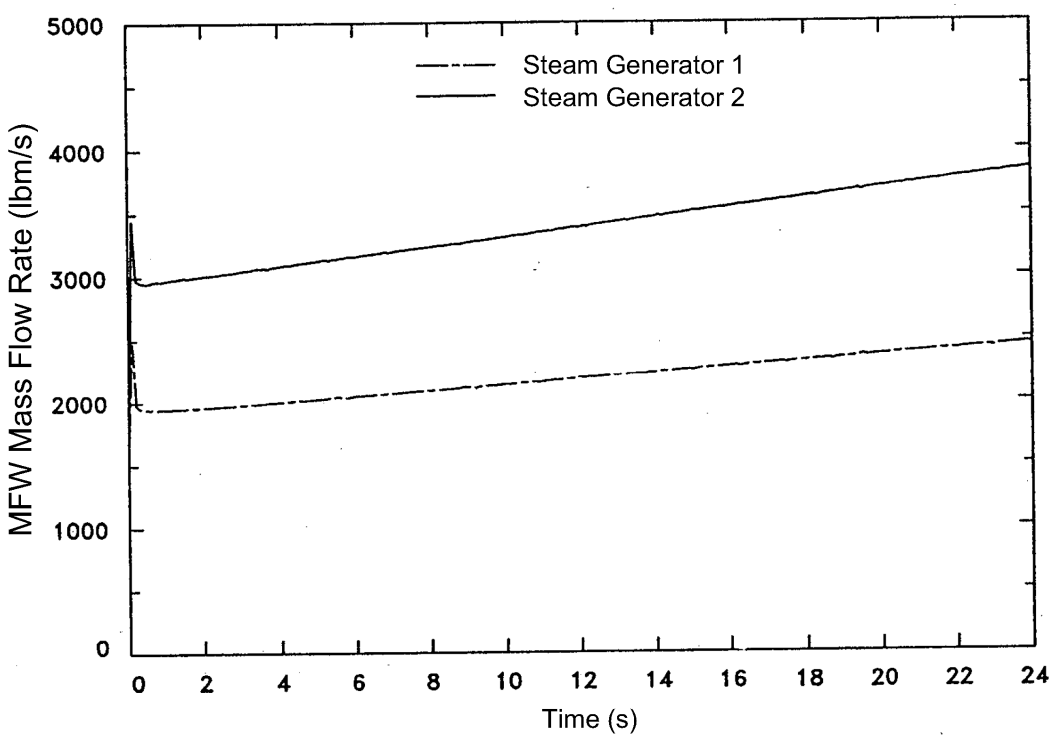


FIGURE 14.1.3–9 MAIN FEEDWATER TEMPERATURE FOR THE INCREASE IN STEAM FLOW EVENT

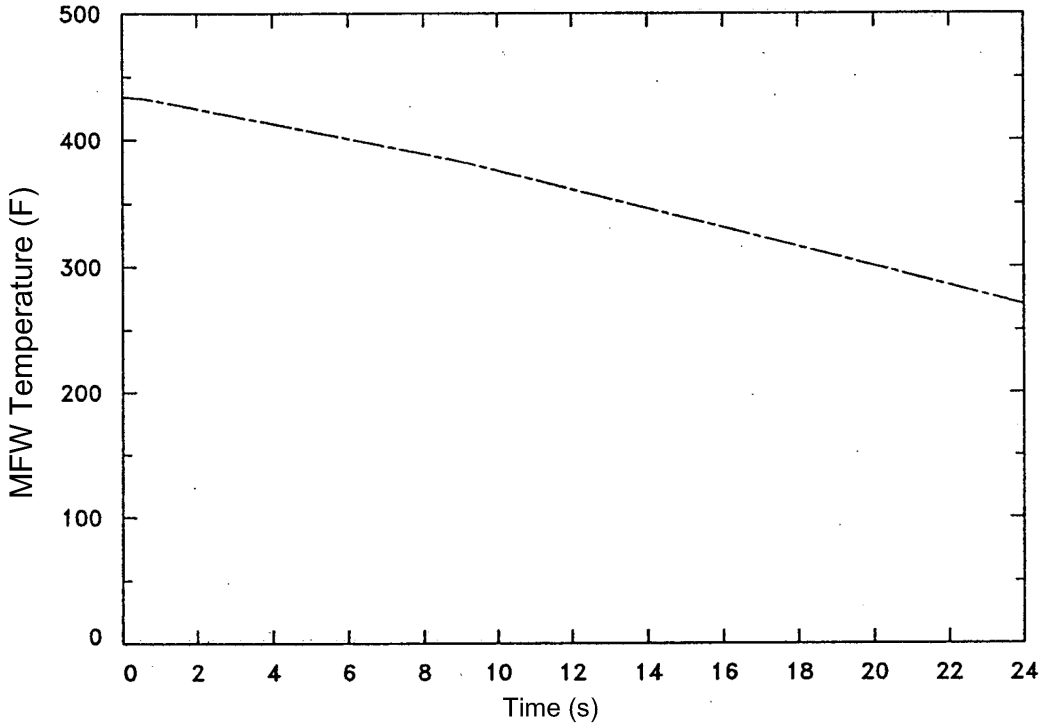


TABLE 14.1.4-1 AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVES

Reactor Operational Modes	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Variable Overpower Trip Local Power Density Trip Thermal Margin/Low Pressure Trip Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Variable Overpower Trip Safety Injection Actuation Signal
3, 4	Technical Specification Requirements on Shutdown Margin, Inherent Negative Doppler Feedback *
5, 6	No Analysis Required; Not a Credible Event

* Provides defense in depth.

TABLE 14.1.4-2 DISPOSITION OF EVENTS FOR THE INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE EVENT

Reactor Operational Modes	Disposition
1-4	Bounded by analyses presented for Event 14.1.3
5, 6	Not a credible event; no analysis required

TABLE 14.1.5.1-1 AVAILABLE REACTOR PROTECTION FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT

PRE-SCRAM ANALYSIS

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Low Reactor Coolant Flow Variable Overpower Trip Local Power Density Trip Thermal Margin/Low Pressure Trip High Containment Pressure Trip Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Low Reactor Coolant Flow Variable Overpower Trip High Containment Pressure Trip Safety Injection Actuation Signal
3-6	Technical Specification Requirements on Shutdown Margin, Inherent Negative Doppler Feedback

TABLE 14.1.5.1-2 DISPOSITION OF EVENTS FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT

PRE-SCRAM ANALYSIS

Reactor Operating Conditions	Disposition
1	Analyze
2	Analyze
3-6	Bounded by the above

TABLE 14.1.5.1-3 S-RELAP5 THERMAL-HYDRAULIC INPUT (PRE-SCRAM STEAM LINE BREAK)

Initial Condition Thermal-Hydraulic Input	HFP
Reactor Power (MW)	2754
Pressurizer Pressure (psia)	2250
Pressurizer Level (%)	65
Cold Leg Coolant Temperature (°F)	549
Total Primary Flow Rate (lbm/sec)	37,640
Secondary Pressure (psia)	876
Core Bypass Flow Rate (lbm/sec) per Loop	724
Main Feedwater Temperature (°F)	432
Steam Generator Mass Inventory (lbm)	167,237

TABLE 14.1.5.1-4 ACTUATION SIGNALS AND DELAYS (PRE-SCRAM STEAM LINE BREAK)

Reactor Trip	Non-Harsh Containment Condition Setpoint	Harsh Containment Condition Setpoint	Delay
Variable Overpower (ceiling)	111.6% of rated (NI power); 114% of rated (thermal power)	Not credited	0.9 s
Low Reactor Coolant Flow	Credited	85% flow	0.65 s
High Containment Pressure	Not applicable	5.83 psig	0.9 s
Low Steam Generator Pressure	658 psia	550 psia	0.9 s
TM/LP (floor)	1728 psia	1700 psia	0.9 s
TM/LP (function)	Evaluated from function given in Technical Specification	Not credited	0.9 s

TABLE 14.1.5.1-5 S-RELAP5 NEUTRONICS INPUT AND ASSUMPTIONS (PRE-SCRAM STEAM LINE BREAK)

Point Kinetics Input	Value
Effective Delayed Neutron Fraction	0.005245
Moderator Temperature Coefficient (pcm/°F)	
Off Site Power Available (Technical Specification most negative limit)	-8 to -32/-30*
Loss of Off Site Power (Technical Specification most positive limit above 70% RTP)	+4
HFP Scram Worth (pcm)	6425
Shutdown Margin Requirement (pcm)	3600
Doppler Coefficient	
Off Site Power Available	Nominal EOC
Loss of Off Site Power	0.80 x least negative value at BOC

Fission Product and Actinide Decay Constants

Default values in ANF-RELAP utilized

*An MTC of -30 pcm/°F is used for the inside containment offsite power available cases.

TABLE 14.1.5.1-6 MDNBR AND PEAK REACTOR POWER LEVEL SUMMARY (PRE-SCRAM STEAM LINE BREAK)

Location of Break	Type of Cooldown	Size of Break	MDNBR	Peak Reactor Power (% of rated)
Outside containment, downstream of check valves	Symmetric	3.25 ft^2	**	129.83%
		3.50 ft^2	**	130.84%
		3.75 ft^2	**	128.19%
Outside containment, upstream of check valve	Asymmetric	1.80 ft^2	**	124.62%
		2.00 ft^2	**	126.05%
		2.20 ft^2	**	126.94%
		2.40 ft^2	**	122.46%
Inside containment, upstream of check valve	Asymmetric	0.05 ft^2	**	134.38%***
		0.10 ft^2	**	135.22%
		0.20 ft^2	**	137.68% *
		0.40 ft^2	**	125.03%
Inside containment, upstream of check valve with loss of off site power	Asymmetric	3.51 ft^2	0.88	106.86%

* The peak LHRs for all pre-scram breaks are bounded by the peak LHR for the 0.20 ft^2 break inside containment and upstream of a check valve.

** The MDNBRs for these pre-scram breaks are bounded by the MDNBR for the 3.51 ft^2 break inside containment, upstream of a check valve with the loss of off site power.

** Inside containment, upstream of check valve offsite power available cases analyzed with -30 pcm/°F except for 0.05 ft^2 case analyzed with MTC of -32 pcm/°F.

TABLE 14.1.5.1-7 LHGR-LIMITING PRE-SCRAM STEAM LINE BREAK SEQUENCE OF EVENTS: HFP 0.20FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT WITH OFFSITE POWER AVAILABLE

Time (sec.)	Event
0.0	Break upstream of main steam line check valves opens
0.0	Turbine control valves open fully
115.8	High containment pressure trip setpoint reached
117.0	Scram signal
117.0	Turbine trips on reactor scram signal
117.3	Reactor power reaches maximum value
117.3	MDNBR occurs

TABLE 14.1.5.1-8 MDNBR-LIMITING PRE-SCRAM STEAM LINE BREAK SEQUENCE OF EVENTS: HFP 3.51FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER

Time (seconds)	Event
0	Break occurs
0	RCPs trip
0	Peak LHGR (kW/ft)
2	Scram signal on low flow trip
3	Scram CEA Insertion begins
3	Maximum Power (Fraction of RTP)
4	MDNBR

FIGURE 14.1.5.1–1 NORMALIZED CORE POWER (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

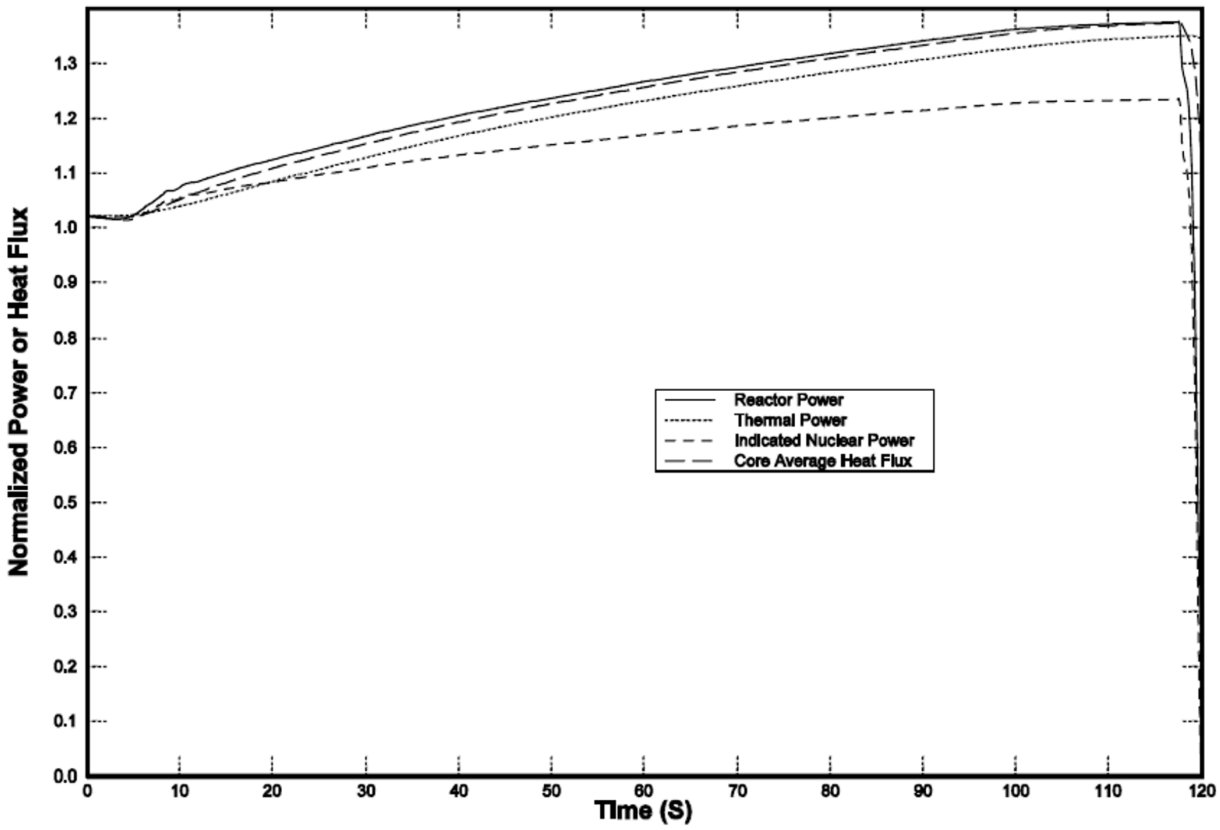


FIGURE 14.1.5.1–2 CORE INLET TEMPERATURES (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

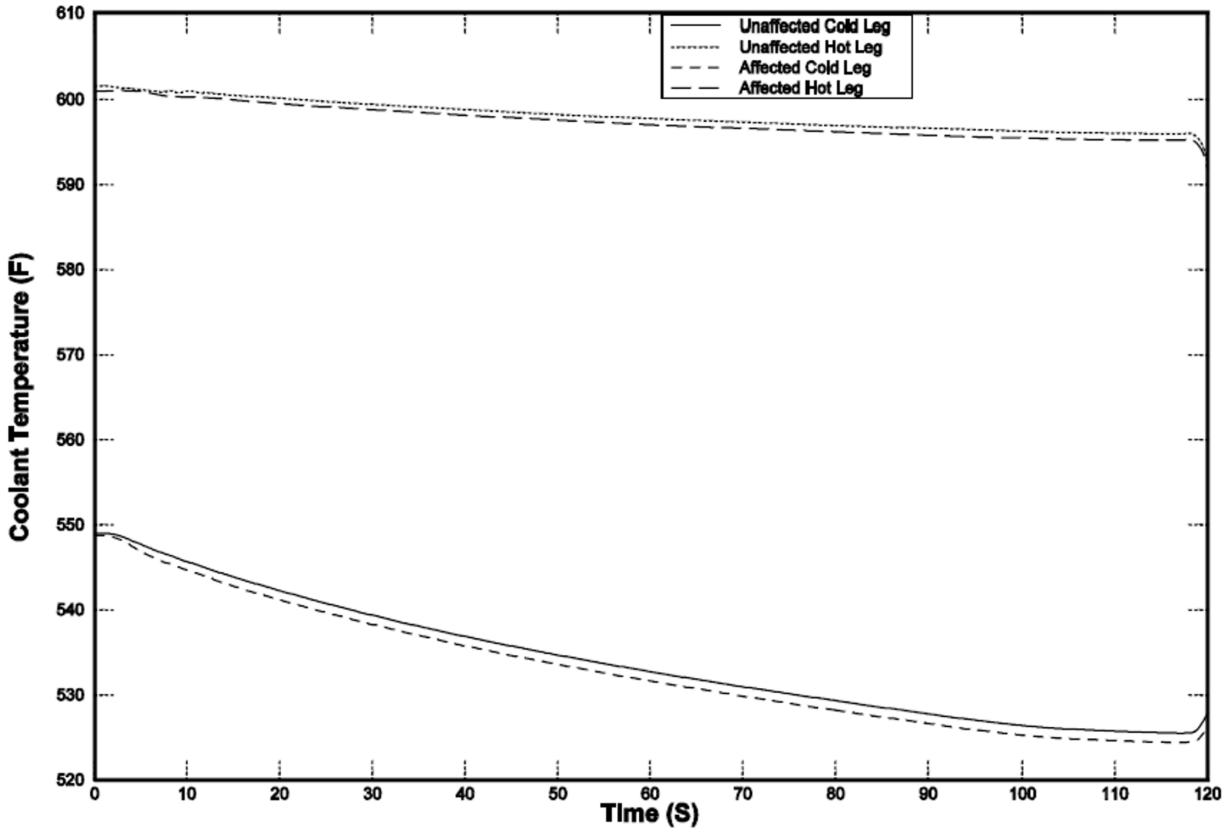


FIGURE 14.1.5.1–3 REACTIVITY FEEDBACK (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

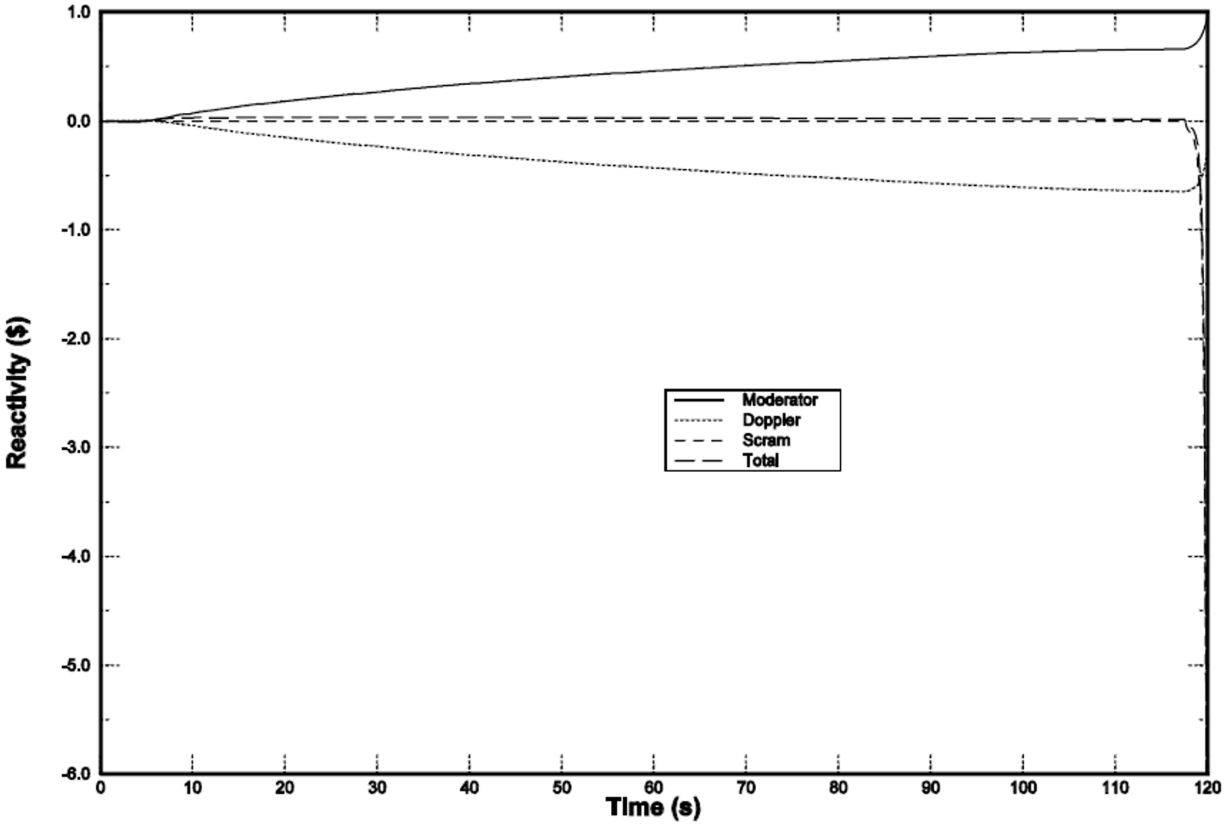


FIGURE 14.1.5.1–4 PRESSURIZER PRESSURE (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

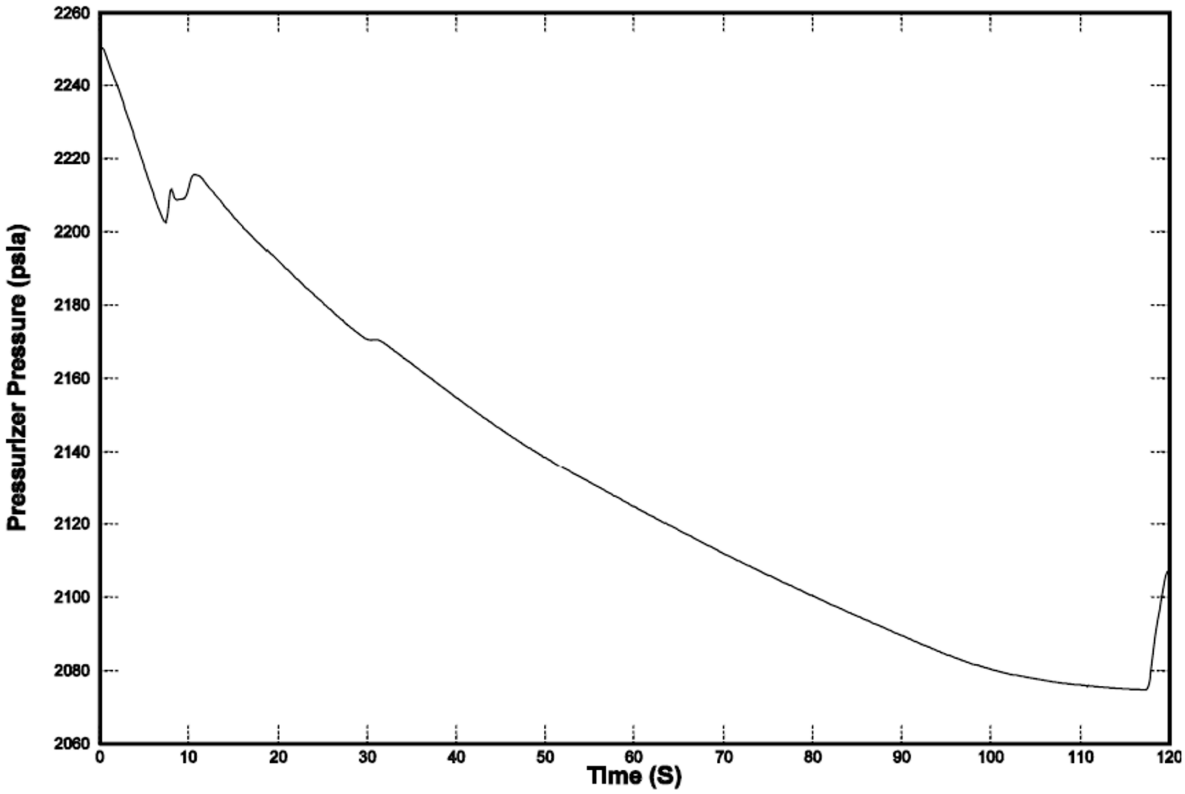


FIGURE 14.1.5.1–5 STEAM GENERATOR PRESSURES (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

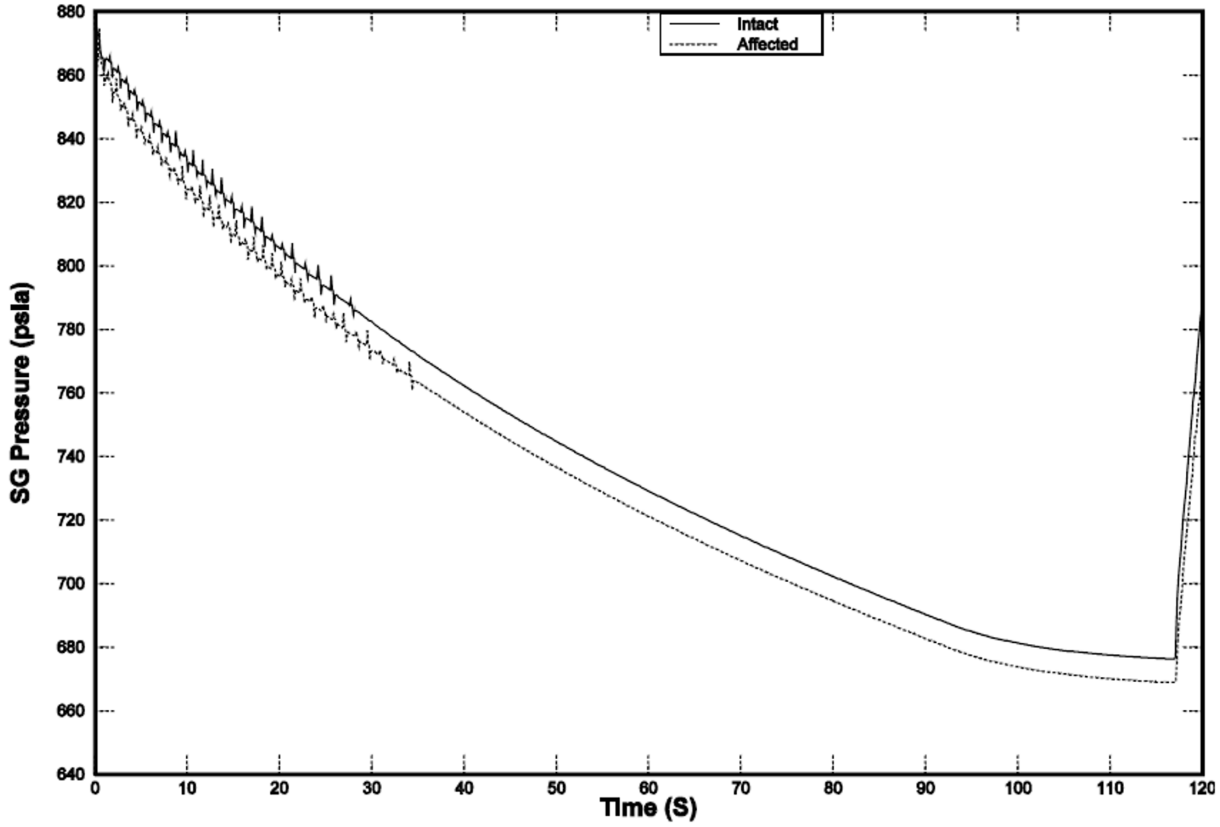


FIGURE 14.1.5.1–6 STEAM MASS FLOW RATES (0.20 FT^2 ASYMMETRIC BREAK INSIDE CONTAINMENT)

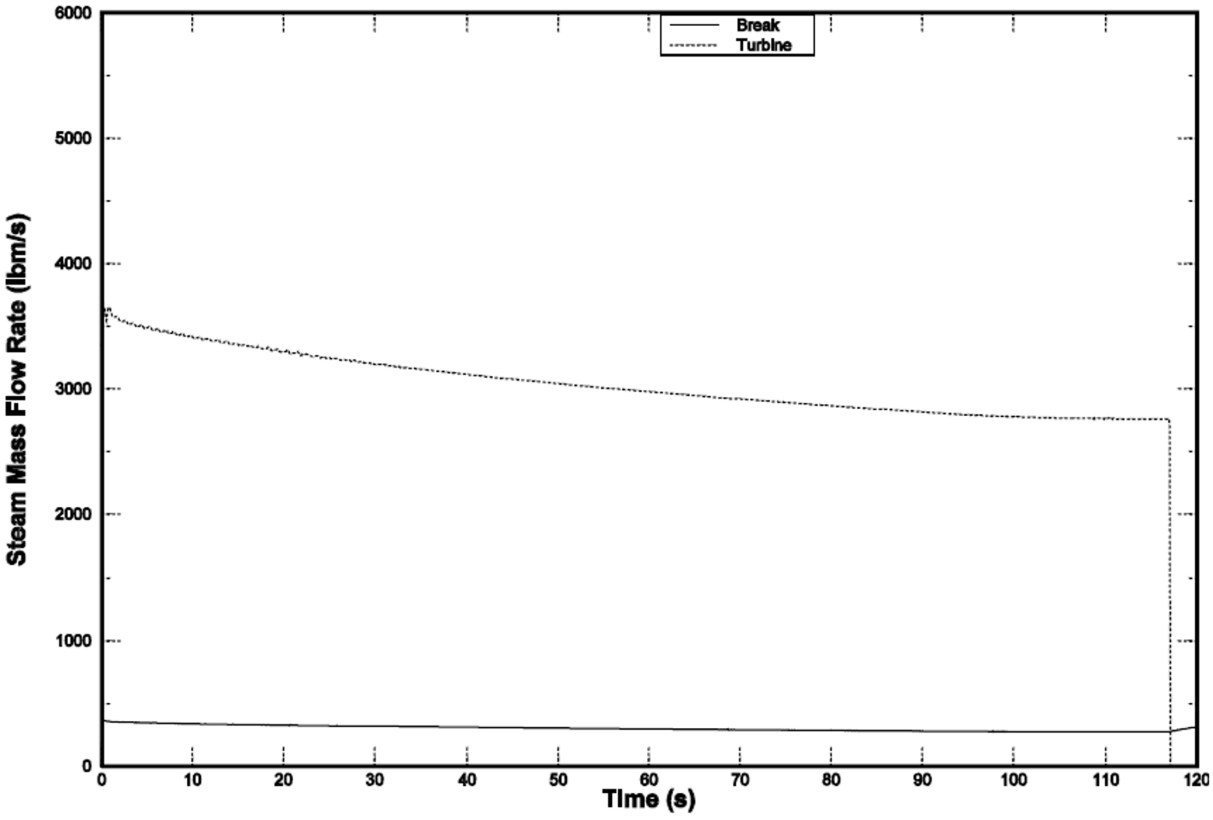


FIGURE 14.1.5.1–7 NORMALIZED POWER AND HEAT FLUX (ASYMMETRIC 3.51 FT^2 BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER)

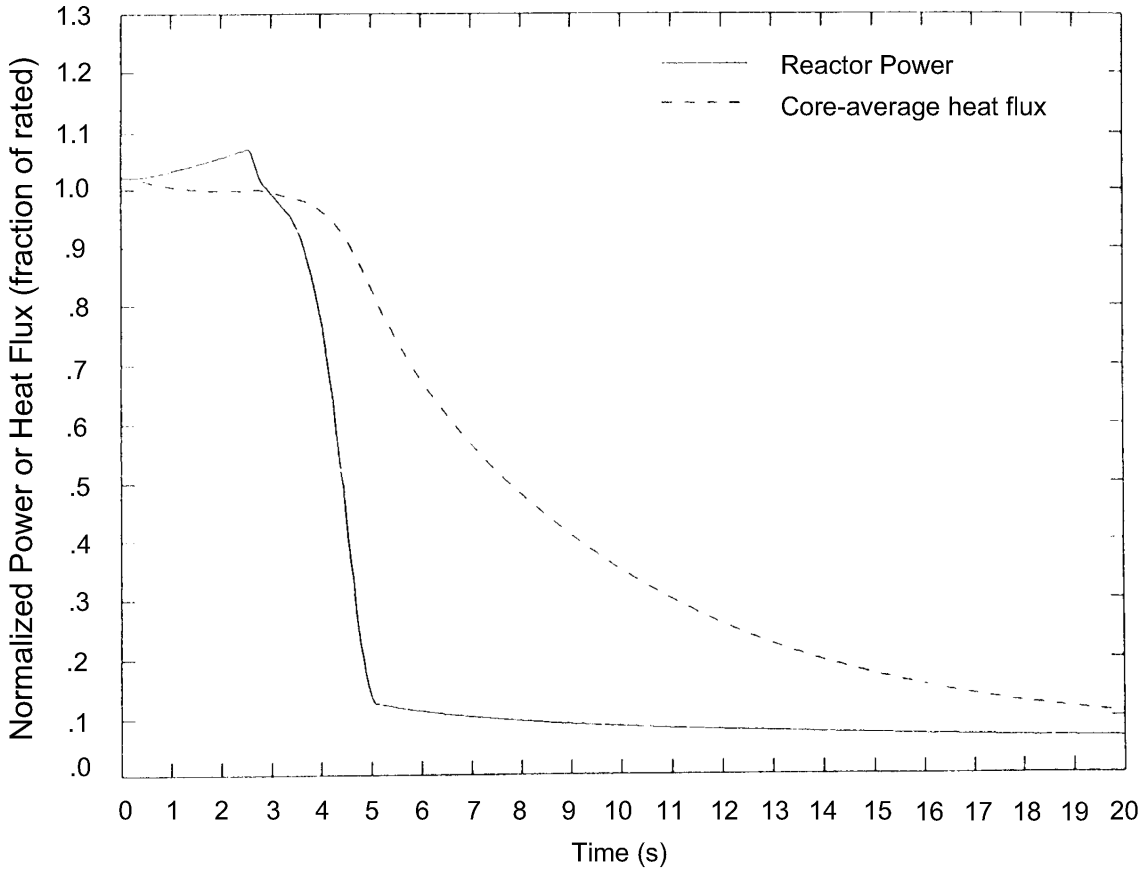


FIGURE 14.1.5.1–8 REACTOR COOLANT TEMPERATURES (ASYMMETRIC 3.51 FT^2 BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER)

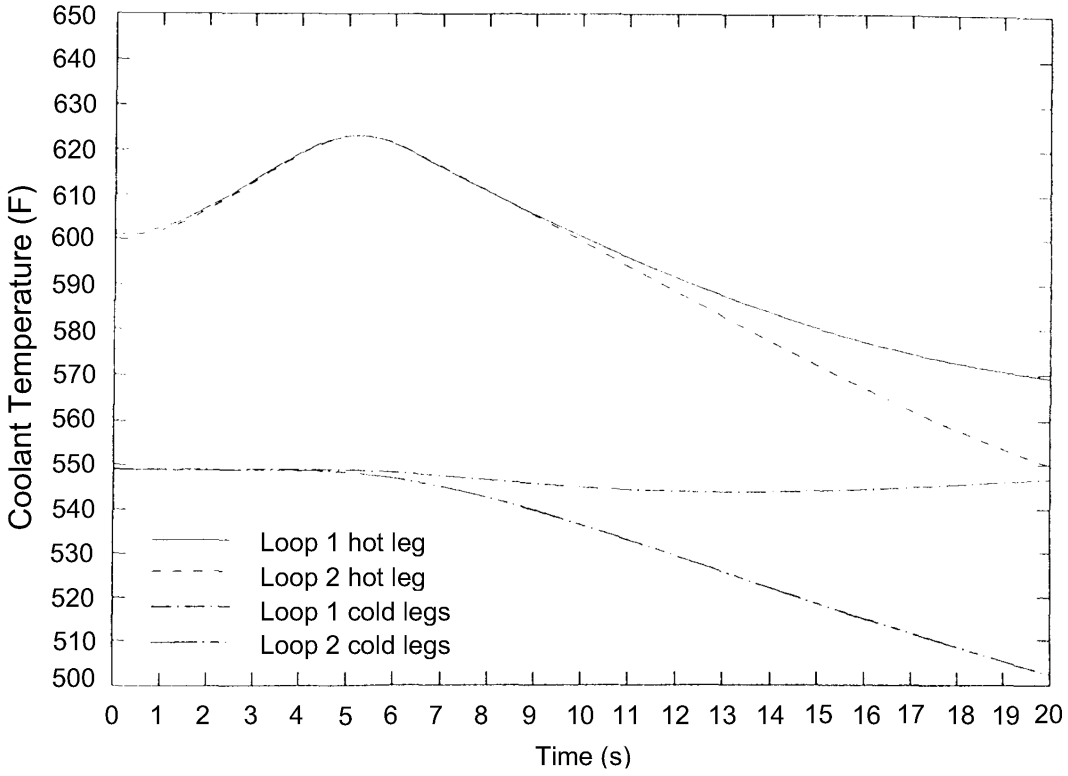


FIGURE 14.1.5.1–9 NORMALIZED REACTOR COOLANT SYSTEM FLOW RATE (ASYMMETRIC 3.51 FT^2 BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER)

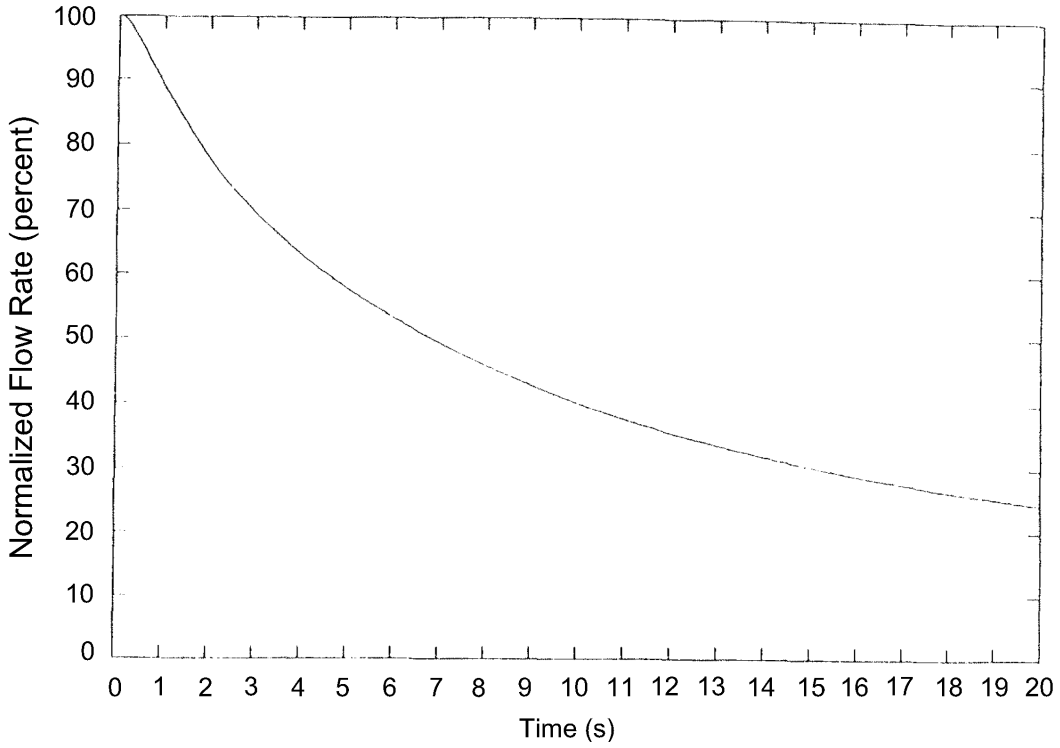


FIGURE 14.1.5.1–10 PRESSURIZER PRESSURE (ASYMMETRIC 3.51 FT^2 BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER)

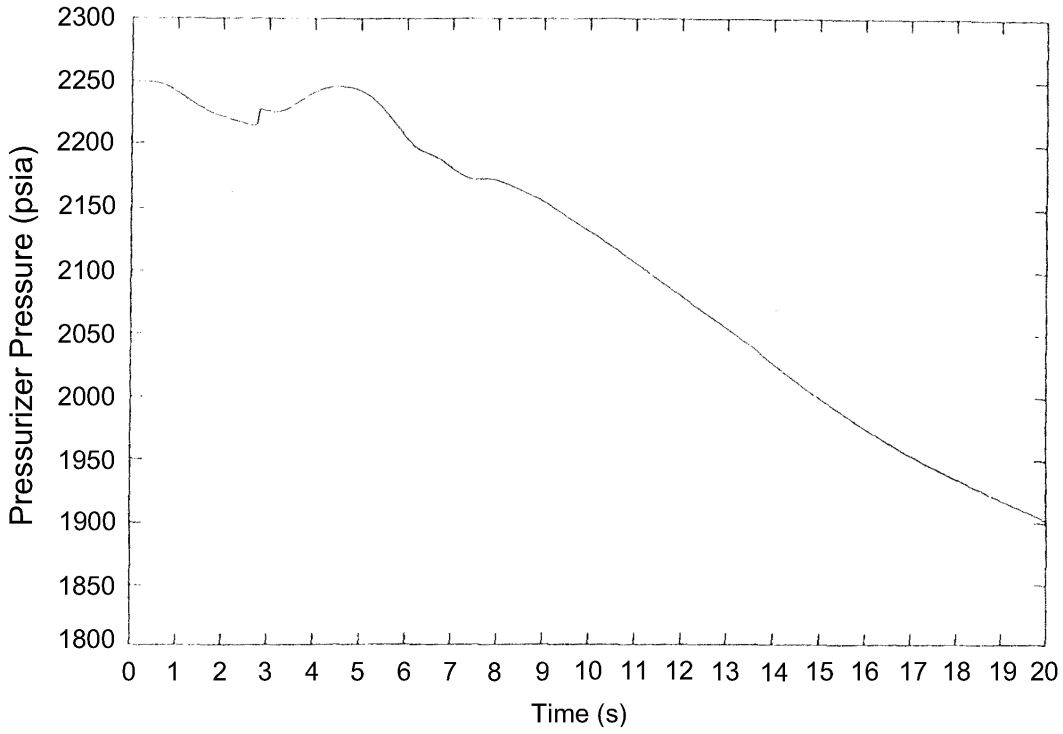


FIGURE 14.1.5.1–11 STEAM GENERATOR PRESSURES (ASYMMETRIC 3.51 FT^2 BREAK INSIDE CONTAINMENT WITH LOSS OF OFFSITE POWER)

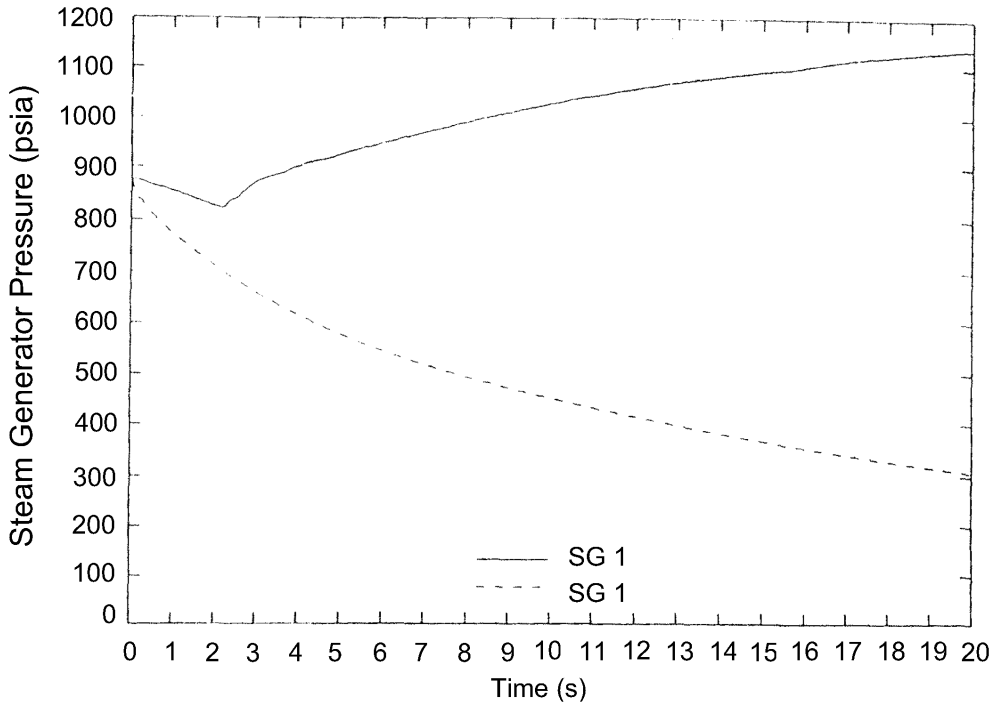


TABLE 14.1.5.2-1 AVAILABLE REACTOR PROTECTION FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT, POST-SCRAM ANALYSIS

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Local Power Density Trip Thermal Margin/Low Pressure Trip High Containment Pressure Trip Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip High Containment Pressure Trip Safety Injection Actuation Signal
3-6	Technical Specification Requirements on Shutdown Margin, Inherent Negative Doppler Feedback

TABLE 14.1.5.2-2 DISPOSITION OF EVENTS FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT, POST-SCRAM ANALYSIS

Reactor Operating Conditions	Disposition
1	Analyze
2	Analyze
3-6	Bounded by the above

TABLE 14.1.5.2-3 SRELAP5 -THERMAL-HYDRAULIC INPUT (POST-SCRAM STEAM LINE BREAK)

Initial Condition Thermal-Hydraulic Input	HFP	HZP
Core Power (MW)	2700	1E-6
Primary Pressure (psia)	2250	2250
Pressurizer Level (%)	65	40
Cold Leg Temperature (°F)	549	532
Primary Flow Rate per Loop (lbm/sec)	18,820	19,241
Secondary Pressure (psia)	881	892
Steam Generator Mass Inventory (lbm)	167,237	253,989
Total Steam Flow (lbm/sec) per Steam Generator	1634	4

Break Characteristics

Minimum Flow Area

Affected Steam Generator (ft^2) 3.51

Unaffected Steam Generator (ft^2) 3.51

Location of Pipe Break Downstream of steam generator integral flow restrictor and upstream of MSIV

Injection Systems	HFP	HZP
Total HPSI Pumps	3	3
Active HPSI Pumps	2	2
Single Failure (No credit for mounted spare)	1 HPSI pump	1 HPSI pump
Active Charging Pumps	0	0
Refueling Water Storage Tank Boron Concentration (ppm)	1720	1720
HPSI Delivery Curve	Figure 14.1.5.2–1	Figure 14.1.5.2–1

TABLE 14.1.5.2-4 ACTUATION SIGNALS AND DELAYS (POST-SCRAM STEAM LINE BREAK)

Parameter Setpoints	Inside Containment	Outside Containment
1. Low Steam Generator Pressure Trip	550 psia	658 psia
2. Low Pressurizer Pressure SIAS	1500 psia	1576 psia
3. Low Steam Generator Pressure MSI	370 psia	478 psia

MSIV Closure

Required Actuation Signal (3) Above

Delay - 6.9 seconds

HPSI Actuation

Required Actuation Signal (2) Above

Delay - 25.0 seconds

Main Feedwater Valve Closure

Required Actuation Signal (3) Above

Delay - 14.0 seconds

Reactor Scram

Required Actuation Signal (1) Above

Delay - 0.9 second instrument delay, 3.0 second insertion time

TABLE 14.1.5.2-5 S-RELAP5 NEUTRONICS INPUT AND ASSUMPTIONS (POST-SCRAM STEAM LINE BREAK)

Point Kinetics Input	Value
Effective Delayed Neutron Fraction	0.005245
Moderator Temperature Coefficient (pcm/°F)	-30
HFP Scram Worth (pcm)	6425.0
Shutdown Margin Requirement (pcm)	3600.0

Stuck Rod Location

Within half-core section cooled by affected loop.

Fission Product and Actinide Decay Constants

Default values in S-RELAP5 utilized.

TABLE 14.1.5.2-6 POST-SCRAM STEAM LINE BREAK ANALYSIS SUMMARY

Initial Power Level	Offsite Power Available	Break Location	Maximum Post-Scram Return to Power (MW)	MDNBR	Maximum LHGR (kW/ft)	Fuel Failure (% of Core)
HFP	No	outside containment	158	4.262	15.13	0.0
HFP	Yes	outside containment	259	1.318	23.09	0.0
HZP	No	outside containment	159	6.522	15.16	0.0
HZP	Yes	outside containment	281	1.517	23.56	0.0

TABLE 14.1.5.2-7 LHGR-LIMITING SEQUENCE OF EVENTS - HZP OFFSITE POWER AVAILABLE

Time (s)	Event
0.	Reactor at HZP
0.+	Double-ended guillotine break. Shutdown reactivity inserted. AFW increased to maximum flow, with 660 gpm to each steam generator.
7.7	MSIV closure trip signal
14.6	MSIVs closed
17.2	SI signal
42.2	SI pumps at rated speed (25 s delay)
331.7	SI lines cleared. Boron begins to enter primary system
333.3	Peak post-scram power reached (281 MW)
600.	Calculation terminated; power decreasing

TABLE 14.1.5.2-8 MDNBR-LIMITING POST-SCRAM STEAM LINE BREAK ANALYSIS SUMMARY

Time (s)	Event
0.	Reactor at HZP
0.+	Double-ended guillotine break. Loss of offsite power. Shutdown reactivity inserted. Full AFW flow started, with 660 gpm to each steam generator.
7.3	MSIV closure trip signal
14.3	MSIVs closed
21.4	SI signal
46.4	SI pumps at rated speed (25 s delay)
332.6	SI lines cleared. Boron begins to enter primary system
352.	Peak post-scram power reached (159.1 MW)
600.	Calculation terminated. Power decreasing.

FIGURE 14.1.5.2–1 ONE PUMP HIGH PRESSURE SAFETY INJECTION SYSTEM DELIVERY VS. PRIMARY PRESSURE (POST-SCRAM STEAM LINE BREAK)

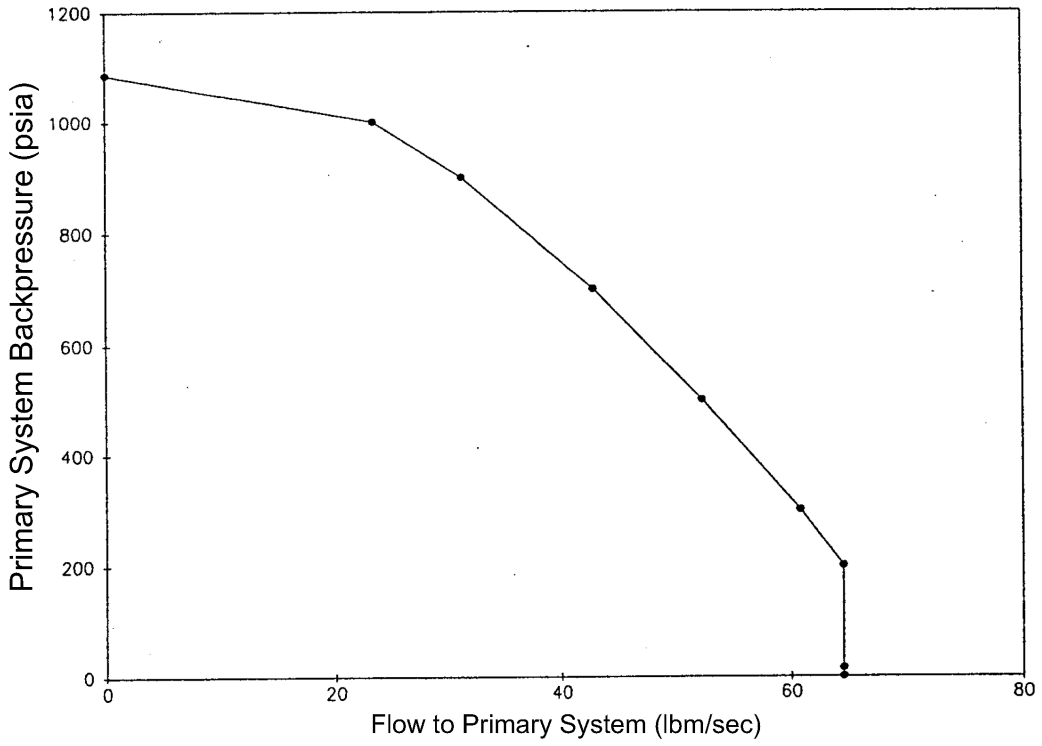


FIGURE 14.1.5.2–2 STEAM GENERATOR BREAK FLOW (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

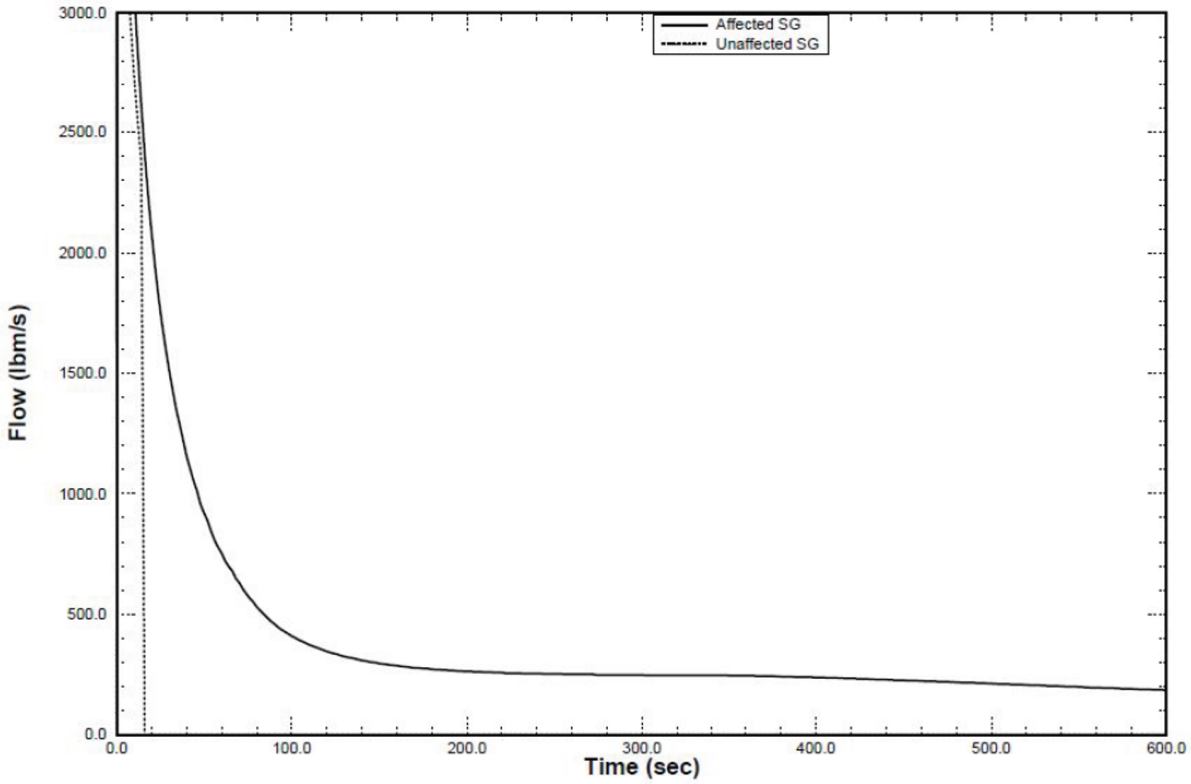


FIGURE 14.1.5.2–3 STEAM GENERATORS' SECONDARY PRESSURES (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

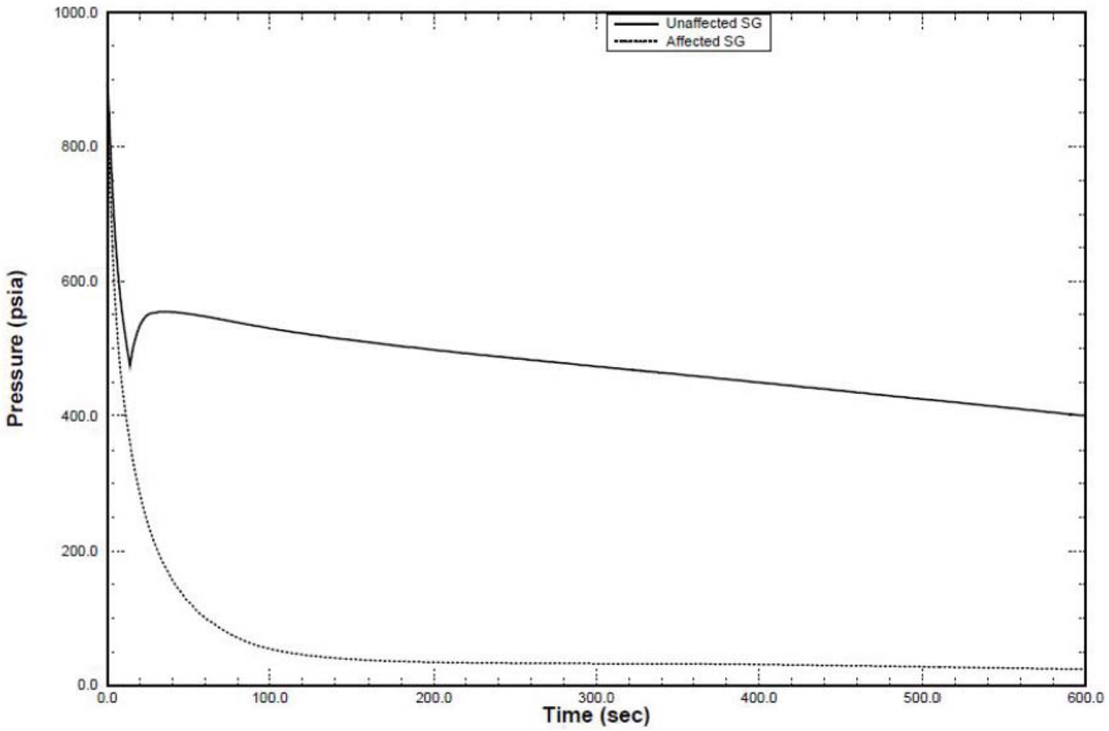


FIGURE 14.1.5.2–4 CORE INLET TEMPERATURES (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

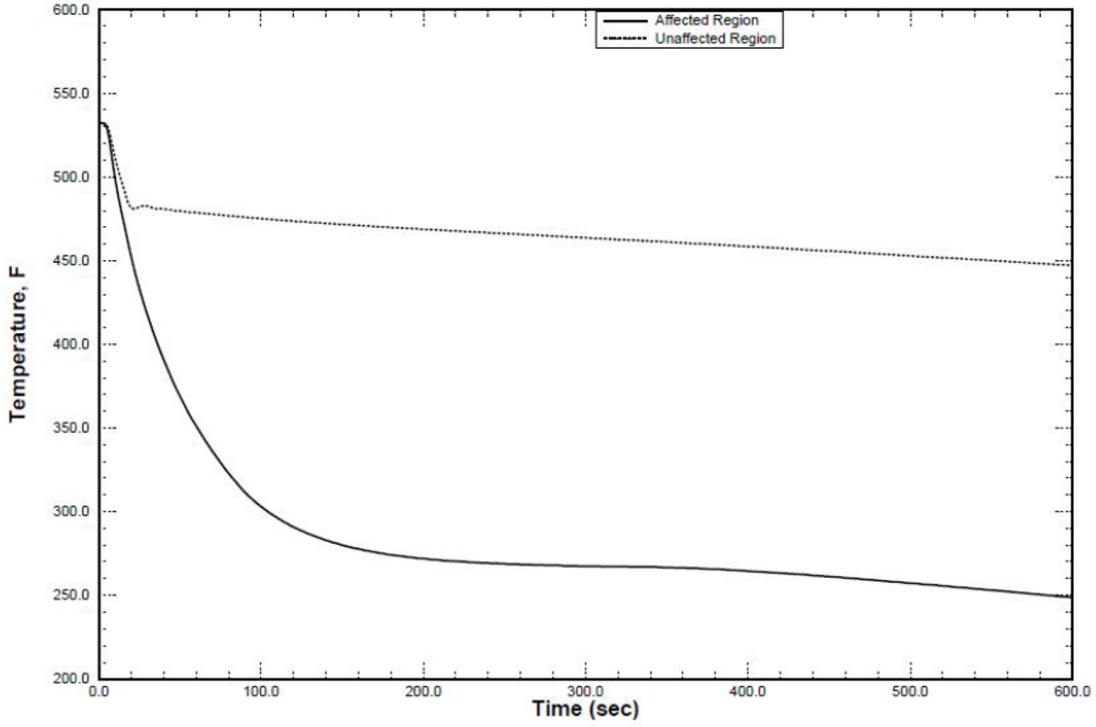


FIGURE 14.1.5.2–5 PRESSURIZER PRESSURE (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

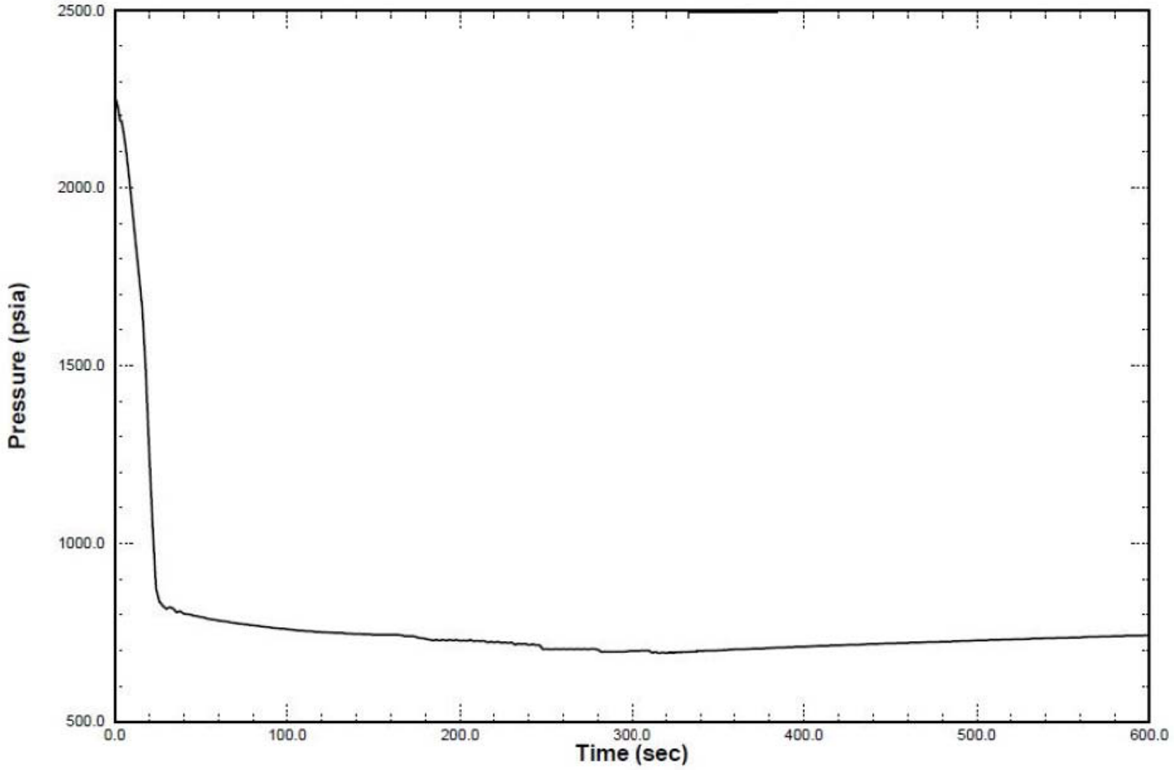


FIGURE 14.1.5.2–6 PRESSURIZER LEVEL (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

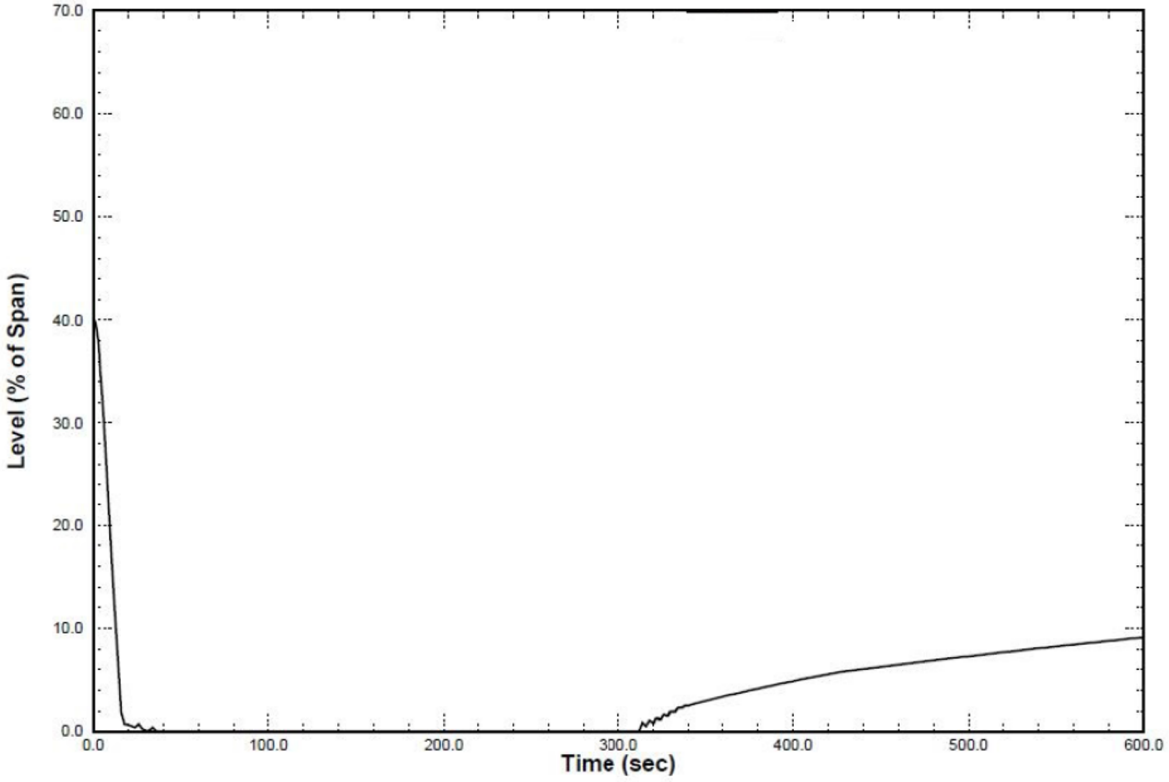


FIGURE 14.1.5.2–7 STEAM GENERATORS' SECONDARY MASS (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

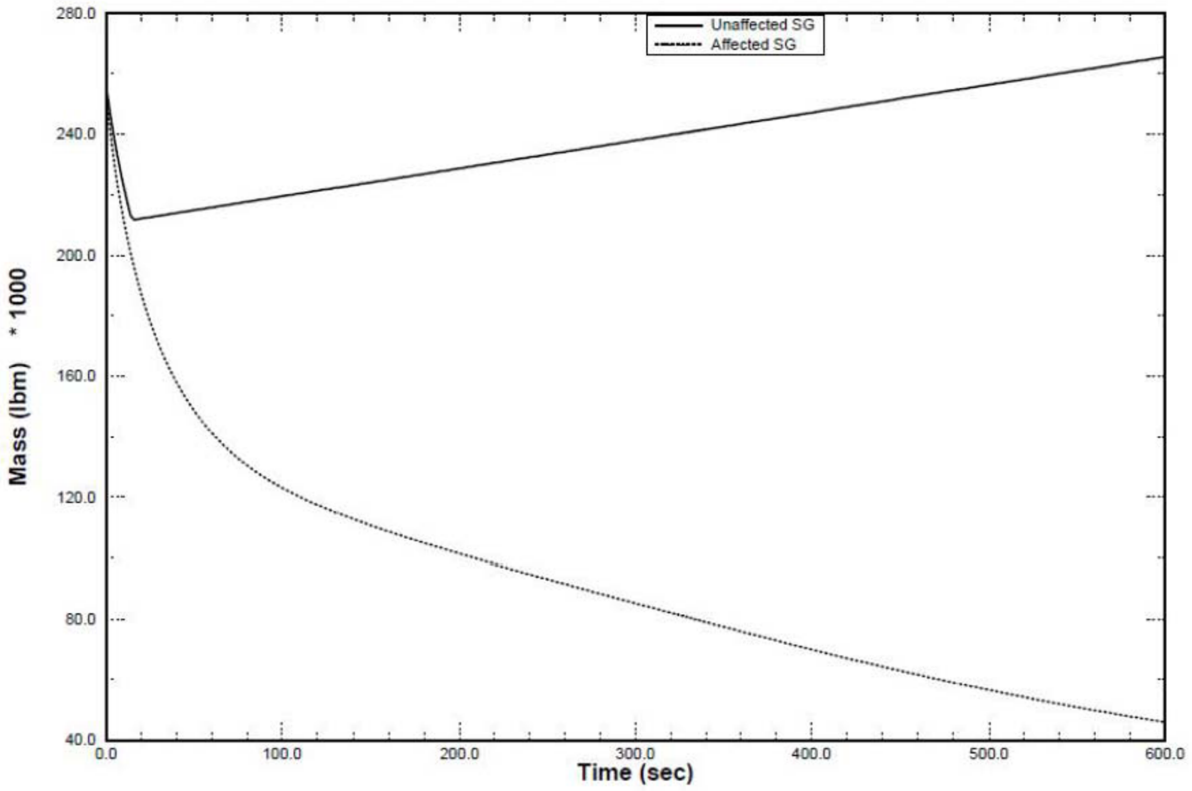


FIGURE 14.1.5.2–8 REACTIVITY COMPONENTS (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

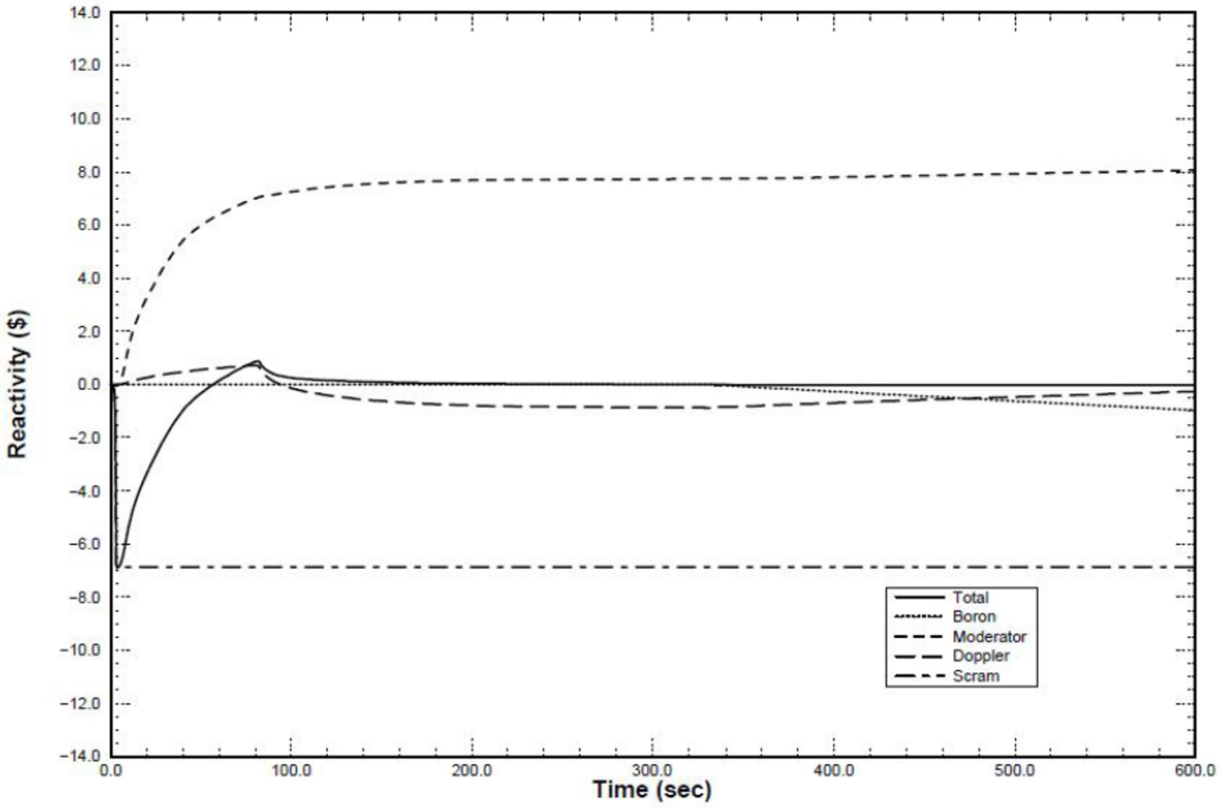


FIGURE 14.1.5.2–9 REACTOR POWER (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH OFFSITE POWER AVAILABLE)

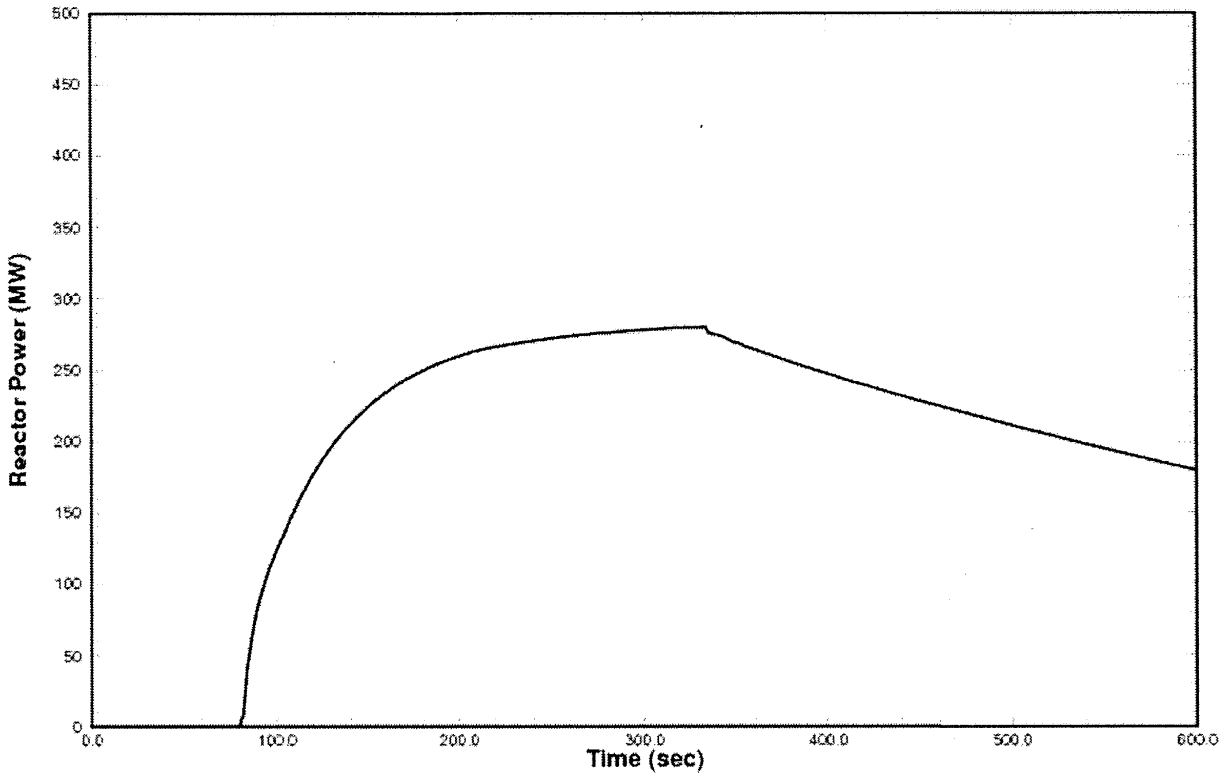


FIGURE 14.1.5.2–10 STEAM GENERATOR BREAK FLOW (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

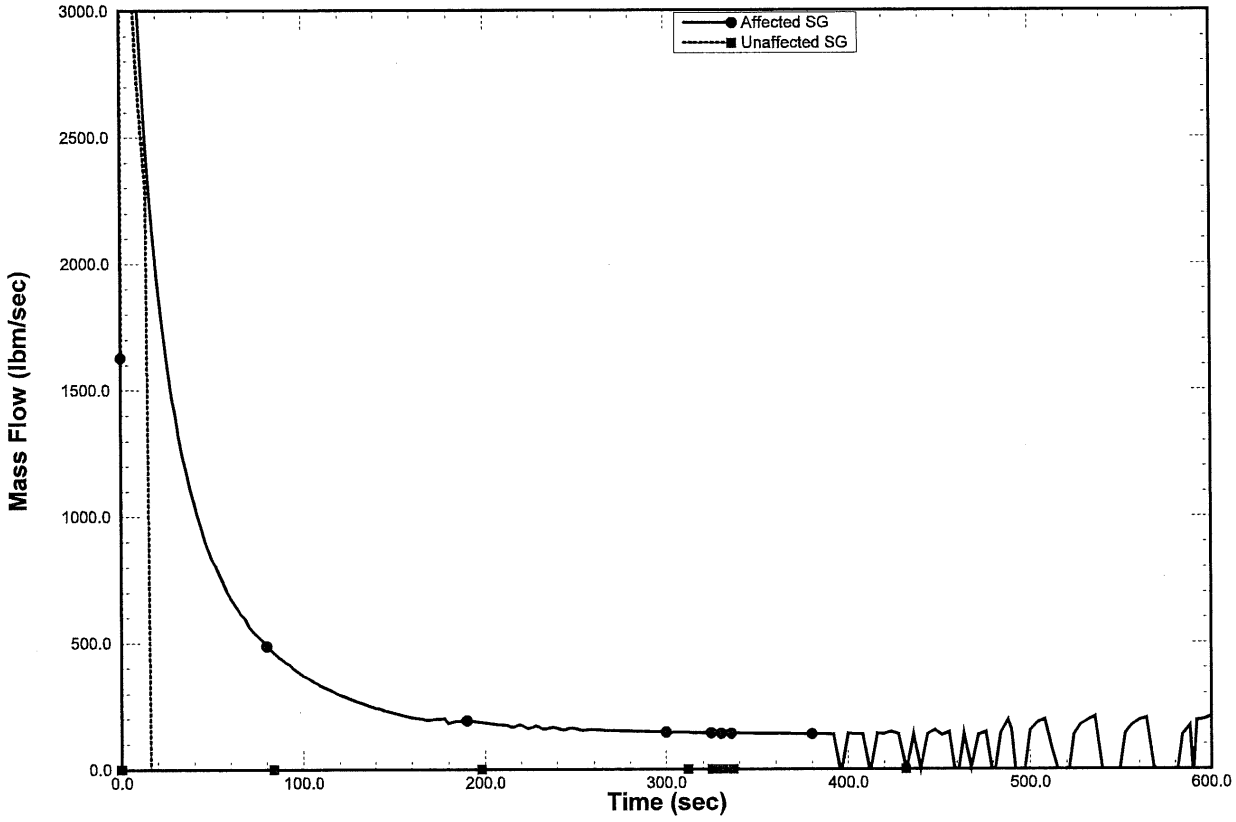


FIGURE 14.1.5.2–11 STEAM GENERATORS' SECONDARY PRESSURES (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

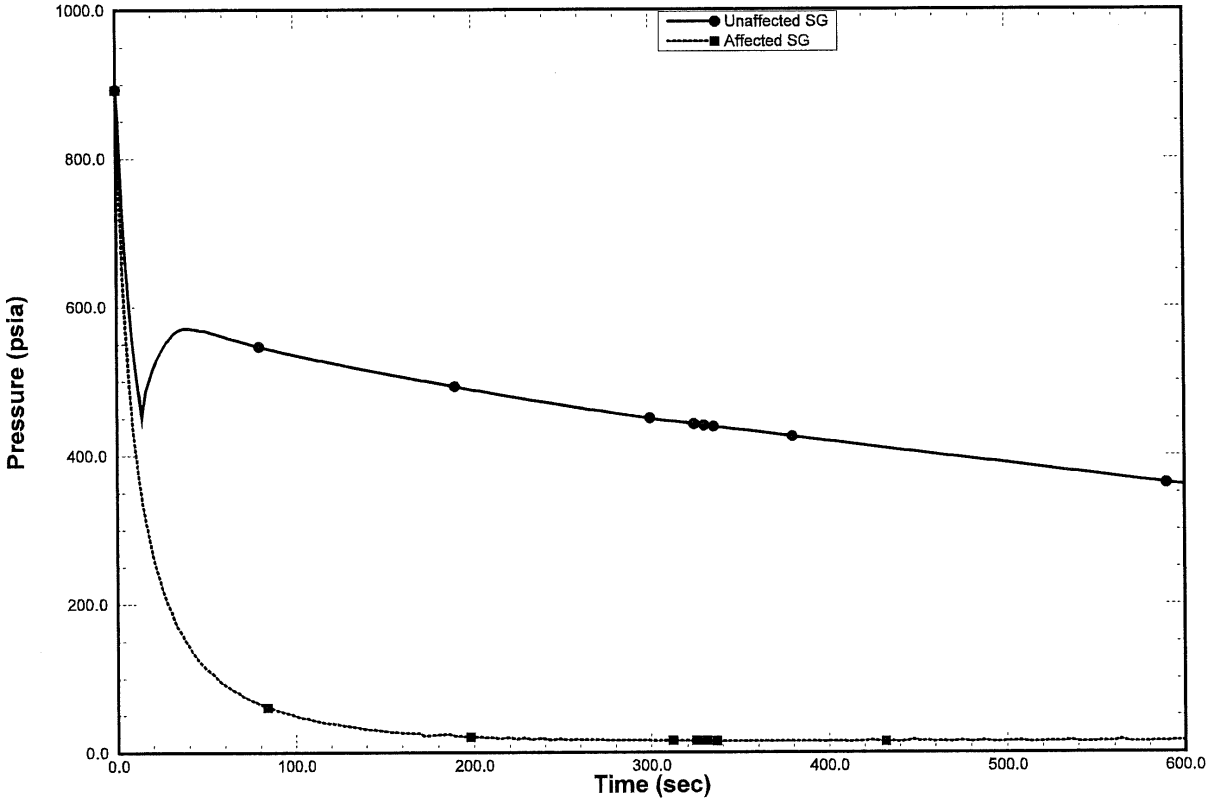


FIGURE 14.1.5.2–12 CORE INLET TEMPERATURES (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

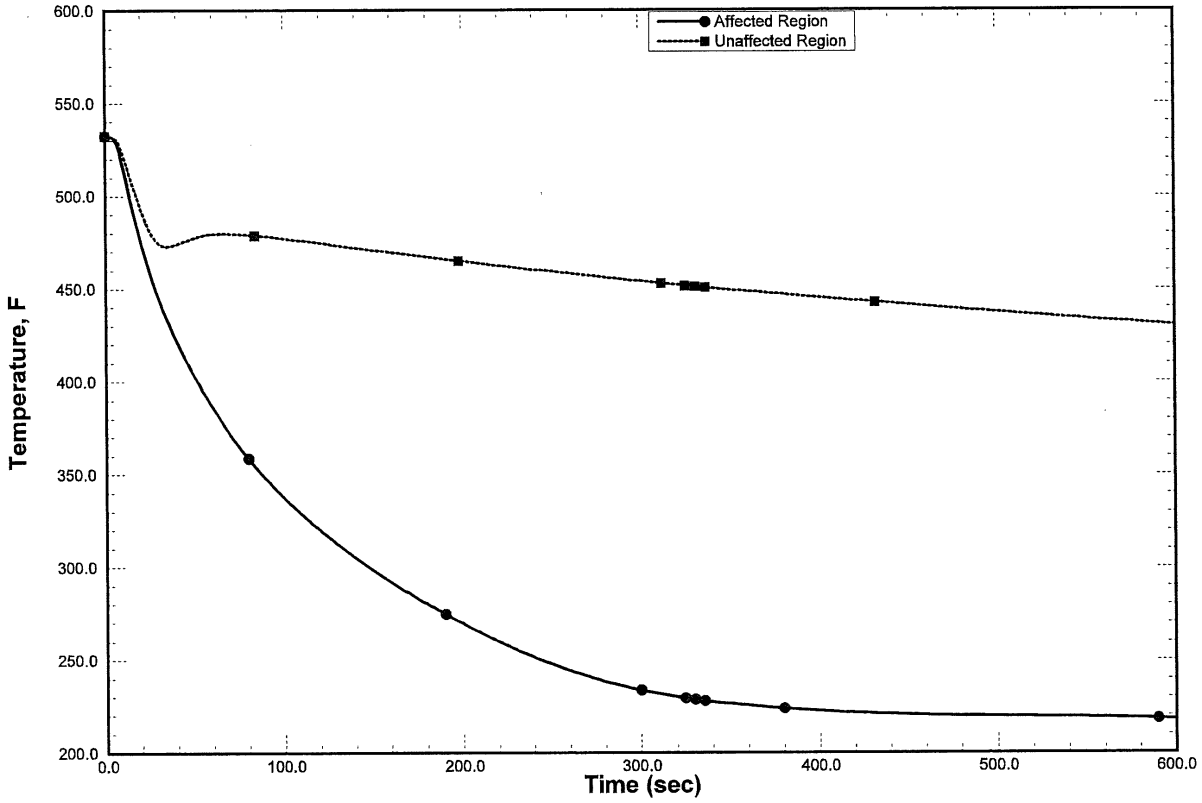


FIGURE 14.1.5.2–13 PRESSURIZER PRESSURE (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

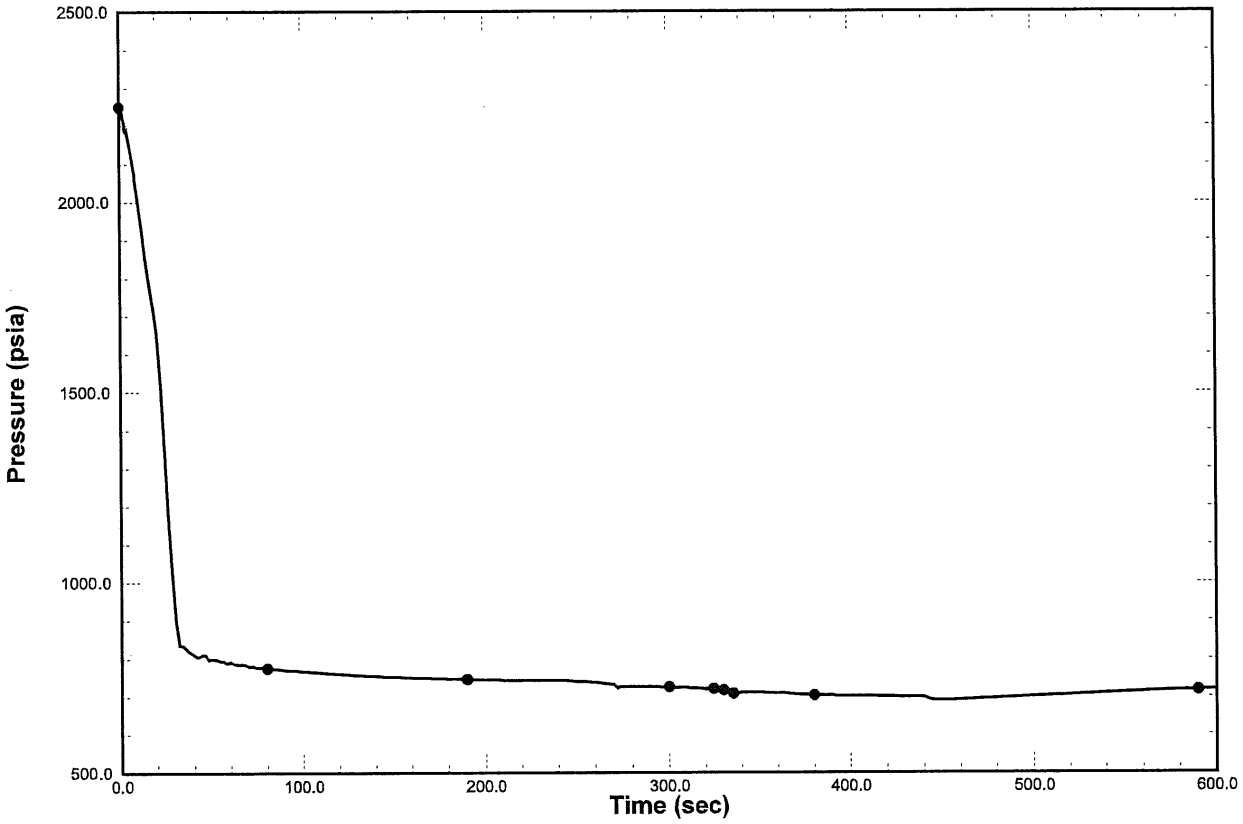


FIGURE 14.1.5.2–14 PRESSURIZER LEVEL (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

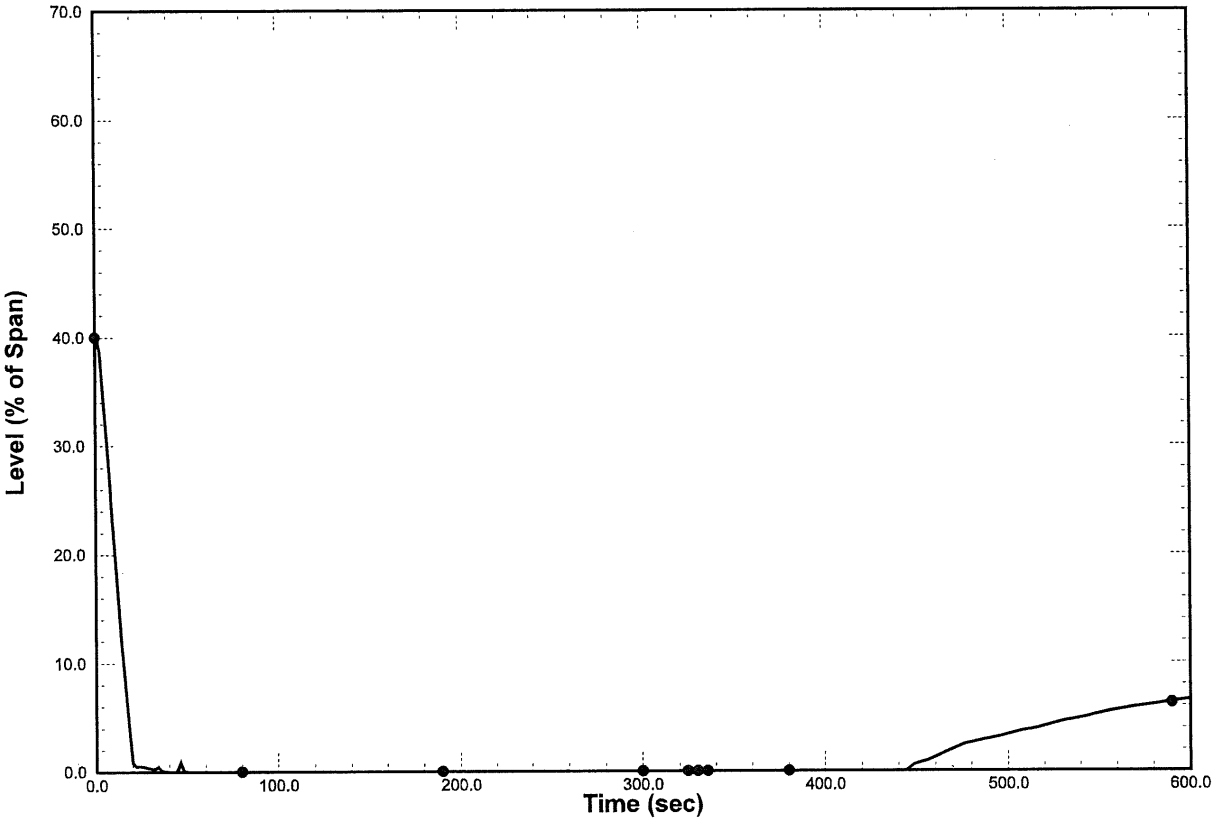


FIGURE 14.1.5.2–15 REACTIVITY COMPONENTS (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

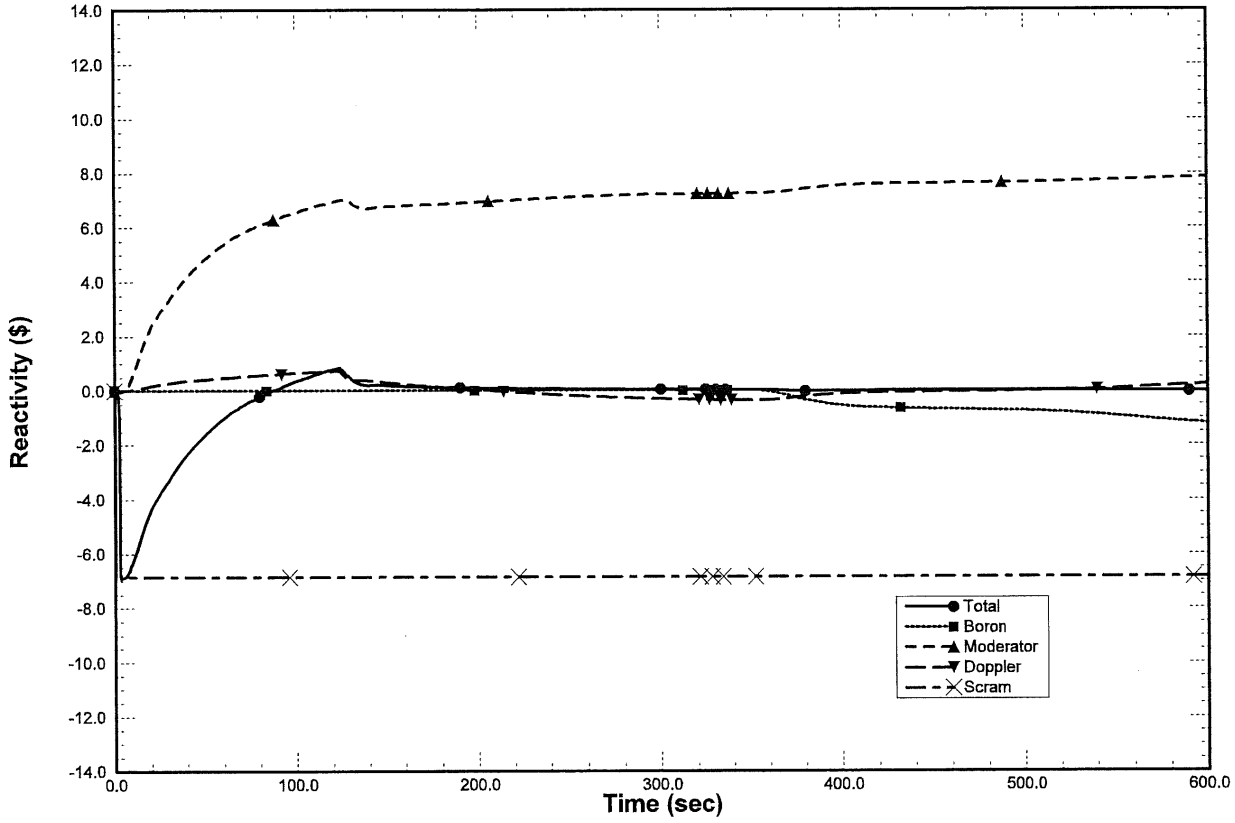


FIGURE 14.1.5.2–16 REACTOR POWER (HZP POST-SCRAM STEAM LINE OUTSIDE CONTAINMENT BREAK WITH LOSS OF OFFSITE POWER)

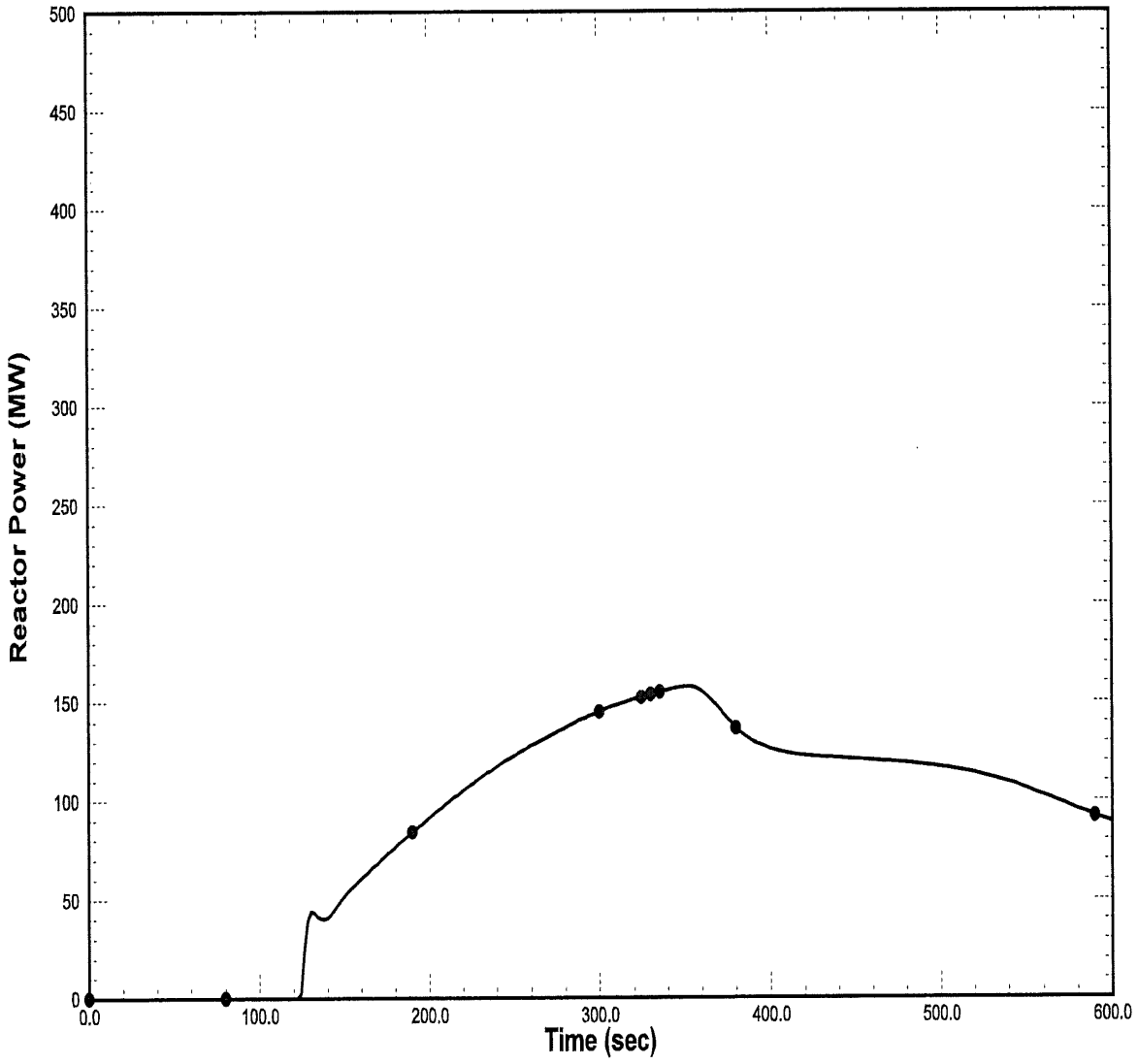


TABLE 14.1.5.3-1 ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS

Core Power Level (MWt)	2754
Primary to Secondary Leak Rate per Steam Generator	150 gpd
Primary Coolant Iodine Concentration	0.5 μCi/gm DE I-131
Secondary Coolant Iodine Concentration	0.05 μCi/gm DE I-131
Primary Coolant Noble Gas Concentration	550 μCi/gm DE Xe-133
Pre-accident Spike Iodine Concentration	30 μCi/gm DE I-131
Failed Fuel Percentage (Containment release only)	3.7%
Peaking Factor	1.69
Reactor Coolant Mass	423,000 lbm
Intact Steam Generator Minimum Mass	91,092 lbm
Site Boundary Breathing Rate (m^3/sec)	
EAB: 0 - 720 hour	3.5E-04
LPZ: 0 - 8 hour	3.5E-04
8 - 24 hour	1.8E-04
24 - 720 hour	2.3E-04
Site Boundary Dispersion Factors (sec/m^3)	
EAB: 0 - 2 hour	3.66E-04
LPZ: 0 - 4 hour	4.80E-05
4 - 8 hour	2.31E-05
8 - 24 hour	1.60E-05
24 - 96 hour	7.25E-06
96 - 720 hour	2.32E-06
Control Room Breathing Rate	3.5E-04 m^3/sec
Control Room Isolation Time - MSLB outside containment (Operator action in the event the break is small enough to not automatically isolate the control room.	4 hours

TABLE 14.1.5.3-1 ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS (CONTINUED)

Control Room Isolation Time - MSLB Inside Containment	140 seconds
Control Room Intake Prior to Isolation	800 cfm
Control Room Inleakage After Isolation	250 cfm
Control Room Emergency Filtered Recirculation Rate (t = 1 hour after isolation)	2,250 cfm
Control Room Intake Dispersion Factors (sec/m^3)	
Containment Enclosure Building - Ground Release	
0 - 2 hour	3.00E-03
2 - 8 hour	1.87E-03
8 - 24 hour	6.64E-04
24 - 96 hour	5.83E-04
96 - 720 hour	4.97E-04
Atmospheric Dump Valves (ADVs) & Enclosure Building Blowout Panels	
0 - 2 hour	7.40E-03
2 - 8 hour	5.71E-03
8 - 24 hour	2.13E-03
24 - 96 hour	1.74E-03
96 - 720 hour	1.43E-03
Turbine Building Blowout Panels	
0 - 2 hour	1.22E-02
2 - 8 hour	8.67E-03
8 - 24 hour	3.77E-03
24 - 96 hour	2.92E-03
96 - 720 hour	2.23E-03

TABLE 14.1.5.3-1 ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS (CONTINUED)

Control Room Free Volume	35,650 ft^3
Control Room Filter Efficiency (particulate/elemental/organic)	90 / 90 / 70 % (1)
Dose Conversion Factors	Federal Guidance Reports 11 and 12

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for filter efficiencies of 90% for all iodine species.

TABLE 14.1.5.3-2 SUMMARY OF MILLSTONE 2 MSLB ACCIDENT DOSES

MSLB	EAB, rem - TEDE	LPZ, rem - TEDE	Control Room, rem - TEDE
MSLB outside Containment			
pre-accident spike	0.1	0.1	2.2
concurrent spike	0.1	0.1	3.0
MSLB in Containment			
3.7% fuel failure	0.2	0.1	1.7

TABLE 14.1.5.3-3 DELETED BY FSARCR PKG FSC 07-MP2-006

14.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

14.2.1 LOSS OF EXTERNAL LOAD

14.2.1.1 Event Initiator

A major loss of load can be initiated as the result of a loss of external electrical load or a turbine trip. Turbine stop valve closure is assumed as the initiator of this event because this is the fastest load rejection which can be postulated which will challenge the plant overpressure and Specified Acceptable Fuel Design Limits (SAFDL) protection features. The assumed fast valve closure time (0.02 sec.) and the assumed unavailability of the steam dump system allow this event to bound the effects of Event 14.2.2 (Turbine Trip — Steam dump system available) and the simultaneous closure of both Main Steam Isolation Valves (MSIV) case of Event 14.2.4 (Closure of the MSIVs — Valve closure time > 0.02 seconds).

14.2.1.2 Event Description

For a full load reduction at power, the primary to secondary heat transfer would be severely diminished because of the increase in secondary side temperature. Initially, in response to the load reduction and diminished energy removal through the secondary system, the primary system temperatures begin to increase. The increasing primary system average temperature causes an insurge into the pressurizer due to the expanding primary fluid. The primary system pressure increases as the pressurizer steam space is compressed by the insurging liquid. Primary system overpressure protection is afforded by the pressurizer power-operated relief valves (PORV) and the primary safety valves. Eventually, the secondary system pressure reaches the opening setpoint of the secondary side safety valves and steam discharge occurs to limit the secondary side pressure rise. Energy removal through the steam generator and pressurizer safety valves mitigates the consequences of the load reduction. However, in analyzing the overpressurization aspects of this event, no credit is taken for the PORVs on the primary system or the turbine bypass system and steam dump system on the secondary system.

14.2.1.3 Reactor Protection

Reactor protection is provided by the high pressurizer pressure, variable overpower, thermal margin/low pressure (TM/LP), and low steam generator water level trips. If the turbine is tripped at the initiation of this event, a direct reactor trip signal would be generated and the effects of this event would be mitigated. However, no credit is taken for a direct reactor trip on turbine trip. Additionally, reactor protection is provided by the primary and secondary safety valves. Because of the potential for increasing the primary system temperatures, with small increases in pressure, this event can challenge the SAFDLs as well as the overpressure criteria mentioned above. Reactor protection for the Loss of External Load event is summarized in Table 14.2.1-1.

14.2.1.4 Disposition and Justification

This event is only credible for rated power and power operating conditions because there is no load on the turbine at other reactor conditions. The consequences of this event for rated power

operation bound the consequences for other reactor conditions because of the maximum stored energy in the primary coolant, minimum initial thermal margin and maximum power to load mismatch which occurs upon loss of load.

Three separate cases of this event are analyzed for Millstone Unit 2 from rated power conditions. Two cases maximize the primary and secondary system overpressurization criteria, respectively, and the third case addresses the fuel design limits. There is no single failure considered which could worsen the results. The disposition of events for the Loss of External Load event is summarized in Table 14.2.1-2.

14.2.1.5 Definition of Events Analyzed

The objectives in analyzing this event are to demonstrate that: the primary pressure relief capacity is sufficient to limit the pressure to less than 110% (2750 psia) of the design pressure, the secondary side pressure relief capacity is capable of limiting the pressure to less than 110% (1100 psia) of design pressure and the minimum Departure From Nucleate Boiling Ratio (DNBR) remains above the safety limit. No credit is taken for direct reactor trip on turbine trip, the turbine bypass system or the steam dump system.

Three cases are analyzed for this event: Two cases addressing the primary and secondary system overpressurization criteria, respectively, and one case addressing the fuel design limits. In the primary overpressurization and fuel design limits cases the input parameters are biased to maximize the increase in reactor power during the transient. The secondary overpressurization case minimizes the increase in reactor power to reduce the pressurizer insurge and delay the reactor trip. For the system overpressurization cases, the parameters and equipment operational states which significantly affect the results of the analysis are selected to maximize the system overpressure, while for the fuel design limit case the parameters and equipment states are selected to reduce the system pressurization and thereby provide a conservative estimate of the minimum DNBR during the transient.

14.2.1.6 Analysis Results

This analysis was performed with AREVA Non-LOCA methodology, Reference 14.2-7, which uses the ANF-RELAP code to calculate the system response, and the XCOBRA-IIIC code to calculate the minimum DNBR.

All cases are initiated with a ramp closure of the turbine control valve in 0.02 seconds. The pressurization of the secondary side results in decreased primary-to-secondary heat transfer, and a substantial rise in primary system temperature. This results in an insurge into the pressurizer, compressing the steam space and pressurizing the primary system. The reactor trips on high pressure. The primary and secondary system safety valve setpoints were modeled with a ± 3% drift allowance, and the flow characteristics were modeled with a 3% allowance for accumulation.

In the primary system overpressurization case, the capacity of the pressurizer safety valves limit the pressurizer pressure to a maximum of 2576 psia. The maximum Reactor Coolant System pressure at the bottom of the reactor vessel is 2717 psia.

For the secondary system overpressurization and the minimum DNBR cases, the transient proceeds in a similar fashion with the primary side pressure being limited by the pressurizer PORV setpoint. The peak secondary system pressure is 1086 psia. The minimum DNBR for this event is bounded by the minimum DNBR of the Section 14.3.1 loss of forced reactor coolant flow event. Because of this, the DNB SAFDL is not violated. This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

The responses of key system variables are given in Figures 14.2.1–1 to 14.2.1–5 for the primary system overpressurization case, Figures 14.2.1–6 to 14.2.1–11 for the secondary system overpressurization case, and Figures 14.2.1–12 to 14.2.1–17 for the minimum DNBR case. The sequence of events for each of these cases is given in Tables 14.2.1-3, 14.2.1-4 and 14.2.1-5, respectively.

The primary and secondary side pressure relief valves have sufficient capacity to limit the respective system pressure to less than 110% (2750 and 1100 psia) of their design pressure.

In the event that one or more of the main steam safety valves are inoperable, analyses have demonstrated that the design pressure limits are met provided thermal power is limited to the values given in Technical Specification 3.7.1.1.

14.2.1.7 Conclusion

The calculated minimum DNBR for the Loss of Load event is above the heat flux correlation safety limit, so the Departure From Nucleate Boiling (DNB) SAFDL is not exceeded in this event. The peak pellet LHR is less than the FCMLHR limit. The maximum primary and secondary system pressures remain below 110% of design pressure. Thus, the Loss of External Load event has been demonstrated to meet all required acceptance criteria.

14.2.2 TURBINE TRIP

14.2.2.1 Event Initiator

This event is initiated by a turbine trip which results in closure of the turbine stop valves and a rapid reduction in energy removal through the steam generators.

14.2.2.2 Event Description

The reactor protection system is designed to generate a reactor trip signal automatically when the turbine is tripped. Following reactor trip, there would be a rapid decrease in the energy being generated in the primary system. This would mitigate the consequences of the turbine trip event. Primary and secondary system overpressurization protection is provided by the code safety valves on both the primary and secondary systems and the secondary atmospheric dump valves. Also, if the condenser was available, the steam bypass system would be activated to reduce the secondary system pressure.

14.2.2.3 Reactor Protection

Reactor protection is provided by the high pressurizer pressure trip, variable overpower trip, TM/LP trip, low steam generator water level trip, and a nonsafety grade reactor trip on turbine trip. Additional protection is also provided by the primary and secondary side safety valves. Reactor protection for the Turbine Trip event is summarized in Table 14.2.2-1.

14.2.2.4 Disposition and Justification

This event is only credible for rated power and power operating conditions since the turbine will either be in tripped condition or there will be no load on the steam generators for other reactor operation conditions. The consequences of this event for rated power operation bound the event consequences for other operating conditions because of the higher initial stored energy in the primary system, maximum power to load mismatch potential, and the reduced SAFDL margin for rated power operation. Because of the limiting assumptions used in the analysis of the consequences of the Loss of External Load (Event 14.2.1), the consequences of the Turbine Trip event are bounded by the consequences of Event 14.2.1, which is analyzed for Millstone Unit 2. The major assumptions used in Event 14.2.1 are the conservatively rapid turbine stop valve closure time, the failure to trip the reactor on turbine trip, and the assumed unavailability of the atmospheric steam dump system. The disposition of events for the Turbine Trip event is summarized in Table 14.2.2-2.

14.2.3 LOSS OF CONDENSER VACUUM

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.2.4 CLOSURE OF THE MAIN STEAM ISOLATION VALVES

14.2.4.1 Event Initiator

The event postulated is the loss of control air to the MSIV operator. Either one or both MSIVs may inadvertently close. The valves are swinging disc-type check valves, installed in a reversed position and held open against steam flow by a pneumatically operating cylinder assembly. The valves are spring-loaded to the closed position.

14.2.4.2 Event Description

The inadvertent MSIV closure is primarily of concern in boiling water reactors as indicated in the Standard Review Plan (Reference 14.2-1), but closure of the MSIVs in a pressurized water reactor would cause a drastic reduction in the load on the reactor. As such, the consequences of a dual MSIV closure are similar to the consequences of Event 14.2.1. Although the valve closure time for the MSIVs is less than 6 seconds, this is much longer than the turbine stop valve closure time assumed in Event 14.2.1 (0.02 seconds); as such, the transient events will proceed somewhat slower and be less severe than in the case of Event 14.2.1.

A single MSIV closure will result in an asymmetric primary temperature distribution. Upon cessation of steam flow, the pressure in the affected steam generator will increase to the opening setpoint of the steam line safety valves. The primary to secondary heat transfer will be diminished, resulting in a heatup of the associated primary side loop. In response to the drop in steam flow through the turbine control valves, the steam flow out of the unisolated steam generator will increase. Depressurization of the steam generator will result, causing an increase in heat removal from the primary. The associated primary side loop will thus experience a cooldown. The side of the core subjected to the cooldown will experience a power rise in the presence of a negative moderator temperature coefficient (MTC).

14.2.4.3 Reactor Protection

Reactor protection for the dual MSIV closure is provided by the high pressurizer pressure trip, variable overpower trip, TM/LP trip, and low steam generator water level trip. Additional protection is provided by the primary and secondary side safety valves.

Reactor protection for the single MSIV closure is provided by the low steam generator level and low steam generator pressure trips. Due to the location of the excore detectors and the asymmetries associated with this event, the variable overpower and TM/LP trip may not get the required 2 out of 4 channels tripped. Further, since one loop will be cooling down and one will be heating up, the pressure may be either increasing or decreasing. Thus, this event cannot take credit for the variable high power, TM/LP or high pressure trips. Additional protection continues to be provided by the primary and secondary side safety valves. Reactor Protection for these events are summarized in Table 14.2.4-1.

14.2.4.4 Disposition and Justification

For simultaneous closure of both MSIVs, the event will progress very similarly to Event 14.2.1. As such, the limiting case is obtained when the event is initiated from rated full power conditions. Due to the decreased heat load and steam flow rates in Modes 2-4 the consequences of the event are bounded by the Mode 1 analysis. The absence of any secondary side heat removal in Modes 5 and 6 eliminates the need to consider the event in these Modes. The turbine stop valve closure time employed in Section 14.2.1 analysis (0.02 sec) is much smaller than the MSIV closure time. Thus, the consequences of Event 14.2.1 will bound those of the dual MSIV closure event.

The asymmetric conditions resulting from the closure of only one of the two MSIVs are similar to that predicted for a steam line break (SLB). That is, the primary coolant loop associated with the closed MSIV experiences a heatup due to the loss of heat sink and the primary coolant loop associated with the open MSIV experiences a cooldown due to the perceived load increase. The temperature increase seen by the hot loop will be limited by the actuation of the steam generator safety valves. The temperature decrease seen by the cooling loop will continue until such time as a reactor trip is generated.

Since the loop experiencing the cooldown will see the larger temperature change, the limiting conditions for the event are at end of cycle (EOC). The EOC MTC is larger in absolute magnitude than the beginning of cycle (BOC) MTC. When the larger MTC is coupled with the larger

temperature change in the cooling loop, a larger overall increase in core power will be predicted. This larger increase in core power will produce the limiting DNB conditions for the event.

Since the asymmetries associated with the event preclude taking credit for the high pressure or variable overpower trip, the single MSIV closure cannot be bounded without analysis by the loss of load, Event 14.2.1.

The limiting single MSIV closure case is that which is initiated from rated power in Mode 1. As for the simultaneous closure of both MSIVs, the reduced heat load and steam flow rates reduce the consequences in all other Modes. For the single MSIV closure case there is also no potential for a post-trip return to power since the remaining MSIV and the turbine stop valves provide redundant means for terminating the remaining steam flow from the unaffected steam generator. Therefore, there is no possibility of a sustained cooldown, thus preventing the addition of sufficient reactivity to the core to return to criticality after reactor trip. There is no single failure considered which could worsen the results. The disposition of events for the Closure of the MSIVs events is summarized in Table 14.2.4-2.

14.2.4.5 Definition of Events Analyzed

As discussed above, the limiting case is obtained when the event is initiated from rated full-power conditions. For simultaneous closure of both MSIVs, the event will progress very similar to Event 14.2.1. The turbine stop valve closure time employed in the Event 14.2.1 analysis (0.02 seconds) is much smaller than the MSIV closure time. Thus, the consequences of Event 14.2.1 will bound those of the dual MSIV closure event.

The objective in analyzing this event is to demonstrate that: the secondary side safety relief capacity is capable of limiting the pressure to less than 110% (1100 psia) of the design pressure, that the minimum DNBR remains above the safety limit, and that the peak LHR is below the centerline melt limit.

The asymmetric conditions resulting from the closure of only one of the two MSIVs is similar to that predicted for an SLB. That is, the primary coolant loop associated with the closed MSIV experiences a heatup due to the loss of heat sink and the primary coolant loop associated with the open MSIV experiences a cooldown due to the perceived load increase. The temperature increase seen by the hot loop will be limited by the actuation of the steam generator safety valves. The temperature decrease seen by the cooling loop will continue until such time as a reactor trip is generated.

Since the loop experiencing the cooldown will see the larger temperature change, the limiting conditions for the event are at EOC. The EOC MTC is larger in absolute magnitude than the BOC MTC. When the larger MTC is coupled with the larger temperature change in the cooling loop, a larger overall increase in core power will be predicted. This larger increase in core power will produce the limiting DNB conditions for the event.

14.2.4.6 Analysis Results

The analysis used EOC initial conditions and assumed that one of the two MSIVs close thereby initiating a heatup on the side of the core associated with the closed MSIV. The side of the core associated with the open MSIV experiences a cooldown corresponding to an approximate doubling of the steam flow. This is because the entire steam demand is shifted to a single steam generator. As the pressure in the steam generator drops, the amount of steam which is supplied by the steam generator is also modeled to decrease. Two sensitivity cases were analyzed. In the first case, at transient initiation, the flow area of the steam control valve was set to a constant value equal to the steady state flow area, thereby simulating the manual mode of operation of the turbine control valves. In this case, the steam flow from a single steam generator with the open MSIV rapidly increases following the initiation of the transient to pick up the load from the isolated steam generator, nearly doubling the initial steam flow through the single MSIV. In the second case, steam flow is modeled as a function of steam generator pressure based on EOC coastdown data. This case simulates the automatic mode of operation of the turbine control valves. The manual mode case, having constant turbine control valve flow area, yielded lower steam flow rates through the transient.

The non-LOCA methodology (Reference 14.2-2) is used to analyze the MSIV closure event. Due to the event asymmetry and the fact that the event proceeds much like a Steam Line Break (SLB) event prior to reactor trip, portions of the pre-scram SLB methodology (Reference 14.2-2) and SLB S-RELAP5 input deck were used to perform this analysis. The input deck was modified to add the MSSV inlet piping and individual MSSVs to the steam line having the closed MSIV. For the single MSIV closure event, the neutronics input required to predict the radial power distribution between the cold and the hot side of the core was, however, redeveloped based on event specific PRISM (Reference 14.2-3) calculations. Core radial power distributions from full-power EOC PRISM cases with differences between the hot-region inlet temperature and the cold-region inlet temperature are used to determine the power split between the halves of the core as a function of the difference in inlet temperatures. Since the temperature differences used in the PRISM cases meets or exceeds the inlet temperature difference calculated by S-RELAP5 during the transient calculation, the power splits used in S-RELAP5 are bounding. The PRISM calculations for the single MSIV closure event are similar to those used in the pre-scram SLB event. The pre-scram SLB and single MSIV closure event analyses both require power distribution data assuming that an all rods out power distribution is appropriate.

The limiting results were obtained from the manual mode turbine control valve case which has lower steam flow rates. The results of the limiting EOC analysis are given in the event summary, Table 14.2.4-3, and in Figures 14.2.4–1 through 14.2.4–5. As indicated in the event summary table the secondary safety valves open early in the transient limiting the temperature rise on the hot side of the core associated with the closed MSIV. The reactor trips on low steam generator pressure which terminates the power rise.

The peak LHR and Minimum Departure from Nucleate Boiling Ratio (MDNBR) are predicted to occur on the cold side of the core. The peak LHR occurs at the time of the reactor trip, and the MDNBR occurs at the time of the peak heat flux. The peak LHR is less than the FCMLHR limit. The minimum DNBR for this event is bounded by the minimum DNBR of the Section 14.3.1 loss

of forced reactor coolant flow event. Thus it is concluded that the DNB limits will not be violated and that fuel failures are precluded during the single MSIV closure event.

The secondary side safety valve setpoints were modeled assuming a +3% tolerance on the opening set pressures, and that the valves would relieve the rated flow at an inlet pressure of 1098 psia. The calculated maximum secondary side pressure is 1093 psia, which is less than 110% (1100 psia) of design pressure.

14.2.4.7 Conclusion

The calculated minimum DNBR for the single MSIV closure event is above the HTP critical heat flux correlation safety limit, so the DNB SAFDL is not exceeded in this event. The peak LHR is less than the FCMLHR limit. The maximum secondary side pressure is below 110% of design pressure. Thus, the single MSIV closure event has been demonstrated to meet all required acceptance criteria.

14.2.5 STEAM PRESSURE REGULATOR FAILURE

Millstone Unit 2 does not have any steam line pressure regulators, so this event is not credible for this plant. No analysis needs to be considered for this event.

14.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.2.7 LOSS OF NORMAL FEEDWATER FLOW

14.2.7.1 Event Initiator

The Loss of Normal Feedwater Flow transient is initiated by a trip of the main feedwater pumps or a malfunction in the feedwater control valves.

14.2.7.2 Event Description

The loss of main feedwater flow will increase the secondary-side temperature and reduce the steam generator heat removal capability because the main feedwater system is supplying subcooled water to the steam generators. The rise in the secondary-side temperature leads to a rise in the primary system coolant temperature. As the primary system temperatures increase, the coolant expands into the pressurizer which increases the pressure by compressing the steam volume.

The temperatures of the secondary sides and primary loops are controlled by the opening and closing of the main steam safety valves and/or the steam dump valves. The long term cooling of the primary system is assured by the secondary-side water inventory supplied by the Auxiliary Feedwater System (AFWS). Two motor-driven auxiliary feedwater (AF) pumps are automatically started upon a steam generator low liquid level signal. If a loss of offsite power occurs, the motor-

driven AF pumps are powered by the emergency diesels. In addition, a turbine-driven AF pump can be manually actuated.

14.2.7.3 Reactor Protection

System overpressure protection is provided by the primary and secondary system safety valves. A reactor trip occurs on low steam generator level with additional reactor protection provided by the high pressurizer pressure trip, variable overpower trip, and the TM/LP trip. Reactor protection for the Loss of Normal Feedwater Flow event is summarized in Table 14.2.7-1.

14.2.7.4 Disposition and Justification

This event is only credible for rated power and power operating conditions because the main feedwater system is not required to provide feedwater to the steam generators for other reactor operating conditions. The consequences of this event for rated power operation bound the consequences for other conditions because of the higher initial stored energy in the primary system, the minimum steam generator inventory, and the greater impact of the loss of feedwater flow on the secondary system.

The near-term pressurization and DNB aspects of this event are bounded by those Events 14.2.1 and 14.3.1, respectively. In Event 14.2.1, reactor trip is delayed until a high pressurizer pressure signal is received. This results in a higher power level at trip, greater pressurization and greater challenge to the SAFDLs than in Event 14.2.7. Long-term pressurization, if it occurs, is very gradual and is arrested by opening of the pressurizer code safety valves.

In Event 14.3.1, the RCPs are tripped as the initiating event. Reactor trip occurs on low coolant flow, and the core flow rate at the time of trip is significantly lower than in the Loss of Normal Feedwater Flow event where the RCPs are tripped coincident with the reactor trip. The core power to flow ratio is much higher for Event 14.3.1, thereby producing a more limiting minimum DNBR.

The Loss of Normal Feedwater event is analyzed to assess the maximum expected pressurizer level swell and the long-term adequacy of the AFWS to restore and maintain steam generator inventory and prevent steam generator dryout. The maximum level swell is examined to assure that the pressurizer does not become water solid. Each case was analyzed using 102% power, maximum allowed positive reactivity feedback, and maximum permitted pressurizer level. The full power initial condition maximizes the core decay heat that must be removed in the post-scram period. A primary concern in simulating this event is to demonstrate adequate long-term cooling capability. The single active failure assumptions reduce heat removal capacity by significantly limiting the amount of AFW flow supplied to the steam generators.

Two single failures considered in the Loss of Normal Feedwater Flow event are the failure of a motor-driven AFW pump to start and the failure of the steam-driven AFW pump to start. Also considered is a loss of offsite power coincident with reactor trip (Reference 14.2-7). The disposition of events for the Loss of Normal Feedwater Flow event is summarized in Table 14.2.7-2.

14.2.7.5 Definition of Events Analyzed

The Millstone Unit 2 AFW system consists of two independent motor-driven pumps which are assumed to start automatically within 240 seconds of AFW system actuation on low-low steam generator level. There is also a steam-driven AFW pump which may be started by operator action. (The operator action is credited 10 minutes following reactor trip in the safety analysis.) The piping configuration allows each pump to supply both steam generators simultaneously. There are two potential single active failures in this configuration: One is the failure of the steam-driven AFW pump to start, and the other is the failure of one of the two motor-driven AFW pumps to start. Because of the differences in pump capacity and actuation times, it is not immediately obvious which single active failure is the most limiting.

Another uncertainty when analyzing the Loss of Normal Feedwater Flow event is the effect of the RCP trip. If the RCPs remain on, the pump heat imposes a significant heat load on the system. If the RCPs are tripped, primary to secondary heat removal capability is degraded due to sole reliance on natural circulation.

The loss of offsite power option (RCP trip), combined with the two single active failure possibilities produces a total of four base cases. The four cases collectively demonstrate compliance with both the pressurizer overfill criterion and the steam generator secondary water inventory criterion when the MSSVs are the sole secondary steam release path. The biases and initial conditions for the cases are identical and are selected to maximize pressurizer level increase and to minimize steam generator level recovery. A fifth case considers the effects of steam dump system operation and determines whether the SG inventory boiloff required to cool the RCS to no load temperature is offset by increased AFW flow at lower SG pressures.

The initiating event for each case is an instantaneous loss of main feedwater.

The analysis is performed with the ANF-RELAP code (Reference 14.2-6). The ANF-RELAP code includes relevant aspects of the mass and energy balance of the primary and secondary systems.

Additional conservative conditions are applied for analysis of each case to present the greatest challenge to the event acceptance criteria. In accordance with AREVA methodology, symmetric tube plugging is modeled for the four base cases utilizing the MSSVs as the sole secondary steam release path. No steam generator tube plugging is applied for the fifth case. This conservatively minimizes post-trip steam generator liquid inventories by producing slightly higher steam generator pressures and consequently lower AFW flows when steam generator pressure is controlled by the steam dumps instead of by the MSSVs.

14.2.7.5.1 Analysis Results

The Loss of Normal Feedwater Flow event is initiated from 102% power with each steam generator at nominal liquid levels. A total instantaneous loss of all Main Feedwater flow initiates the event. When loss of offsite power assumptions are applied, the loss of offsite power and reactor coolant pump trip is assumed to occur at scram. The reactor trips on the steam generator

low water level signal. The turbine trips one second after the reactor trips. Cases one through four were evaluated assuming an analytical low SG level reactor trip setpoint of 34%. Case five was evaluated with an analytical low level trip setpoint of 43%.

The cases where off site power is available and the RCPs maintain forced coolant flow through the primary system produce the lowest steam generator inventories. The loss of one of the two motor-driven AFW pumps combined with post-trip RCS temperature control using the steam dumps resulted in a minimum steam generator liquid mass inventory of 5,540 lbs per steam generator at 654 seconds. The steam generator level steadily recovers from this minimum level, thus ensuring continued heat removal. An event summary is presented in Table 14.2.7-3. The transient responses are presented in Figures 14.2.7–1 through 14.2.7–6. The transient execution time was 1800 seconds.

The cases where offsite power is assumed to be lost coincident with the reactor trip, and primary to secondary heat transfer is achieved via natural circulation, generated the highest pressurizer levels. The loss of one of the two motor-driven AFW pumps produced the maximum pressurizer level of 76.3% at 43 seconds. Sufficient steam volume remains to preclude the expulsion of liquid from the pressurizer safety valves. An event summary is presented in Table 14.2.7-4. The transient responses are presented in Figures 14.2.7–6 through 14.2.7–10. The transient execution time was 2400 seconds.

14.2.7.6 Conclusions

A loss of normal feedwater event does not result in the violation of SAFDLs, peak pressurizer pressure does not exceed 110% of the design rating and primary liquid is not expelled through the pressurizer safety valves. Adequate cooling water is supplied by the AFWS to allow a safe and orderly plant shutdown and to prevent steam generator dryout. Thus, the loss of normal feedwater event has been demonstrated to meet all required acceptance criteria.

14.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT

This event is not in the current licensing basis for Millstone Unit 2 and, therefore, is not analyzed.

14.2.9 REFERENCES

14.2-1 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

14.2-2 “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” EMF-2310(P)(A), Revision 1, Framatome-ANP, May 2004.

14.2-3 “Reactor Analysis Systems for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,” EMF-96-029 (P)(A), Siemens Power Corporation, January 1997.

14.2-4 Technical Specifications for Millstone Unit 2, Docket Number 50-336.

14.2-5 “Advanced Nuclear Fuels Methodology for Pressurized Water Reactors - Analysis of Chapter 15 Events,” ANF-84-73(P)(A), Rev. 5, Advanced Nuclear Fuels Corp., October 1990.

14.2-6 “ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events,” ANF-89-151(P)(A), Siemens Power Corporation, May 1992.

14.2-7 “Millstone Unit No. 2 Loss of Normal Feedwater Flow Transient with Reduced Auxiliary Feedwater Flow,” EMF-98-015, Rev. 1, Siemens Power Corporation, December 1998.

14.2-8 “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” EMF-92-153(P)(A), Revision 1, Siemens Power Corporation, January 2005.

TABLE 14.2.1-1 AVAILABLE REACTOR FOR THE LOSS OF EXTERNAL LOAD EVENT

Reactor Operational Mode	Reactor Protection
1	High Pressurizer Pressure Trip Variable Overpower Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip
2	High Pressurizer Pressure Trip Variable Overpower Trip Low Steam Generator Water Level Trip
3-6	No Analysis Required; Not a Credible Event

TABLE 14.2.1-2 DISPOSITION OF EVENTS FOR THE LOSS OF EXTERNAL LOAD EVENT

Reactor Operational Mode	Reactor Protection
1	Analyze
2	Bounded by the above, no analysis required
3-6	No Analysis Required; Not a Credible Event

TABLE 14.2.1-3 EVENT SUMMARY FOR THE LOSS OF EXTERNAL LOAD EVENT (PRIMARY OVERPRESSURIZATION CASE)

Event	Time (seconds)
Turbine Trip	0.0
First MSSV opens	4.3
Primary pressure reaches high pressure trip setpoint	4.9
Reactor scrams	6.3
Pressurizer safety valves open	7.4
Peak pressurizer pressure	7.4
Primary system peak pressure	7.9
Last MSSV opens	9.5
End of calculation	20.0

TABLE 14.2.1-4 EVENT SUMMARY FOR THE LOSS OF EXTERNAL LOAD EVENT (SECONDARY OVERPRESSURIZATION CASE)

Event	Time (seconds)
Turbine Trip	0.0
First MSSV opens	2.9
Pressurizer spray begins	3.2
Pressurizer PORV opens	4.9
Primary pressure reaches high-pressure trip setpoint	4.9
Pressurizer safety valves open	5.0
Peak pressurizer pressure	5.0
Reactor scrams	6.3
Last MSSV opens	10.1
Steam Generator secondary peak pressure	10.2
End of calculation	20.0

TABLE 14.2.1-5 EVENT SUMMARY FOR THE LOSS OF EXTERNAL LOAD EVENT (MINIMUM DEPARTURE FROM NUCLEATE BOILING RATIO CASE)

Event	Time (seconds)
Turbine Trip	0.0
Pressurizer spray on	3.4
First MSSV opens	4.3
Primary pressure reaches high pressure trip setpoint	4.9
Pressurizer PORVs open	4.9
Primary safety valves open	4.9
Peak pressurizer pressure	5.0
Reactor scrams	6.3
End of calculation	20.0

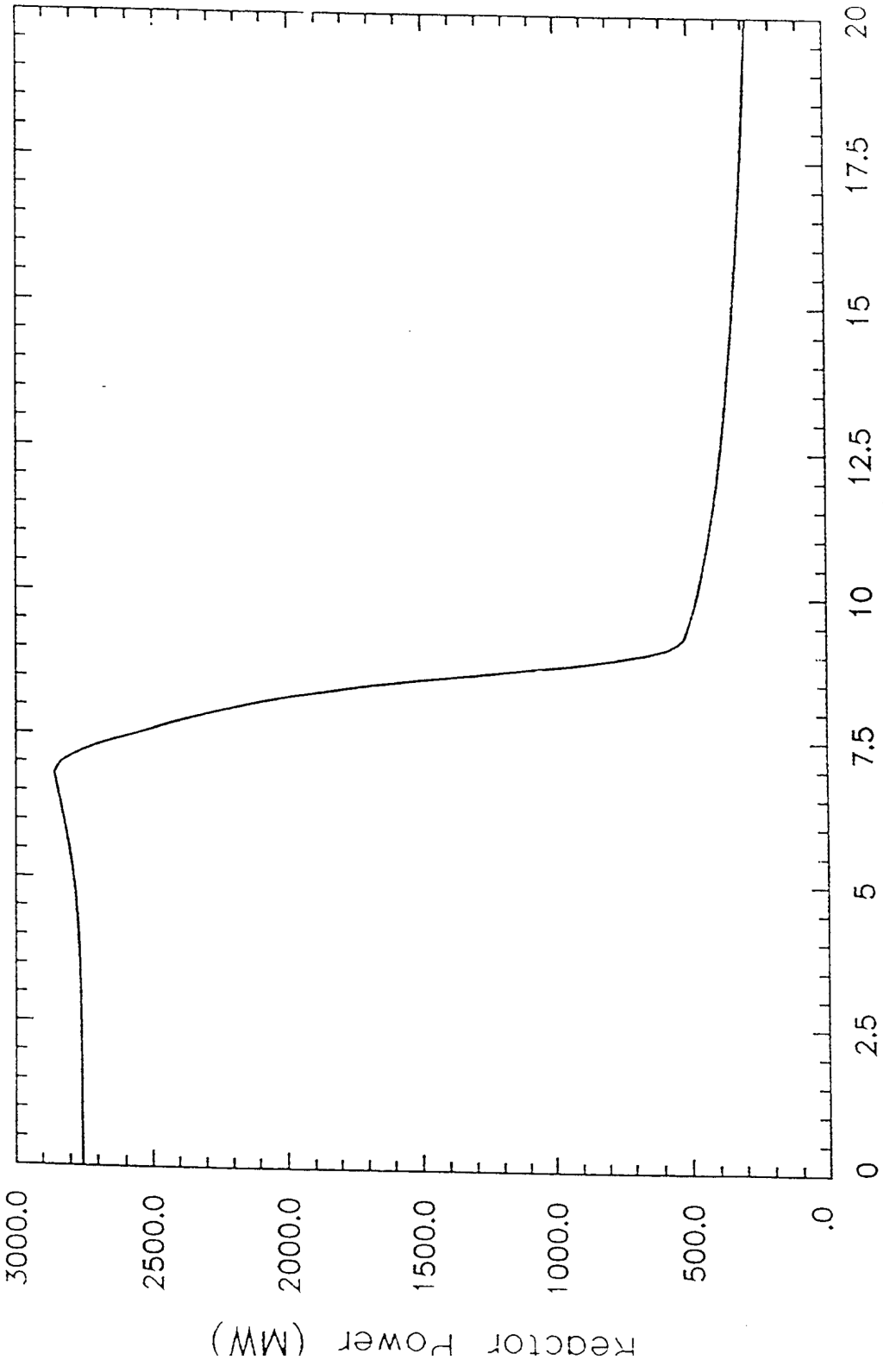


FIGURE 14.2.1–1 REACTOR POWER LEVEL FOR LOSS OF EXTERNAL LOAD (PRIMARY OVERPRESSURIZATION CASE)

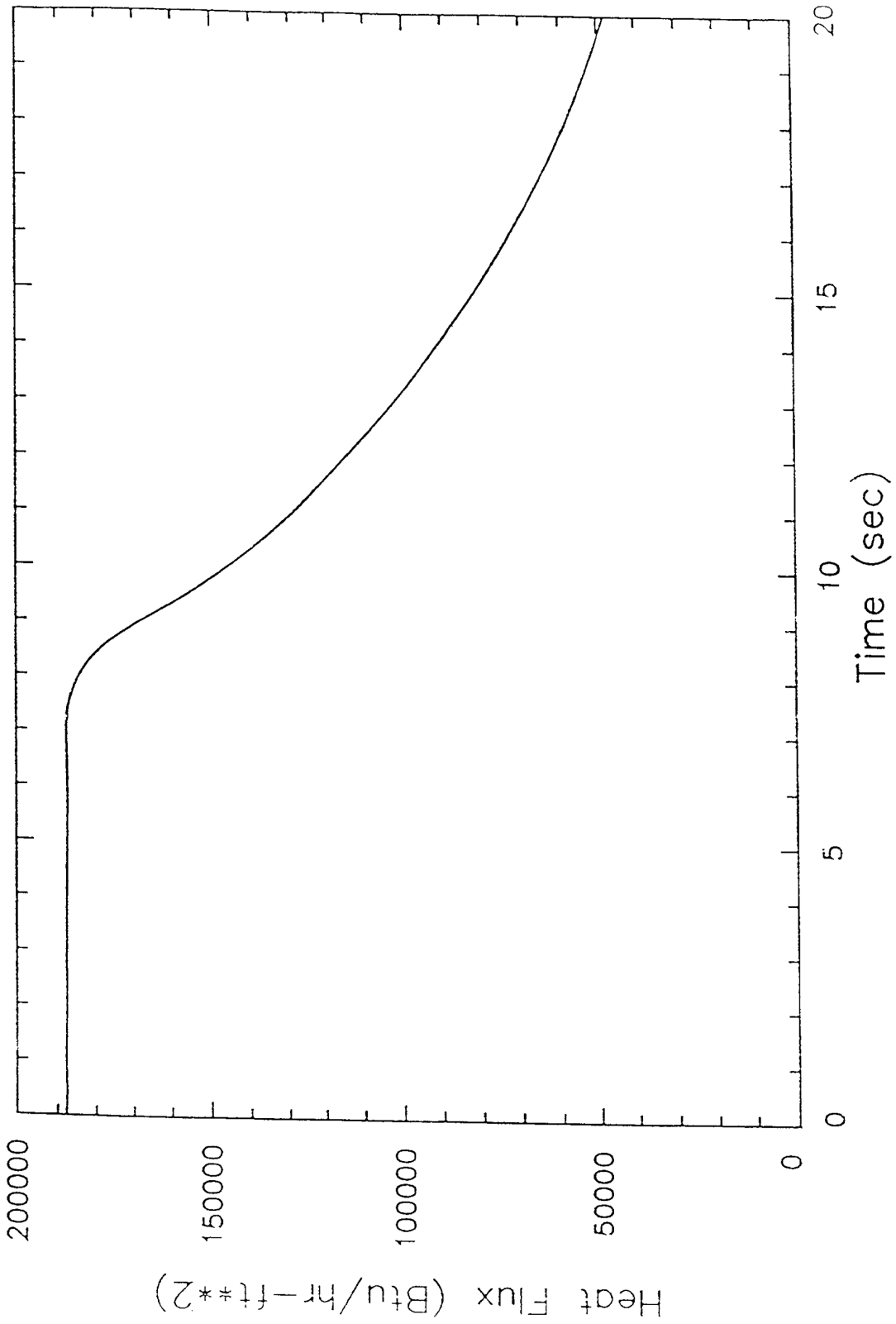


FIGURE 14.2.1–2 CORE AVERAGE HEAT FLUX FOR LOSS OF EXTERNAL LOAD (PRIMARY OVERPRESSURIZATION CASE)

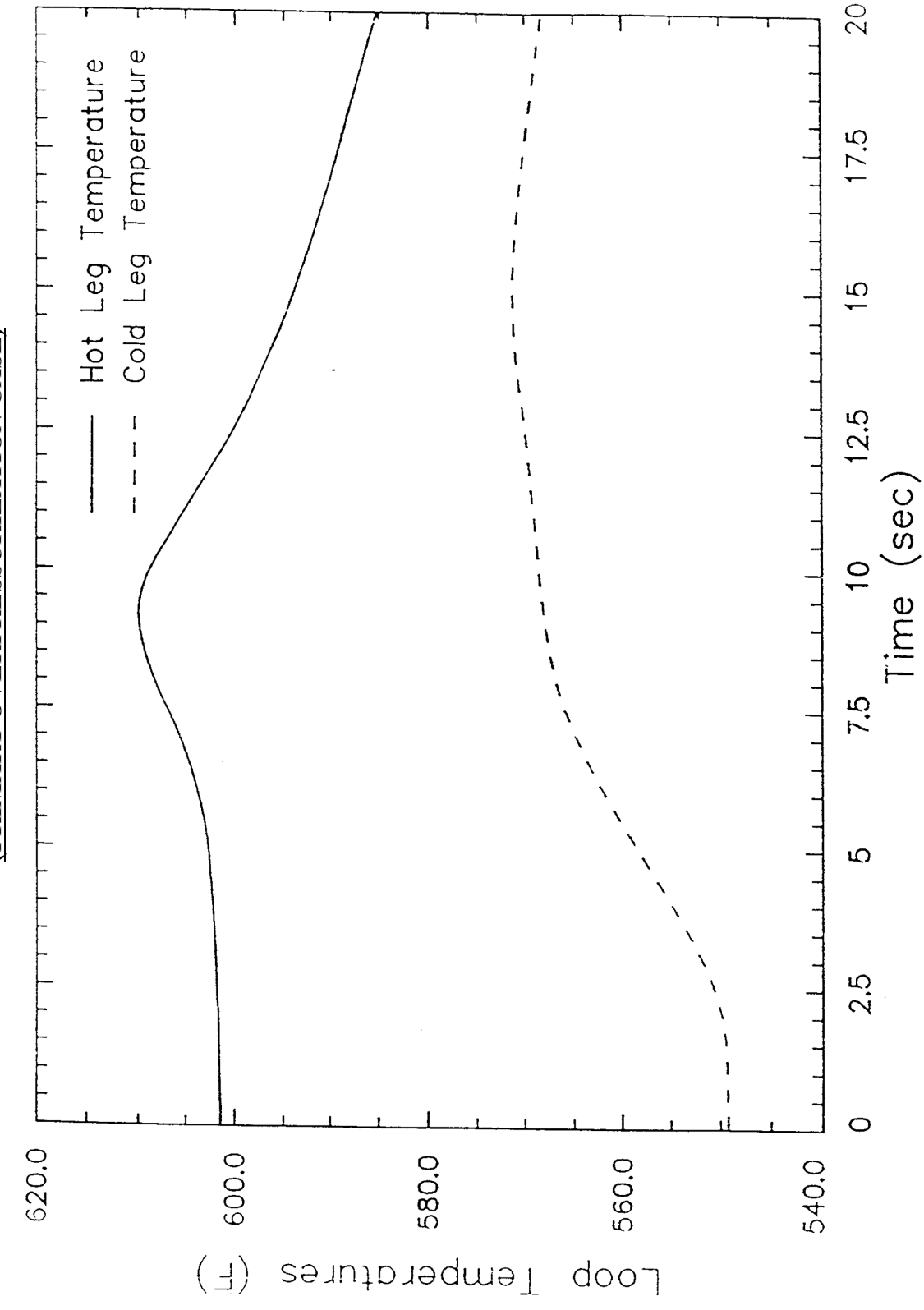


FIGURE 14.2.1–3 REACTOR COOLANT SYSTEM TEMPERATURES FOR LOSS OF EXTERNAL LOAD (PRIMARY OVERPRESSURIZATION CASE)

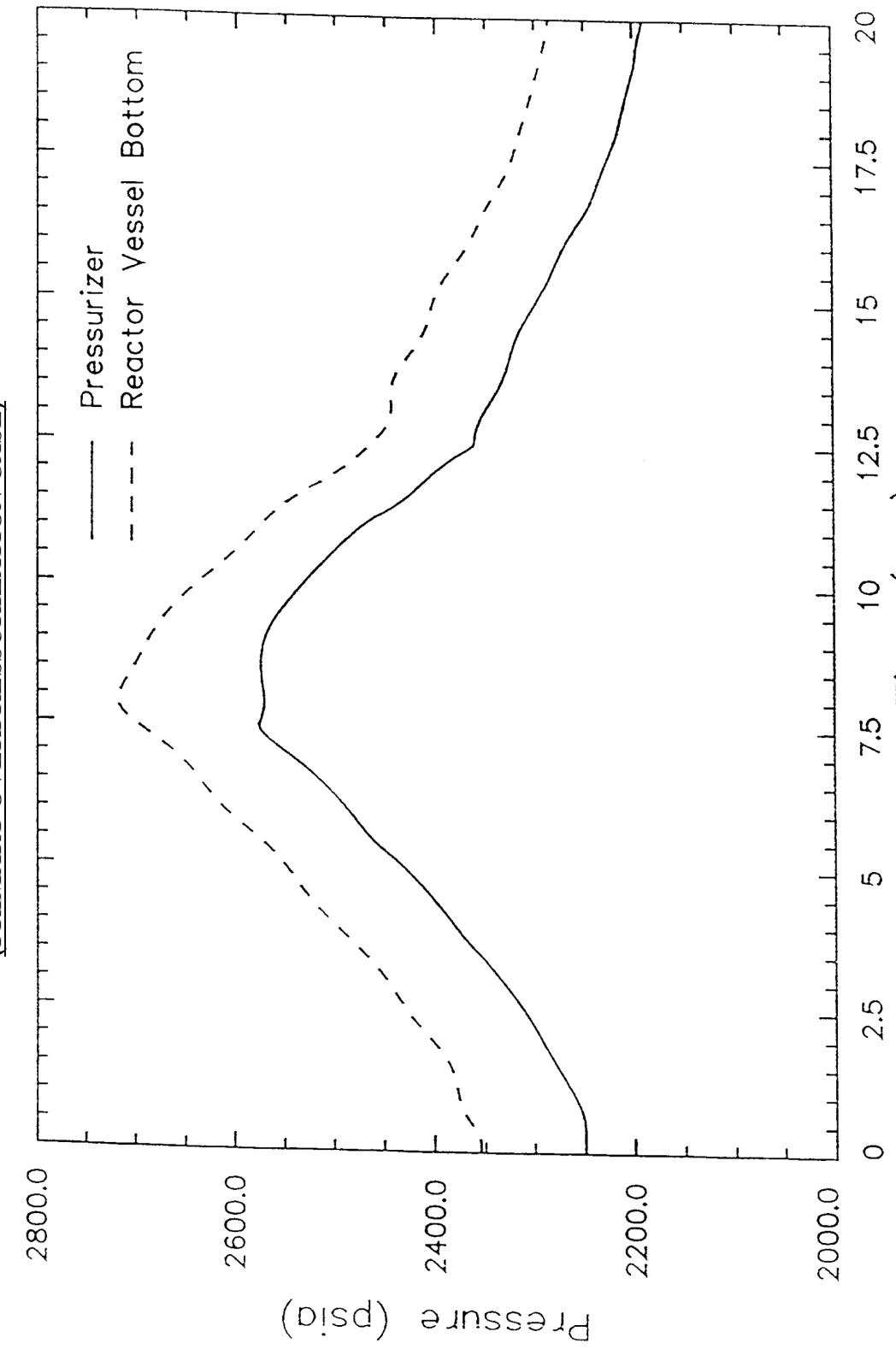


FIGURE 14.2.1–4 PRIMARY SYSTEM PRESSURES FOR LOSS OF EXTERNAL LOAD (PRIMARY OVERPRESSURIZATION CASE)

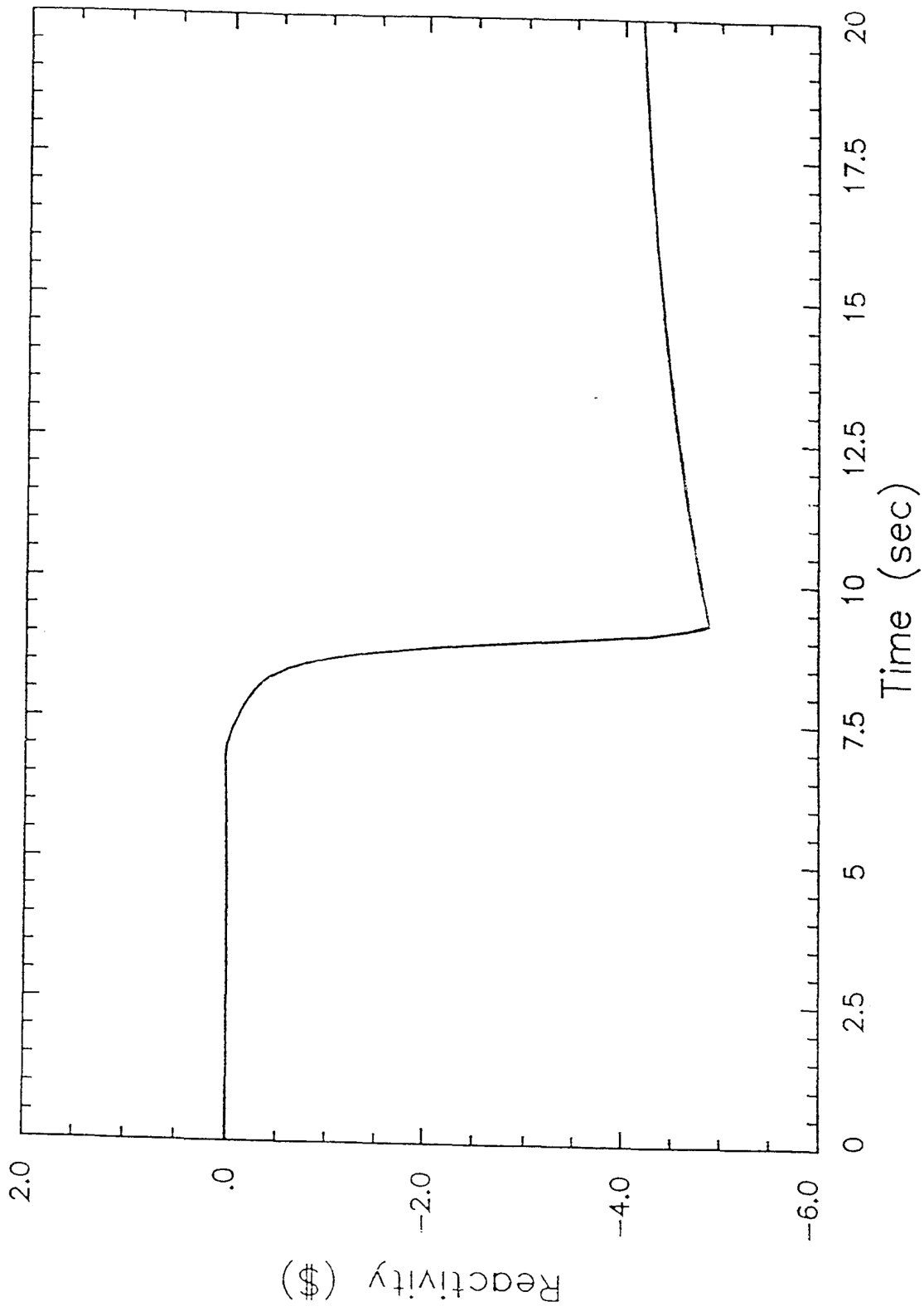


FIGURE 14.2.1–5 TOTAL REACTIVITY FOR LOSS OF EXTERNAL LOAD (PRIMARY OVERPRESSURIZATION CASE)

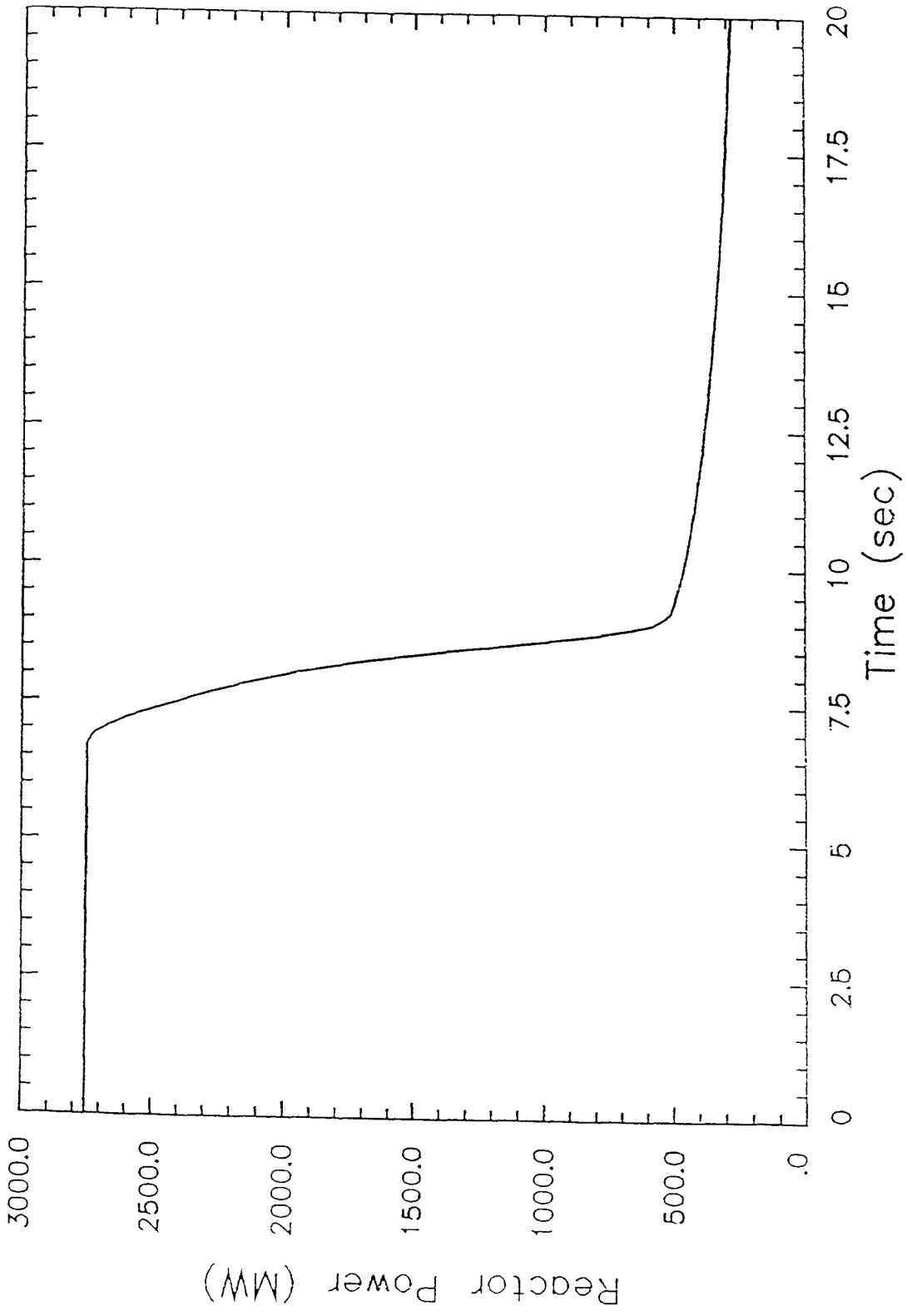


FIGURE 14.2.1–6 REACTOR POWER LEVEL FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

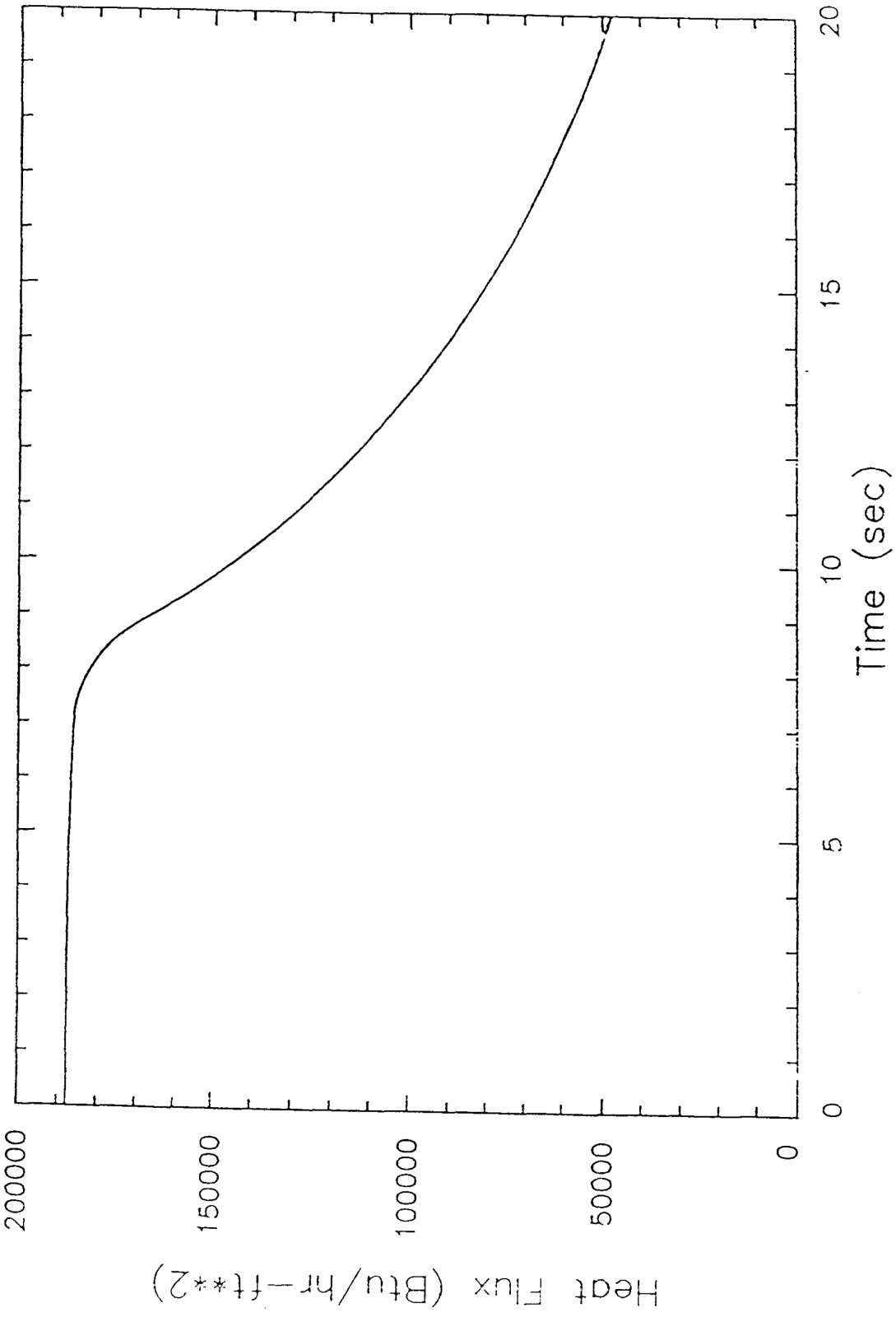


FIGURE 14.2.1–7 CORE AVERAGE HOT FLUX FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

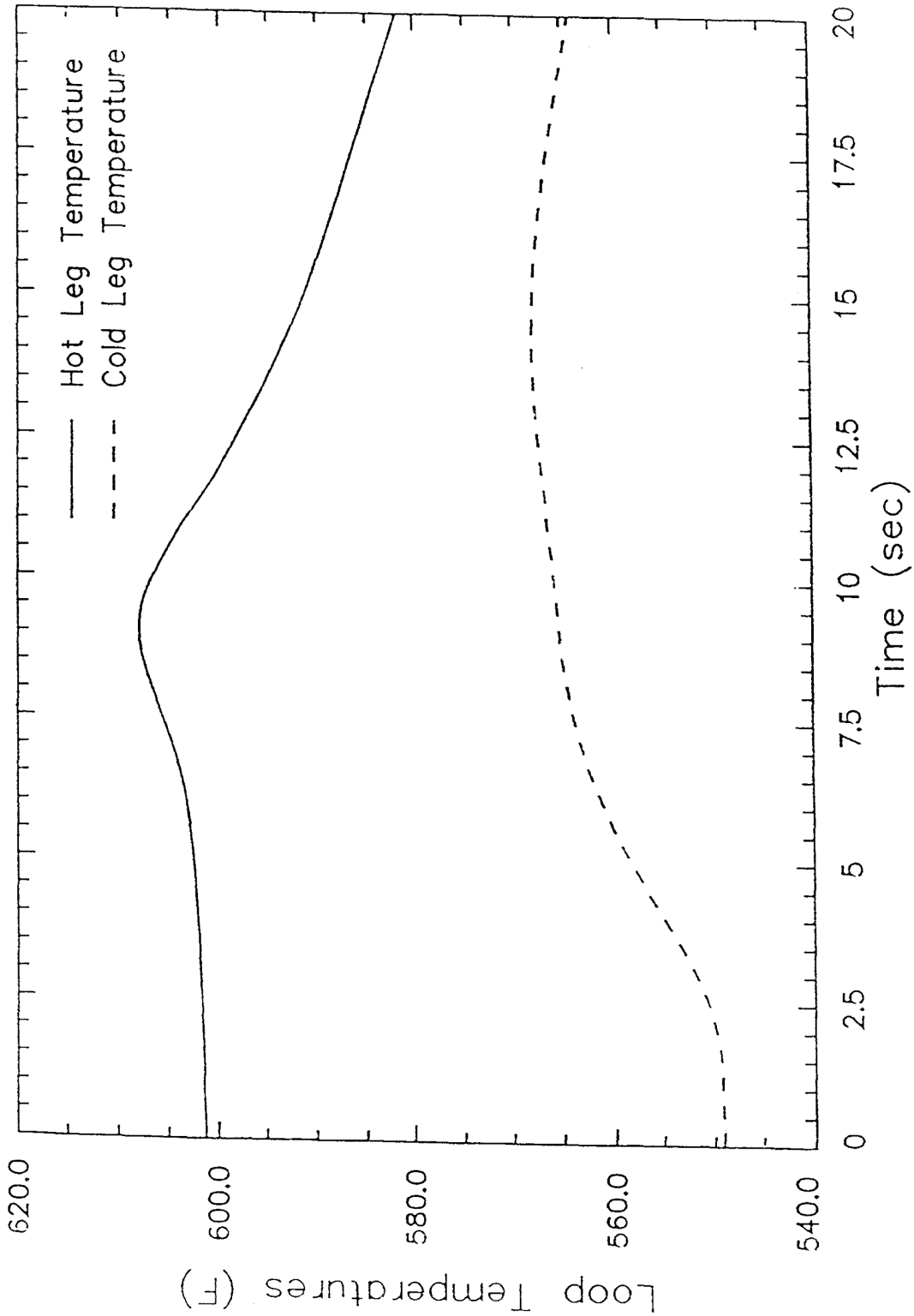


FIGURE 14.2.1–8 REACTOR COOLANT SYSTEM TEMPERATURES FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

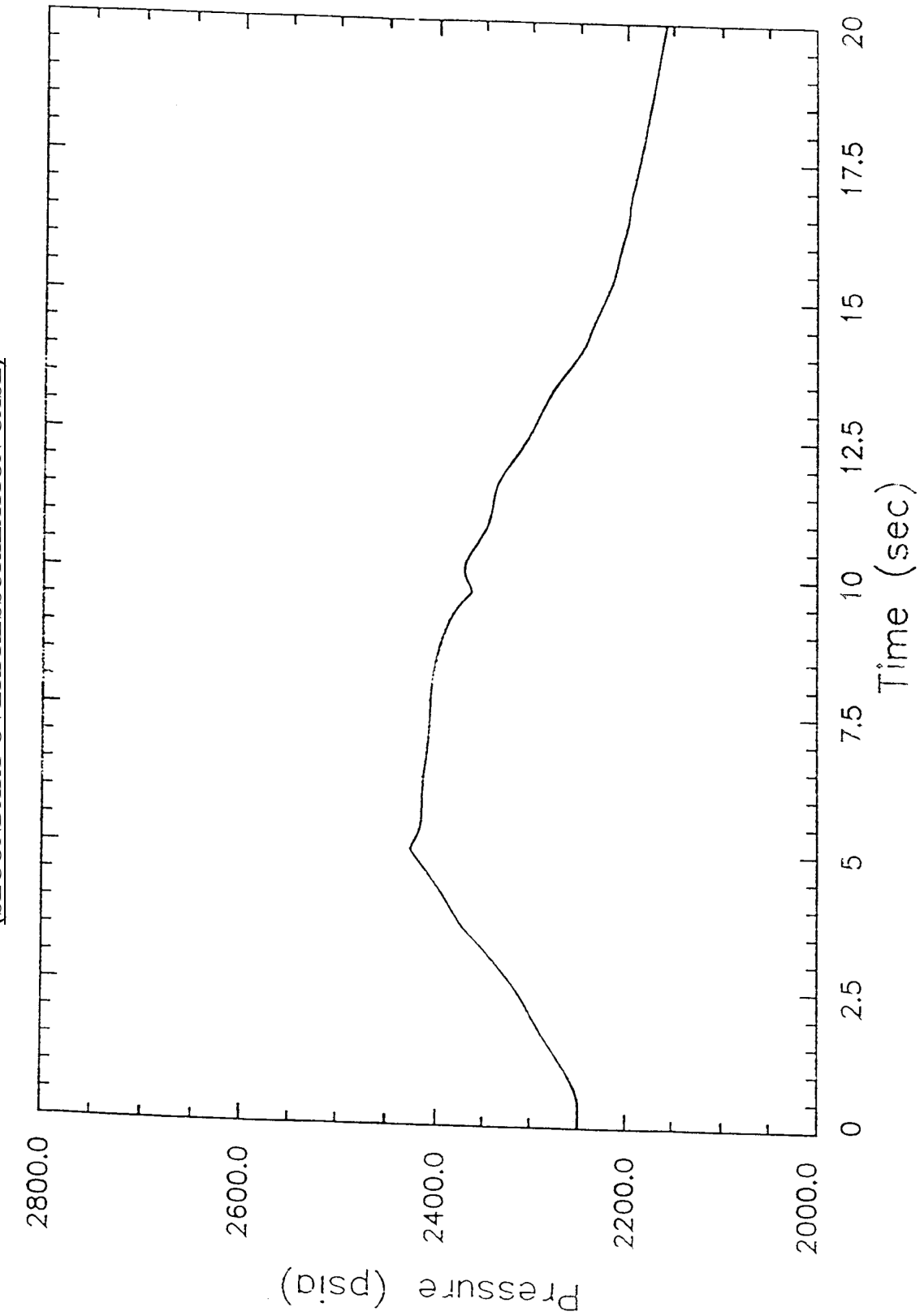


FIGURE 14.2.1–9 PRESSURIZER PRESSURE FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

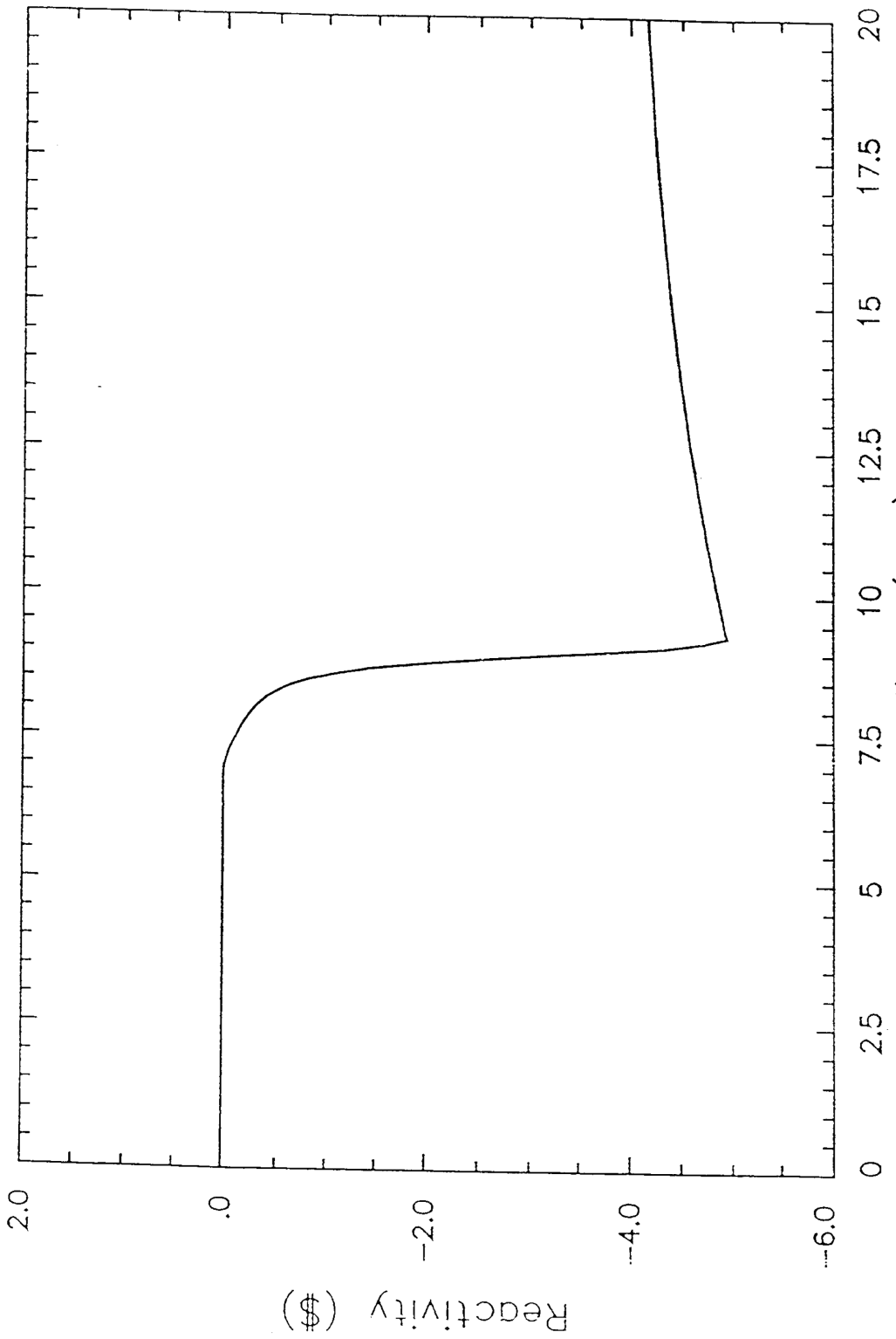


FIGURE 14.2.1–10 TOTAL REACTIVITY FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

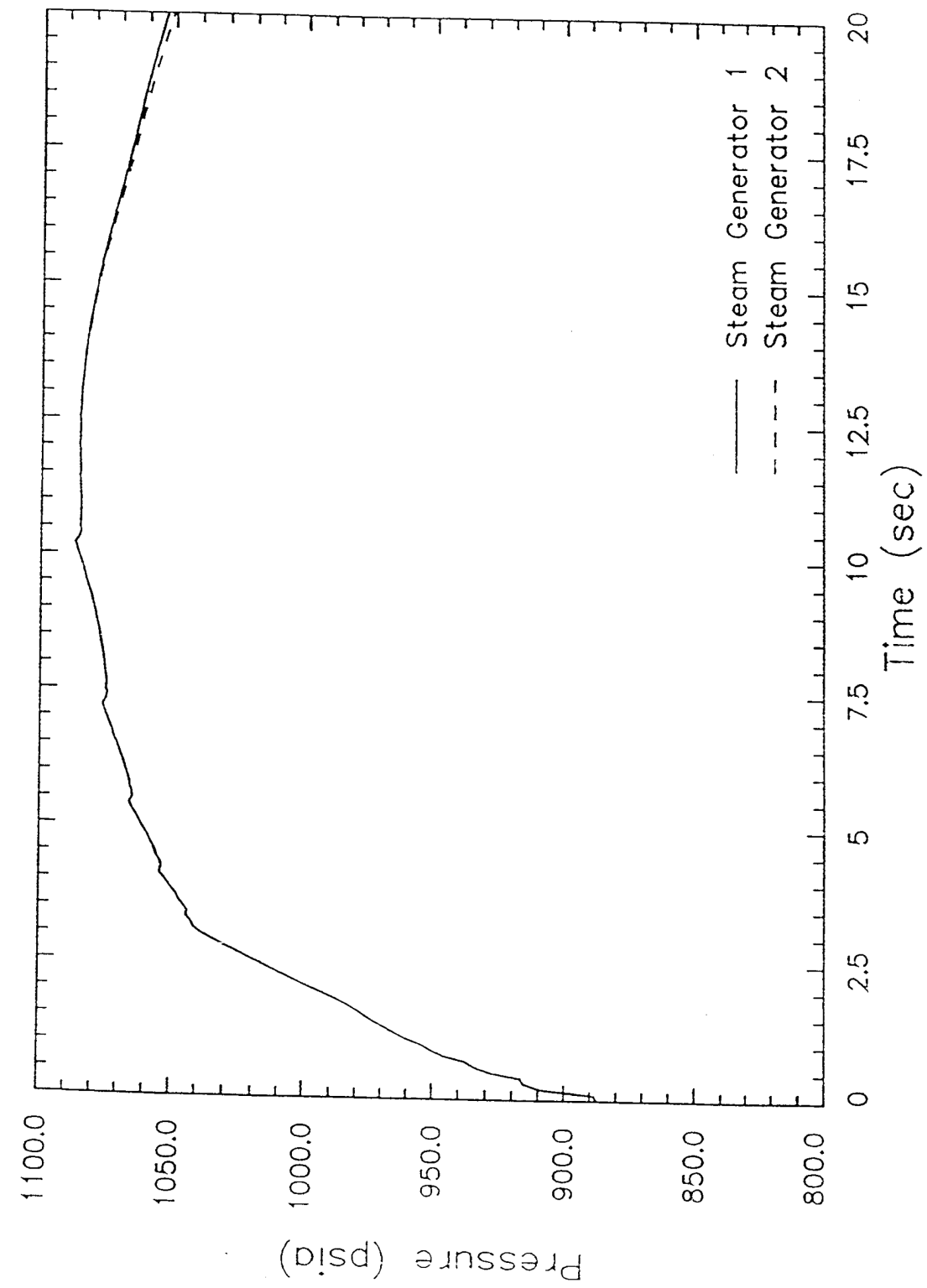


FIGURE 14.2.1–11 MAXIMUM SECONDARY SYSTEM PRESSURES FOR LOSS OF EXTERNAL LOAD (SECONDARY OVERPRESSURIZATION CASE)

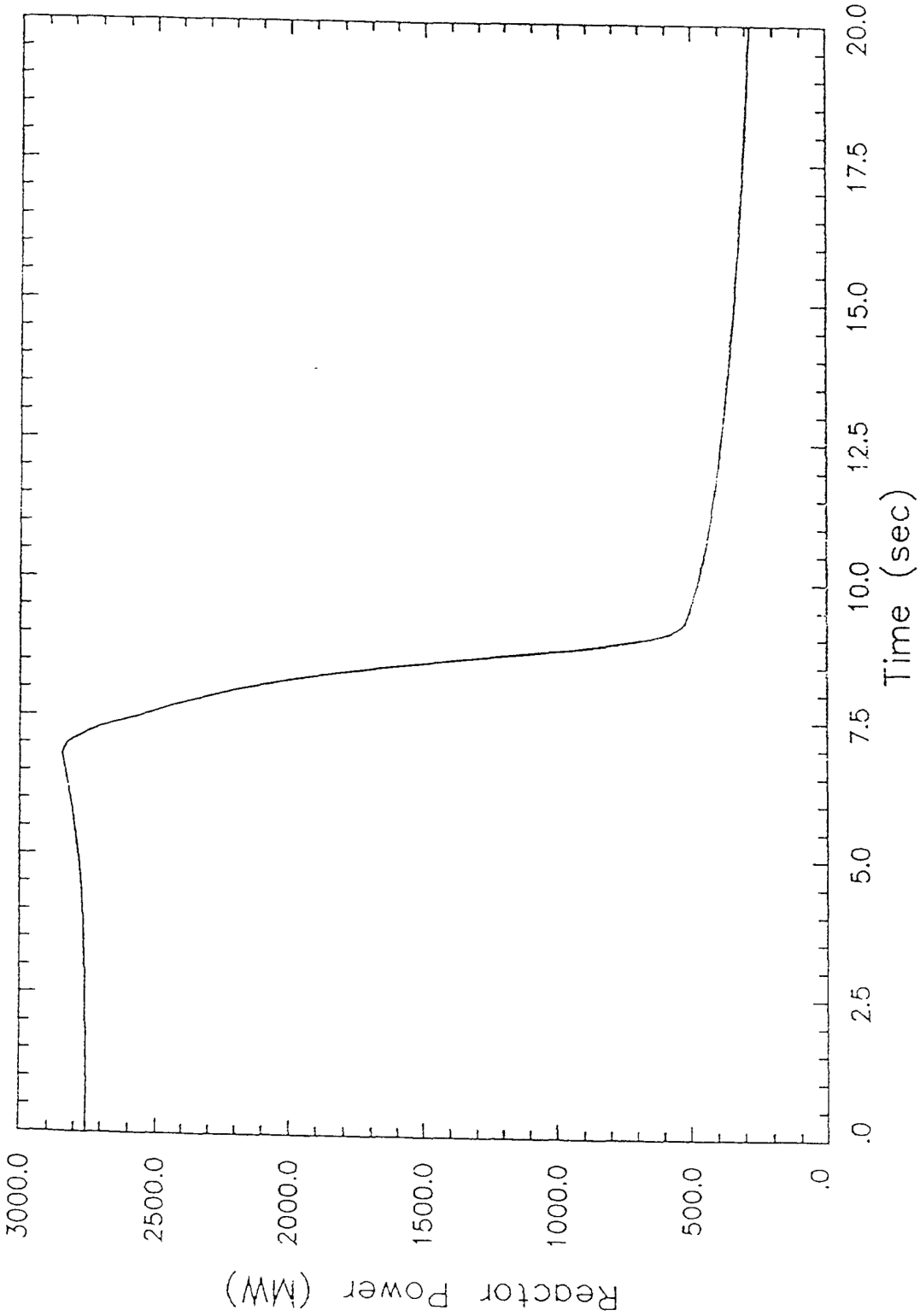


FIGURE 14.2.1–12 REACTOR POWER LEVEL FOR LOSS OF EXTERNAL LOAD (MDNBR CASE)

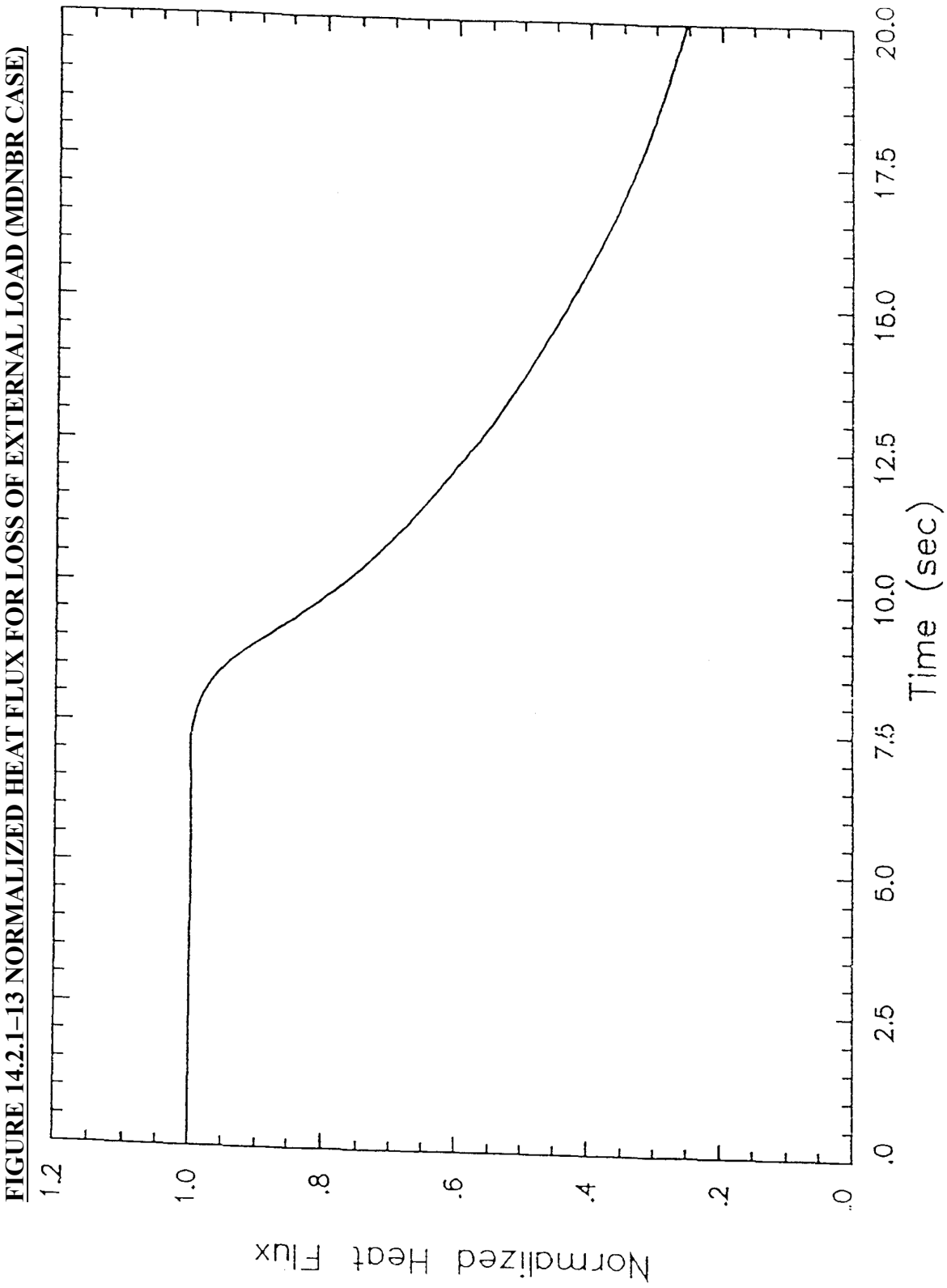


FIGURE 14.2.1–13 NORMALIZED HEAT FLUX FOR LOSS OF EXTERNAL LOAD (MDNBR CASE)

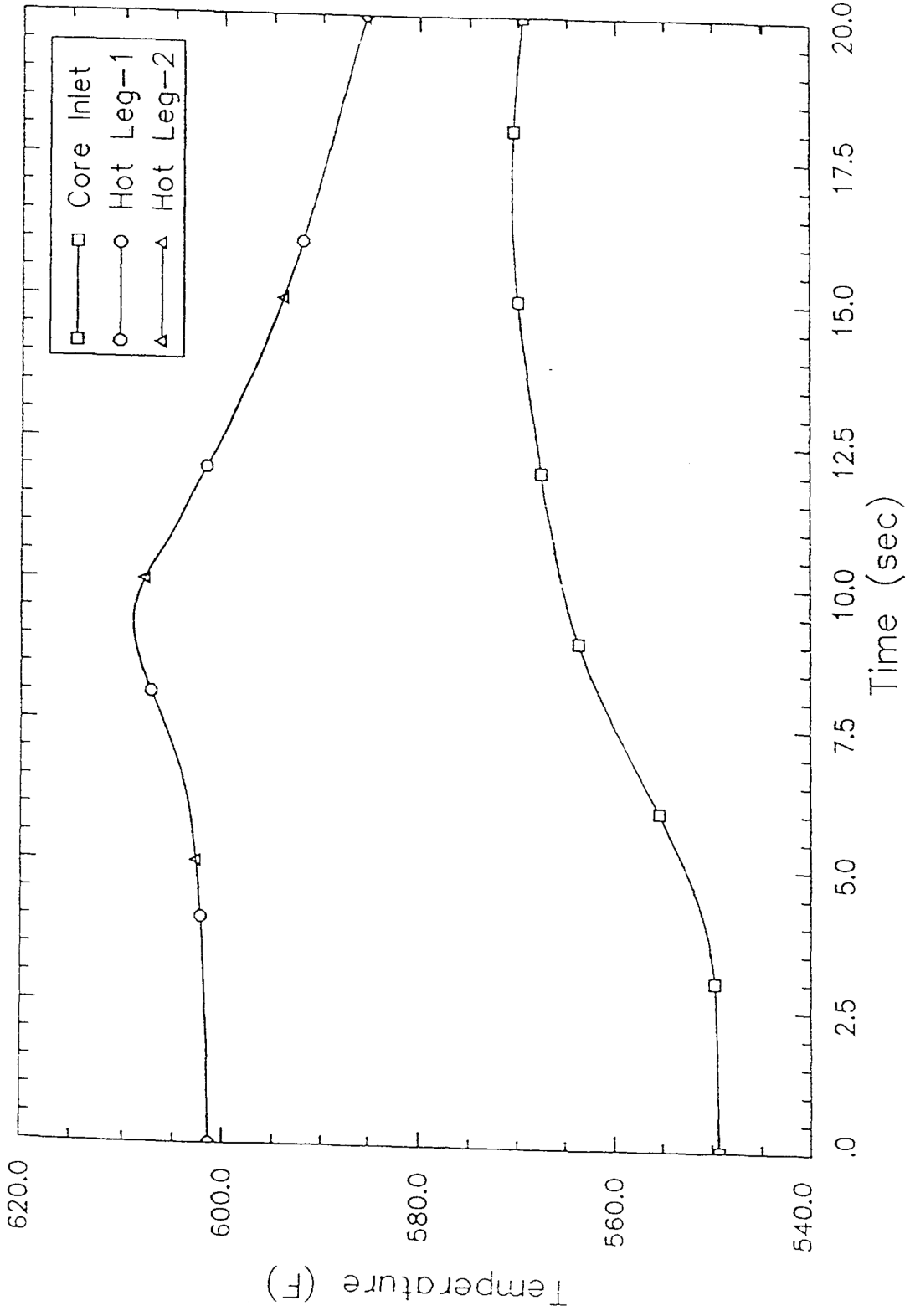


FIGURE 14.2.1–14 REACTOR COOLANT SYSTEM TEMPERATURE FOR LOSS OF EXTERNAL LOAD (MDNBR CASE) [MPS-2 FSAR]

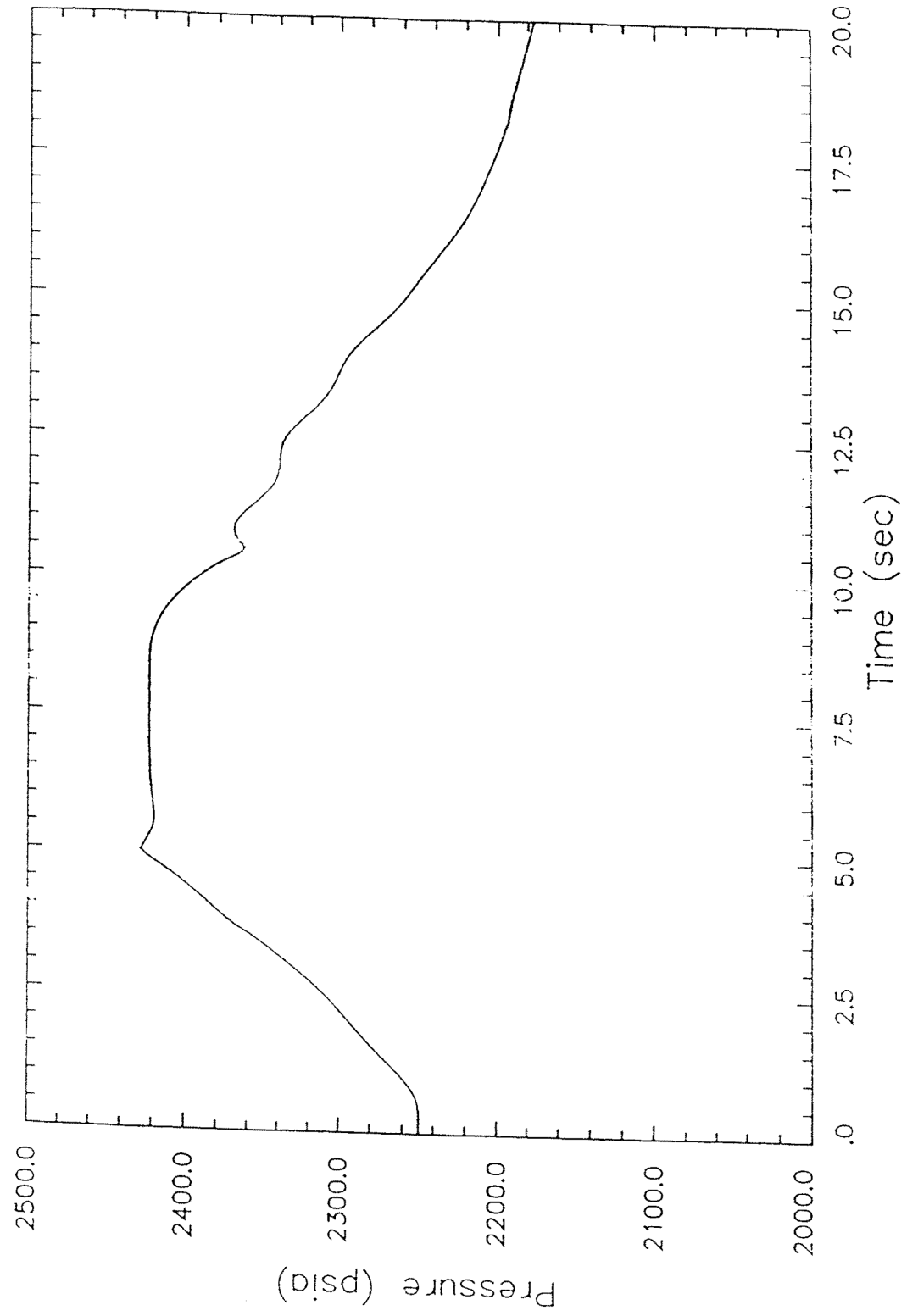


FIGURE 14.2.1–15 PRESSURIZER PRESSURE FOR LOSS OF EXTERNAL LOAD (MDNBR CASE)

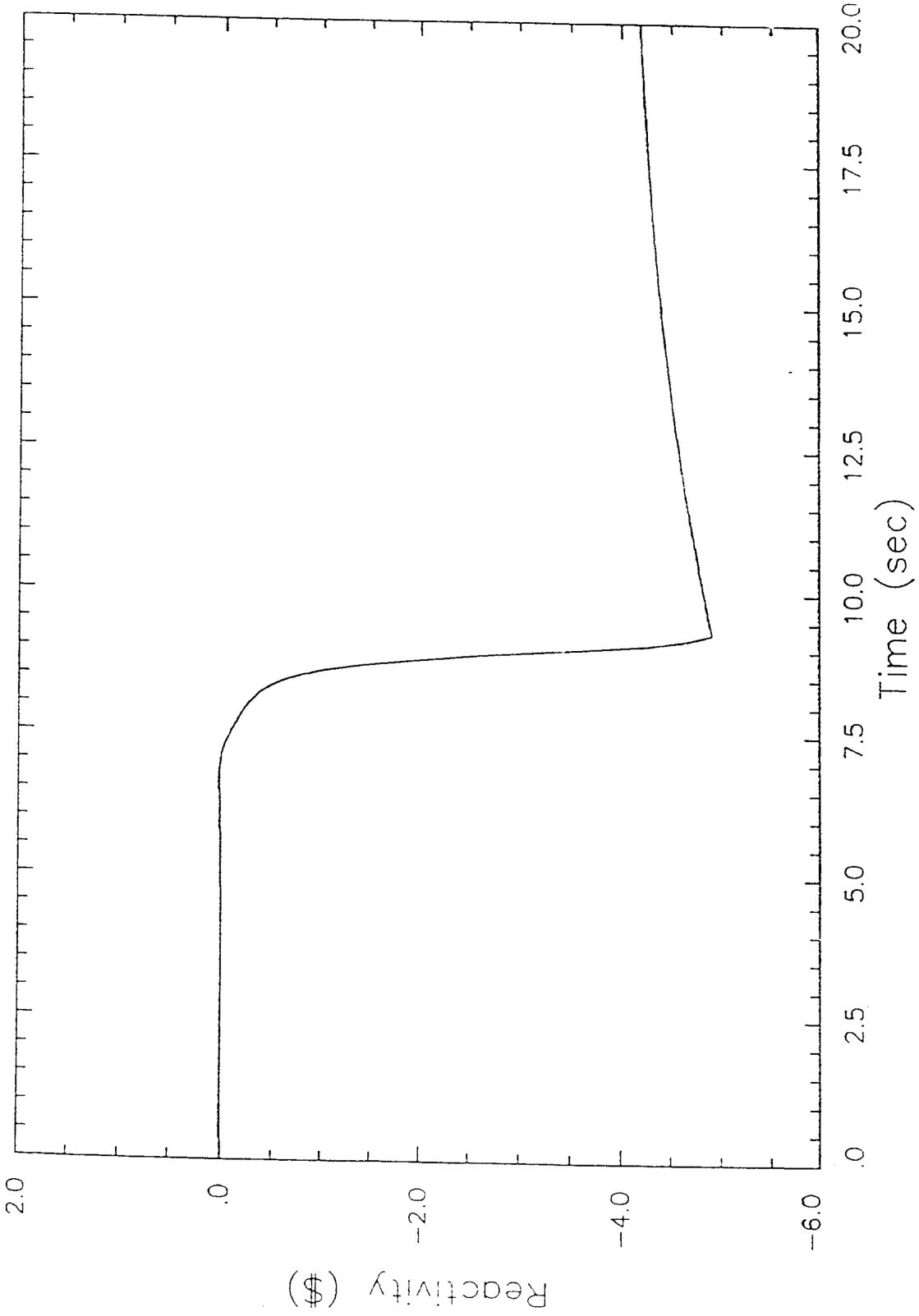


FIGURE 14.2.1–16 TOTAL REACTIVITY FOR LOSS OF EXTERNAL LOAD (MDNBR CASE)

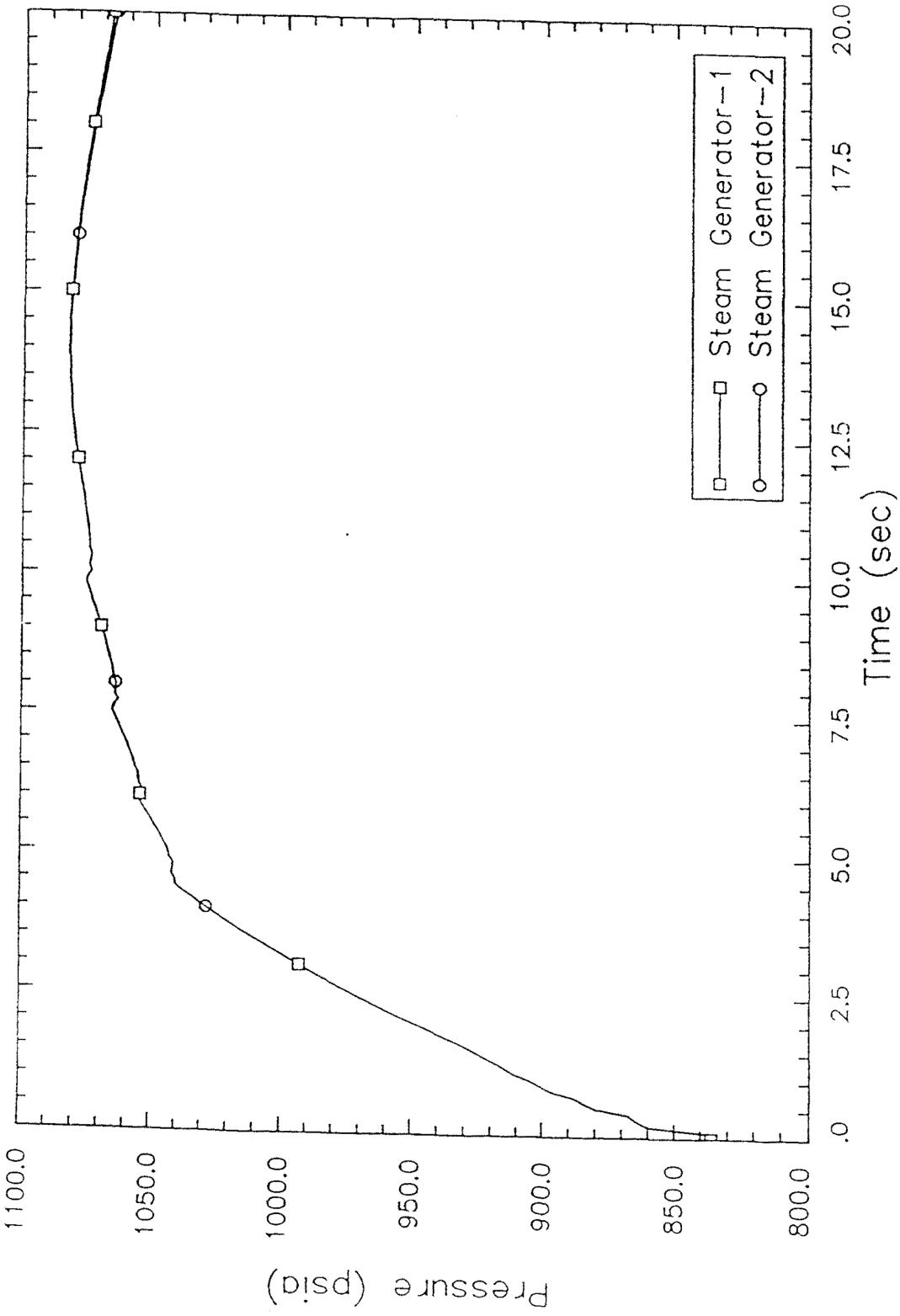


FIGURE 14.2.1–17 MAXIMUM SECONDARY SYSTEM PRESSURE FOR LOSS OF EXTERNAL LOAD (MDNBR CASE)

TABLE 14.2.2-1 AVAILABLE REACTOR PROTECTION FOR THE TURBINE TRIP EVENT

Reactor Operational Mode	Reactor Protection
1	High Pressurizer Pressure Trip Nonsafety Grade Reactor Trip on Turbine Trip Variable Overpower Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip
2	High Pressurizer Pressure Trip Variable Overpower Trip Low Steam Generator Water Level Trip
3-6	No Analysis Required; Not a Credible Event

TABLE 14.2.2-2 DISPOSITION OF EVENTS FOR THE TURBINE TRIP EVENT

Reactor Operational Mode	Disposition
1	Bounded by Event 14.2.1 for the rated power operating condition (number 1).
2	Same as above.
3-6	No Analysis Required; Not a Credible Event.

TABLE 14.2.4-1 AVAILABLE REACTOR PROTECTION FOR THE CLOSURE OF THE MAIN STEAM ISOLATION VALVES EVENTS

Reactor Operational Mode	Reactor Protection
1	High Pressurizer Pressure Trip Variable Overpower Trip Thermal Margin/Low Pressure Trip Low Steam Generator Water Level Trip Low Steam Generator Pressure
2	High Pressurizer Pressure Trip Variable Overpower Trip Low Steam Generator Water Level Trip Low Steam Generator Pressure
3	Variable Overpower Trip
4-6	No Analysis Required

TABLE 14.2.4-2 DISPOSITION OF EVENTS FOR THE CLOSURE OF THE MAIN STEAM ISOLATION VALVES EVENTS

Reactor Operational Mode	Disposition
1	Dual MSIV closure; Bounded by Event 14.2.1 Single MSIV Closure; Analyze
2-6	Bounded by Mode 1

TABLE 14.2.4-3 EVENT SUMMARY FOR THE MAIN STEAM ISOLATION VALVE CLOSURE EVENT (LOWER STEAM FLOW CASE)

Event	Time (seconds)
Reactor at full power	0.0
One MSIV closes instantaneously, flow though other MSIV increases	0.0
First MSSV opens	4.1
Peak pressurizer pressure	12.5
Peak Steam Generator secondary pressure	12.5
Last MSSV opens	12.6
Steam Generator pressure reaches low-pressure trip setpoint	23.8
Reactor scrams (beginning of CEA insertion)	25.2
Peak reactor power	25.3
End of calculation	40.0

FIGURE 14.2.4–1 REACTOR POWER LEVEL FOR MSIV CLOSURE (LOWER STEAM FLOW CASE)

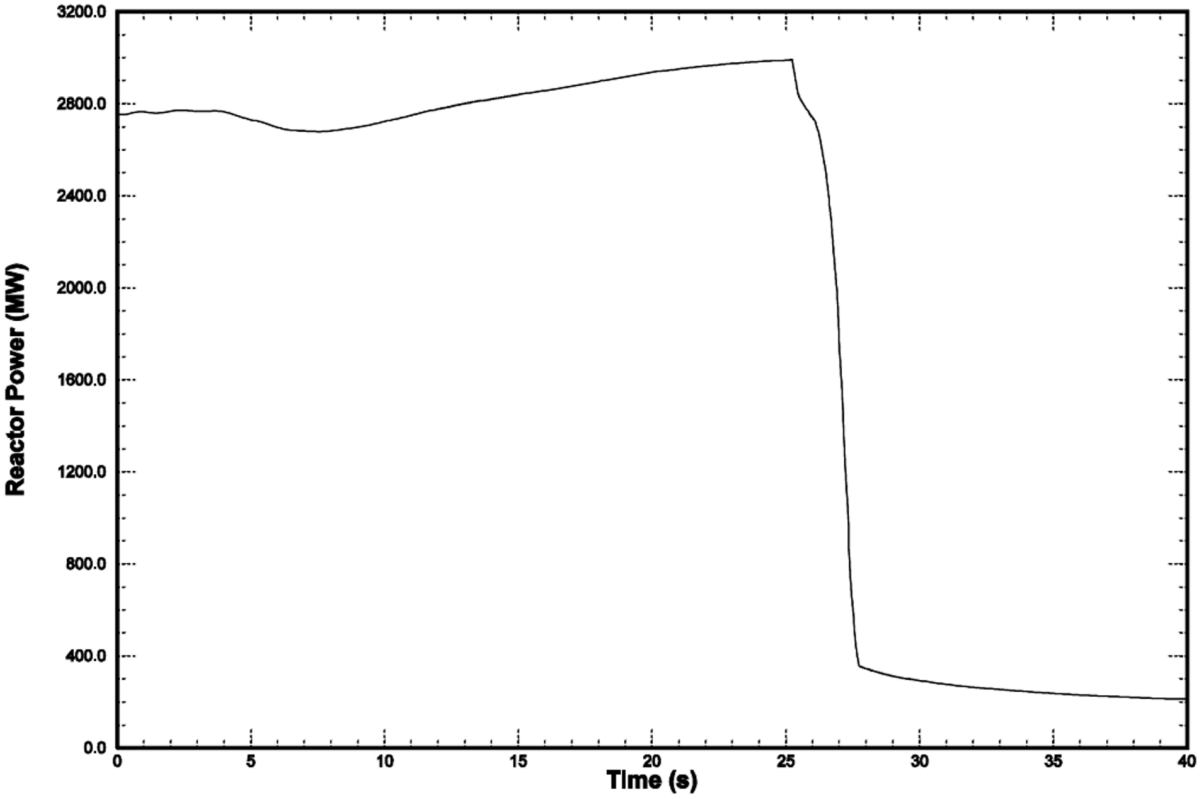


FIGURE 14.2.4–2 REACTOR COOLANT SYSTEM TEMPERATURES FOR MSIV CLOSURE (LOWER STEAM FLOW CASE)

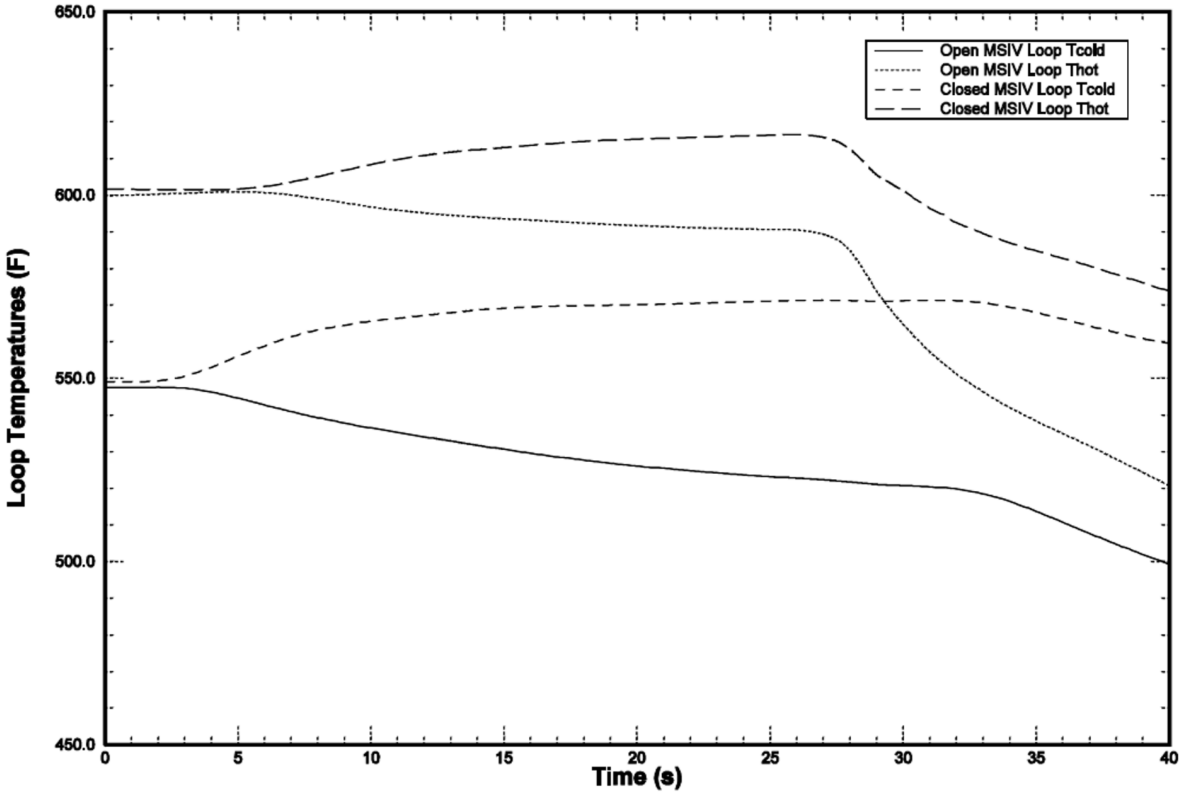


FIGURE 14.2.4–3 PRESSURIZER PRESSURE FOR MSIV CLOSURE (LOWER STEAM FLOW CASE)

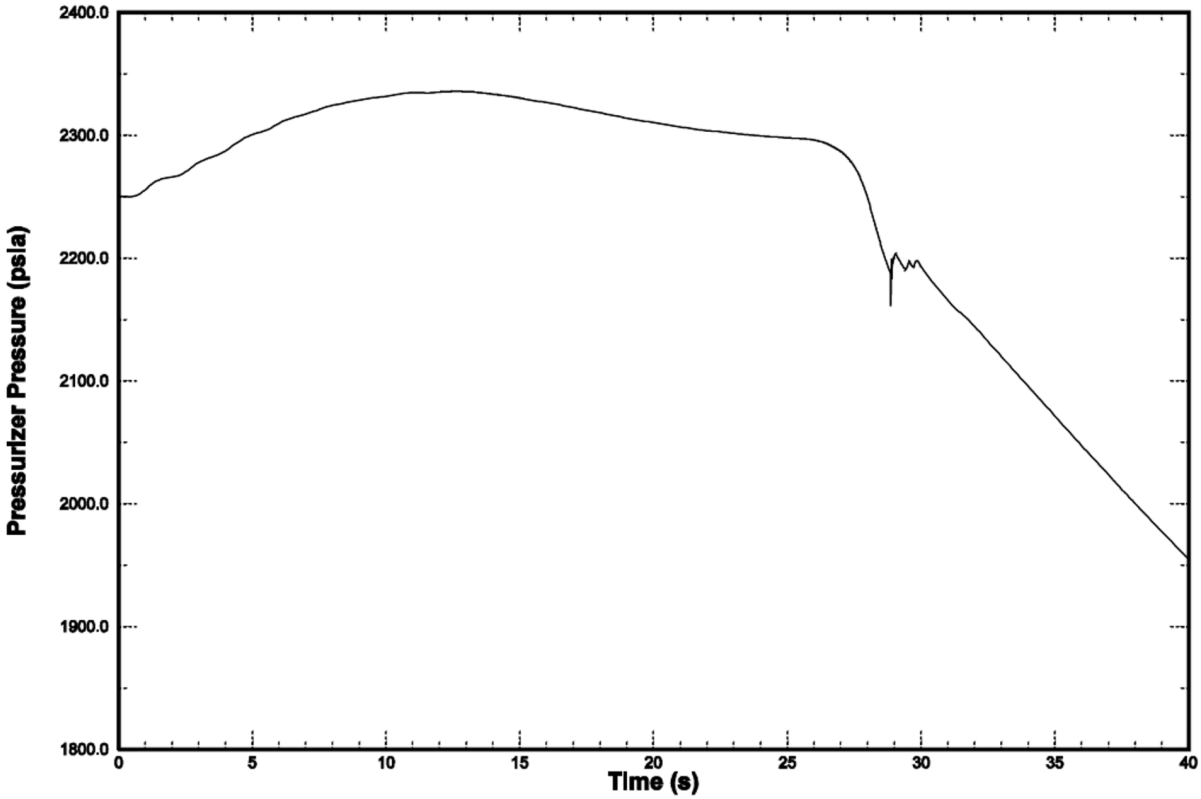


FIGURE 14.2.4–4 ISOLATED STEAM GENERATOR PRESSURE AT BOTTOM OF BOILER REGION FOR MSIV CLOSURE (LOWER STEAM FLOW CASE))

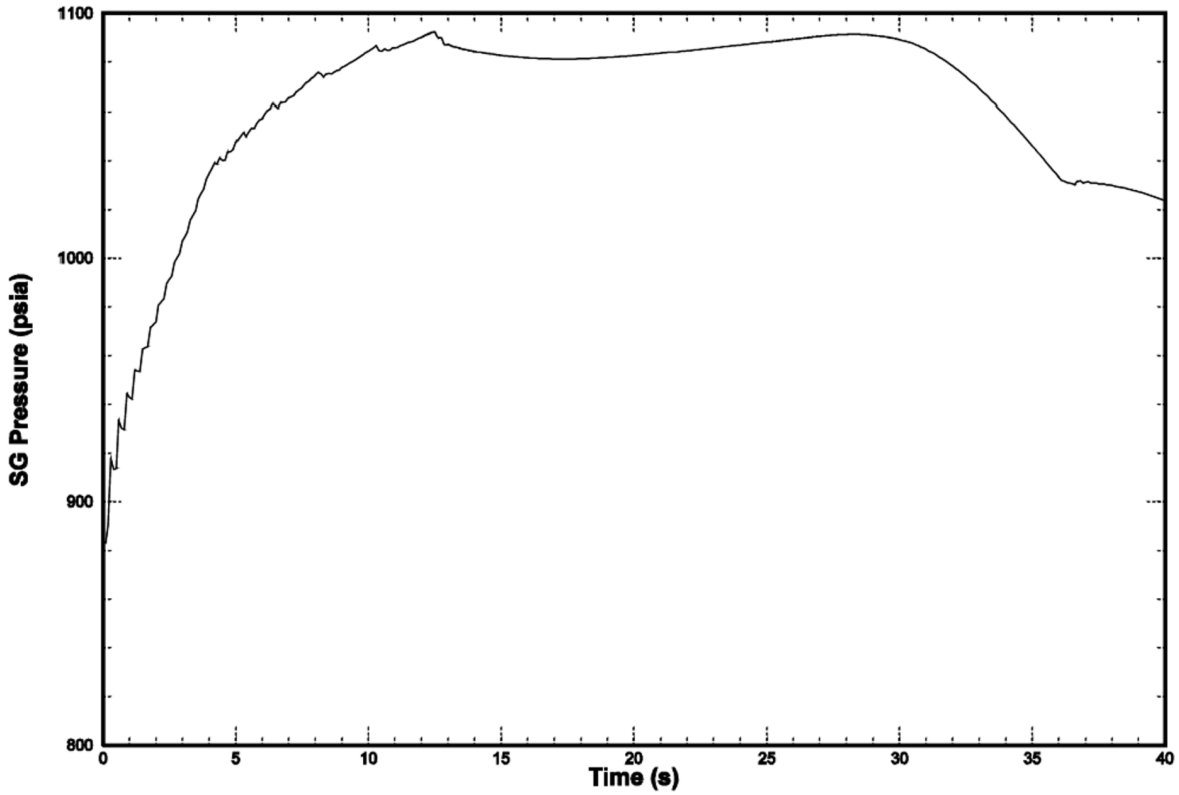


FIGURE 14.2.4–5 OPEN MSIV STEAM GENERATOR STEAM DOME PRESSURE FOR MSIV CLOSURE (LOWER STEAM FLOW CASE))

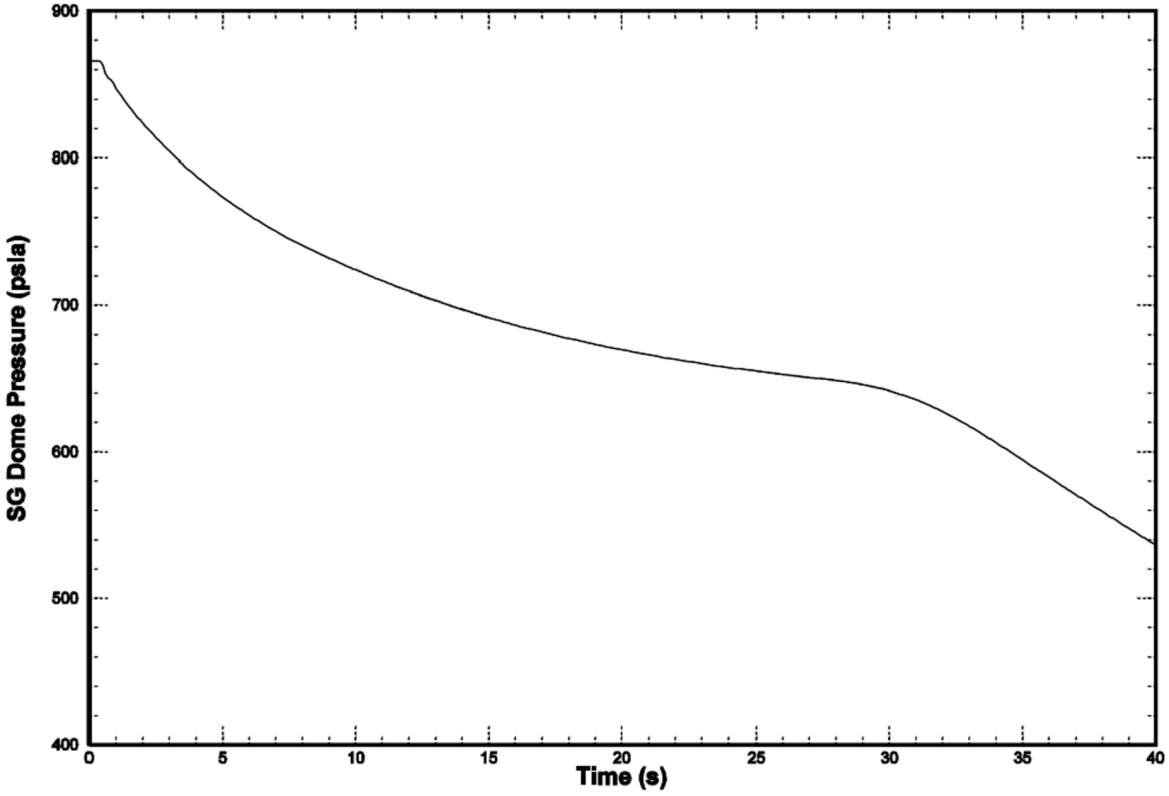


TABLE 14.2.7-1 AVAILABLE REACTOR PROTECTION FOR THE LOSS OF NORMAL FEEDWATER FLOW EVENT

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Water Level Trip High Pressurizer Pressure Trip Thermal Margin/Low Pressure Trip Variable Overpower Trip
2	High Pressurizer Pressure Trip Variable Overpower Trip Low Steam Generator Water Level Trip
3	Variable Overpower Trip
4-6	No Analysis Required; Not a Credible Event

TABLE 14.2.7-2 DISPOSITION OF EVENTS FOR THE LOSS OF NORMAL FEEDWATER FLOW EVENT

Reactor Operating Conditions	Disposition
1	Analyze to assess maximum pressurizer level swell and long term adequacy of AFW. Pressurization and DNB aspects bounded by Event 14.2.1.
2, 3	Bounded by the above, no analysis required.
4-6	No Analysis Required; Not a Credible Event

TABLE 14.2.7-3 SEQUENCE OF EVENTS FOR MINIMUM STEAM GENERATOR INVENTORY CASE: ONE MOTOR-DRIVEN AFW PUMP FAILS TO START WITH OFFSITE POWER AND STEAM DUMPS

Time (seconds)	Event
0	Total loss of main feedwater
20	Pressurizer spray actuates
27.9	Reactor trip signal on low steam generator water level
28.8	Control rods begin to drop
29.9	Main turbine trip
32	Maximum pressurizer level, 73%
38	Peak steam generator pressure (993 psia)
48	AFW actuation signal on low-low Steam Generator water level
53	Charging flow initiated in response to pressurizer level program
58	Steam generator blowdown isolated
288	Train “A” motor-driven AFW pump starts
628	Steam-driven AFW pump starts
654	Minimum Steam generator liquid inventory occurs
1800	End of calculation

TABLE 14.2.7-4 SEQUENCE OF EVENTS FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START

Time (seconds)	Event
0	Total loss of main feedwater
34.1	Reactor trip signal on low steam generator water level
35.1	Control rods begin to drop; RCPs tripped
36.1	Main turbine trip
39	Pressurizer PORV cycles open/closed
41	AFW actuation signal on low-low steam generator water level signal
43	Maximum pressurizer level, 76%
46	Peak steam generator pressure (1055 psia)
51	steam generator blowdown isolated
73	Charging flow on (pressurizer level below program)
170	Charging flow off (pressurizer level at program setpoint)
281	Train “A” motor-driven AFW pump starts
635	Steam-driven AFW pump starts
730	Minimum steam generator liquid inventory occurs
757	Maximum post-trip RCS average temperature (571°F)
2400	End of calculation

FIGURE 14.2.7–1 REACTOR COOLANT SYSTEM LOOP TEMPERATURES FOR MINIMUM STEAM GENERATOR INVENTORY CASE: OFFSITE POWER AVAILABLE, "B" MOTOR-DRIVEN AFW PUMP FAILS TO START

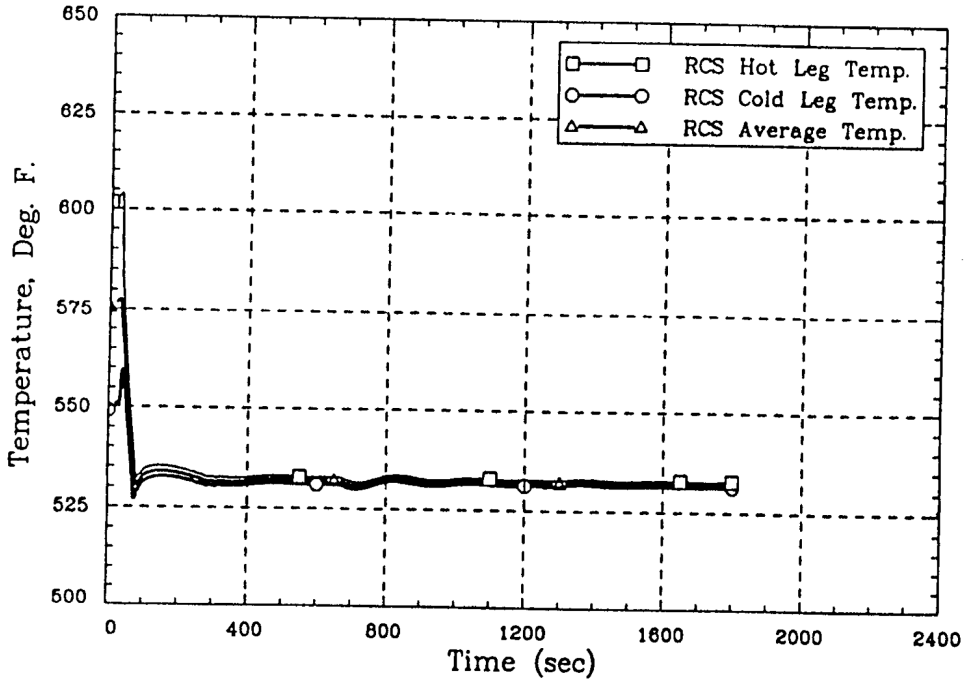


FIGURE 14.2.7–2 STEAM GENERATOR DOME PRESSURE FOR MINIMUM STEAM GENERATOR INVENTORY CASE: OFFSITE POWER AVAILABLE, "B" MOTOR-DRIVEN AFW PUMP FAILS TO START

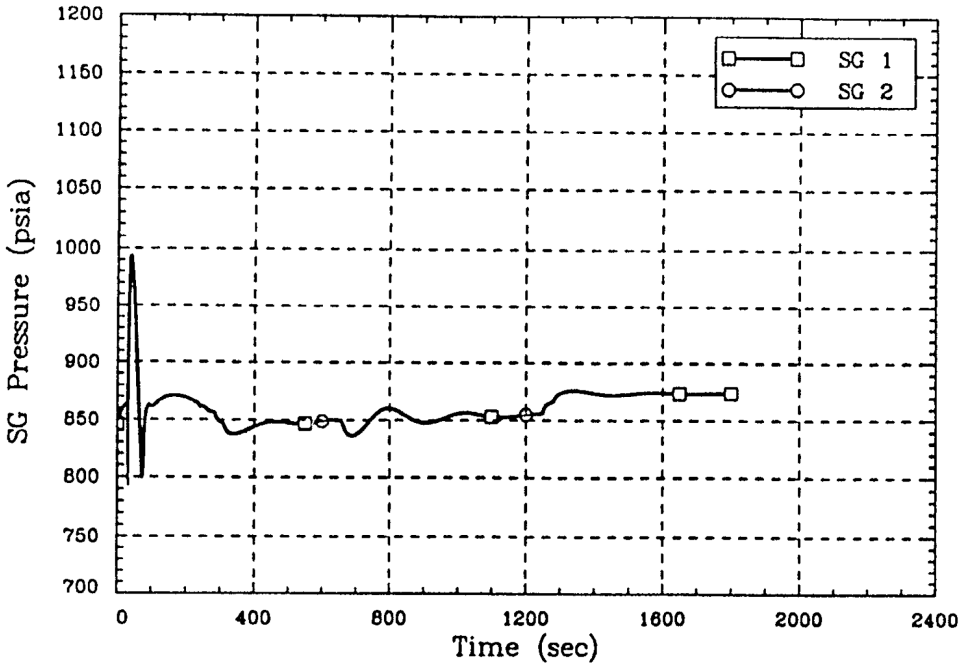


FIGURE 14.2.7–3 PRESSURIZER LEVEL FOR MINIMUM STEAM GENERATOR INVENTORY CASE: OFFSITE POWER AVAILABLE, "B" MOTOR-DRIVEN AFW PUMP FAILS TO START

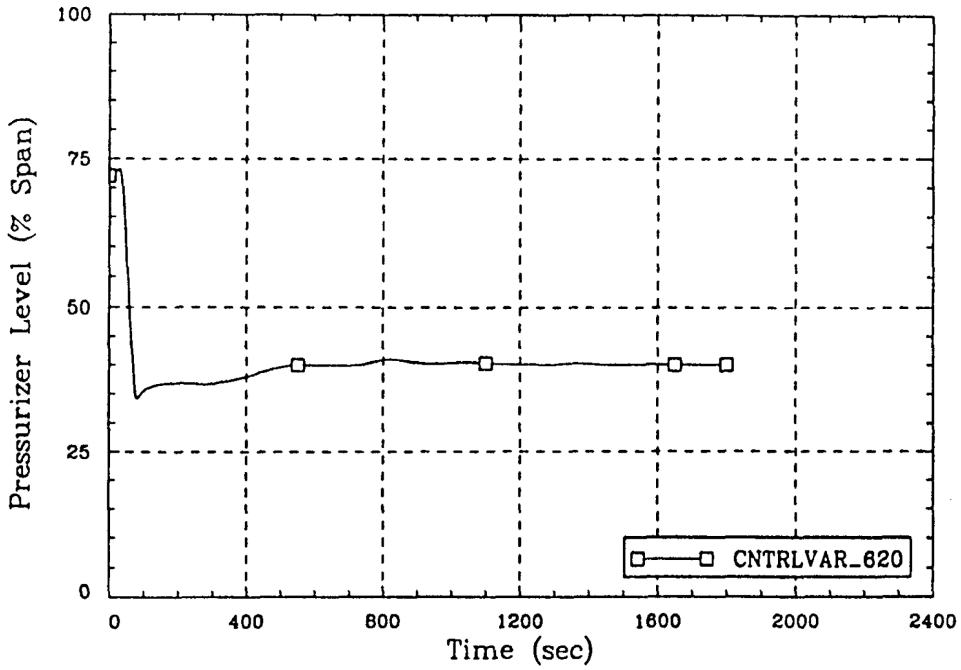


FIGURE 14.2.7–4 STEAM GENERATOR FOR LIQUID MASS INVENTORY FOR MINIMUM STEAM GENERATOR INVENTORY CASE: OFFSITE TO POWER AVAILABLE, "B" MOTOR-DRIVEN AFW PUMP FAILS TO START

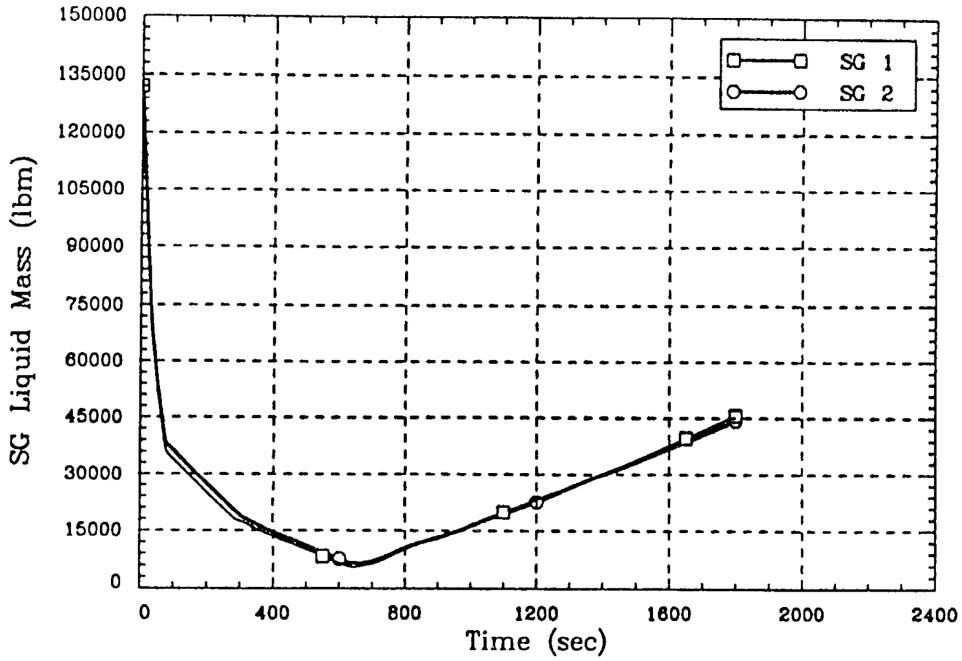


FIGURE 14.2.7–5 STEAM GENERATOR COLLAPSED LIQUID LEVEL FOR MINIMUM STEAM GENERATOR INVENTORY CASE: OFFSITE TO POWER AVAILABLE, "B" MOTOR-DRIVEN AFW PUMP FAILS TO START

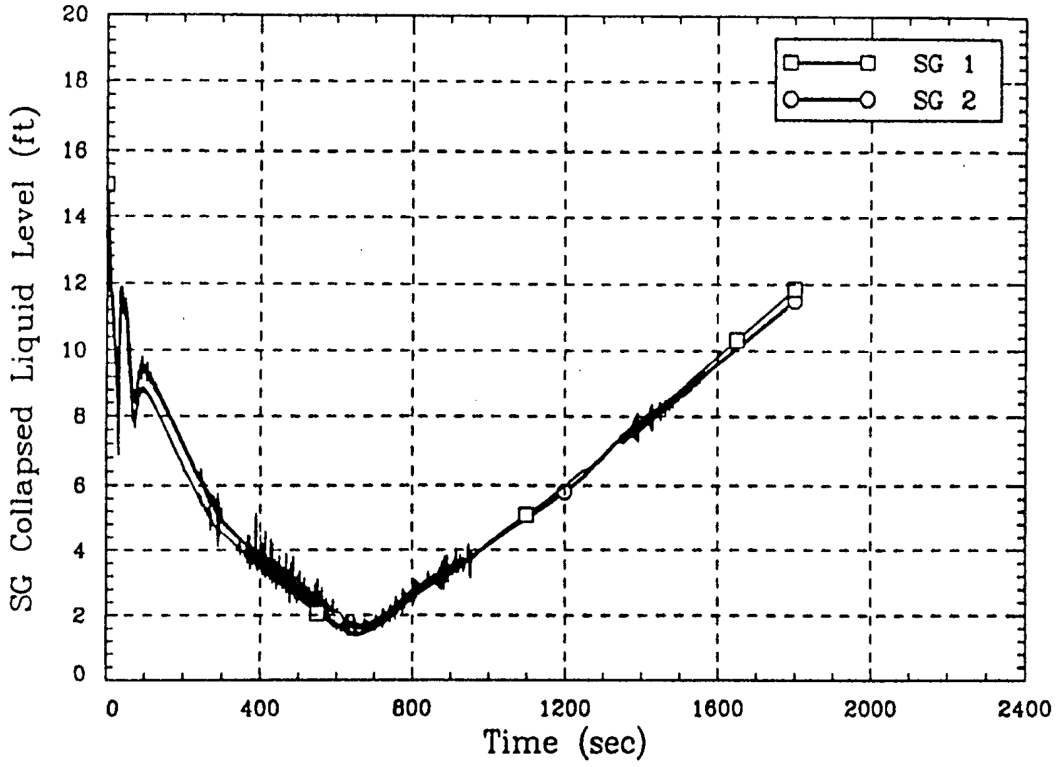


FIGURE 14.2.7–6 REACTOR COOLANT SYSTEM LOOP TEMPERATURES FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START

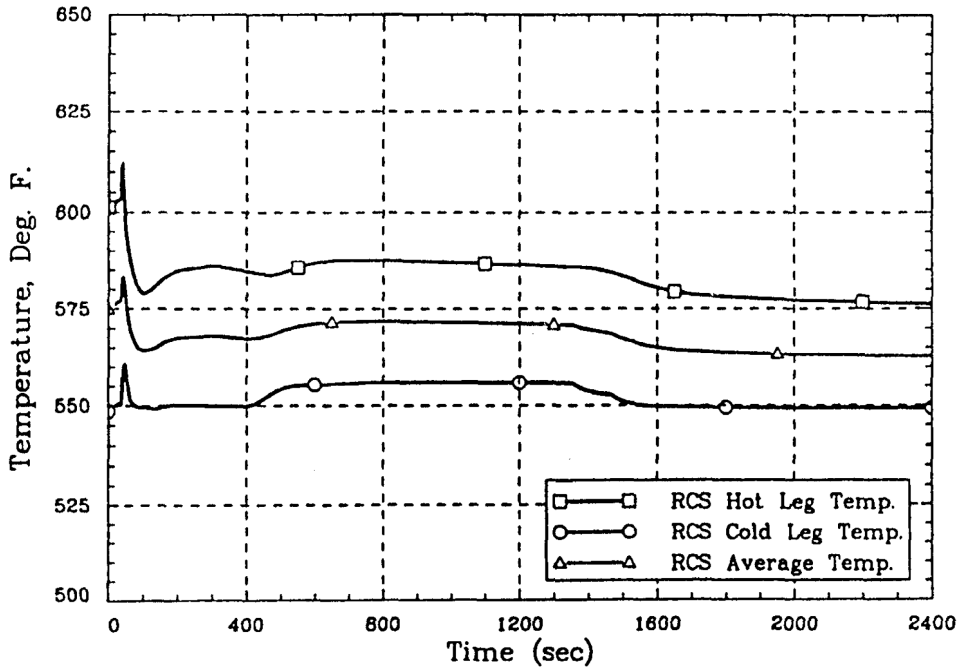


FIGURE 14.2.7–7 STEAM GENERATOR DOME PRESSURE FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START

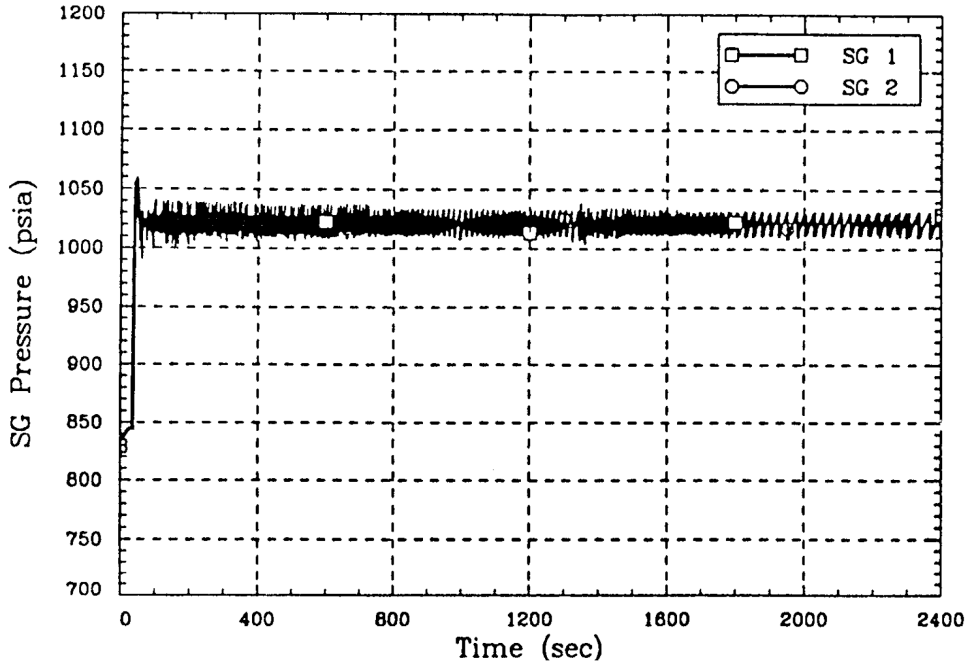


FIGURE 14.2.7–8 PRESSURIZER LEVEL FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START

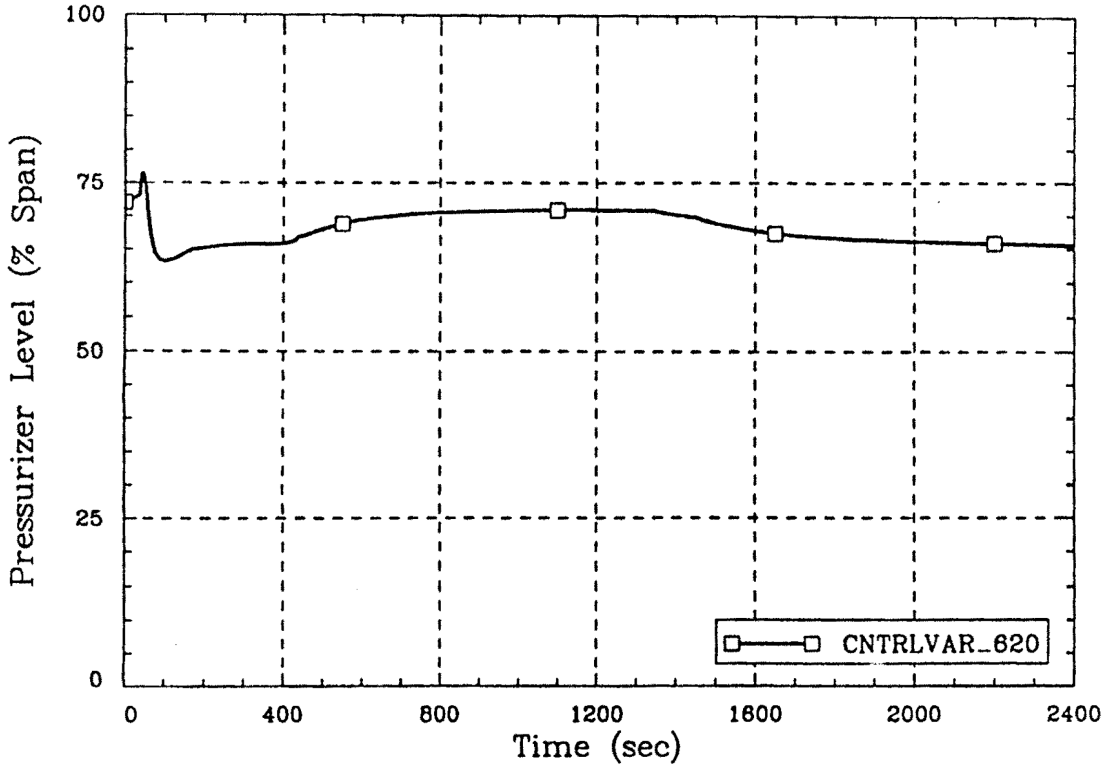


FIGURE 14.2.7–9 STEAM GENERATOR LIQUID MASS INVENTORY FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START

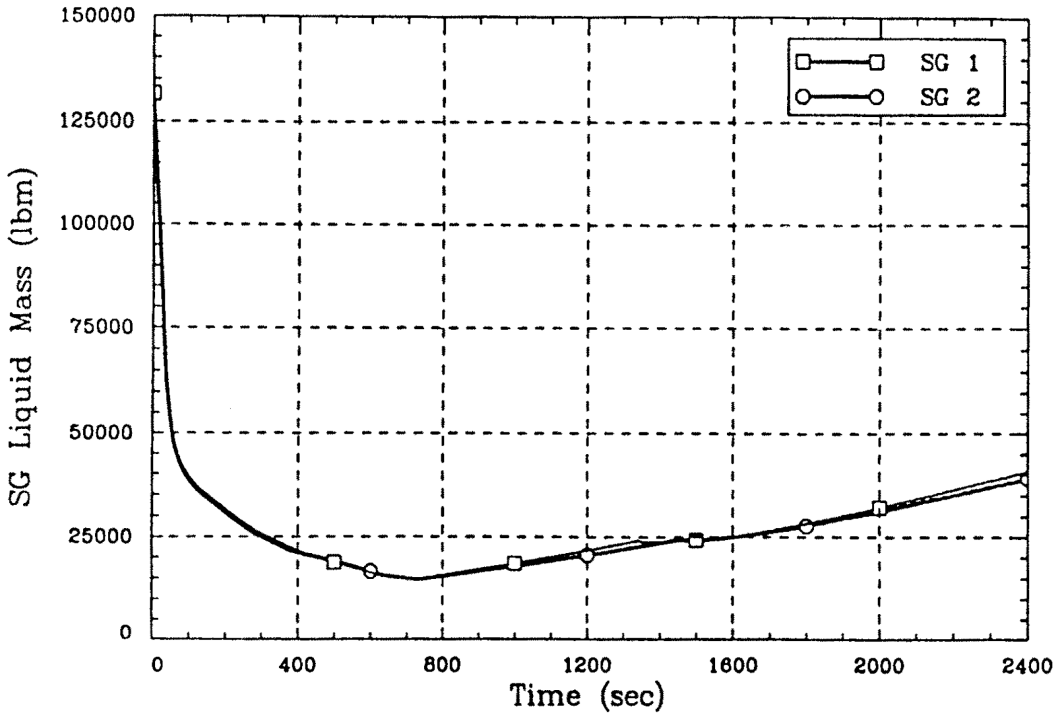
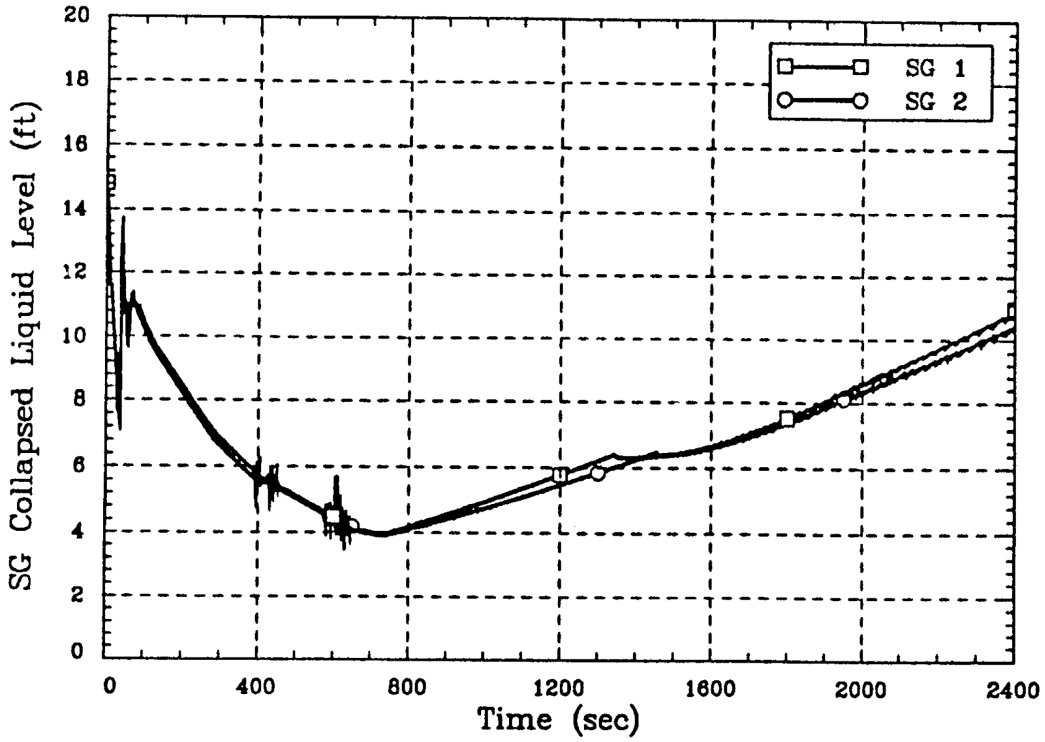


FIGURE 14.2.7–10 STEAM GENERATOR COLLAPSED LIQUID LEVEL FOR MAXIMUM PRESSURIZER LEVEL CASE: LOSS OF OFFSITE POWER, ONE MOTOR-DRIVEN AFW PUMP FAILS TO START



14.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW

14.3.1 LOSS OF FORCED REACTOR COOLANT FLOW

14.3.1.1 Event Initiator

The loss of forced reactor coolant flow in the primary system may result from a mechanical or electrical failure in a main reactor coolant pump (RCP) or in the power supply to these pumps. Forced coolant flow may be completely or partially lost. The limiting event initiator is that which results in the trip of all four RCPs.

14.3.1.2 Event Description

The Loss of Forced Reactor Coolant Flow transient is initiated by a loss of the electrical power supplied to or a mechanical failure in a reactor coolant system (RCS) pump. These failures may result in a complete or partial loss of forced coolant flow. The immediate result of the loss of forced coolant flow is an increase in the coolant temperature as it flows through the reactor core. Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to Departure From Nucleate Boiling (DNB) limits.

14.3.1.3 Reactor Protection

Reactor protection is provided by the following reactor trips:

1. Low reactor coolant flow;

2. Thermal margin/low pressure (TM/LP); and

3. High pressurizer pressure trip.

Reactor protection for the Loss of Forced Reactor Coolant Flow event is summarized in Table 14.3.1-1.

14.3.1.4 Disposition and Justification

The power sources for the main RCPs are the most likely initiator for a loss of flow event involving more than one pump. A mechanical or electrical fault in one of the pumps will only result in a single pump loss of forced coolant flow transient. The normal power supplies for the pumps are from two buses which receive power from the main generator. Two pumps, in opposite loops, are powered from each bus. If there is a generator trip, the pumps are automatically transferred to a bus supplied from the external power lines. A generator trip with the failure of this transfer could result in a loss of power to all four pumps.

In the case of four pump operation, two situations must be considered: two pump coastdown and a total loss of forced coolant flow. Considering first the total loss of flow cases, the consequences of this postulated event are bounded by rated power operation.

The rated power case is bounding because of the reduced DNB margin for this initial state combined with the highest power to flow ratio during coastdown.

For the two pump loss of flow cases, the magnitude of the coastdown is less severe than the four pump coastdown, and the consequences of this event are bounded by the four pump loss of flow event. For the two pump flow coastdown cases, there is always some degree of forced reactor coolant flow. These events are, therefore, not as challenging as the four pump coastdown events. A comparison of the governing parameters indicates that these events are bounded by the four pump loss of flow event from full rated power conditions.

In summary, the four pump loss of flow event is the bounding event for the 14.3.1 events in all modes of operation. The only active system challenged is the reactor protection system (RPS) which is redundant and single failure proof.

The disposition of events for the Loss of Forced Reactor Coolant Flow event is summarized in Table 14.3.1-2.

14.3.1.5 Definition of Events Analyzed

This event is analyzed from full-power initial conditions. The core thermal margins are minimized at full power conditions resulting in this being the bounding mode of operation for this event. One case is analyzed for this event to assess the challenge to the DNB Specified Acceptable Fuel Design Limit (SAFDL).

The loss of coolant flow immediately results in a loss of system heat rejection capacity. This causes the primary system coolant temperature to increase. The objective of selecting input and biasing is to minimize Departure From Nucleate Boiling Ratio (DNBR). The event analysis is, therefore, biased to minimize pressure which minimizes DNBR. The steam bypass and the atmospheric dump valves are both assumed not to operate, which again most challenges the DNB SAFDL.

14.3.1.6 Analysis Results

The transient is initiated by tripping all four primary coolant pumps. As the pumps coast down, the core flow is reduced, causing a reactor scram on low flow. No credit was taken for the RCP under speed trip. As the flow coasts down, primary temperatures increase. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures.

The deterministic Minimum Departure from Nucleate Boiling Ratio (MDNBR) may violate the fuel design limit for this event. Because of this, the DNBR consequences of this event were evaluated using AREVA statistical setpoint methodology (Reference 14.3-1). The event Minimum Departure From Nucleate Boiling Ratio (MDNBR) was shown to be greater than thermal margin limits. This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated

The responses of key system variables for the deterministic case are given in Figures 14.3.1–1 to 14.3.1–7. The sequence of events is given in Table 14.3.1-3.

14.3.1.7 Conclusion

The statistical setpoint analysis demonstrates that the MDNBR limit is not penetrated by the Loss of Forced Reactor Coolant Flow event. Maximum peak pellet LHR for this event is below the FCMLHR limit.

14.3.2 FLOW CONTROLLER MALFUNCTION

There are no flow control devices on the primary RCS of Millstone Unit 2. This event is therefore not credible and need not be analyzed.

14.3.3 REACTOR COOLANT PUMP ROTOR SEIZURE

14.3.3.1 Event Initiator

This event is initiated by an instantaneous seizure of an RCP rotor.

14.3.3.2 Event Description

The RCP seizure causes an immediate reduction in RCS flow rate. As in the Loss of Forced Coolant Flow event (Event 14.3.1), the impact of losing an RCS pump is a decrease in the active flow rate in the reactor core and an increase in core temperatures. Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to DNB limits. A pressurization of the primary system will also occur due to the heatup of the primary coolant which causes a rapid insurge into the pressurizer.

14.3.3.3 Reactor Protection

Reactor protection for the RCP rotor seizure event is provided by the low reactor coolant flow trip, TM/LP trip, and the high pressurizer pressure trip.

Reactor protection for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 14.3.3-1.

14.3.3.4 Disposition and Justification

This event is a concern for only rated power and power operating conditions because for other reactor operating conditions there is sufficient thermal margin so there will not be a challenge to the fuel design limits. The core heat flux to flow ratio is an excellent indicator of the potential DNB challenge for a loss of flow event. The highest ratios for this event are predicted to occur during the first few seconds of the transient from full-power rated operating conditions. The consequences of this event will therefore be bounded by a pump rotor seizure event initiated from full-power rated conditions. There is no single failure considered which could worsen the results.

The disposition of events for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 14.3.3-2.

14.3.3.5 Definition of Events Analyzed

One case is analyzed for this event to maximize the challenge to the DNB limit. The bounding operating mode for this event is full-power initial conditions.

14.3.3.6 Analysis Results

The locked rotor analysis assumes the locked pump loss coefficient given by the homologous pump curves at zero pump speed. The sequence of events is given in Table 14.3.3-3 and the responses of key system variables are given in Figures 14.3-3–1 to 14.3-3–7 for the deterministic case. This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

The DNBR consequences of the Loss of Flow event (14.3.1) were evaluated using AREVA statistical setpoint methodology (Reference 14.3-1), and the MDNBR was shown to be greater than thermal design limits. Because the Rotor Seizure event (14.3.3) is inherently similar to and has a MDNBR greater than the Loss of Flow event, penetration of thermal design limits is precluded for the Rotor Seizure event, as well.

14.3.3.7 Conclusion

The MDNBR limits are not exceeded by this event. The peak LHR is less than the FCMLHR limit.

14.3.4 REACTOR COOLANT PUMP SHAFT BREAK

This event is not in the current licensing basis for Millstone Unit 2 and is, therefore, not analyzed.

14.3.5 REFERENCES

14.3-1 “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors,” EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000.

TABLE 14.3.1-1 AVAILABLE REACTOR PROTECTION FOR THE LOSS OF FORCED REACTOR COOLANT FLOW EVENT

Reactor Operational Mode	Reactor Protection
1 (4 pump operation)	Low Reactor Coolant Flow Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip
2 (4 pump operation)	High Pressurizer Pressure Trip Technical Specification requirements on number of operating pumps
3-6 (less than 4 pump operation)	High Pressurizer Pressure Trip Technical Specification requirements on number of operating pumps

TABLE 14.3.1-2 DISPOSITION OF EVENTS FOR THE LOSS OF FORCED REACTOR COOLANT FLOW EVENT

Reactor Operational Mode	Disposition
1	Analyze
2-6	Bounded by the above, no analysis required

TABLE 14.3.1-3 EVENT SUMMARY FOR THE LOSS OF FORCED REACTOR COOLANT FLOW

Event	Time (seconds)
Initiate Four Pump Coastdown	0.00
Letdown Flow Valve Open	0.00
Reactor Scram Signal	1.29
Rod Insertion Begins	2.44
Peak Power	2.45
MDNBR	3.7
Peak Core Average Temperature	4.10
Peak Pressurizer Pressure	5.53
Steam Line Safety Valves Open	7.35

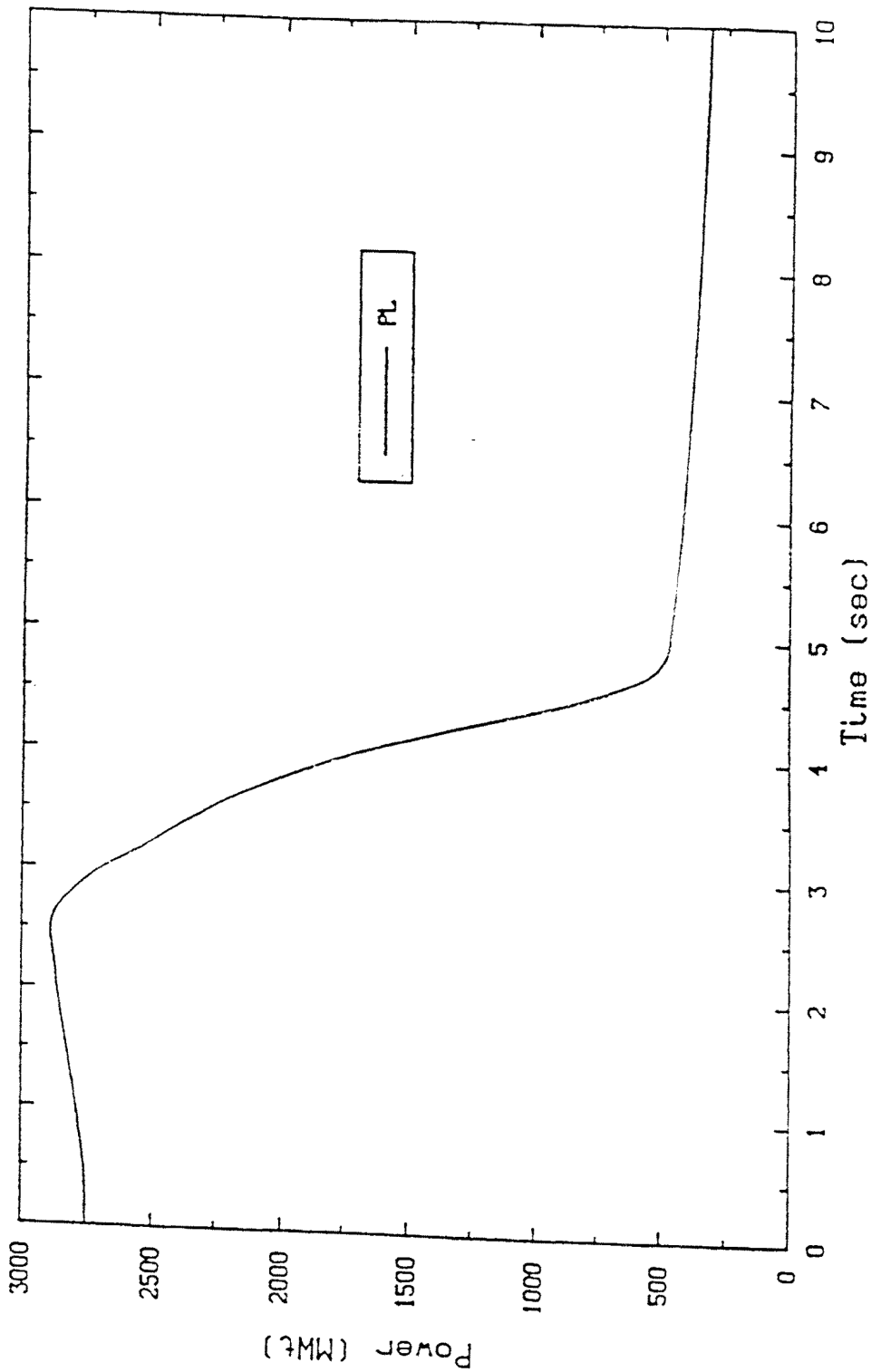


FIGURE 14.3.1–1 REACTOR POWER LEVEL FOR LOSS OF FORCED REACTOR COOLANT FLOW

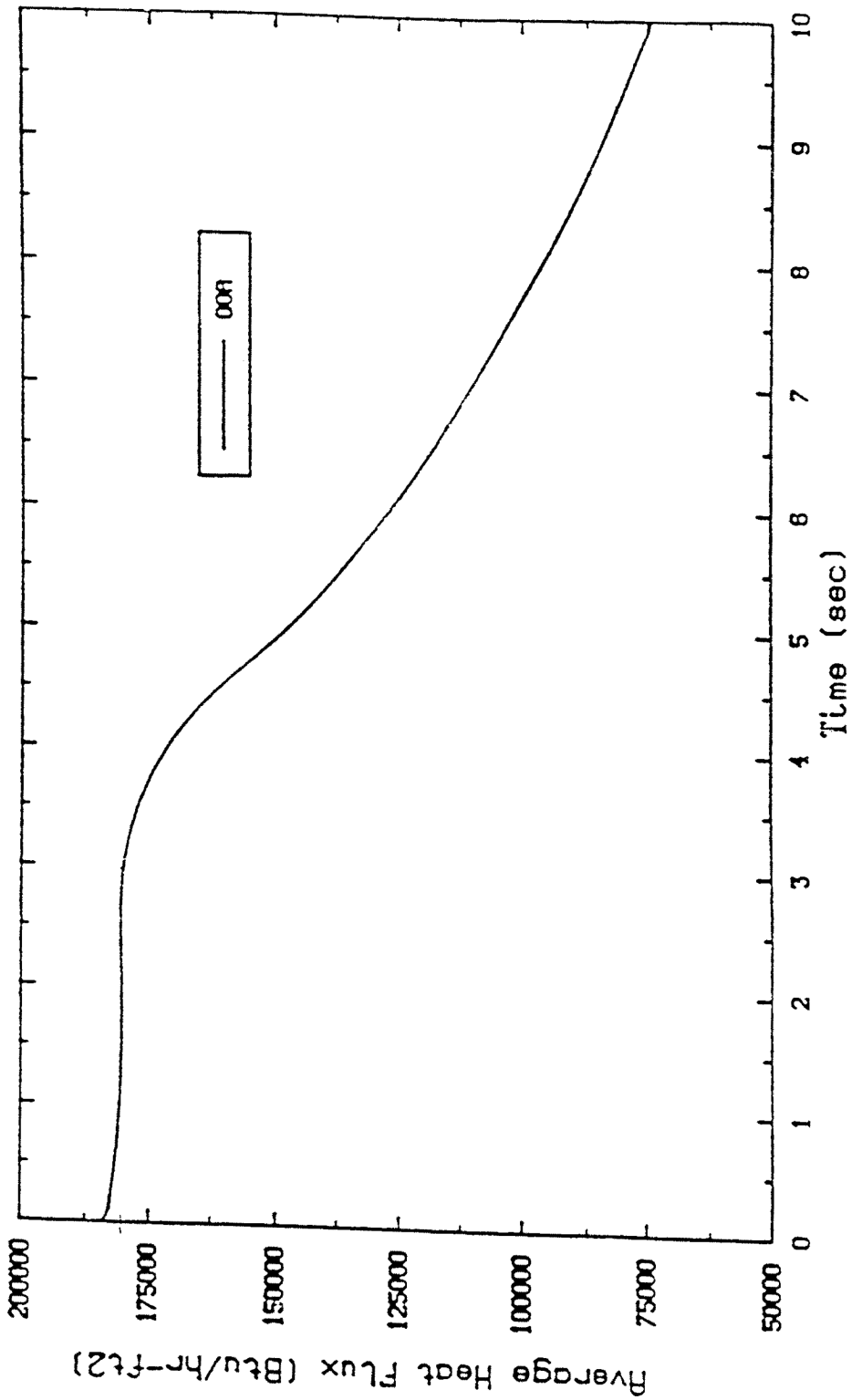


FIGURE 14.3.1–2 CORE AVERAGE HEAT FLUX FOR LOSS OF FORCED REACTOR COOLANT FLOW

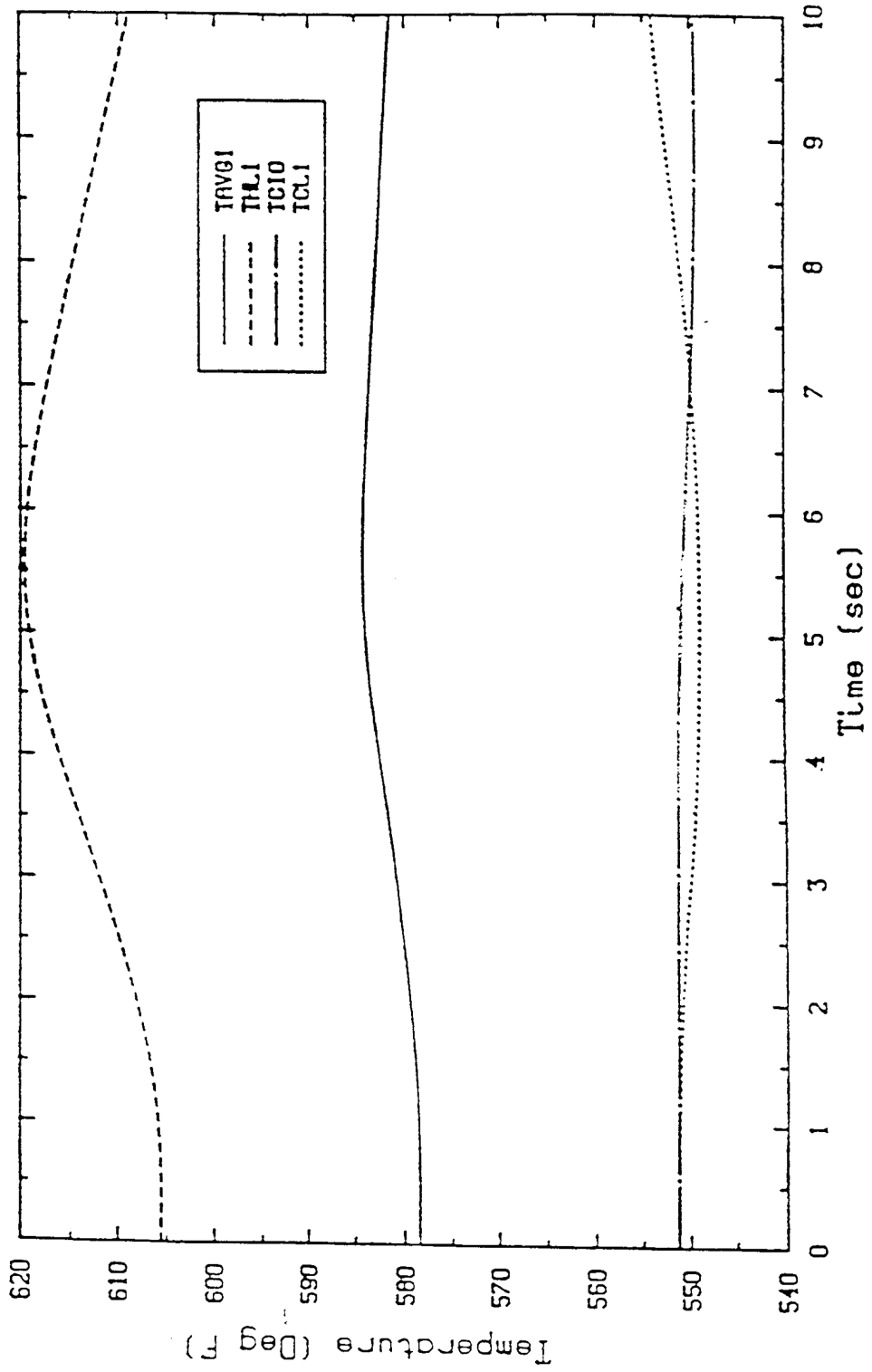


FIGURE 14.3.1–3 REACTOR COOLANT SYSTEM TEMPERATURE FOR LOSS OF FORCED REACTOR COOLANT FLOW

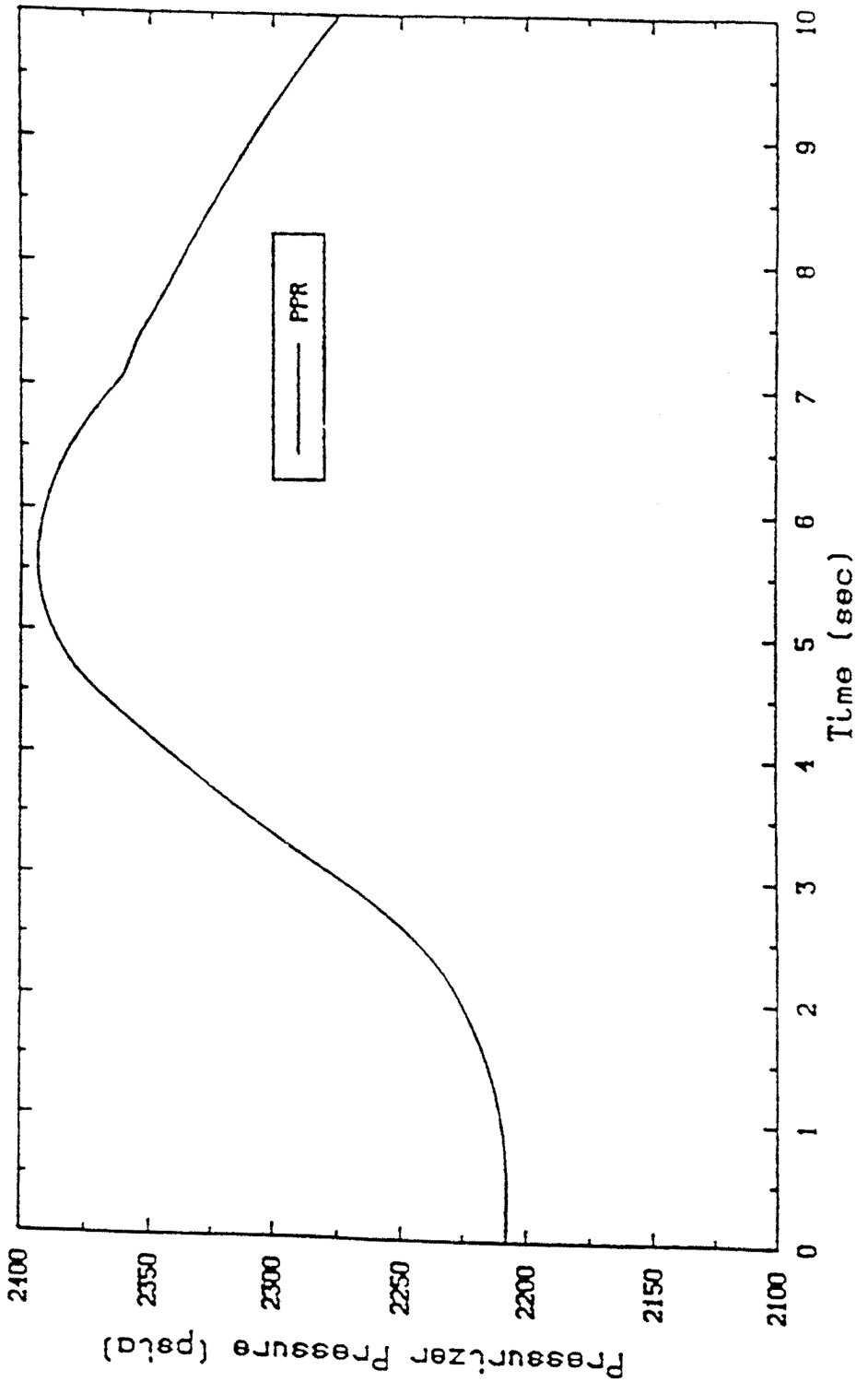


FIGURE 14.3.1–4 PRESSURIZER PRESSURE FOR LOSS OF FORCED REACTOR COOLANT FLOW

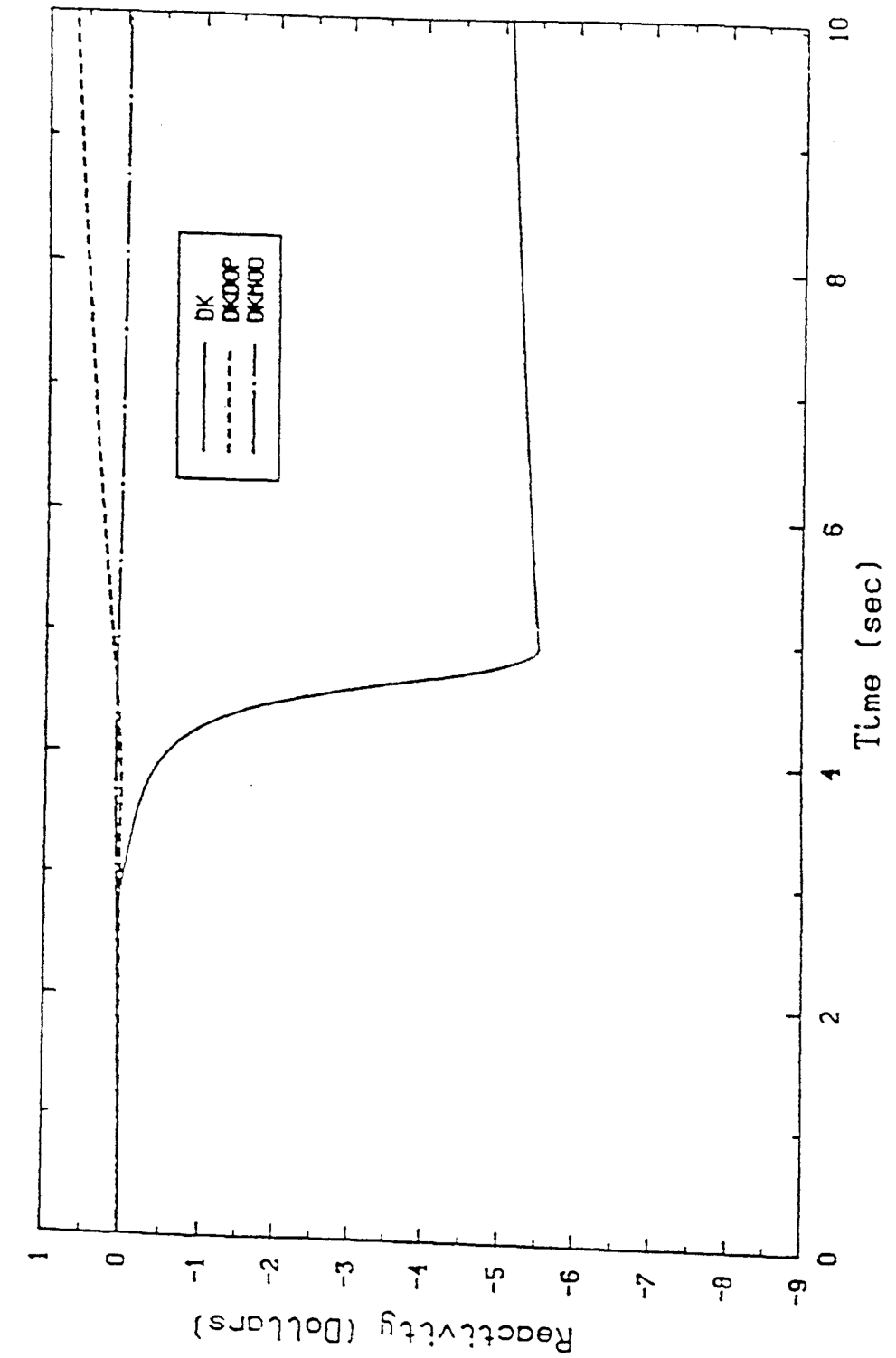


FIGURE 14.3.1–5 REACTIVITIES FOR LOSS OF FORCED REACTOR COOLANT FLOW

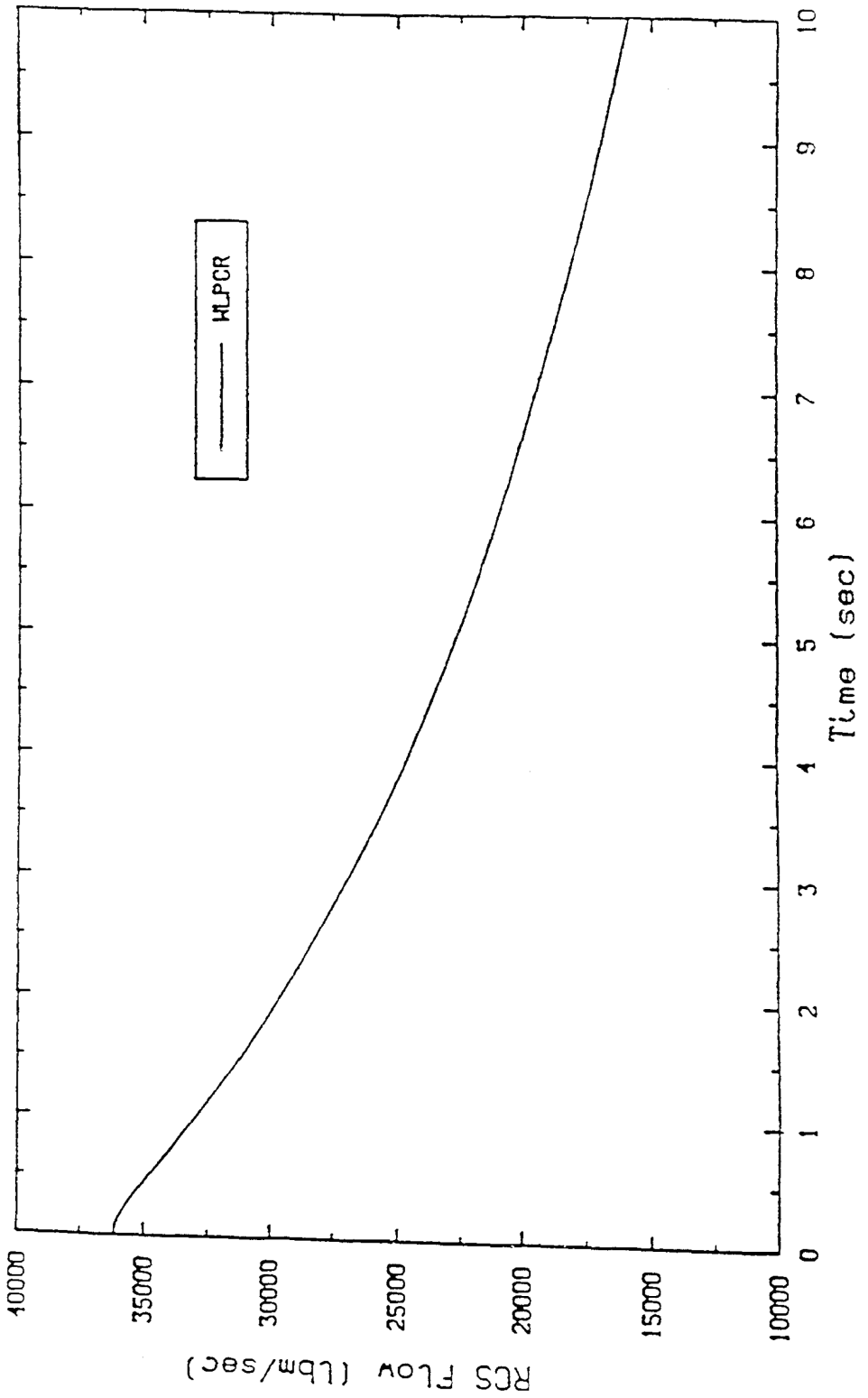


FIGURE 14.3.1–6 PRIMARY COOLANT FLOW RATE FOR LOSS OF FORCED REACTOR COOLANT FLOW

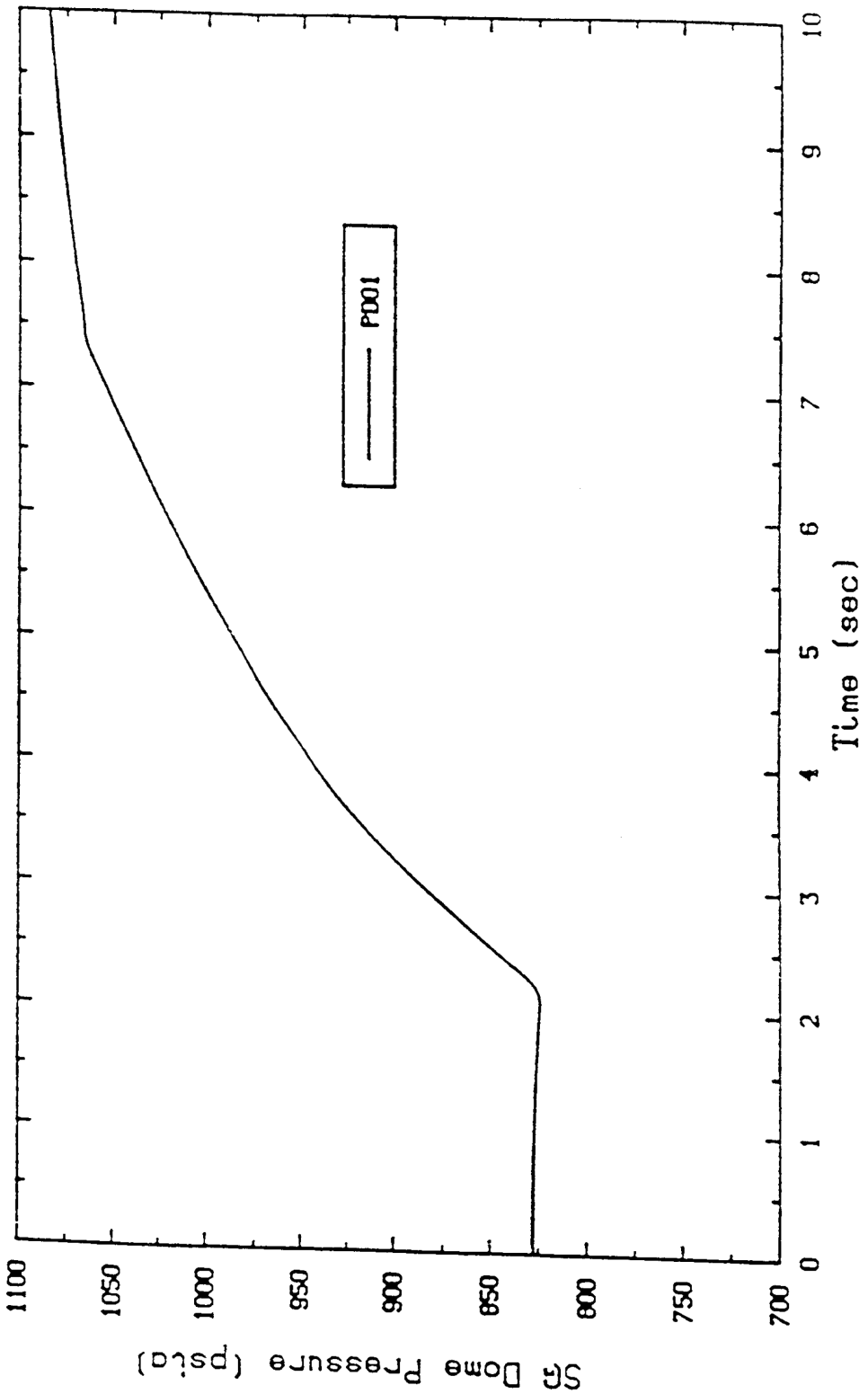


FIGURE 14.3.1–7 SECONDARY PRESSURE FOR LOSS OF FORCED REACTOR COOLANT FLOW

TABLE 14.3.3-1 AVAILABLE REACTOR PROTECTION FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

Reactor Operational Mode	Reactor Protection
1	Low Reactor Coolant Flow Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip
2	High Pressurizer Pressure Trip Available Thermal Margin (1)
3-6	Available Thermal Margin (1)

* Defense In Depth

TABLE 14.3.3-2 DISPOSITION OF EVENTS FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

Reactor Operational Mode	Disposition
1	Analyze
2	Bounded by the above
3-6	No analysis required

TABLE 14.3.3-3 EVENT SUMMARY FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE

Event	Time (seconds)
Reactor Coolant Pump Rotor Seizes	0.00
Reactor Scram Signal	0.08
Rod Insertion Begins	1.23
Peak Power	1.23
MDNBR	1.7
Peak Core Average Temperature	1.90
Peak Pressurizer Pressure	3.68
Steam Line Safety Valves Open	5.30
Peak Steam Dome Pressure	6.35

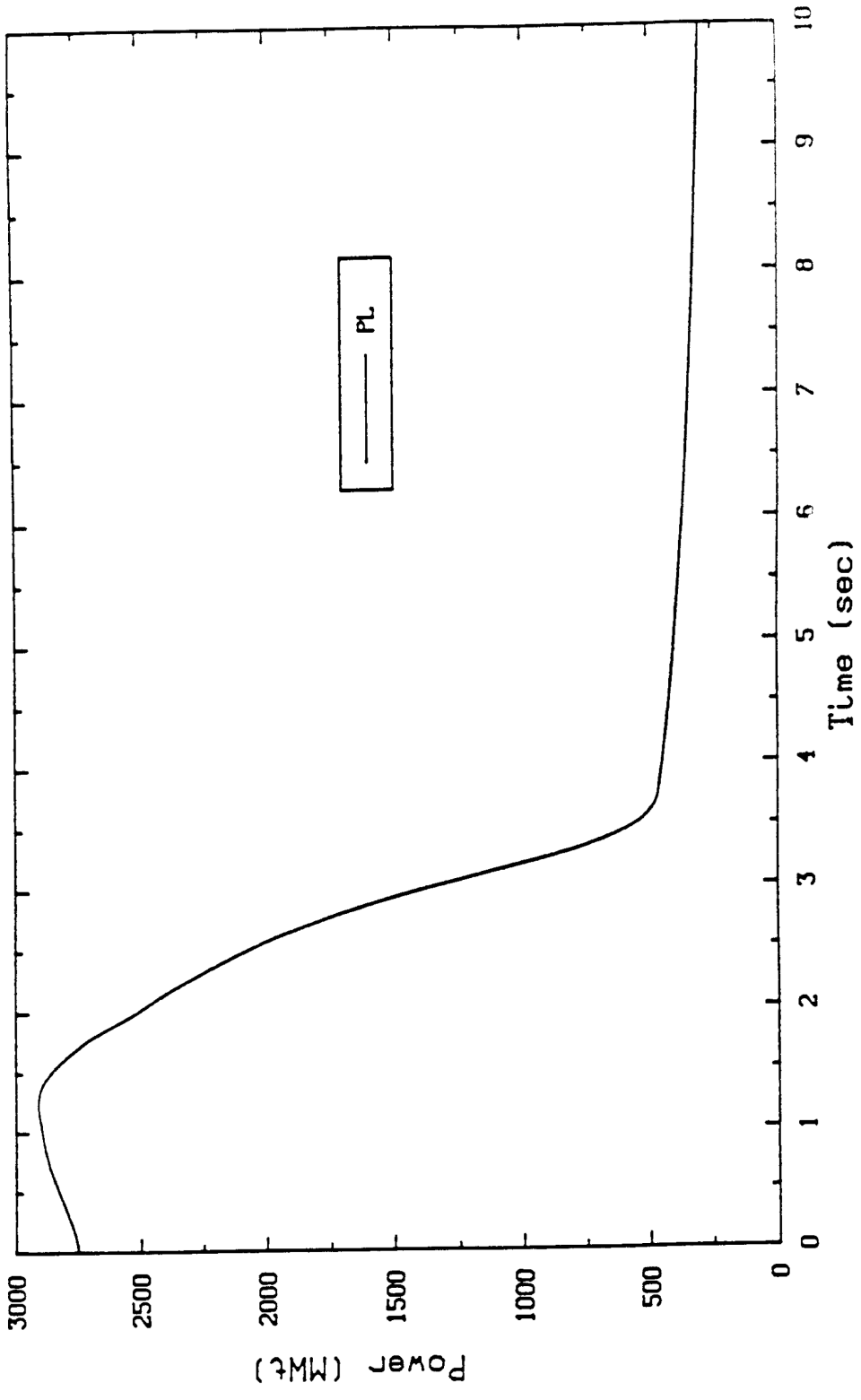


FIGURE 14.3-3–1 REACTOR POWER LEVEL FOR REACTOR COOLANT PUMP ROTOR SEIZURE

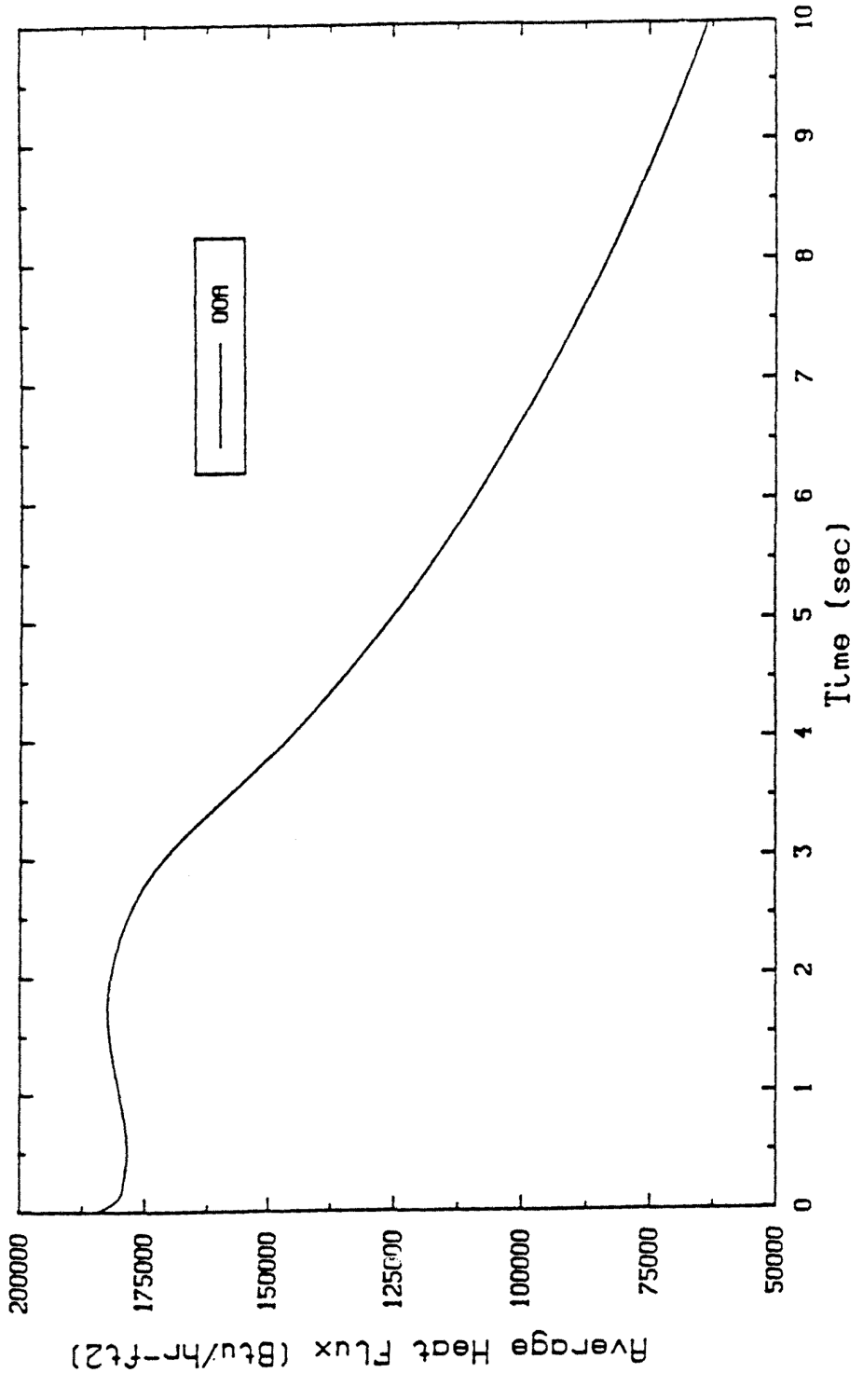


FIGURE 14.3-3–2 CORE AVERAGE HEAT FLUX FOR REACTOR COOLANT PUMP ROTOR SEIZURE

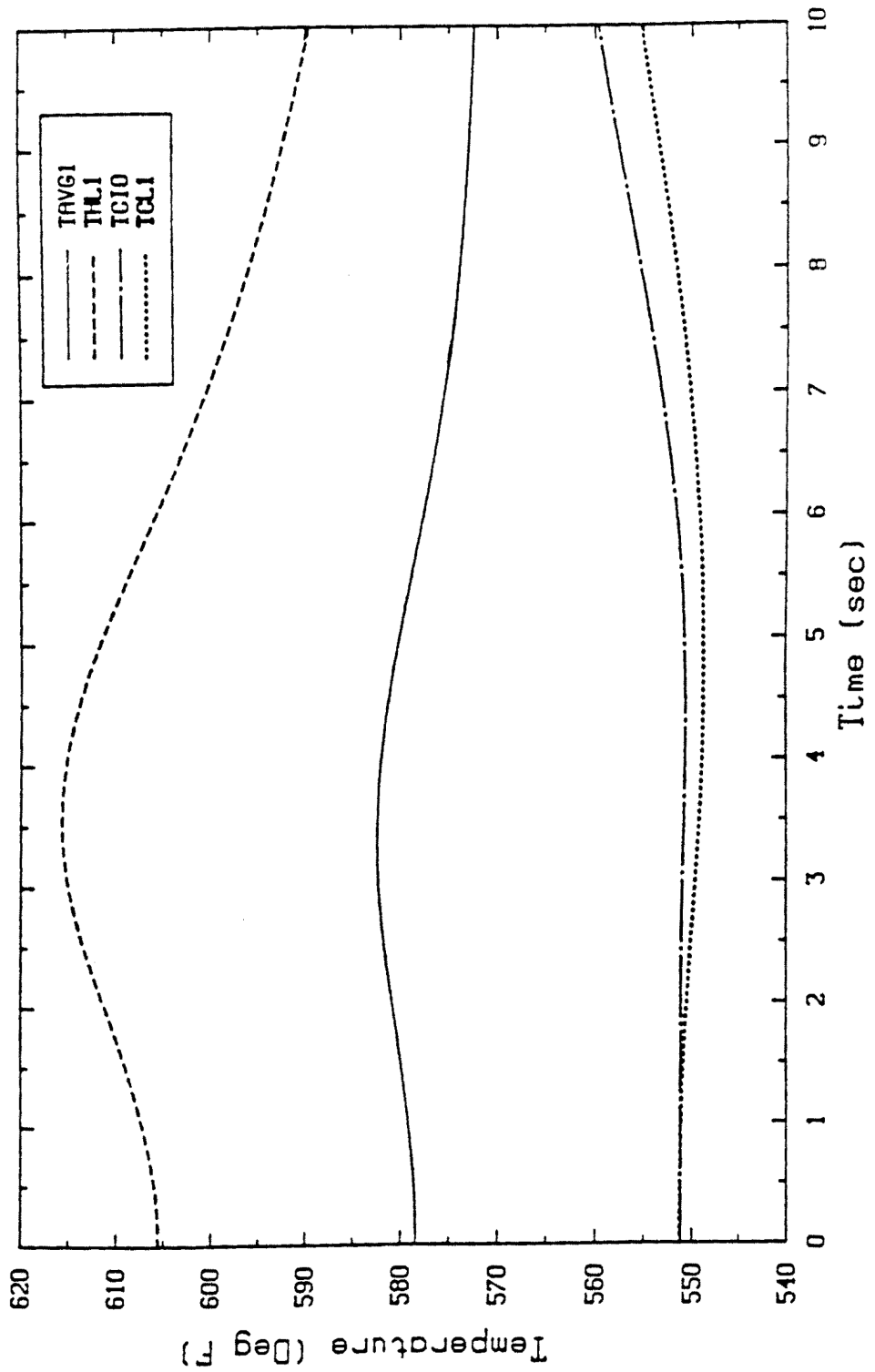


FIGURE 14.3-3–3 REACTOR COOLANT SYSTEM TEMPERATURES FOR REACTOR COOLANT PUMP ROTOR SEIZURE

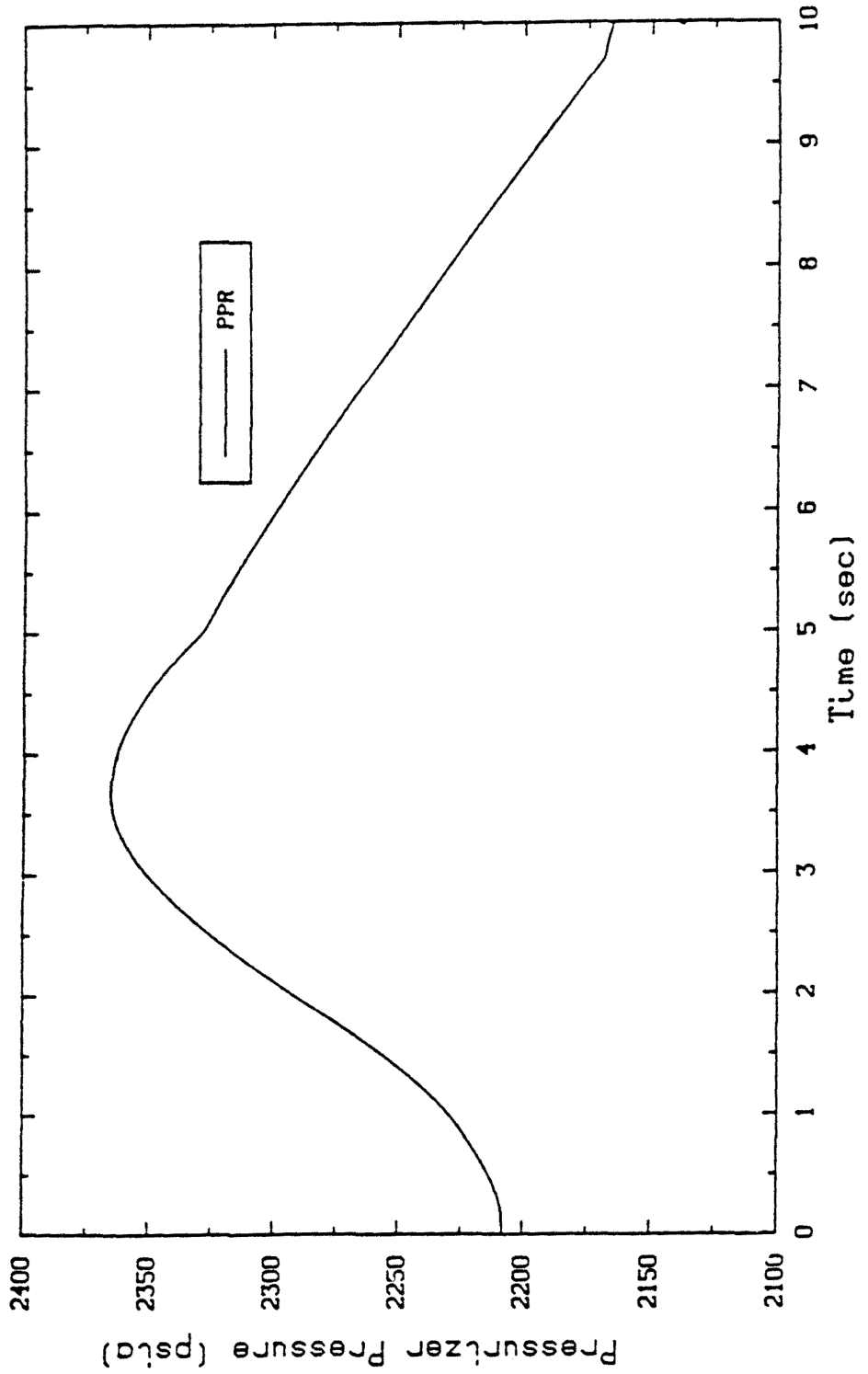


FIGURE 14.3-3–4 PRESSURIZER PRESSURE FOR REACTOR COOLANT PUMP ROTOR SEIZURE

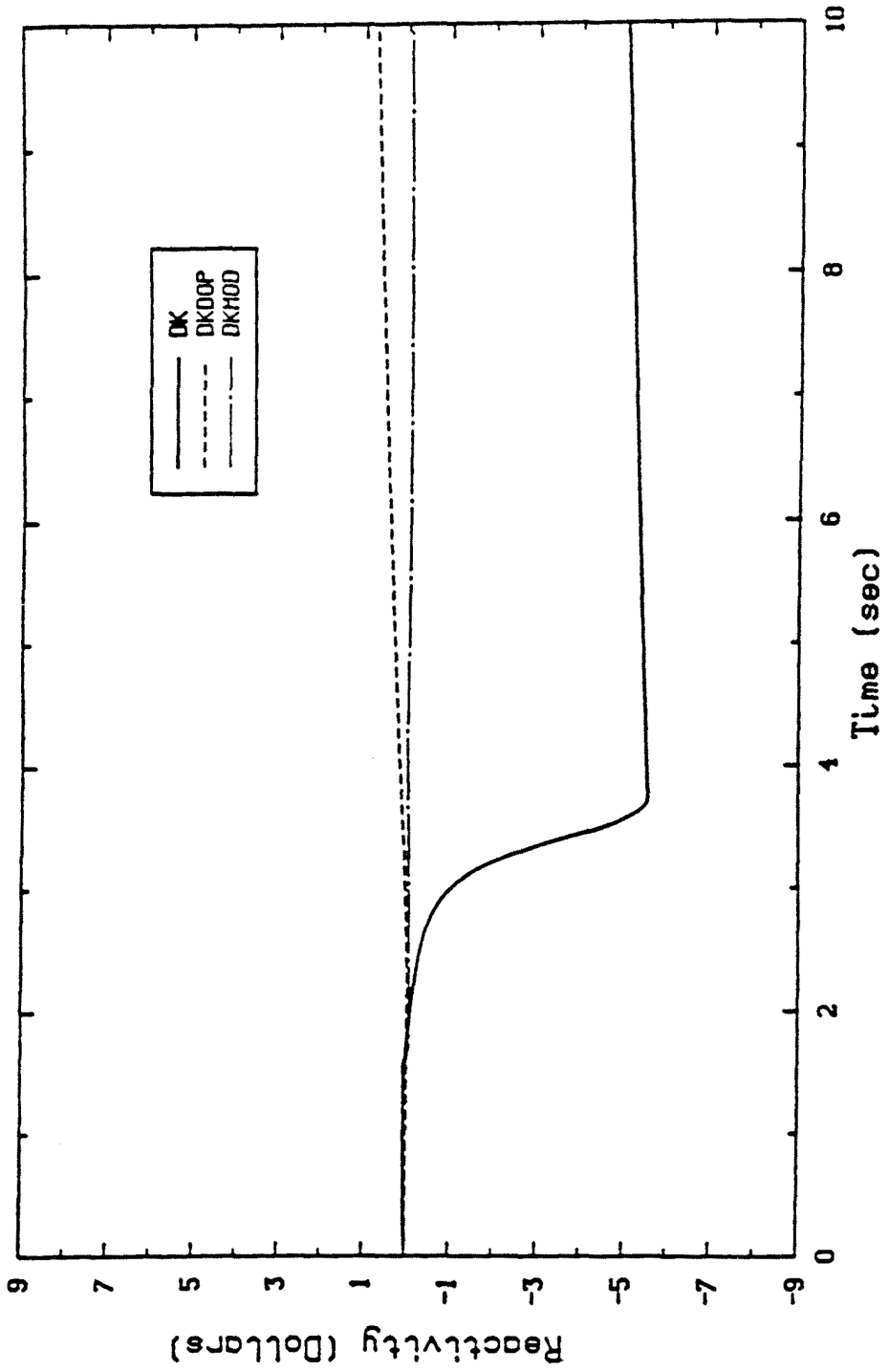


FIGURE 14.3-3–5 REACTIVITIES FOR REACTOR COOLANT PUMP ROTOR SEIZURE

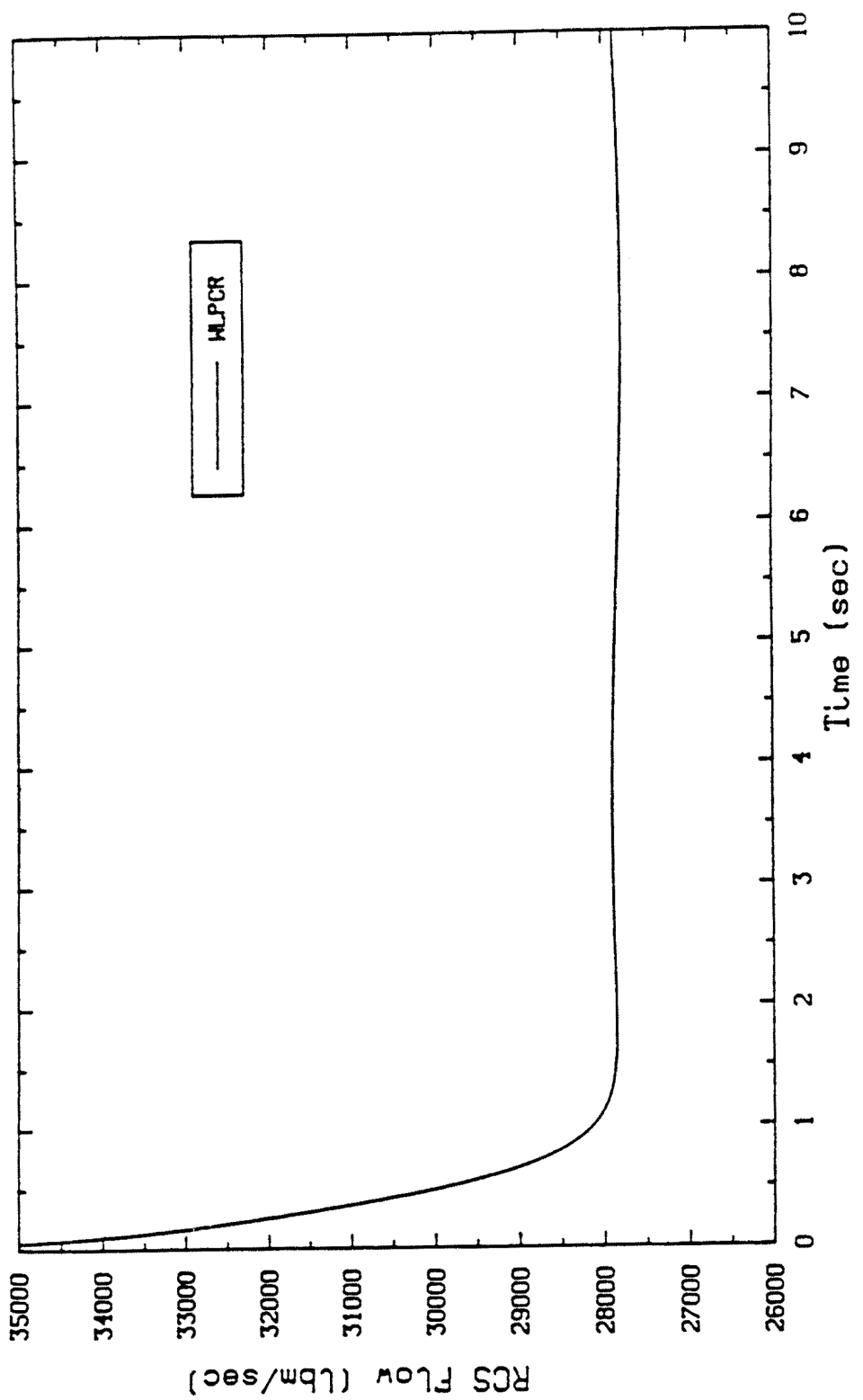


FIGURE 14.3-3–6 PRIMARY COOLANT FLOW RATE FOR REACTOR COOLANT PUMP ROTOR SEIZURE

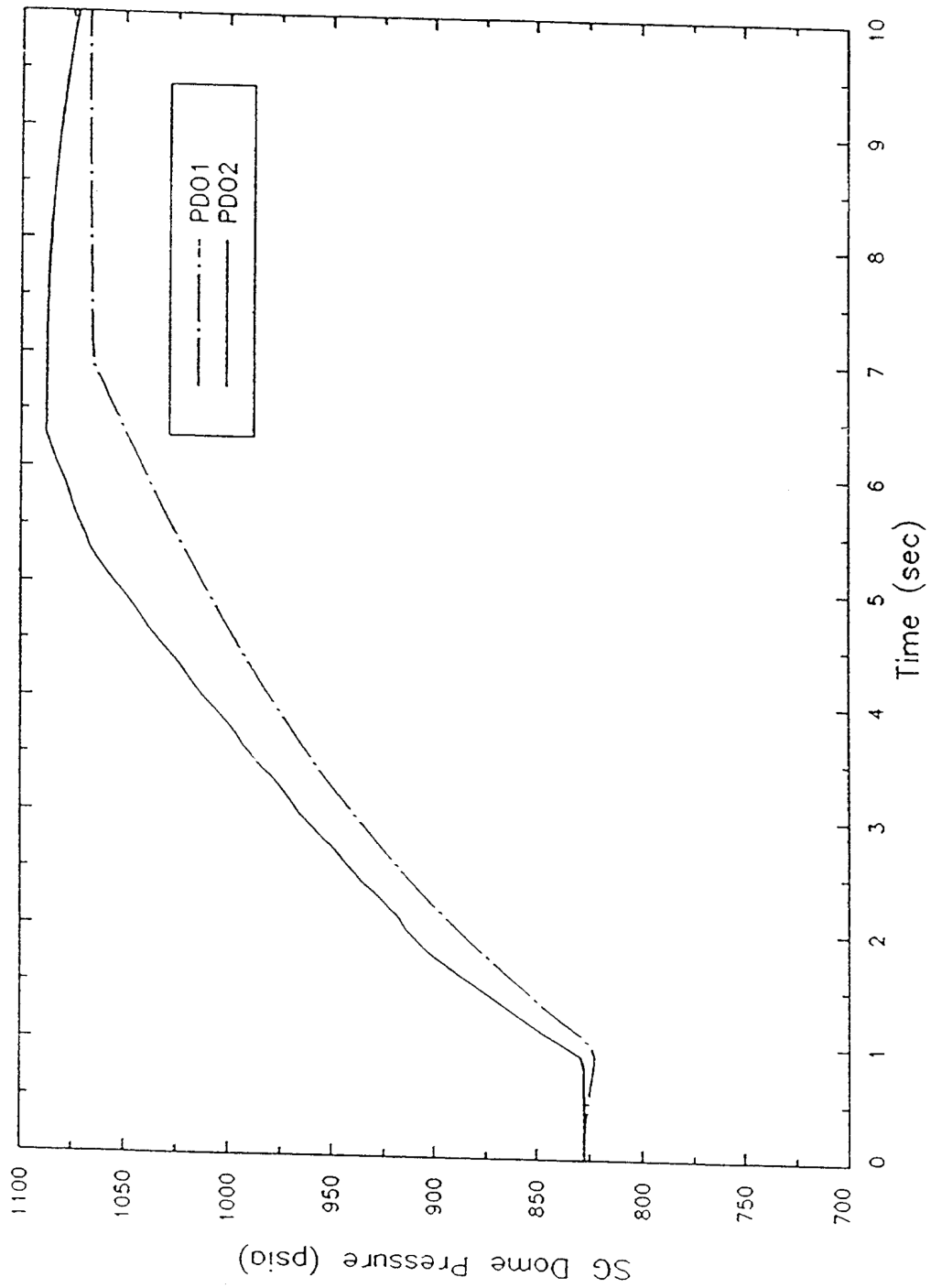


FIGURE 14.3-3–7 SECONDARY PRESSURE FOR REACTOR COOLANT PUMP ROTOR SEIZURE

14.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

14.4.1 UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW-POWER STARTUP CONDITION

14.4.1.1 Event Initiator

Initiated by the uncontrolled withdrawal of the control rod/banks in sequence, this event results in the insertion of positive reactivity and consequently a power excursion. This event could be caused by a malfunction in the reactor control or rod control systems. The consequences of the bank withdrawal from operating Modes 2-6 are considered in this event category; the consequences at rated power and power operating initial conditions are considered in Event 14.4.2.

The control rods are wired together into preselected bank configurations. These circuits prevent the control rods from being withdrawn in other than their respective banks. Power is supplied to the banks in such a way that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence.

14.4.1.2 Event Description

This event is initiated by the uncontrolled withdrawal of control rod banks in sequence. This withdrawal adds positive reactivity to the core which leads to a power excursion. As the control banks are withdrawn, the positive reactivity insertion causes a significant core power increase as the reactor approaches prompt criticality. Low coolant flow rates in the core, combined with a rapid surge of power and pronounced radial and axial power peaking, represent a challenge both to the Departure from Nucleate Boiling (DNB) and fuel centerline melt acceptance criteria. The DNB acceptance criteria may also be challenged by the reduced Reactor Coolant System (RCS) pressure for a Mode 3 initial condition. Doppler reactivity feedback from the negative Doppler coefficient limit the power excursion until the transient can be terminated by the Reactor Protection System (RPS).

14.4.1.3 Reactor Protection

The power transient is eventually terminated (as well as the control rod withdrawal) by the RPS on one of the following signals:

1. Variable overpower trip or

2. High pressurizer pressure trip.

Reactor protection for the Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Startup Condition event is summarized in Table 14.4.1-1.

14.4.1.4 Disposition and Justification

The Technical Specifications (Reference 14.4-1) for Millstone Unit 2 require that the control rod drives be deenergized in Modes 4-6 whenever the RCS boron concentration is less than the refueling requirement. A rod withdrawal from these modes is therefore not considered a credible event.

During Mode 3 operations, the control rod drive mechanism may be energized provided the Technical Specification requirements that 4 RCPs are operating, the RCS temperature is greater than 500°F, the RCS pressure is greater than 2000 psia, and the variable overpower trip is operable are met. Consequently, a rod withdrawal from these operating conditions within this mode is possible. The greatest power rise for this event is obtained when it is initiated from the lowest power.

Therefore, the event initiated from a Mode 3 condition at 2000 psia will bound all other low-power or subcritical cases. The only active system challenged in this event is the RPS, which is redundant and single failure proof.

The disposition of events for the Uncontrolled Control Rod/Bank Withdrawal from a Subcritical or Low-Power Startup Condition event is summarized in Table 14.4.1-2.

14.4.1.5 Definition of Events Analyzed

As discussed in Section 14.4.1.4, the event was analyzed from a Mode 3 initial condition at 2000 psia. These conditions will bound all other low power or subcritical cases. Axial and radial power distributions for various control rod configurations, ranging from the critical configurations to all control rods fully withdrawn, were considered. Conservative system conditions were used in the analysis to bound potential initial conditions for the transient. Four coolant pumps were considered to be in operation, consistent with the Technical Specification minimum for Mode 3 operation with Control Element Assembly (CEA) drives energized and shutdown requirements met. Credit is taken only for the variable overpower trip and high pressurizer pressure trips; other trips are not modeled. The variable overpower trip setpoint was conservatively set to the zero-power initial condition of 14.6% of rated thermal power plus 12.62% to account for uncertainties in the variable overpower trip minimum setpoint, power calibration, calorimetric power, and for power decalibration allowance. The variable overpower trip delay was set to a conservatively large value of 0.7 seconds. Biased beginning-of-cycle (BOC) kinetics values were assumed to maximize the reactor power during the transient.

14.4.1.6 Analysis Results

The event is initiated from Mode 3 with both shutdown CEA banks fully withdrawn and all regulating CEA banks fully inserted. The resultant power excursion results in a fuel temperature increase and negative Doppler reactivity feedback which limits the peak power. The transient is terminated when control rods are inserted upon a variable high power trip. The responses of key system parameters are plotted in Figures 14.4.1–1 to 14.4.1–5. The sequence of events is given in Table 14.4.1-3.

The Minimum Departure from Nucleate Boiling Ratio (MDNBR) is evaluated for conditions at the time of peak clad surface heat flux and accounts for elevated zero-power peaking. The calculated MDNBR was well above the 95/95 acceptance criterion for the HTP DNB correlation limit. This ensures that, with 95% probability and 95% confidence, DNB is not expected to occur. The peak fuel centerline temperature is calculated to be well below the melting point. Thus, no fuel failures are predicted to occur.

14.4.1.7 Conclusion

No fuel failures are predicted for this event. Therefore, the event meets the applicable acceptance criteria.

14.4.2 UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL AT POWER

14.4.2.1 Event Initiator

This event is initiated by an uncontrolled control rod/bank withdrawal from power operating conditions.

14.4.2.2 Event Description

This event is initiated by an uncontrolled withdrawal of a control bank, causing a positive reactivity addition to the reactor core. This positive reactivity addition causes an increase in the core power and primary coolant system temperatures. Due to the increasing power and temperatures, the DNB limits are challenged.

14.4.2.3 Reactor Protection

The challenge to the fuel design limits is terminated by the automatic action of the RPS which terminates the bank withdrawal and inserts negative reactivity to terminate the power transient. The automatic action of the RPS is initiated as the result of one of the following signals:

1. Variable overpower trip;
2. Local power density (LPD) trip;
3. Thermal margin/low pressure (TM/LP) trip; or
4. High pressurizer pressure trip.

Reactor protection for the Uncontrolled Control Rod/Bank Withdrawal at Power event is summarized in Table 14.4.2-1.

14.4.2.4 Disposition and Justification

This event is designed to address the safety challenge posed by an uncontrolled control rod/bank withdrawal transient from power conditions. This event addresses all the power operating conditions and the rated power operating conditions. It is performed to test the adequacy of the variable overpower and TM/LP trip setpoints in mitigating the challenge to the Specified Acceptable Fuel Design Limits (SAFDL).

A rod withdrawal initiated from lower powers will provide less of a challenge to the SAFDLs due to increased initial thermal margin, a lesser amount of setpoint overshoot, and a decreased variable overpower trip setpoint resulting in a greater thermal margin at trip. The event initiated from full power will then bound those initiated from lower power conditions. This event will therefore be analyzed at full power for conditions ranging from BOC to end of cycle (EOC) for a spectrum of reactivity insertion rates. The only active system challenged by this event is the RPS, which is redundant and single failure proof.

The disposition of events for the Uncontrolled Control Rod/Bank Withdrawal at Power event is summarized in Table 14.4.2-2.

14.4.2.5 Definition of Events Analyzed

The analysis evaluates the consequences of an uncontrolled control rod bank withdrawal from rated power. A spectrum of reactivity insertion rates were evaluated in order to bound events ranging from a slow dilution of the primary system boron concentration to the fastest allowed control bank withdrawals. Specifically, the analysis encompasses reactivity insertion rates from 4 x 10^{-6} to 4 x 10^{-4} delta rho/sec.

14.4.2.6 Analysis Results

The uncontrolled control bank withdrawal transients were analyzed for full-power conditions (100% of rated). The limiting uncontrolled control rod bank withdrawal at 100% power occurred with EOC kinetics at an insertion rate of 4 x 10^{-6} delta rho/sec. The MDNBR was calculated to be above the CHF correlation limit. This transient tripped on a TM/LP signal. The maximum peak pellet linear heat rate (LHR) occurs in a 100% power case which uses BOC kinetics. The variable high power and LPD trips ensure the maximum peak pellet LHR is less than the FCMLHR limit.

The sequence of events for the Uncontrolled Bank Withdrawal transient is given in Table 14.4.2-3. The transient response of key system variables are given in Figures 14.4.2–1 to 14.4.2–6.

14.4.2.7 Conclusion

Reactivity insertion transient calculations demonstrate that the DNBR limit will not be penetrated during any credible reactivity insertion transient at full power. The maximum peak pellet linear heat generation rate for this event is less than the FCMLHR limit. Applicable acceptance criteria are therefore met, and the adequate functioning of the TM/LP trip is demonstrated.

14.4.3 CONTROL ROD MISOPERATION

The control rod misoperation event encompasses a number of transients resulting from different event initiators. The specific events addressed under this event category include the following:

1. Dropped control rod or control rod bank;
2. Dropped part-length control rod;
3. Malpositioning of the part-length control rod group;
4. Statically misaligned control rod/control rod bank;
5. Single control rod withdrawal;
6. Reactivity control device removal error during refueling; and
7. Variations in reactivity load to be compensated by burnup or on-line refueling.

14.4.3.1 Dropped Control Rod/Bank

14.4.3.1.1 Event Initiator

The Dropped Control Rod/Bank event is initiated by a de-energized control rod drive mechanism or by a malfunction associated with a control rod bank.

14.4.3.1.2 Event Description

A Dropped Control Rod/Bank event is initiated by a deenergized control element drive mechanism (CEDM) or another failure in the control rod system. The reactor power initially drops in response to the insertion of negative reactivity. However, the local peaking increases due to the local effect on the power distribution. The reactor core will attempt to return to a new equilibrium at the original power level as a result of moderator and Doppler reactivity feedback. Because of the increased peaking and the potential return to the initial power level, the Dropped Control Rod/Bank event poses a challenge to the DNB margin.

14.4.3.1.3 Reactor Protection

If the amount of reactivity is large enough to cause a significant reduction in core power, a reactor trip could be generated by the variable overpower trip prior to returning to full power. Reactor protection for the Dropped Control Rod/Bank event is summarized in Table 14.4.3.1-1.

14.4.3.1.4 Disposition and Justification

Since the control rod drive mechanisms are deenergized in Modes 4-6 and reactor power is limited to zero percent with $k_{eff} < 0.99$ in Mode 3, there will be no consequences of this event for these modes.

Ultimately, the consequences of this event are a return to power at elevated peaking conditions. Thus, the worst case is obtained when the combination of final power level, increased peaking, and core inlet temperature are maximized. This occurs for cases initiated from full-power. The full-power case thus bounds all other power operation conditions.

For a single dropped control rod, a reactor trip is not expected. Thus, a DNB evaluation assuming a return to full-power at maximum dropped rod peaking will be performed to demonstrate that the SAFDLs are not violated.

The return to power for a dropped control bank is limited by the capacity of the turbine control valve. In response to a decrease in the secondary side steam flow resulting from a drop in core power, the turbine valve will throttle open in an attempt to maintain a constant load demand. If the reactivity worth of the dropped control bank is sufficiently large, the turbine valve will not have enough excess capacity for the reactor to return to full power. The lower power level could be offset by the higher peaking factor associated with a dropped control bank. It should be noted that the operator will have multiple indications that a dropped rod/bank has occurred via CEA deviation alarms and rod bottoming signals. The only active system challenged in this event is the RPS, which is redundant and single failure proof.

The disposition of events for the Control Rod Misoperation (Dropped Control Rod/Bank) event is summarized in Table 14.4.3.1-2.

14.4.3.1.5 Definition of Events Analyzed

The analysis evaluates the consequences of this event from rated power conditions. A spectrum of dropped control rod/bank cases were analyzed at full power with increased radial peaking and Technical Specifications minimum primary coolant flow. Radial peaking augmentation factors for the Dropped Control Rod/Bank event are calculated at full power for different exposure conditions. Bounding radial peaking augmentation factors were used in the analysis. In addition, bounding values of control rod and bank worth were used.

14.4.3.1.6 Analysis Results

The sequence of events for the limiting dropped control rod/bank case is given in Table 14.4.3.1-3. The transient response or key system parameters are given in Figures 14.4.3.1–1 to 14.4.3.1–4. The limiting case, both from the standpoint of MDNBR and peak LHR, was with a 1079.8 pcm dropped control bank.

Upon transient initiation, the core power decreased in response to the negative reactivity insertion resulting from the dropped control bank. The primary coolant and fuel temperatures decreased in

response to the falling power. Positive moderator and Doppler reactivity feedback resulted in the return to a new steady state power level. The initial decrease in the primary coolant temperatures resulted in the VHP trip being reset to a power level of 89.3% of RTP. The measured DT power exceeded the VHP trip setpoint at 88.3 seconds. Throughout the transient, the core inlet mass flow rate increased and the primary-side pressure decreased due to changes in the moderator density resulting from the primary-side cooldown.

The peak pellet LHR is calculated to be less than the FCMLHR limit. The minimum DNBR for this event is bounded by the minimum DNBR of the Section 14.3.1 loss of forced reactor coolant flow event, which is greater than thermal margin limits.

14.4.3.1.7 Conclusion

All of the cases analyzed were above the HTP 95/95 DNB safety limit and below the peak LHGR limit. Therefore, no fuel failure is predicted to occur. Applicable acceptance criteria for the Dropped Control Rod/Bank event are therefore met for Millstone Unit 2.

14.4.3.2 Dropped Part-Length Control Rod

All part-length control rods have been removed from the Millstone Unit 2 core. Therefore, this event is not applicable.

14.4.3.3 Malpositioning of the Part-Length Control Rod Group

All part-length control rods have been removed from the Millstone Unit 2 core. Therefore, this event is not applicable.

14.4.3.4 Statically Misaligned Control Rod/Bank

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.4.3.5 Single Control Rod Withdrawal

14.4.3.5.1 Event Initiator

This event is initiated by the inadvertent withdrawal of a single CEA from the core. No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single CEA from the inserted CEA bank during full power operation. Procedures are available to permit the operator to withdraw a single CEA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event can occur only as the result of multiple wiring failures or multiple operator actions in disregard of available event indication.

In the extremely unlikely event of simultaneous electrical failures which could result in single CEA withdrawal, the rod position indicators and deviation alarms would indicate the relative positions of the assemblies in the bank. Withdrawal of a single CEA by operator action, whether

deliberate or by a combination of errors, would similarly result in the same visual indications. The CEA Motion Inhibit prevents further rod control motion upon detection of CEA malpositioning.

14.4.3.5.2 Event Description

The withdrawal of a single full-length CEA is initiated by the inadvertent withdrawal of a single control rod. The ensuing reactivity insertion causes core power to increase. In the event that the secondary steam dump control system does not respond to the increased power production, secondary system temperature and pressure will increase, causing a corresponding increase in primary coolant temperature. This increase in primary coolant temperature occurs slowly enough that the pressurizer pressure control system, if available, is capable of suppressing the primary pressure increase. The degradation of coolant conditions coupled with the power increase is essentially the same as expected for slow CEA bank withdrawals at power and may approach DNB conditions in the hot channel.

The single CEA withdrawal is distinguished from the withdrawal of a CEA bank by a severe radial power redistribution. High radial power peaking is localized in the region of the single withdrawn CEA and may, in severe cases, surpass the design limits. Thus, assemblies in the immediate vicinity of the withdrawn CEA may experience boiling transition for a short time period. Some fuel damage might occur.

14.4.3.5.3 Reactor Protection

The challenge to the fuel design limits is terminated by the automatic action of the RPS which terminates the CEA withdrawal and inserts negative reactivity to terminate the power transient. The automatic action of the RPS is initiated as the result of one of the following signals:

1. Variable overpower trip;

2. LPD trip;

3. TM/LP trip; or

4. High pressurizer pressure trip.

Reactor protection for the Single Control Rod Withdrawal event is summarized in Table 14.4.3.5-1.

14.4.3.5.4 Disposition and Justification

The overall system response to the withdrawal of a single CEA will be identical to the response to a slow withdrawal of a CEA bank. The only difference will be that the core will experience localized peaking in the vicinity of the withdrawn CEA that is not present if an entire bank is withdrawn. Therefore, the disposition of the single CEA withdrawal will be identical to that of the CEA bank withdrawal.

The disposition of the low to zero power bank withdrawal is addressed in Event 14.4.1. The disposition of the bank withdrawal from power operating conditions is addressed in Event 14.4.2. The disposition of events for the Single Control Rod Withdrawal event is summarized in Table 14.4.3.5-2.

14.4.3.5.5 Definition of Events Analyzed

This event was analyzed at rated power conditions. Radial peaking augmentation factors to account for localized peaking redistribution were utilized in the assessment of the challenge to MDNBR limits.

14.4.3.5.6 Analysis Results

Radial peaking augmentation factors for single control rod withdrawal events are calculated at full power for different exposure conditions. Bounding radial peaking augmentation factors were used in the analysis. In addition, bounding value of control rod worth was used.

The MDNBR calculation is performed with the methodology described in Reference 14.4-5.

The MDNBR for the single rod withdrawal event is greater than the HTP correlation limit. The peak pellet LHR is calculated to be less than the FCMLHR limit. Therefore, no fuel failure is predicted to occur.

14.4.3.5.7 Conclusion

The maximum peak LHR for the single rod withdrawal event is such that fuel centerline melt is not expected. In addition, the minimum DNBR is greater than the limit. Thus, no fuel failure is predicted to occur.

14.4.3.6 Reactivity Control Device Removal Error During Refueling

Millstone Unit 2 has no reactivity control devices which are used during refueling and could inadvertently be removed. Boron dilution during refueling is considered in Event 14.4.6. Therefore, this event is not applicable.

14.4.3.7 Variations in Reactivity Load to be Compensated by Burnup or On-Line Refueling

This event considered the anticipated variations in the reactivity load of the reactor, to be compensated by means of action such as buildup and burnup of xenon poisoning, fuel burnup, on-line refueling, fuel followers, temperature moderator and void coefficients.

Provisions for xenon changes and fuel burnup are described in Chapter 3. On line refueling will not be performed on Millstone Unit 2. The core design does not include fuel followers. The safety analyses are based upon the most adverse combination of temperature, moderator and void coefficients. Therefore, this event has no significant consequences and is not analyzed.

14.4.4 STARTUP OF AN INACTIVE LOOP

14.4.4.1 Event Initiator

This event is initiated by the startup of an inactive reactor coolant pump (RCP).

14.4.4.2 Event Description

Each primary coolant loop is equipped with two single-suction centrifugal pumps, one per cold leg, which are located between the steam generator outlet and the reactor vessel inlet nozzles. A nonreversing mechanism is provided to prevent reverse rotation of the pump rotor. This feature also limits backflow through the pump under nonoperating conditions. Note that there is no backflow in the hot leg (or steam generator) associated with the side of the plant that has the inactive RCP. The inadvertent actuation of an inactive pump would therefore lead to a decrease in moderator temperature and, with a negative moderator coefficient, an increase in core reactivity with a potential increase in core power level.

14.4.4.3 Reactor Protection

Reactor protection for this event is afforded by Technical Specification requirements on shutdown margin and RCP operation. Reactor protection for the Startup of an Inactive Loop event is summarized in Table 14.4.4-1.

14.4.4.4 Disposition and Justification

This event is not credible in operating Modes 1 and 2 because Technical Specifications require all four RCPs to be operating. It is not credible in Mode 6 due to administrative procedures requiring that the pumps be prevented from starting.

Technical specification requirements on shutdown margin in Modes 3-5 are such that any reactivity insertion due to an inactive loop start is not great enough to reach criticality. Thus, the consequences of this event in Modes 3-5 are minimal and no analysis is required. The disposition of events for the Startup of an Inactive Loop event is summarized in Table 14.4.4-2.

14.4.5 FLOW CONTROLLER MALFUNCTION

Millstone Unit 2 does not have any flow control devices on the primary reactor coolant loops so this event is not credible and does not need to be analyzed.

14.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

14.4.6.1 Event Initiator

A dilution of the primary system boron concentration can occur as a result of adding primary grade water into the RCS via the Chemical and Volume Control System (CVCS). The greatest dilution rate occurs for operation of the CVCS charging pumps. The three available charging pumps can inject water into the primary system at a maximum rate of 147 gpm. For Modes 4, 5, and 6, only two charging pumps are to be operable for a maximum rate of 98 gpm.

14.4.6.2 Event Description

A boron dilution event can occur when demineralized water is added to the primary coolant system via the CVCS resulting in decreasing boron concentration in the primary system coolant. This dilution of the primary system coolant boron concentration results in the addition of positive reactivity to the core. This event can lead to an erosion of shutdown margin for subcritical initial conditions, or a slow power excursion for at-power conditions. A boron dilution at rated or power operating conditions behaves in a manner similar to a slow uncontrolled rod withdrawal transient (Event 14.4.2).

14.4.6.3 Reactor Protection

Reactor protection for the boron dilution event during operating Modes 3-6 is provided by Technical Specification shutdown margin requirements, administrative procedures, and sufficient time for the operator to take the appropriate action in the unlikely event that a boron dilution should occur. Reactor protection for the reactor critical, power operation, and rated power operating conditions is provided by various trips and operator response time. Reactor protection for the CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant event is summarized in Table 14.4.6-1.

14.4.6.4 Disposition and Justification

For boron dilutions in reactor Modes 1-6, the challenge to the SAFDLs is very similar to that of slow control rod withdrawals and can be bounded by the consequences of control rod withdrawal events as analyzed for events 14.4.2 and 14.4.1. A spectrum of control rod withdrawal reactivity addition rates is considered for Events 14.4.2 and 14.4.1, so the range of reactivity addition rates will be established to encompass the predicted reactivity addition rates for boron dilution events in Modes 1-6.

There must be 15 minutes (modes 3, 4, and 5) or 30 minutes (mode 6) from the onset of the dilution prior to a complete erosion of shutdown margin. The disposition of events for the CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant event is summarized in Table 14.4.6-2.

14.4.6.5 Definition of Events Analyzed

The boron dilution analysis evaluates the time to criticality caused by the dilution of the primary system boron and the subsequent loss of shutdown margin. This analysis determines the shutdown cooling system flow rate needed to meet the time criteria for Refueling (Mode 6), Cold Shutdown (Mode 5), and Hot Shutdown (Mode 4). The systems that would be involved in the boron dilution event, depending upon the mode of operation are the RCS, the shutdown cooling system and the CVCS.

The major differences between the operating modes are the system parameters which affect the rate at which boron dilution occurs and the boron mixing model used once the demineralized water is injected into the lower plenum of the reactor vessel. Parameters such as charging pump capacity and primary system water volume affect the dilution rate. The mixing model used depends on whether the RCPs are operating and whether the shutdown cooling system discharge is to both cold legs when the dilution is postulated to occur.

For the six modes of operation, two mixing models are used: (1) instantaneous mixing and (2) dilution front. The use of the instantaneous mixing and dilution front models is consistent with EMF-2310(P)(A) (Reference 14.4-4).In the instantaneous mixing mode, the diluting water is assumed to uniformly mix with the entire RCS volume immediately upon injection into the primary system. Instantaneous mixing is assumed to occur if one or more RCPs are in operation.

A dilution front model is used to simulate operation of the shutdown cooling system when the main RCPs are not running. It is conservatively assumed that the diluted water from the shutdown cooling system discharged to the lower plenum will not immediately mix with the entire reactor coolant due to the relatively low flow rate. Rather, the boron dilution occurs locally at the charging/shutdown cooling mixing location. The diluted mixture then flows through the RCS system until it reaches the mixing location where further dilution occurs. Thus, the RCS boron concentration can be viewed as a series of dilution fronts traveling through the RCS.

Symmetric and asymmetric variations of the dilution front model are considered. In the symmetric flow variation, diluting water is assumed to be injected into two or more cold legs located on opposite loops of the RCS. In the asymmetric flow variation, diluting water is assumed to be injected into only one cold leg. The asymmetric flow variation is more limiting than the symmetric flow variation. In either case, the time to criticality is reduced if the shutdown cooling system flow is reduced.

The boron dilution analysis also included calculations to determine the maximum and minimum reactivity insertion rates during Full Power Operation (Mode 1). Reference 14.4-4 indicates that in the event of an unplanned Boron dilution during power escalation while in Startup (Mode 2), the plant status is such that minimal impact will result. The plant will slowly escalate in power and will activate a power-related trip. Boron dilution analysis does not need to include calculations to determine the maximum and minimum reactivity insertion rates during Startup (Mode 2) under this methodology. These values are used to confirm that the reactivity insertion rates used for the uncontrolled rod withdrawal analysis remain bounding of the boron dilution event.

14.4.6.6 Analysis Results

Table 14.4.6-3 presents the minimum shutdown cooling flow for Modes 4 through 6 required to avoid complete erosion of shutdown margin within the required time. These results are based on the asymmetric dilution front model. For Modes 4 to 6, the reactor vessel is filled to the mid-plane of the coolant loop nozzles. This is the minimum expected RCS volume during shutdown cooling operations. For these modes, only two charging pumps were considered to be operable. A reactivity shutdown margin of 3.6% was used for all operating modes except Mode 6. For Mode 6, a K_{eff} of 0.95 was used. The results indicate that as long as the shutdown cooling system flow remains above the flowrates provided in Table 14.4.6-3, the operator response time criteria will be satisfied.

Table 14.4.6-4 presents the results of the calculated time to loss of shutdown margin for Modes 1 through 5 based on the use of the instantaneous mixing assumption. The erosion of shutdown margin following trip in Mode 1 is bounded by the Mode 2 Boron Dilution analysis, as indicated in Reference 14.4-4. For Modes 2 and 3, three charging pumps were assumed to be operable and only two charging pumps were considered to be operable for Modes 4 and 5. The results indicate that the time to loss of shutdown margin criteria is satisfied for all cases in which an RCS pump is in operation.

For Mode 1, the Reactivity Insertion Rate associated with a Boron dilution event is bounded by the rates of reactivity insertion from a rod withdrawal event.

14.4.6.7 Conclusions

The results of the boron dilution analysis show that there is not a complete erosion of shutdown margin within 15 minutes for modes 1 through 5 and 30 minutes in mode 6. The operator can initiate reboration to recover the shutdown margin. For all modes there is adequate time for the operator to manually terminate the source of dilution flow.

14.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.4.8 SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

14.4.8.1 Event Initiator

This accident is initiated by a failure in the control rod drive pressure housing which could result in the rapid ejection of a control rod.

14.4.8.2 Event Description

This event is initiated by a failure in the CEDM pressure housing causing a rapid ejection of the affected control rod. This results in a rapid loss of negative reactivity causing a nuclear power

transient. In addition to the power transient, the ejected rod results in a highly perturbed power distribution which, coupled with the power transient, could possibly lead to localized fuel damage. Also, the rapid nuclear power excursion can result in a significant short-term heatup of the coolant with a resultant RCS pressure increase, although on the long-term the RCS will depressurize due to the break in the reactor coolant pressure boundary.

14.4.8.3 Reactor Protection

Reactor protection for the Spectrum of Control Rod Ejection Accidents is summarized in Table 14.4.8-1. Doppler feedback inherent in the fuel also limits the nuclear power excursion.

14.4.8.4 Disposition and Justification

This event is not a concern in Modes 4-6 as all control rods are required to be fully inserted per Technical Specifications and no one CEA possesses enough reactivity worth to overcome the minimum allowed shutdown margin. The fuel energy content is maximized by starting from rated power initial conditions, so the consequences of this event are bounding for power operating initial conditions. However, because of the complex interaction of the ejected rod worth and ejected peaking factor (which are maximized at hot zero power (HZP) operating conditions, and Doppler feedback effects, it is difficult to bound the consequences of the event for either rated power or HZP operating conditions without additional analysis. Therefore, the consequences of this event are analyzed for both rated power and HZP operating conditions. Separate evaluations are performed for deposited enthalpy, DNBR, and RCS pressurization for each of these initial operating conditions.

In addition to the rod ejection, this event is characterized by a small-break loss-of-coolant accident (SBLOCA) as the failure of the pressure housing is assumed to result in a breach of the primary coolant pressure boundary. The short-term aspects of the event are dominated by the rod ejection, while the long-term aspects are dominated by the SBLOCA. The limiting SBLOCA is evaluated in Event 14.6.5 and is typically a cold leg break. In the rod ejection, the break is more characteristic of a hot leg break and therefore will be bounded by the SBLOCA. Also in the rod ejection, a much earlier reactor trip occurs, resulting in lower powers and temperatures than in Event 14.6.5. It is concluded that the long-term aspects of the rod ejection are bounded by those of Event 14.6.5 for small breaks. Thus, only the short-term rod ejection consequences need be evaluated. Note also that the limiting 14.6.5 event occurs for rated power operating conditions.

The disposition of events for the Spectrum of Control Rod Ejection Accidents is summarized in Table 14.4.8-2.

14.4.8.5 Definition of Events Analyzed

Due to the complex interaction of the ejected rod worth, ejected peaking factor and Doppler feedback effects, it is difficult to bound the consequences of the event for either rated power or HZP operating conditions without analysis. Therefore, each of these conditions were evaluated at both BOC and EOC for deposited enthalpy, DNBR and pressurization concerns.

For the evaluation of the DNBR and pressurization consequences, concurrent loss of offsite power is assumed. No credit is taken for the variable overpower trip in the analysis of the pressurization consequences of a control rod ejection.

14.4.8.6 Analysis Results

The hot full-power (HFP) control rod ejection event was determined to deposit more energy into the primary system than the event initiated from HZP. Therefore, in terms of the event acceptance criteria, the HFP event poses a greater challenge than the HZP event. For this analysis, the event was assumed to initiate from HFP at 102% of rated full power.

To assess the acceptability of the outcome of an HFP rod ejection event, two cases were examined. The first case determines the maximum pressurization potential of the primary system during this event. The second case evaluates the MDNBR. For both the maximum pressurization and minimum DNB case, BOC and EOC kinetics were employed to establish the respective limiting cases.

The limiting minimum DNB case is calculated to occur for BOC kinetics. Core boundary conditions used to evaluate the DNBR conservatively account for depressurization due to the postulated breach in the CEDM housing. The MDNBR calculation is performed with the methodology described in Reference 14.4-5. As a consequence of this event, less than 11.5% of the fuel rods are calculated to fail due to penetration of DNBR limits. The responses of key system parameters are shown in Figures 14.4.8–1 to 14.4.8–6.

The maximum pressurization case occurs for EOC kinetics. The peak RCS pressure at the bottom of the reactor vessel remains below 110% of the pressure vessel design limit. The peak RCS pressure is conservatively calculated to be 2748 psia. Key system parameters for the overpressure case are plotted in Figures 14.4.8–7 to 14.4.8–12.

The sequence of events for the Control Rod Ejection transient is given in Tables 14.4.8-3 and 14.4.8-4.

The deposited enthalpy portion of the rod ejection accident has been evaluated with the procedures developed in the Generic Rod Ejection Analysis (Reference 14.4-3). The ejected rod worths and hot pellet peaking factors were calculated using the PRISM code. No credit was taken for the power flattening effects of Doppler or moderator feedback in the calculation of ejected rod worths or resultant peaking factors. The calculations performed used a full-core three-dimensional PRISM model. The pellet energy deposition resulting from an ejected rod was conservatively evaluated explicitly for BOC and EOC conditions. The rod ejection accident was found to result in an energy deposition of less than the 280 cal/g limit as stated in Regulatory Guide 1.77. The significant parameters for the analyses, along with the results, are summarized in Table 14.4.8-5.

14.4.8.7 Conclusion

The maximum RCS pressure does not exceed 110% of the design pressure. Less than 11.5% of the core will experience fuel failure due to penetration of DNBR limits. Deposited enthalpy is less than the limit of 280 cal/g.

14.4.8.8 Radiological Consequences

Two release paths are considered independently for the rod ejection accident. Each release path is considered independently as the only one available. The actual doses for the accident would be a composite of doses resulting from portions of the release going out the two different pathways.

Scenario 1: A postulated CREA that releases the failed fuel activity into the RCS, which is released in its entirety, into containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment and then released at containment Technical Specification leak rate. Releases occur via the Millstone stack and the Unit 2 containment.

Scenario 2: A postulated CREA that releases the failed fuel activity into the RCS which is then transmitted to the secondary side via steam generator tube leakage. The condenser is assumed to be unavailable due to a loss of offsite power. Releases occur from both steam generators via the MSSVs and the ADVs.

Scenario 1: Containment Release

Subsequent to a CREA, the following activity is assumed to be instantaneously released and to homogeneously mix in the free volume of containment.

- 11.5% of the core gap activity and consideration of a peaking factor of 1.69

The activity transport model assumes full enclosure building bypass until the enclosure building has achieved drawdown to negative pressure. After the drawdown time, 1.4% of the activity that leaks from the containment completely bypasses the enclosure building for the first 24 hours of the accident, after which it is reduced by 50% for the duration of the accident.

Scenario 2: Release via the MSSVs/ADVs

Following a CREA, the activity available for release via the secondary system consists of:

- Steam Generator tube leakage containing 11.5% of the core activity and consideration of a peaking factor of 1.69

It is assumed that offsite power is lost, therefore the main condenser is not available and releases are via the MSSVs/ADVs from both steam generators. The releases from both steam generators continue for about 16 hours until shutdown cooling commences. During this 16 hour period, reactor coolant is assumed to leak into the steam generators at a maximum leak rate of 150 gpd

per steam generator. The noble gases that enter the steam generators are released directly to the environment without holdup while the iodine activity is released from the steam generators in proportion to the steaming rate and the partition factor.

The assumptions used for the radiological consequences of a CREA are listed in Table 14.4.8-6. The radiological consequences of a CREA are presented in Table 14.4.8-7 and are within the guidelines of 10 CFR 50.67 and Regulatory Guide 1.183.

14.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BOILING WATER REACTOR)

Millstone Unit 2 is not a Boiling Water Reactor and as such this event is not applicable.

14.4.10 REFERENCES

14.4-1 Technical Specifications for Millstone Unit 2, Docket Number 50-336.

14.4-2 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

14.4-3 “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” XN-NF-78-44(A), Exxon Nuclear Company, October 1983.

14.4-4 SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” EMF-2310(P)(A), Revision 1, Framatome ANP Inc., June 2004.

14.4-5 “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors,” EMF-1961(P)(A), Revision 0, Siemens Power Corporation, July 2000.

TABLE 14.4.1-1 AVAILABLE REACTOR PROTECTION FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW-POWER STARTUP CONDITION EVENT

Reactor Operational Mode	Reactor Protection
1	Considered as FSAR Event 14.4.2
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit on Variable Overpower Pretrip Alarm (1)
3	Variable Overpower Trip
	Rod Withdrawal Prohibit on Variable Overpower Pretrip Alarm (1)
4-6	Not a Credible Event; No Analysis Required

(1) Control grade. Provides defense in depth. Not credited in any safety analysis event.

TABLE 14.4.1-2 DISPOSITION OF EVENTS FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW-POWER STARTUP CONDITION EVENT

Reactor Operational Mode	Disposition
1	Considered as FSAR Event 14.4.2
2	Bounded by Mode 3
3	Analyze at 2,000 psia
4-6	Not a Credible Event; No Analysis Required

TABLE 14.4.1-3 EVENT SUMMARY FOR THE UNCONTROLLED BANK WITHDRAWAL FROM LOW-POWER EVENT

Event	Time (seconds)
Bank Withdrawal Begins	0.00
Variable Overpower Trip Setpoint Reached	37.6
Scram Reactivity Insertion Begins	38.8
Peak Nuclear Power (155.7% of rated)	39.0
Peak Core Heat Flux (58.0% of rated)	40.0
Peak Reactor Vessel Upper Plenum Coolant Temperature	41.2

FIGURE 14.4.1–1 REACTOR POWER LEVEL FOR LOW POWER BANK WITHDRAWAL

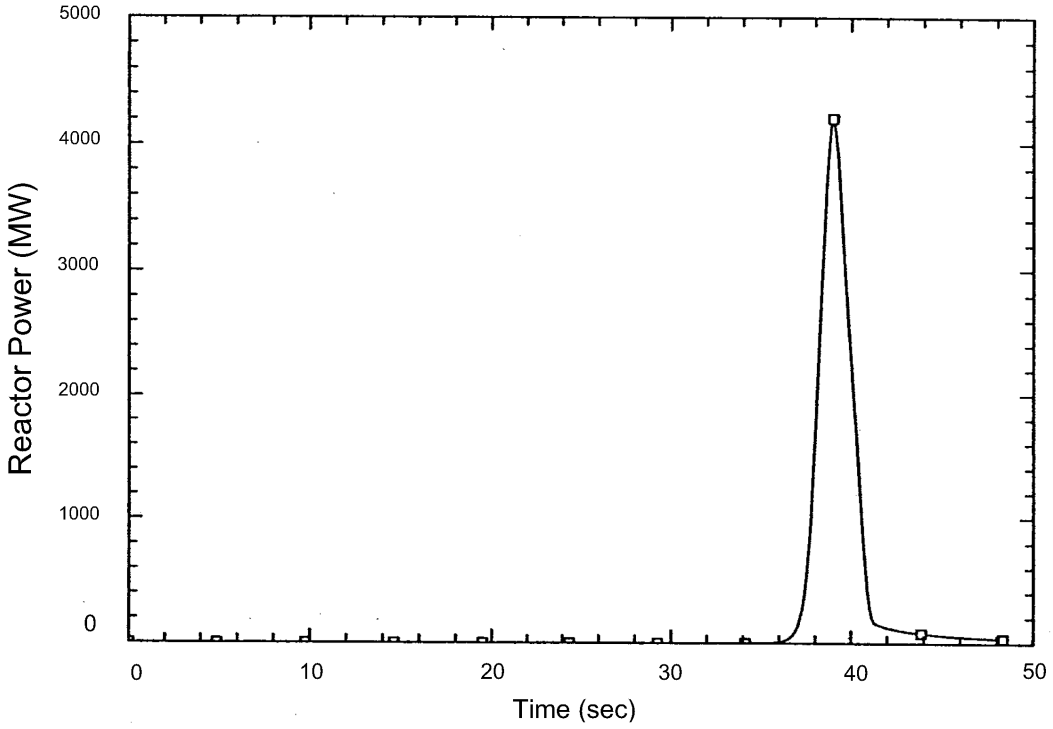


FIGURE 14.4.1–2 CORE AVERAGE HEAT FLUX FOR LOW POWER BANK WITHDRAWAL

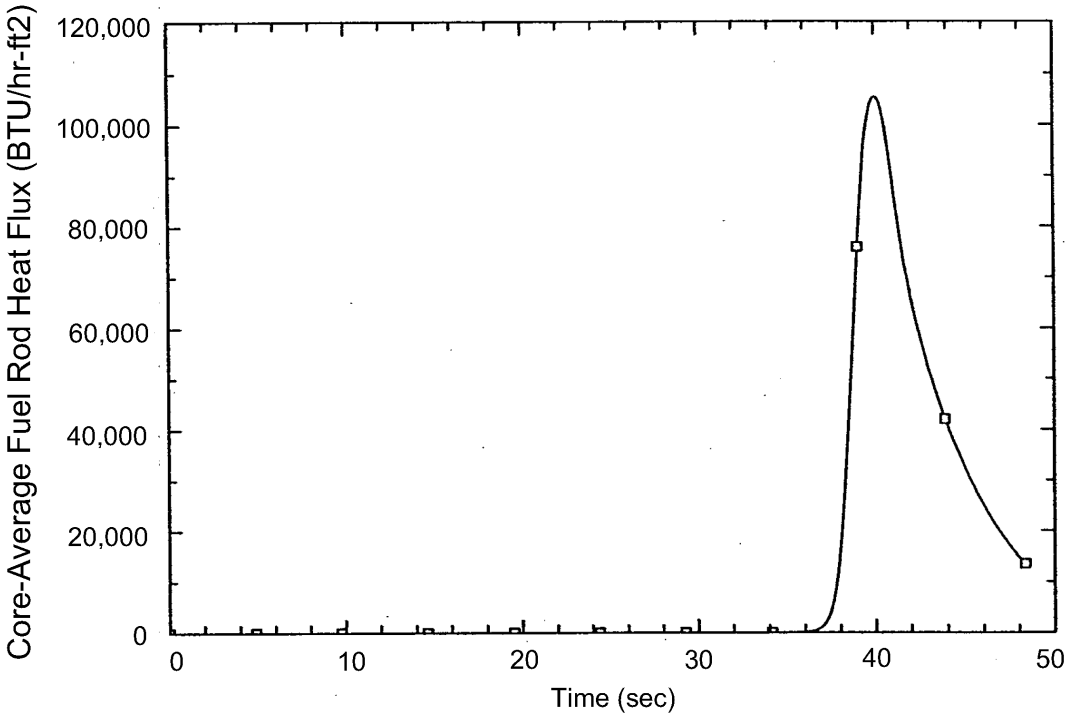


FIGURE 14.4.1–3 REACTOR COOLANT TEMPERATURES FOR LOW POWER BANK WITHDRAWAL

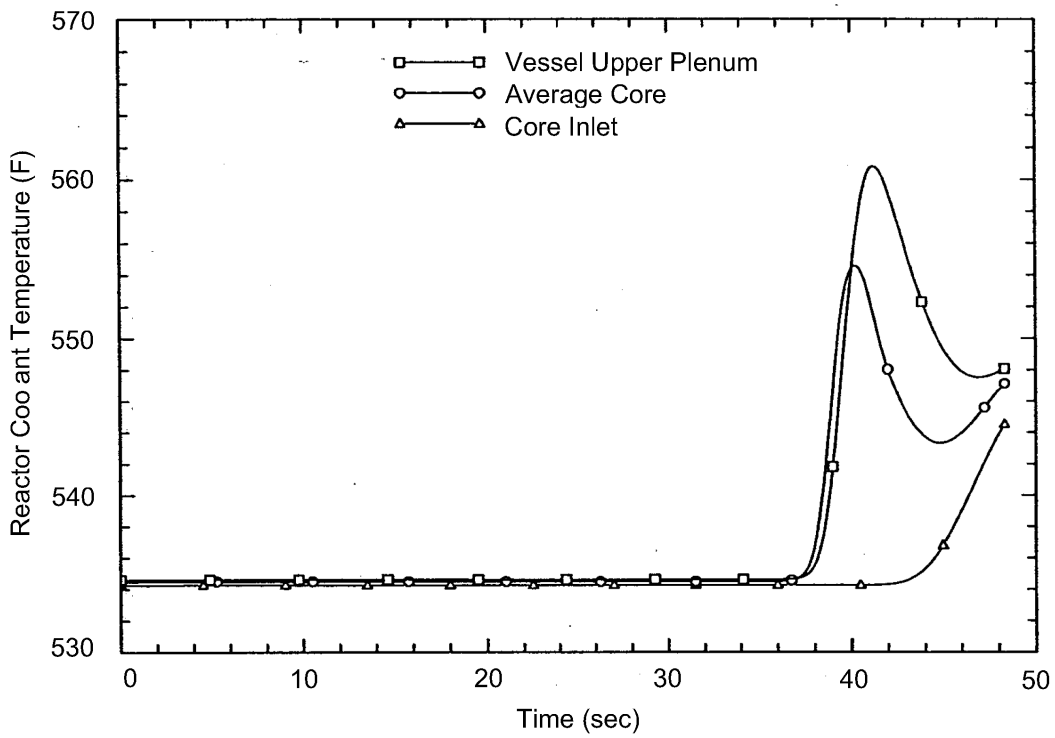


FIGURE 14.4.1–4 PRESSURIZER PRESSURE FOR LOW POWER BANK WITHDRAWAL

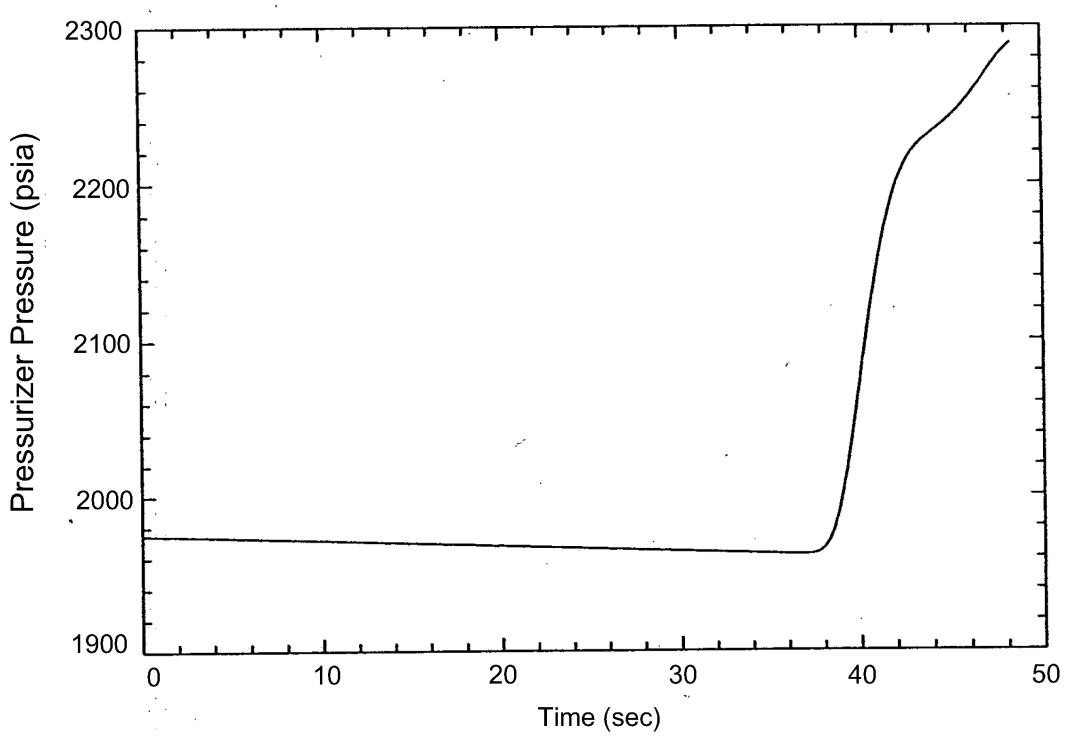


FIGURE 14.4.1–5 REACTIVITIES FOR LOW POWER BANK WITHDRAWAL

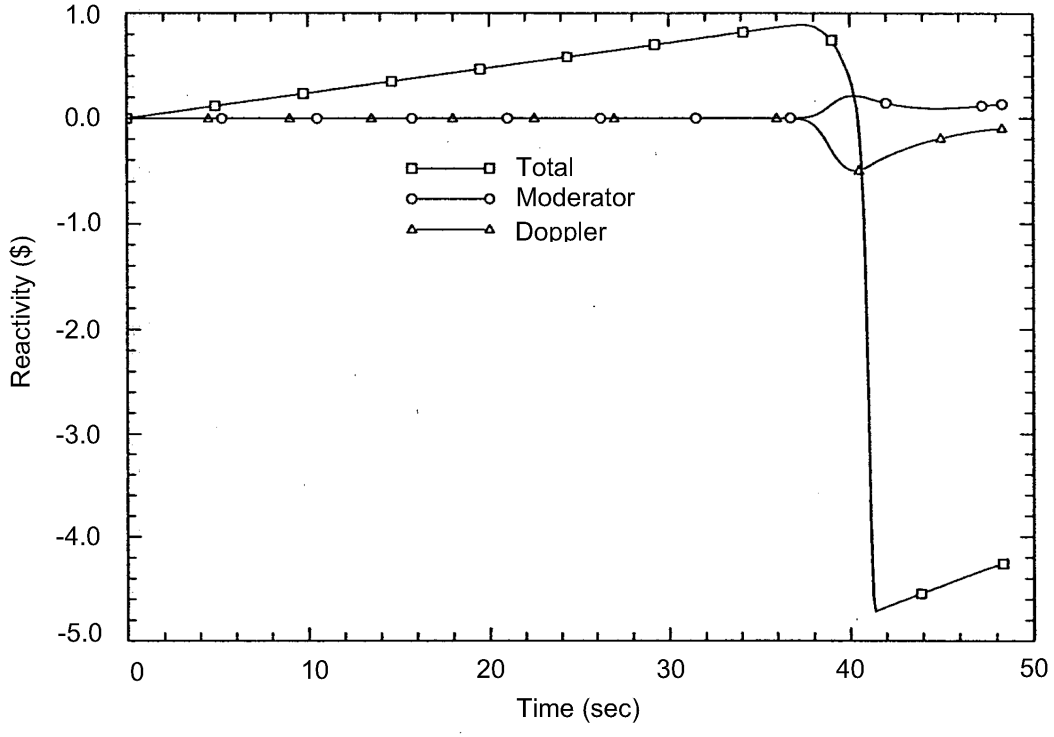


TABLE 14.4.2-1 AVAILABLE REACTOR PROTECTION FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL AT POWER EVENT

Reactor Operational Mode	Reactor Protection
1	Variable Overpower Trip
	Local Power Density Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit Action on Variable Overpower or TM/LP Pretrip Alarm (1)
2-6	Not considered in this section

(1) Control grade. Provides defense in depth. Not credited in any safety analysis event.

TABLE 14.4.2-2 DISPOSITION OF EVENTS FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL AT POWER EVENT

Reactor Operational Mode	Disposition
1	Analyze at rated power
2-6	No analysis required; not considered in this section

TABLE 14.4.2-3 EVENT SUMMARY FOR THE UNCONTROLLED ROD/BANK WITHDRAWAL EVENT FOR THE LIMITING 100% POWER CASE

Event	Time (seconds)
Start Rod Withdrawal	0.00
Letdown Flow Valve Open	0.00
TM/LP Trip Signal	511.96
Turbine Stop Valve Closed	512.88
Peak Power Level	513.32
MDNBR	513.36
Peak Core Average Temperature	513.39
Steam Line Safety Valves Open	514.94
Peak Steam Dome Pressure	516.96

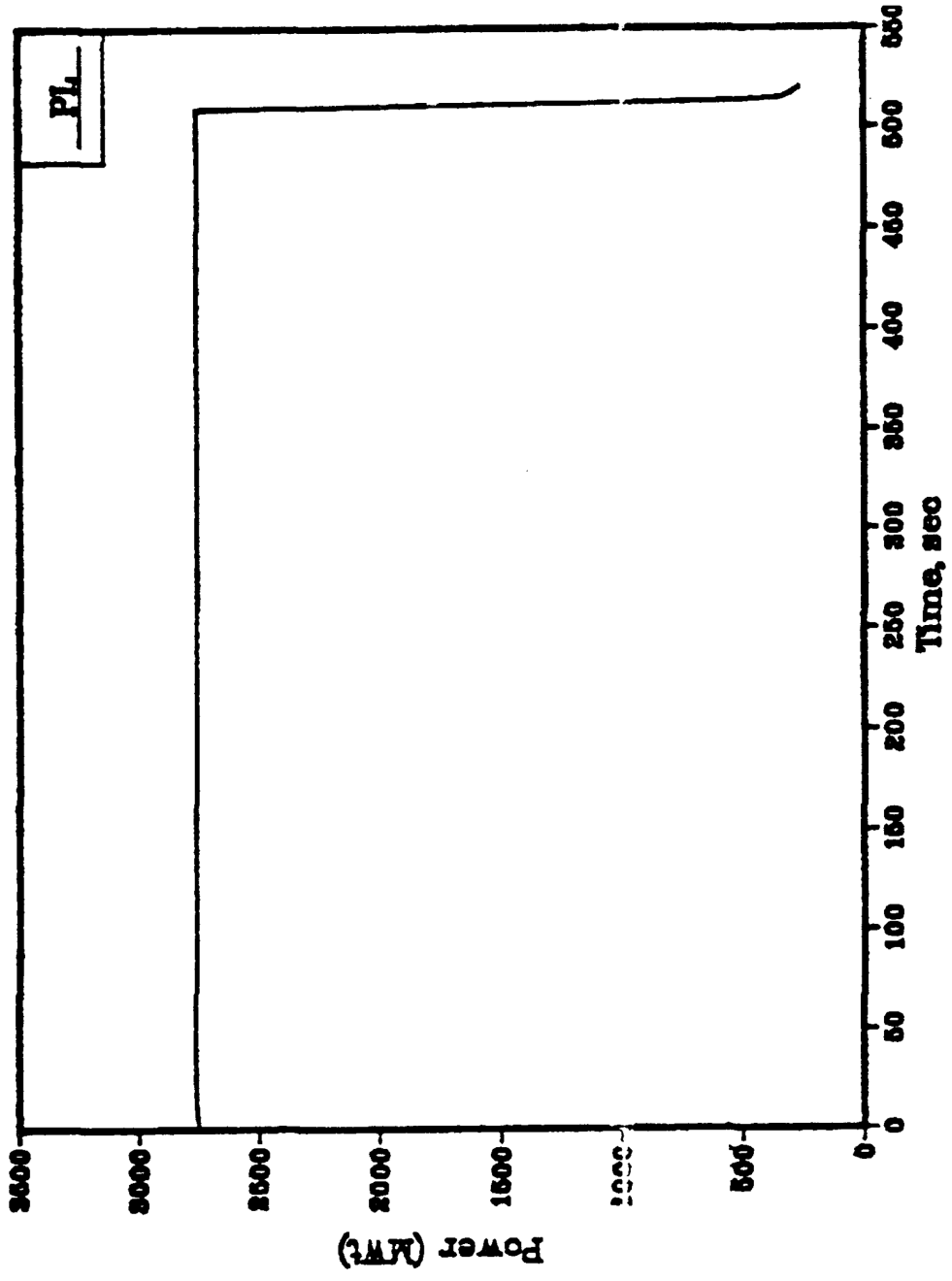


FIGURE 14.4.2–1 REACTOR CORE POWER FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

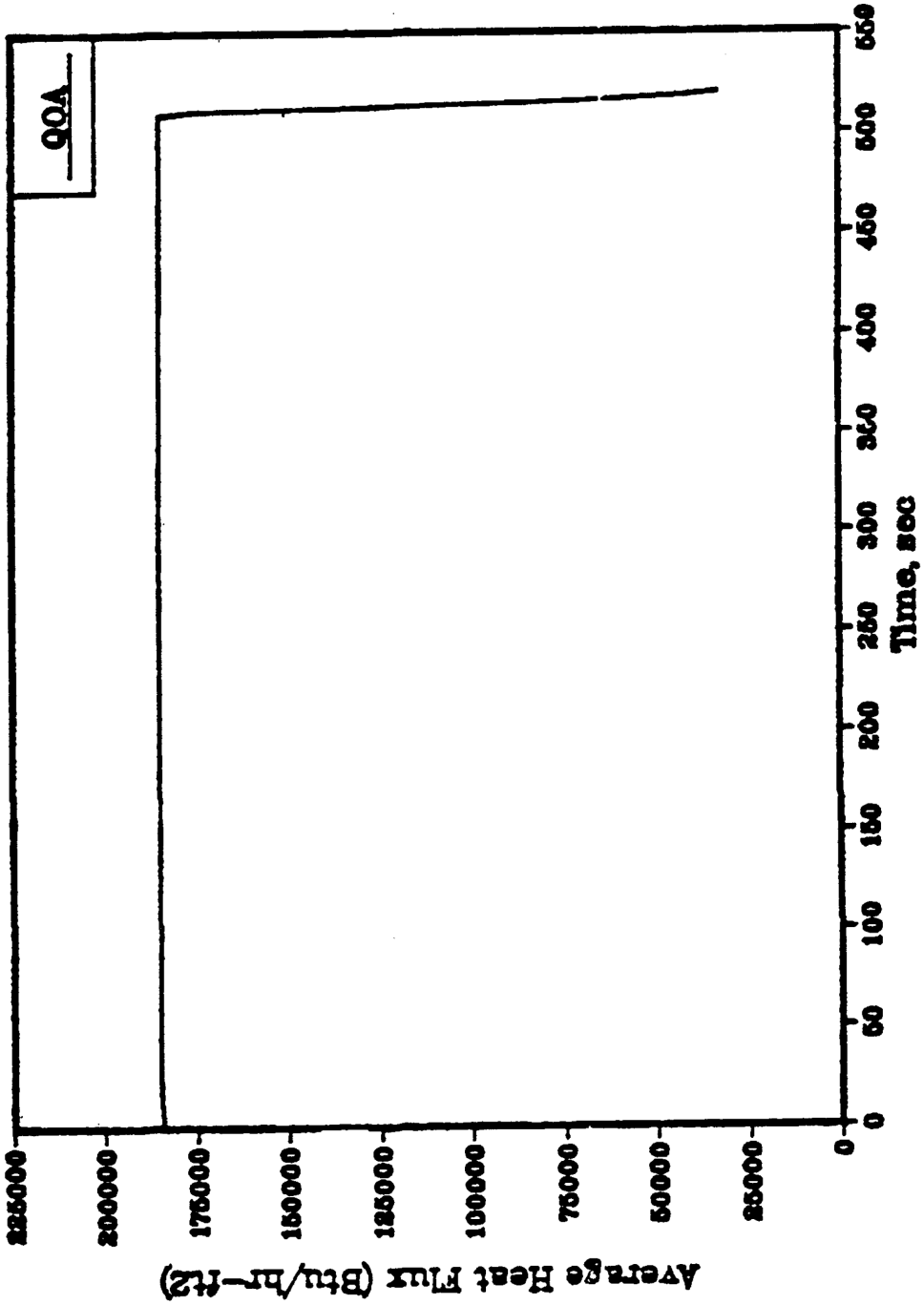


FIGURE 14.4.2–2 CORE AVERAGE HEAT FLUX FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

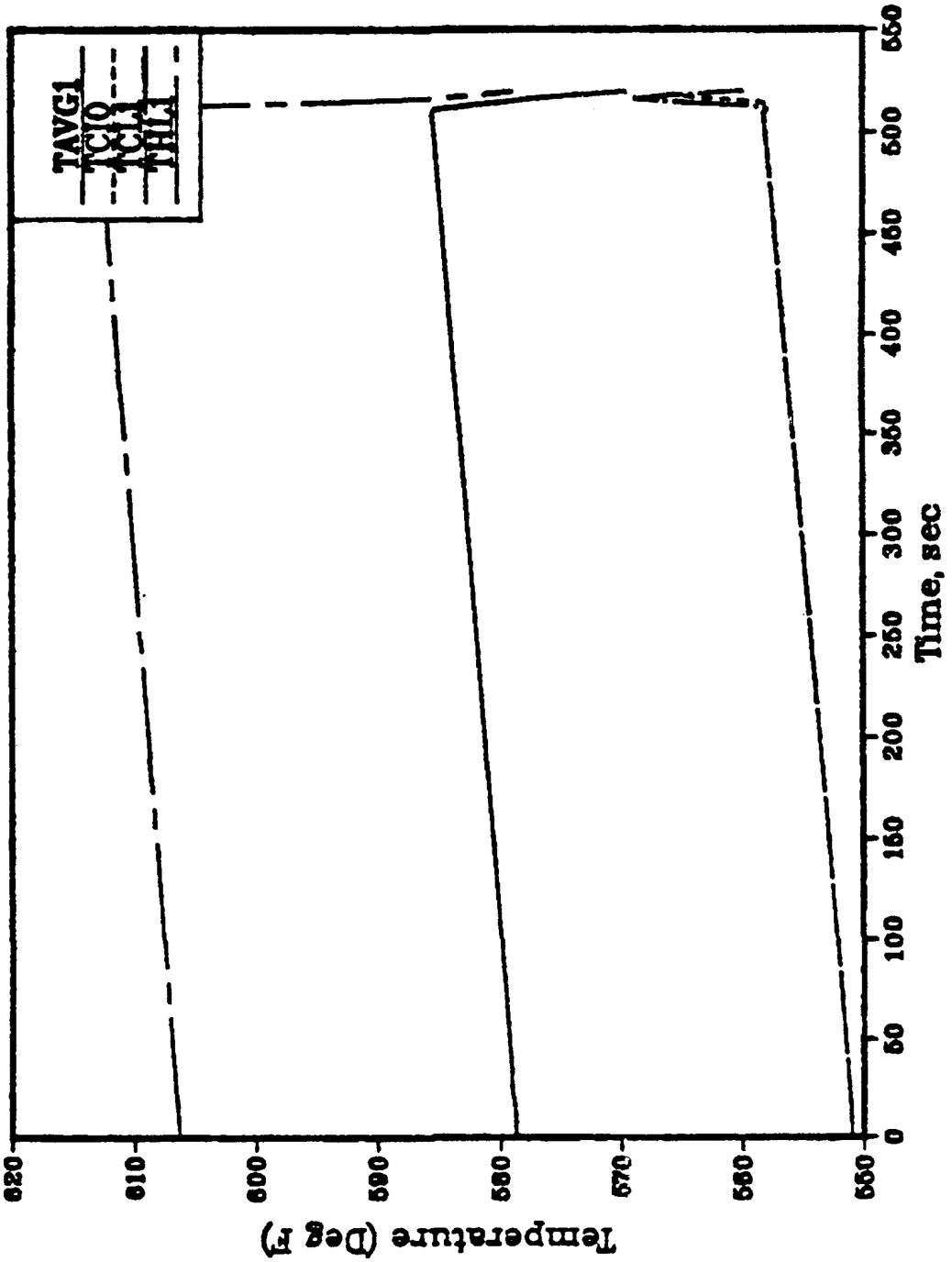


FIGURE 14.4.2–3 REACTOR COOLANT SYSTEM TEMPERATURES FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

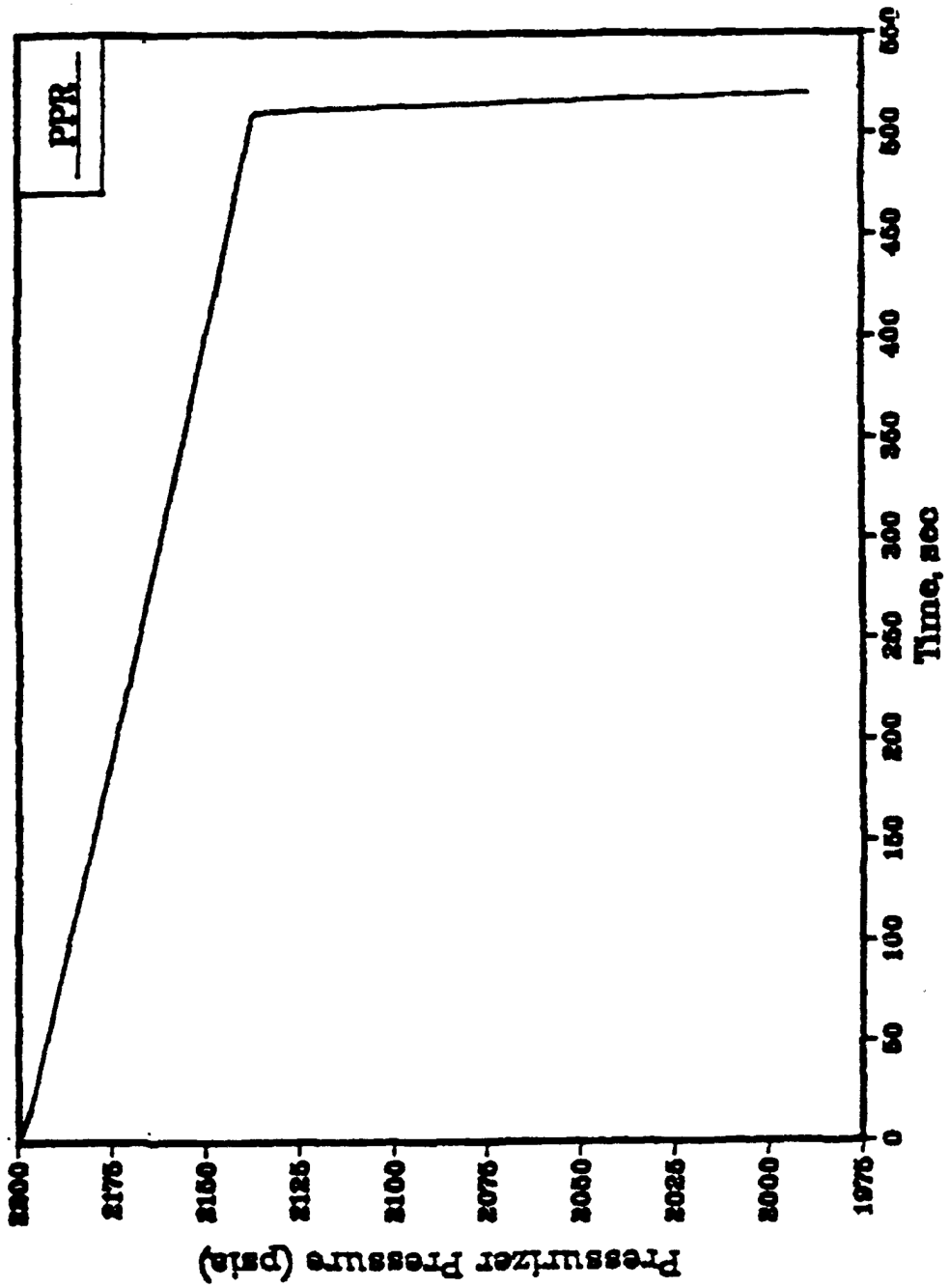


FIGURE 14.4.2–4 PRESSURIZER PRESSURE FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

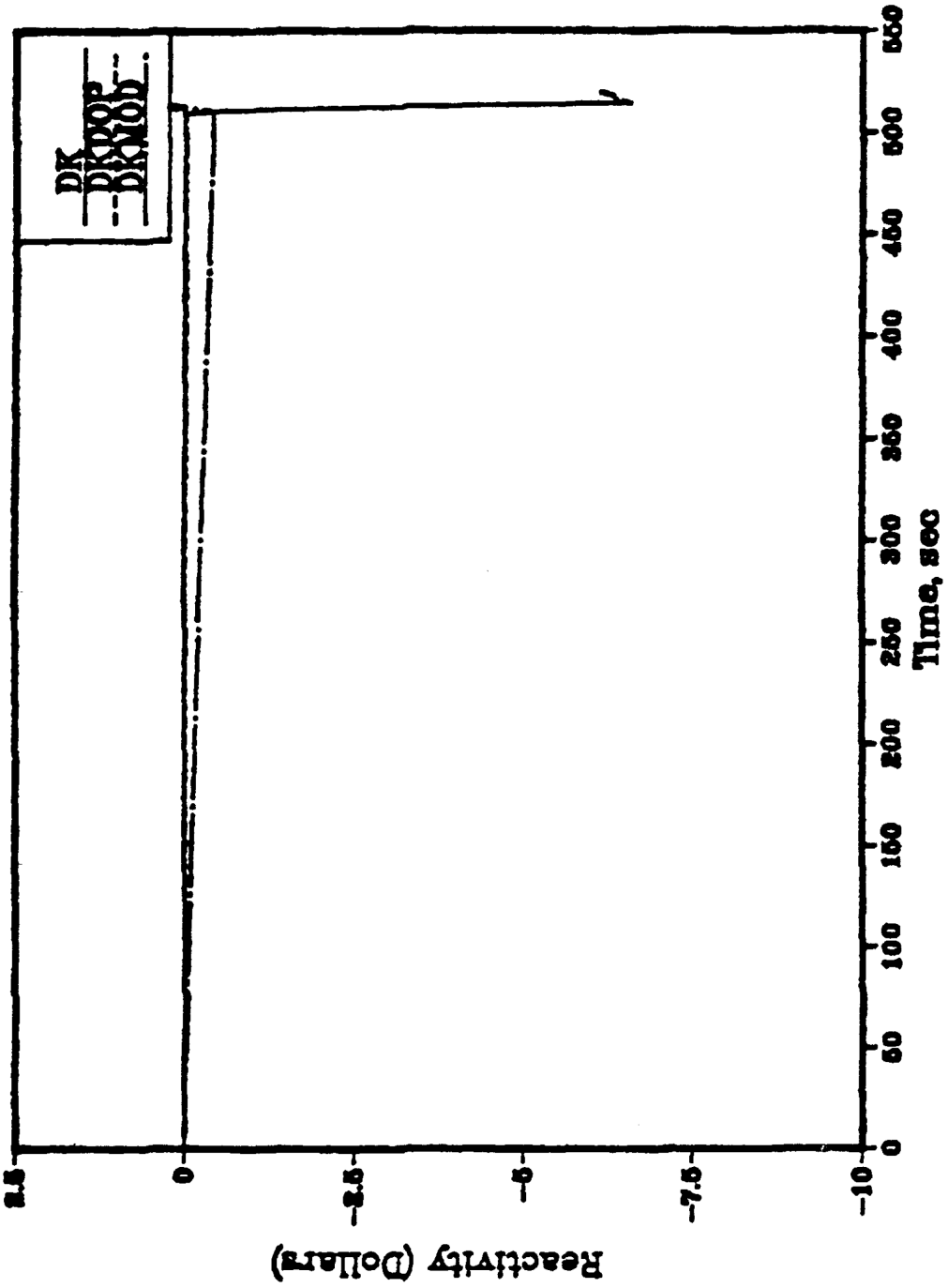


FIGURE 14.4.2–5 REACTIVITIES FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

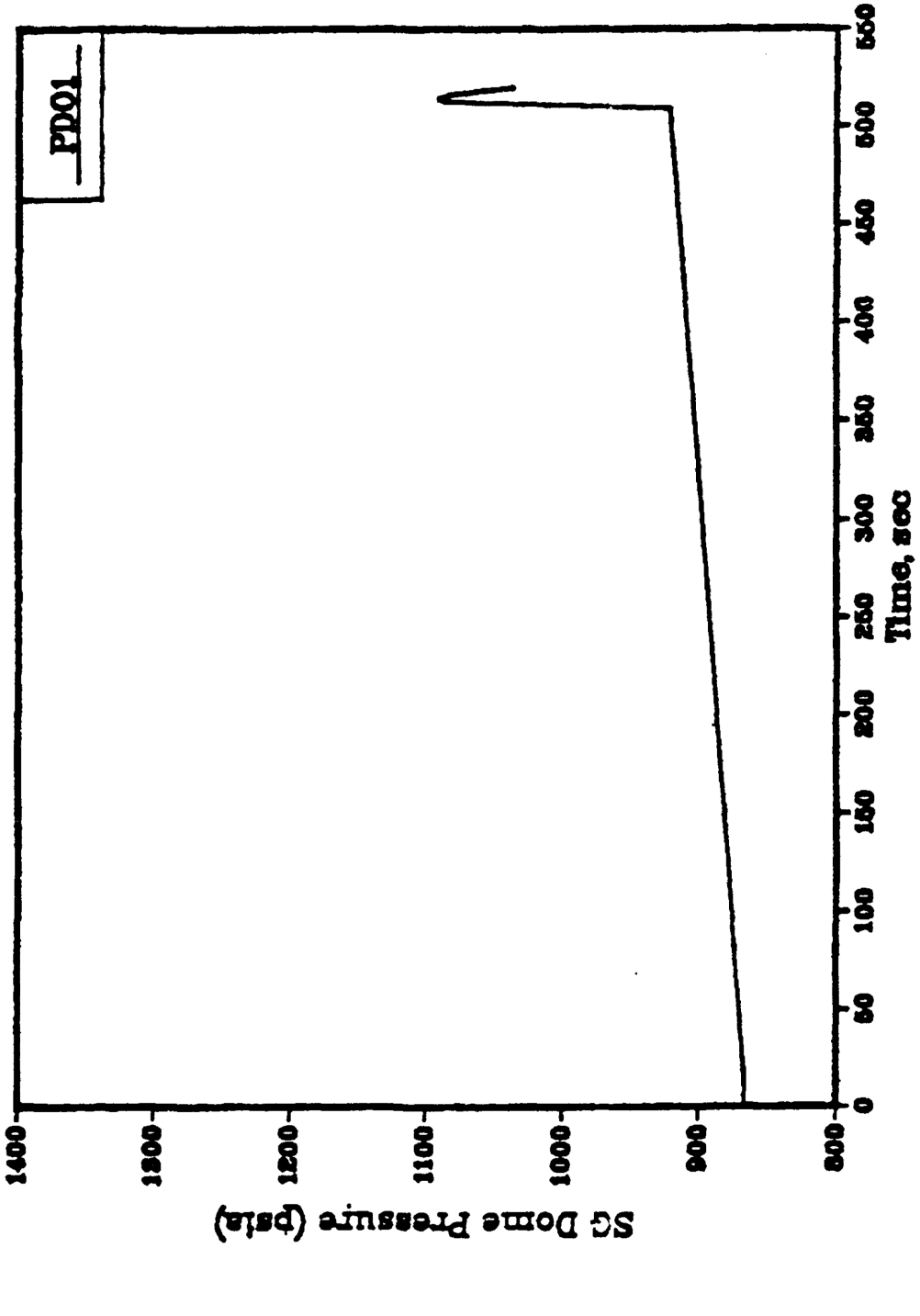


FIGURE 14.4.2–6 SECONDARY PRESSURE FOR AN UNCONTROLLED BANK WITHDRAWAL AT POWER

TABLE 14.4.3.1-1 AVAILABLE REACTOR PROTECTION FOR THE DROPPED CONTROL ROD/BANK EVENT

Reactor Operational Mode	Reactor Protection
1	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	Local Power Density Trip
	Available Thermal Margin (1)
2	Variable Overpower Trip
	Available Thermal Margin (1)
3-6	No Significant Consequences for these Reactor Operational Modes

(1) Provides defense in depth

TABLE 14.4.3.1-2 DISPOSITION OF EVENTS FOR THE DROPPED CONTROL ROD/ BANK EVENT

Reactor Operational Mode	Disposition
1	Analyze at rated power
2	Bounded by the above; no analysis required
	Available Thermal (1)
3-6	No analysis required

(1) Provides defense in depth.

TABLE 14.4.3.1-3 EVENT SUMMARY FOR THE LIMITING DROPPED CONTROL ROD/BANK CASE

Event	Time (seconds)
Minimum Worth Control Rod Drops into Core	0.0
DT Power Exceeds VHP Trip Setpoint	88.3
MDNBR and Peak LHGR occur	89.5
Analysis Terminated	100.0

FIGURE 14.4.3.1–1 REACTOR POWER LEVEL FOR THE LIMITING DROPPED CONTROL ROD/BANK CASE

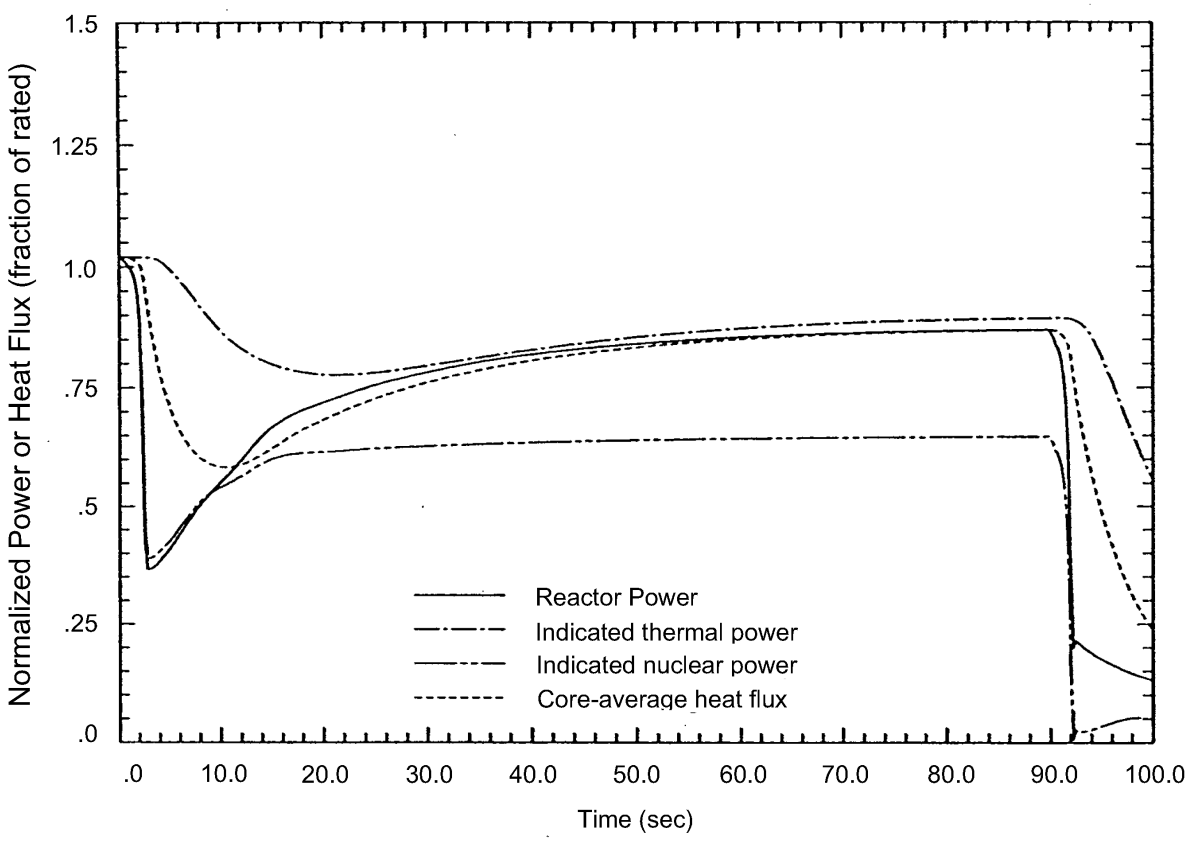


FIGURE 14.4.3.1–2 REACTOR COOLANT SYSTEM TEMPERATURES FOR THE LIMITING DROPPED CONTROL ROD/BANK CASE

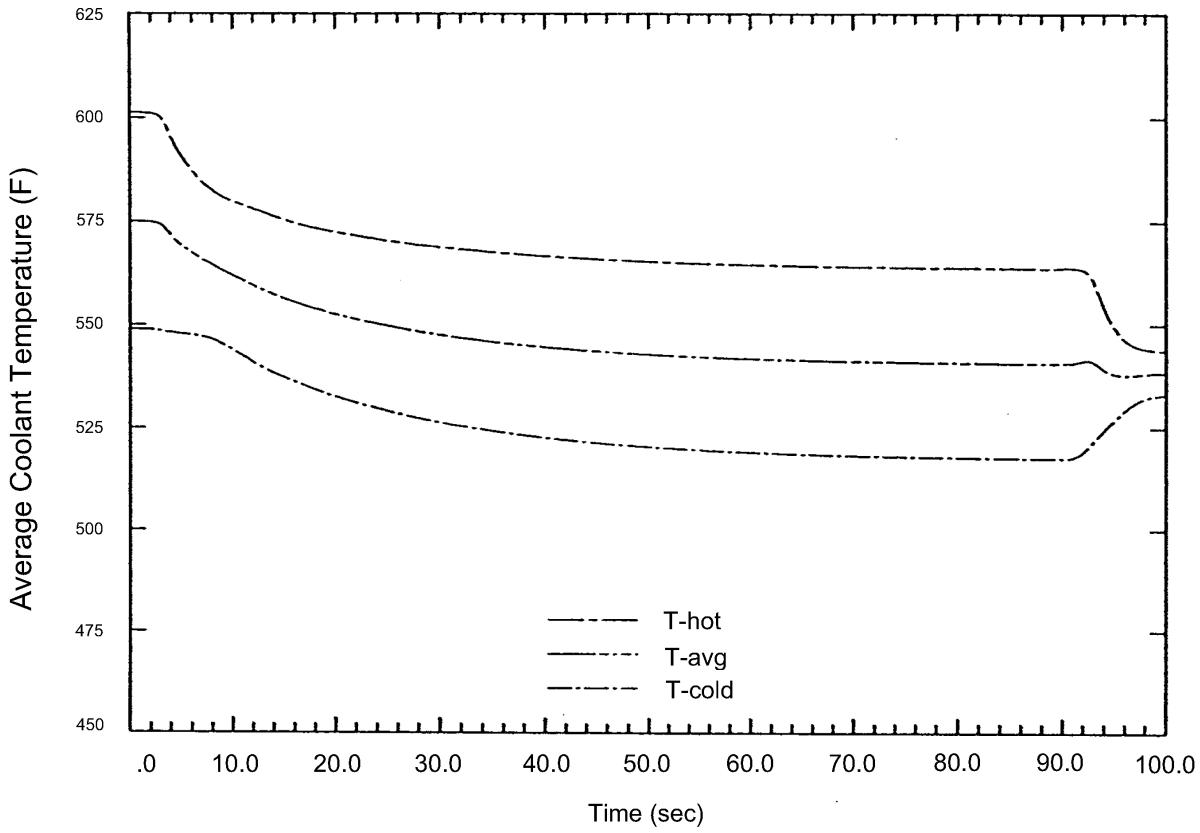


FIGURE 14.4.3.1–3 PRESSURIZER PRESSURE FOR THE LIMITING DROPPED CONTROL ROD/BANK CASE

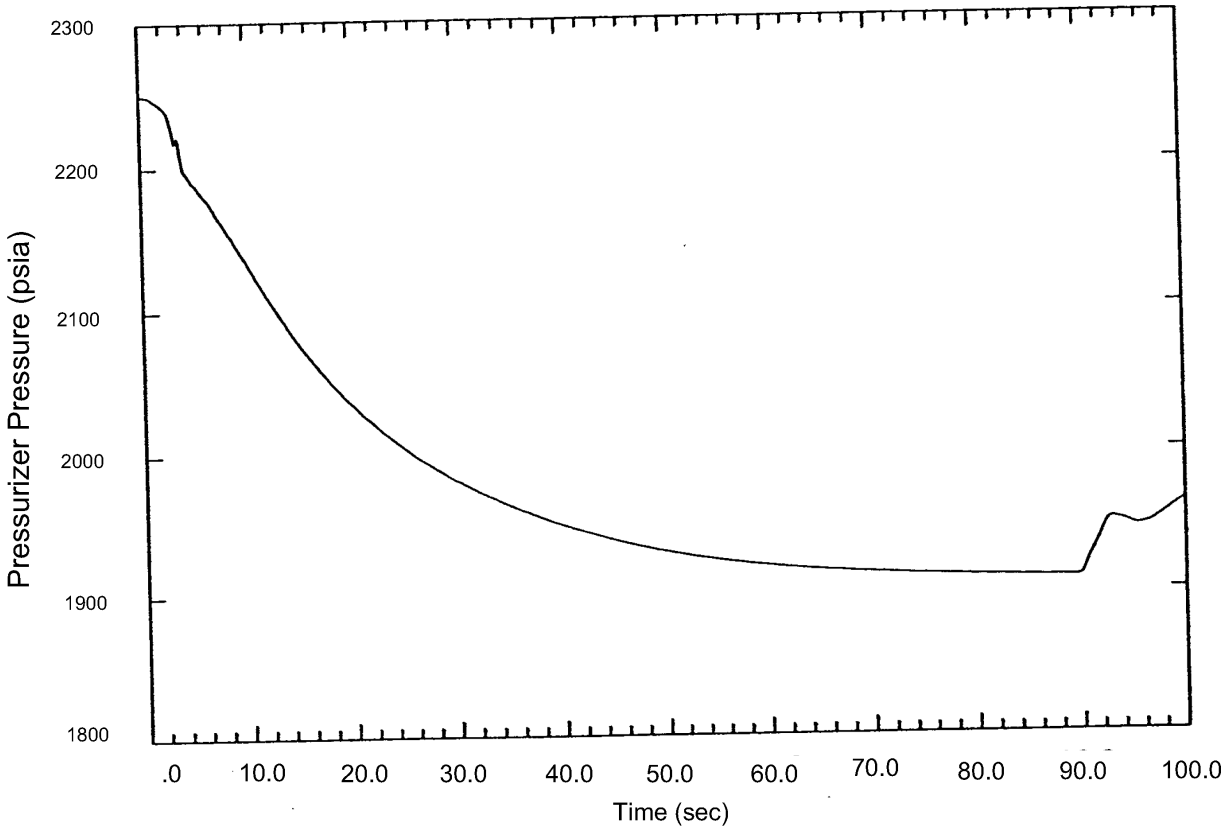


FIGURE 14.4.3.1–4 SECONDARY PRESSURE FOR THE LIMITING DROPPED CONTROL ROD/BANK CASE

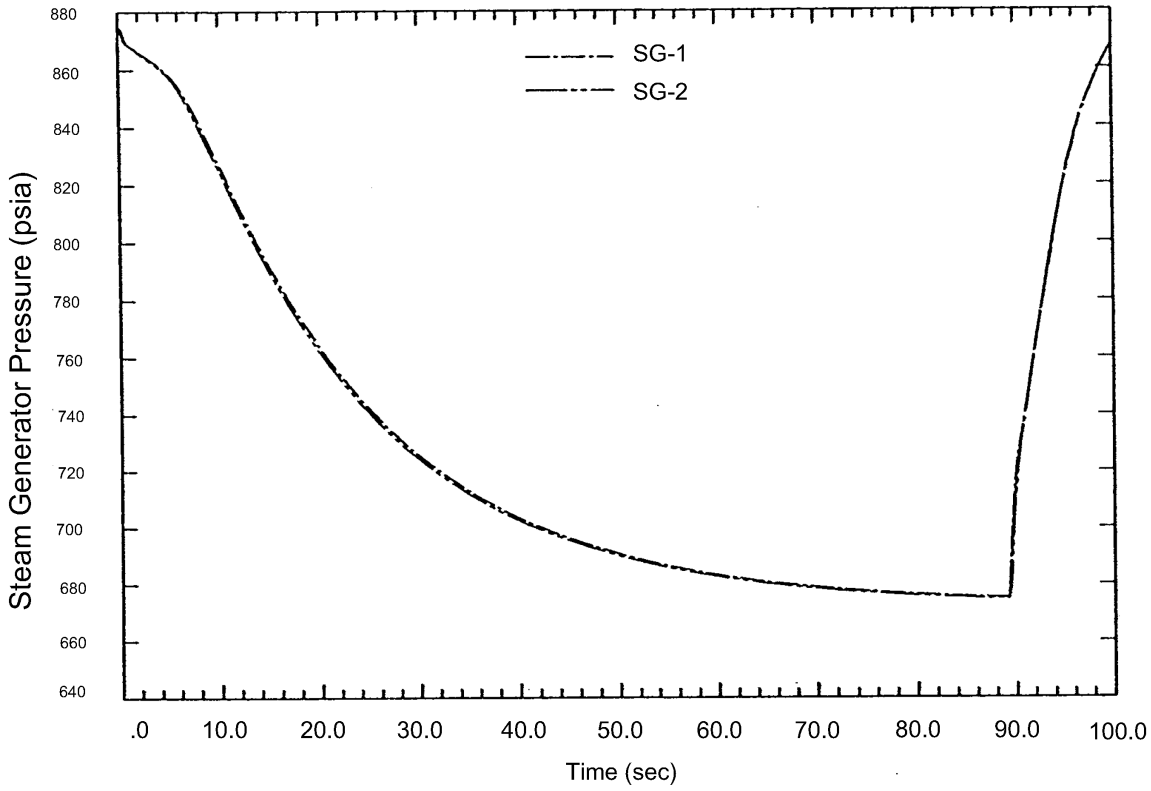


TABLE 14.4.3.5-1 AVAILABLE REACTOR PROTECTION FOR THE SINGLE CONTROL ROD WITHDRAWAL EVENT

Reactor Operational Mode	Reactor Protection
1	Variable Overpower Trip Local Power Density Trip Thermal Margin/Low Pressure Trip High Pressurizer Pressure Trip Rod Withdrawal Prohibit Action on Variable Overpower or TM/LP Pretrip Alarm (1)
2	Variable Overpower Trip High Pressurizer Pressure Trip Rod Withdrawal Prohibit on Variable Overpower Pretrip Alarm (1)
3	Variable Overpower Trip Rod Withdrawal Prohibit on Variable Overpower Pretrip Alarm (1)
4-6	Not a Credible Event; No Analysis Required

(1) Control grade. Provides defense in depth. Not credited in any safety analysis event.

TABLE 14.4.3.5-2 DISPOSITION OF EVENTS FOR THE SINGLE CONTROL ROD WITHDRAWAL EVENT

Reactor Operational Mode	Disposition
1	Analyze at rated power
2	Bounded by Mode 1 (1)
3	Bounded by the above
4-6	Not a Credible Event; No Analysis Required

(1) Mode 2 operation is dispositioned as bounded by Mode 1 operation because the MDNBR for an Uncontrolled Bank Withdrawal from Startup (Event 14.4.1) was found to be bounded by the MDNBR for an Uncontrolled Bank Withdrawal at Power (Event 14.4.2).

TABLE 14.4.4-1 AVAILABLE REACTOR PROTECTION

Reactor Operational Mode	Reactor Protection
1, 2, 6	Not Applicable
3-5	Technical Specification Requirements on Shutdown Margin and Reactor Coolant Pump Operation

TABLE 14.4.4-2 DISPOSITION OF EVENTS FOR THE STARTUP OF AN INACTIVE LOOP EVENT

Reactor Operational Mode	Disposition
1, 2, 6	Not Applicable
3-5	No analysis required; minimal consequences

TABLE 14.4.6-1 AVAILABLE REACTOR PROTECTION FOR CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT EVENT

Reactor Operational Mode	Reactor Protection
1	Local Power Density Trip
	Variable Overpower Trip
	Thermal Margin / Low Pressure Trip
	High Pressurizer Pressure Trip
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
3-6	Technical Specification Shutdown Margin Requirements
	Administrative Procedures
	Operator Response Time

TABLE 14.4.6-2 DISPOSITION OF EVENTS FOR THE CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT EVENT

Reactor Operational Mode	Disposition
1-6	Analyze for loss of shutdown margin

TABLE 14.4.6-3 SUMMARY OF RESULTS FOR THE BORON DILUTION EVENT ASYMMETRIC DILUTION FRONT MODEL

Reactor Operational Mode	Minimum Analytical Required SDC Flow to Satisfy Licensing Criteria for Operator Response Time (gpm) (1)	Licensing Criteria for Operator Response Time (minutes)
Mode 6	600	30
Mode 5	600	15
Mode 4	600	15

(1) The required flows do not include SDC flow measurement uncertainties. Plant procedures maintain the SDC flowrate at a higher value to account for instrument uncertainty.

TABLE 14.4.6-4 SUMMARY OF RESULTS FOR THE BORON DILUTION EVENT INSTANTANEOUS MIXING MODE

Reactor Operational Mode	Calculated Time to Loss of Shutdown Margin (minutes)	Licensing Criteria for Time to Loss of Shutdown Margin (minutes)
Mode 5	62	15
Mode 4	77	15
Mode 3	62	15
Mode 2	62	15

TABLE 14.4.8-1 AVAILABLE REACTOR PROTECTION FOR THE SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Reactor Operational Mode	Reactor Protection
1	Variable Overpower Trip
	Thermal Margin / Low Pressure Trip
	High Pressurizer Pressure Trip
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
3	Variable Overpower Trip
4-6	No Reactor Protection Required; Ejected Rod Worth Less than the Technical Specification Minimum Shutdown Margin. No Significant Consequence for this Operating Condition.

TABLE 14.4.8-2 DISPOSITION OF EVENTS FOR THE SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Reactor Operational Mode	Disposition
1	Analyze for short-term response. Long-term bounded by Event 14.6.5
2, 3	Analyze
4-6	No analysis required

TABLE 14.4.8-3 EVENT SUMMARY FOR A CONTROL ROD EJECTION (MAXIMUM PRESSURIZATION CASE)

Event	Time (seconds)
Control Rod Ejects	0.00
Charging Pumps On	0.00
Pressurizer Heaters On	0.00
Reactor Scram Signal	4.23
Peak Power	5.61
Pressurizer Safety Valves Open	5.94
Peak Core Average Temperature	6.53
Peak Pressurizer Pressure	7.08
Steam Line Safety Valves Open	9.60
Peak Steam Dome Pressure	11.63

TABLE 14.4.8-4 EVENT SUMMARY FOR A CONTROL ROD EJECTION MINIMUM DEPARTURE FROM NUCLEATE BOILING RATIO CASE

Event	Time (seconds)
Control Rod Ejects	0.00
Letdown Valve Open	0.00
Reactor Scram Signal	3.23
Peak Power	4.10
Peak Core Average Heat Flux	4.22
MDNBR	4.22
Peak Core Average Temperature	4.64
Peak Pressurizer Pressure	5.42
Steam Line Safety Valves Open	7.53
Peak Steam Dome Pressure	9.56

TABLE 14.4.8-5 DNBOUNDING BEGINNING OF CYCLE/END OF CYCLE EJECTED ROD ANALYSIS

	HFP		HZP	
	Value	**Contribution[(a)] to Energy Deposition**	**Value**	**Contribution [(a)] to Energy Deposition**
A.	103.4 cal/g	---	18.0 cal/g	---
B. Generic Initial Fuel Enthalpy	40.8 cal/g	---	16.7 cal/g	---
C. Delta Initial Fuel Enthalpy	62.6 cal/g	62.6 cal/g	1.3 cal/g	1.3 cal/g
D. Maximum Control Rod Worth	200 pcm	131.4 cal/g	800 pcm	109.1 cal/g
E. Doppler Coefficient	-0.8 pcm/°F	1.17	-0.8 pcm/°F	1.31
F. Delayed Neutron Fraction	0.0045	1.06	0.0045	1.28
G. Power Peaking Factor	6.0	---	14.0	---
Total Fuel Enthalpy		240.6 cal/g [(b)]		185.1 cal/g [(b)]

(a) The contribution to the total pellet energy deposition is a function of initial fuel enthalpy, maximum control rod worth, Doppler coefficient, and delayed neutron fraction. The energy deposition contribution values and factors are derived from data calculated in the generic analysis of the control rod ejection transient document XN-NF-78-44.

(b) Total pellet energy deposition (cal/g) is calculated by the equation: Total (cal/g) = (C+D) * (E) * (F).

TABLE 14.4.8-6 CREA RADIOLOGICAL ANALYSIS ASSUMPTIONS

Core Power Level	2754 Mwt
Failed Fuel Percentage	11.5%
Peaking Factor	1.69
Percentage of Core Activity in Gap	10% Noble Gas
	10% Halogens
Composition of Iodine in the Core Gap (particulate/elemental/organic)	
	95 / 4.85 / 0.15 % - containment
	0 / 97 / 3 % - secondary side
Reactor Coolant Mass	430,000 lbs
Steam Generator Minimum Mass	100,000 lbs/SG
Containment Free Volume	1.899E6 ft^3
Containment Leak Rate	0-1 day: 0.5% vol per day
	1-30 days: 0.25% vol per day
Enclosure Building Bypass Fraction	1.4%
Time Before Enclosure Building Filtration System (EBFS) is Fully Functional	170 seconds
Enclosure Building Filter Efficiency (particulate/elemental/organic)	70 / 70 / 70% (1)
Site Boundary Meteorology	X/Q's
Ground Level Release	
EAB: 0 - 2 hr	3.66E-04
LPZ: 0 - 4 hr	4.80E-05
4 - 8 hr	2.31E-05

TABLE 14.4.8-6 CREA RADIOLOGICAL ANALYSIS ASSUMPTIONS (CONTINUED)

8 - 24 hr	1.60E-05
24 - 96 hr	7.25E-06
96 - 720 hr	2.32E-06
Millstone Stack Release (includes fumigation):	
EAB: 0 - 2 hr	1.00E-04
LPZ: 0- 4 hr	2.69E-05
4 - 8 hr	1.07E-05
8 - 24 hr	6.72E-06
24 - 96 hr	2.46E-06
96 - 720 hr	5.83E-07
Control Room Breathing Rate	3.5E-04 m^3/sec
Control Room Isolation Time post-accident	105 seconds
Control Room Intake Prior to Isolation	800 cfm
Control Room Inleakage During Isolation	200 cfm
Control Room Emergency Filtered Recirculation Rate (from 1 hour after isolation)	2,250 cfm

Control Room Intake Dispersion Factors (sec/m^3)

	Ground	Millstone Stack	ADV
0 - 2 hr	3.00E-3	2.51E-4	7.40E-3
2 - 4 hr	1.87E-3	2.51E-4	5.71E-3
4 - 8 hr	1.87E-3	1.96E-5	5.71E-3
8 - 24 hr	6.64E-4	5.46E-6	2.13E-3
24 - 96 hr	5.83E-4	3.43E-7	1.74E-3
96 - 720 hr	4.97E-4	6.44E-9	1.43E-3

Control Room Free Volume	35,656 ft^3
Control Room Filter Efficiency (particulate/elemental/organic)	70 / 70 / 70 %(1)
Dose Conversion Factors	Federal Guideline Reports 11 and 12

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for filter efficiencies of 90% for all iodine species.

TABLE 14.4.8-7 RADIOLOGICAL CONSEQUENCES OF A CREA

CREA	EAB, rem-TEDE	LPZ, rem-TEDE	Control Room, rem-TEDE
Containment Release	5.4 E -01	4.7 E -01	1.6E+00
Secondary Side Release	7.9 E -01	1.8 E -01	3.9E+00

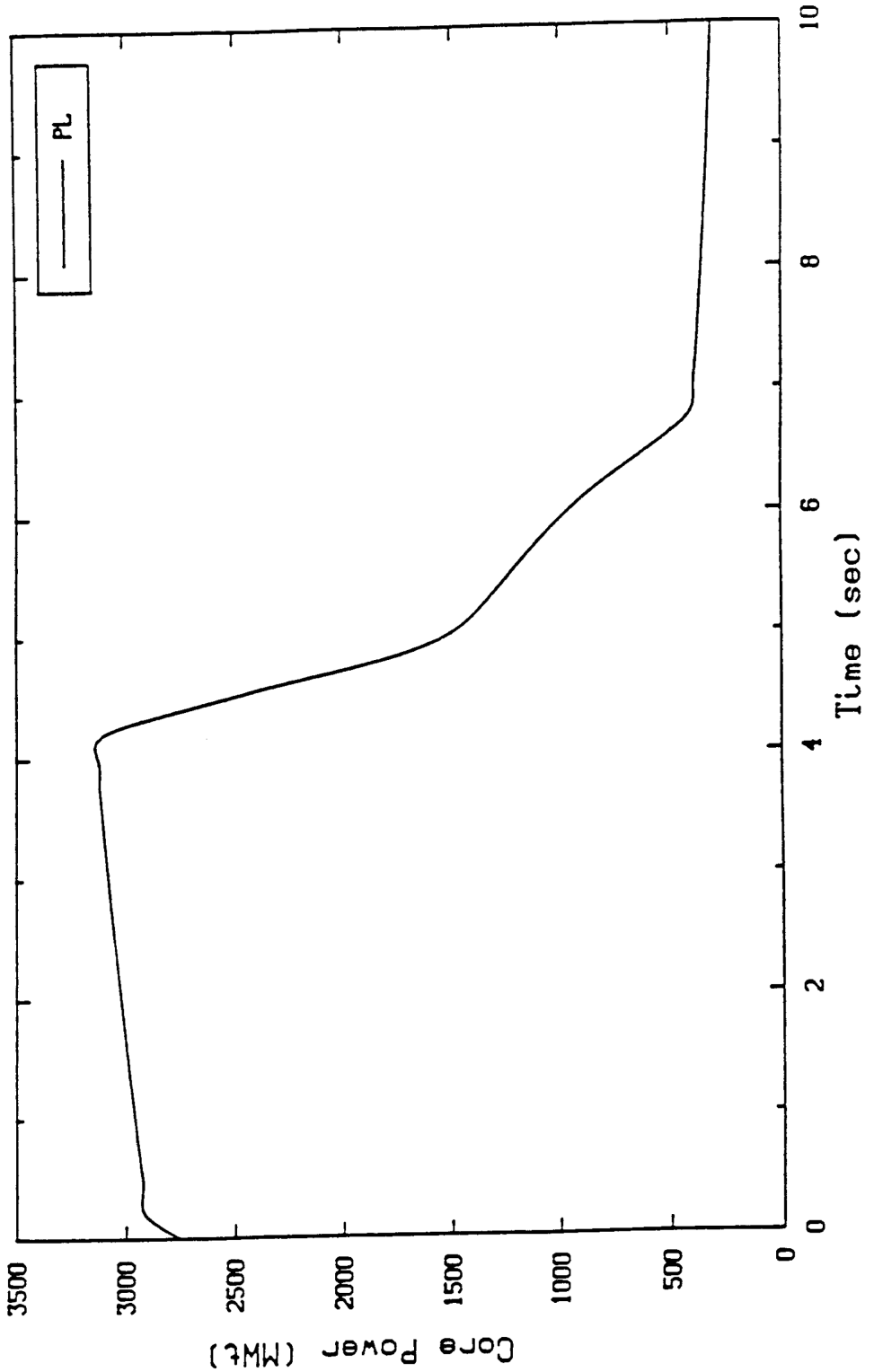


FIGURE 14.4.8–1 CORE POWER FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

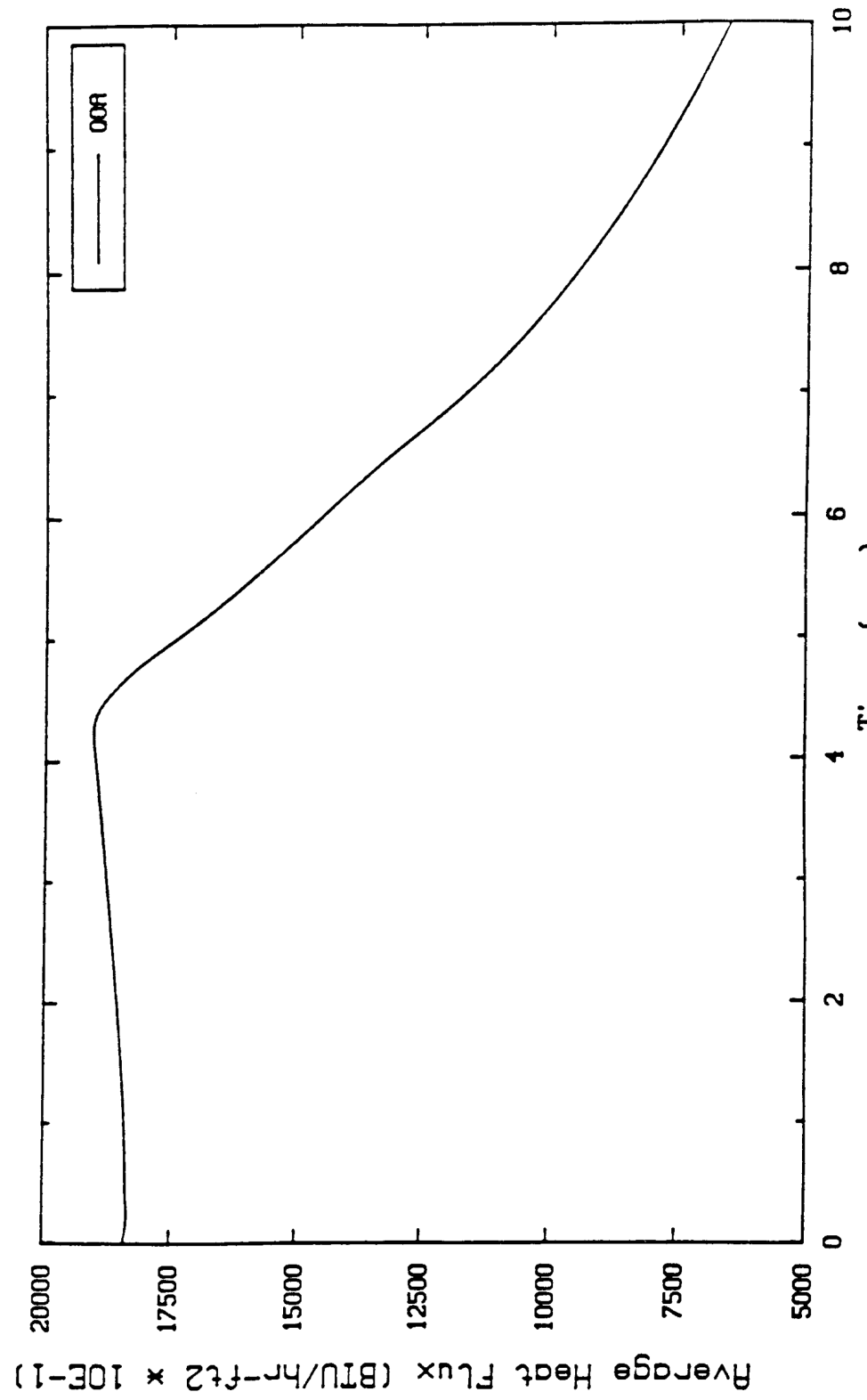


FIGURE 14.4.8–2 CORE AVERAGE HEAT FLUX FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

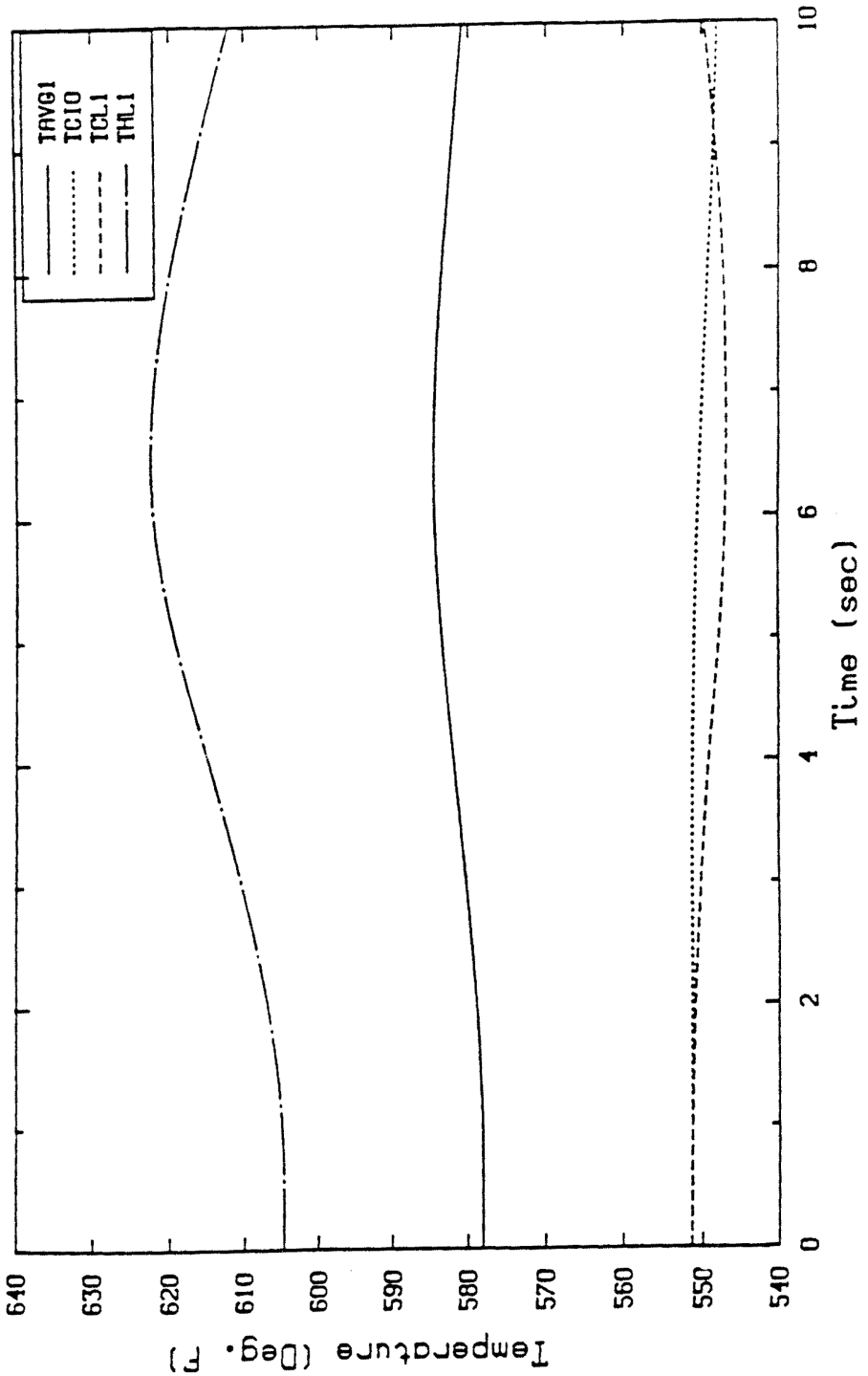


FIGURE 14.4.8–3 REACTOR COOLANT SYSTEM TEMPERATURES FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

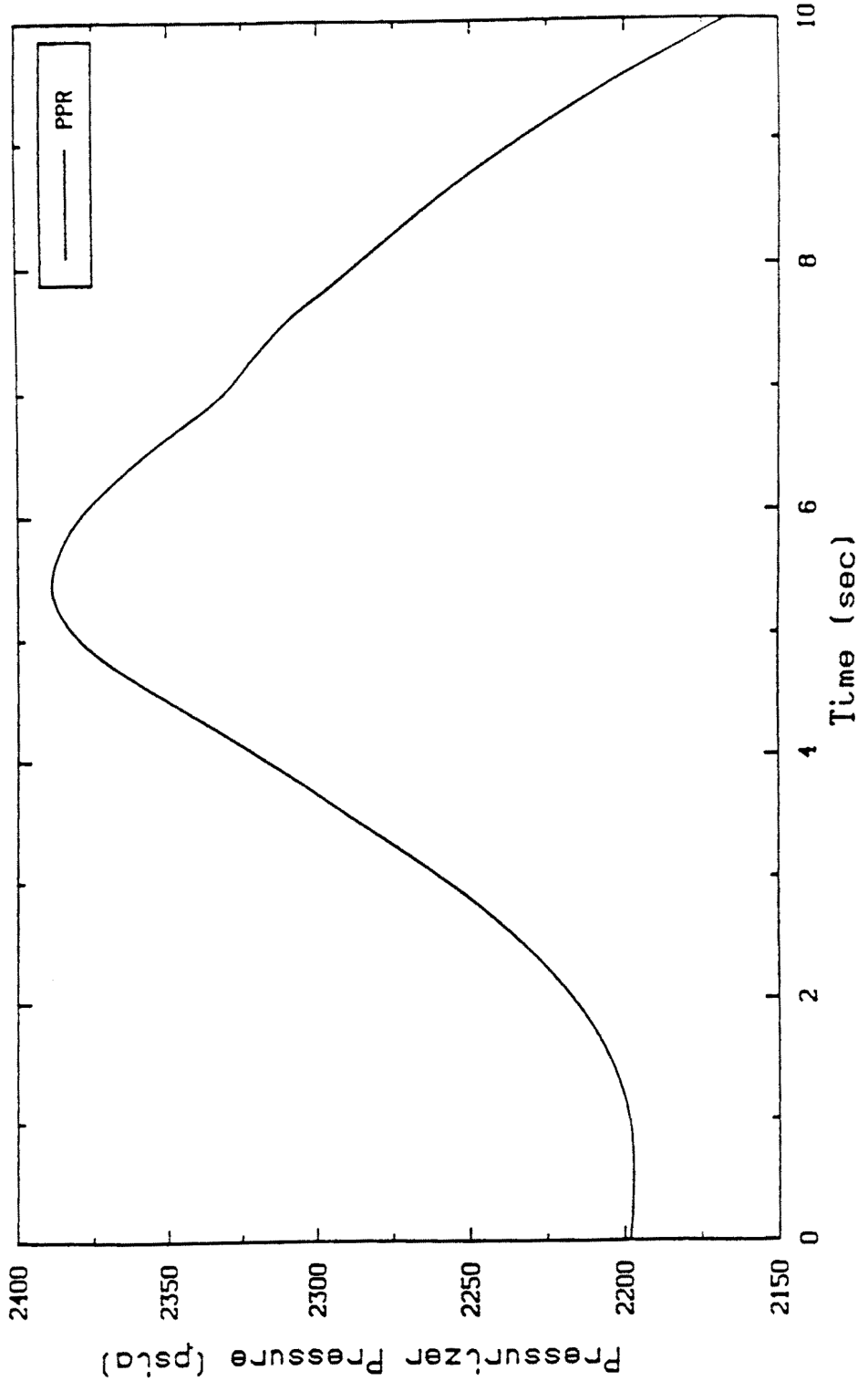


FIGURE 14.4.8–4 PRESSURIZER PRESSURE FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

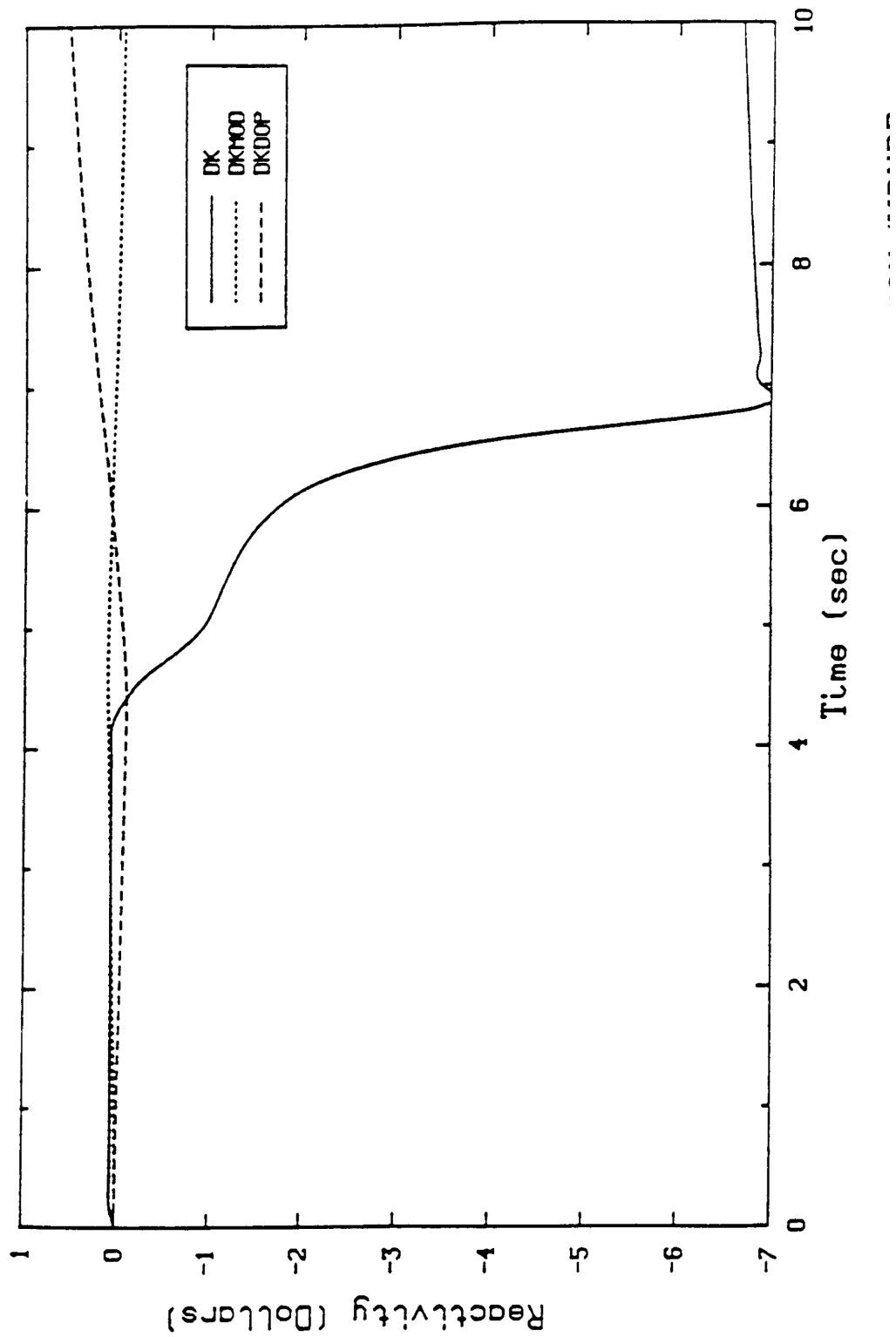


FIGURE 14.4.8–5 REACTIVITIES FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

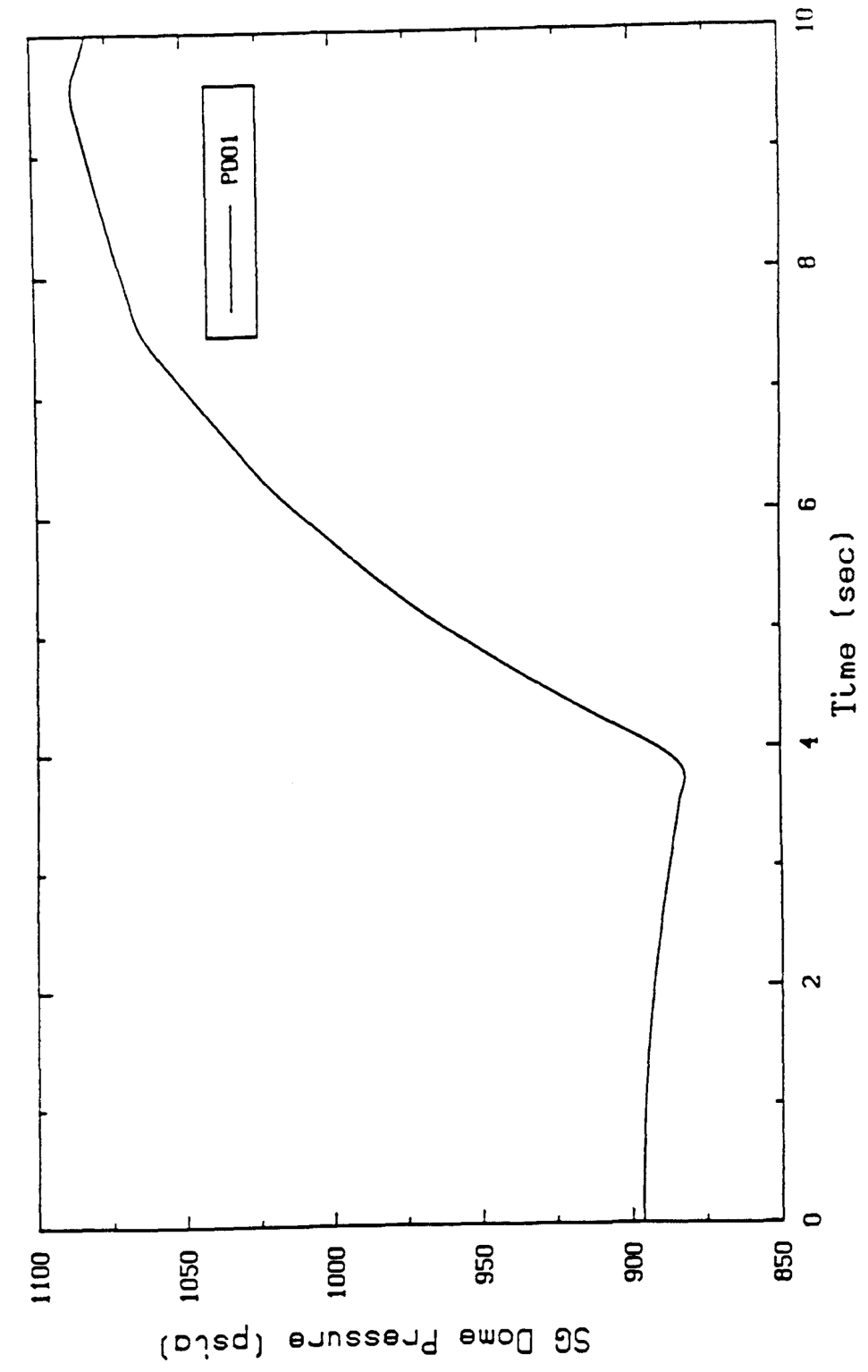


FIGURE 14.4.8–6 SECONDARY PRESSURE FOR A CEA EJECTION (MINIMUM DEPARTURE FOR NUCLEATE BOILING RATIO CASE)

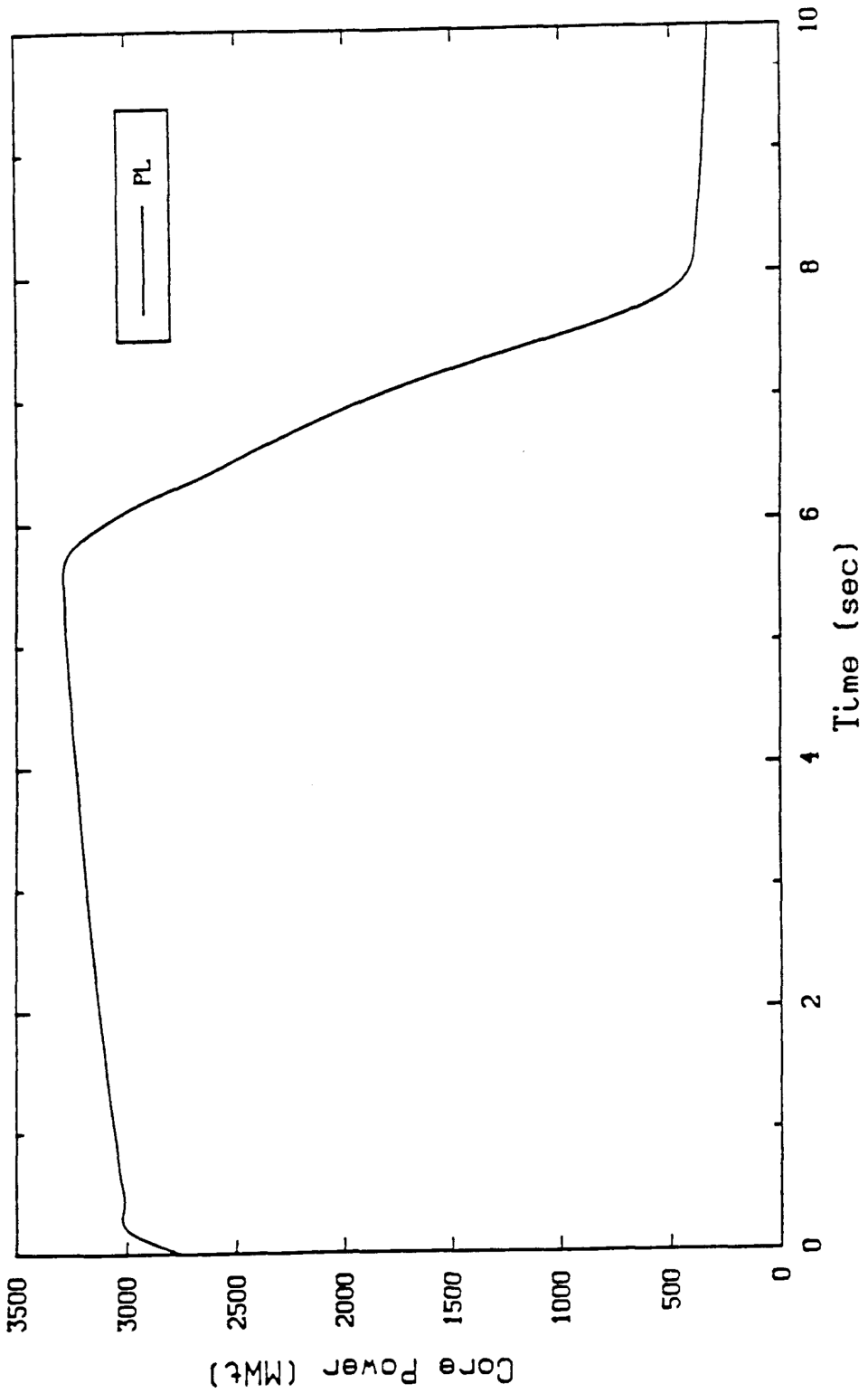


FIGURE 14.4.8–7 CORE POWER FOR A CEA EJECTION (OVERPRESSURE)

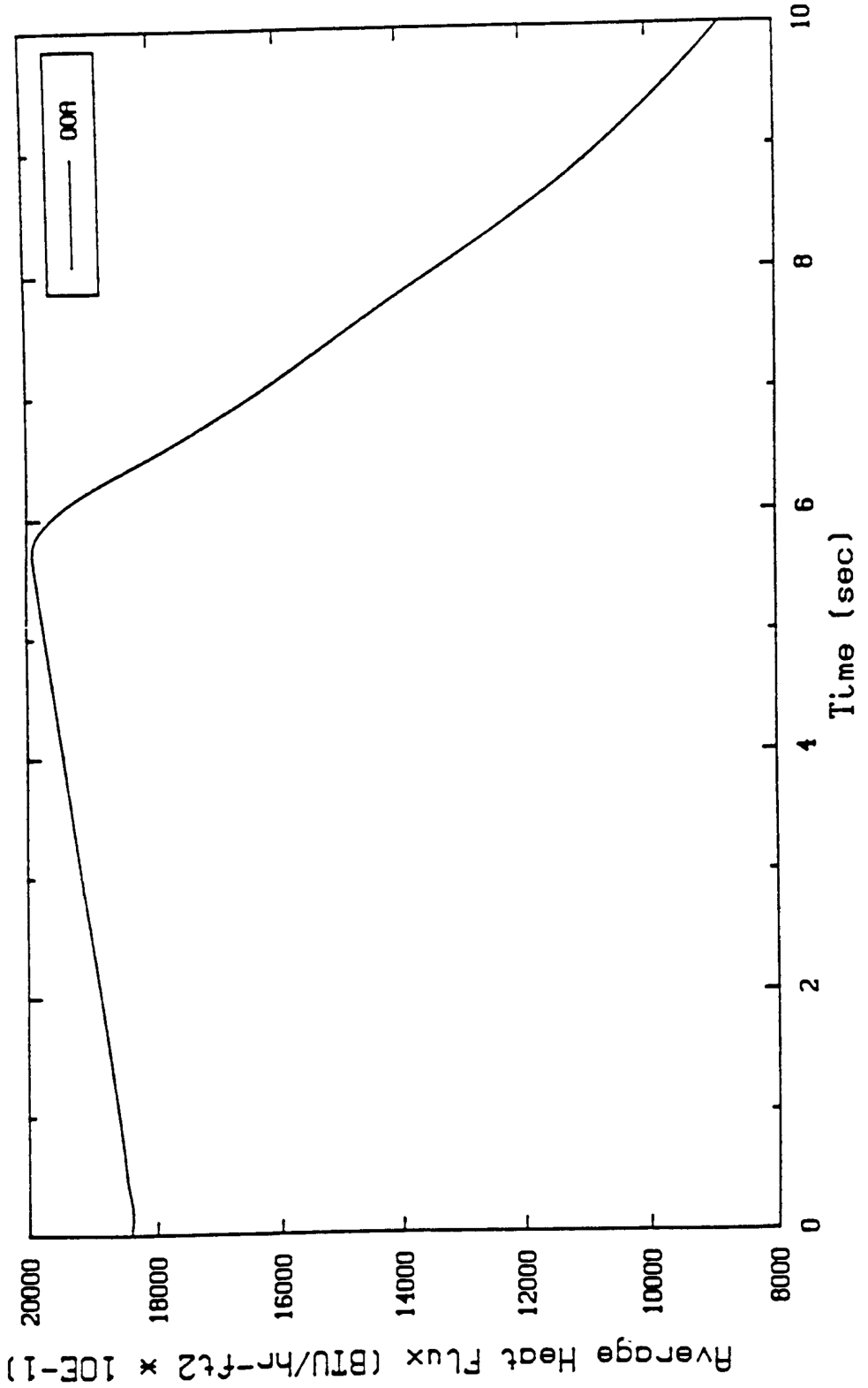


FIGURE 14.4.8–8 CORE AVERAGE HEAT FLUX FOR A CEA EJECTION (OVERPRESSURE)

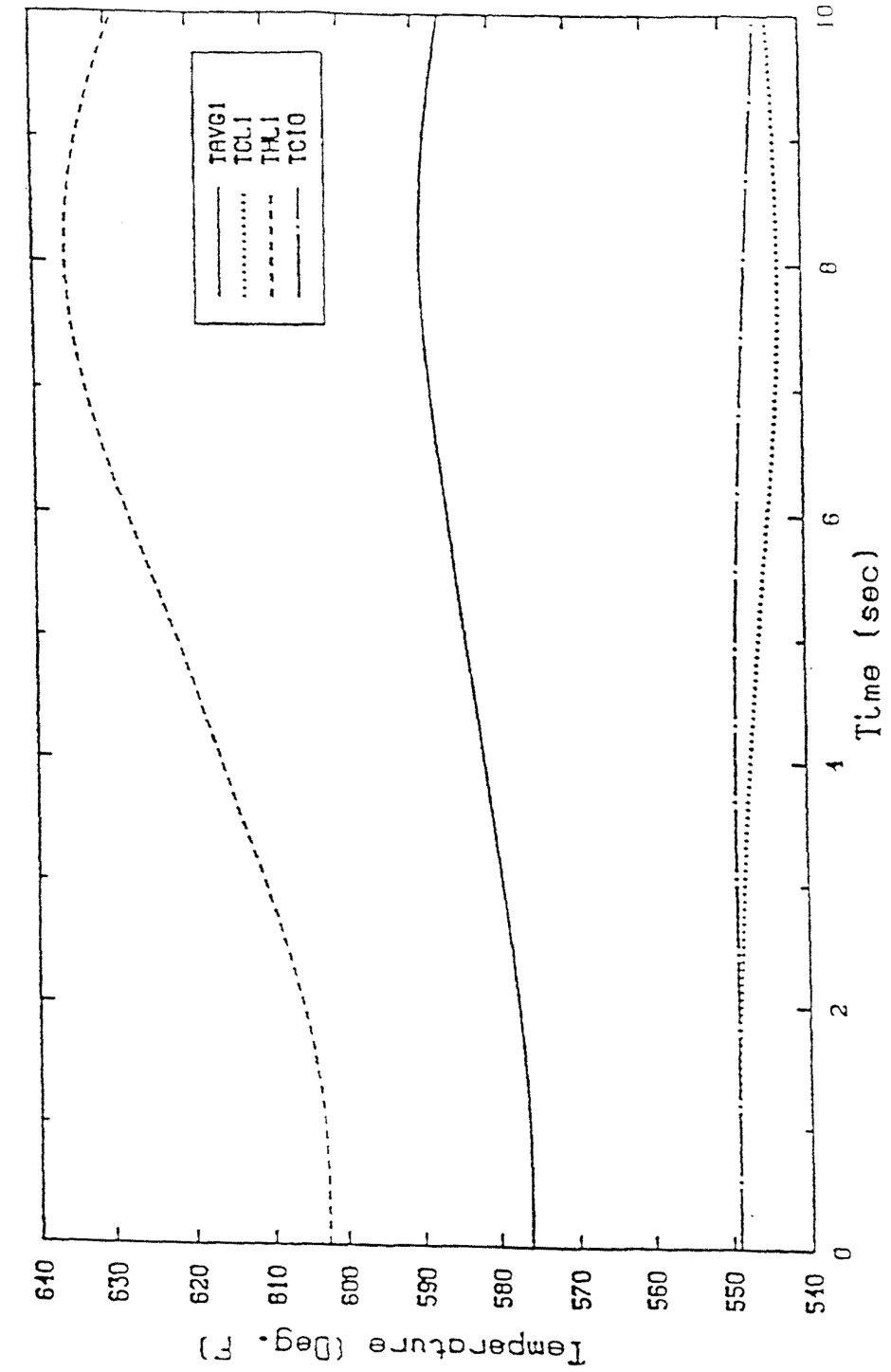


FIGURE 14.4.8–9 PRIMARY SYSTEM TEMPERATURES FOR A CEA EJECTION (OVERPRESSURE)

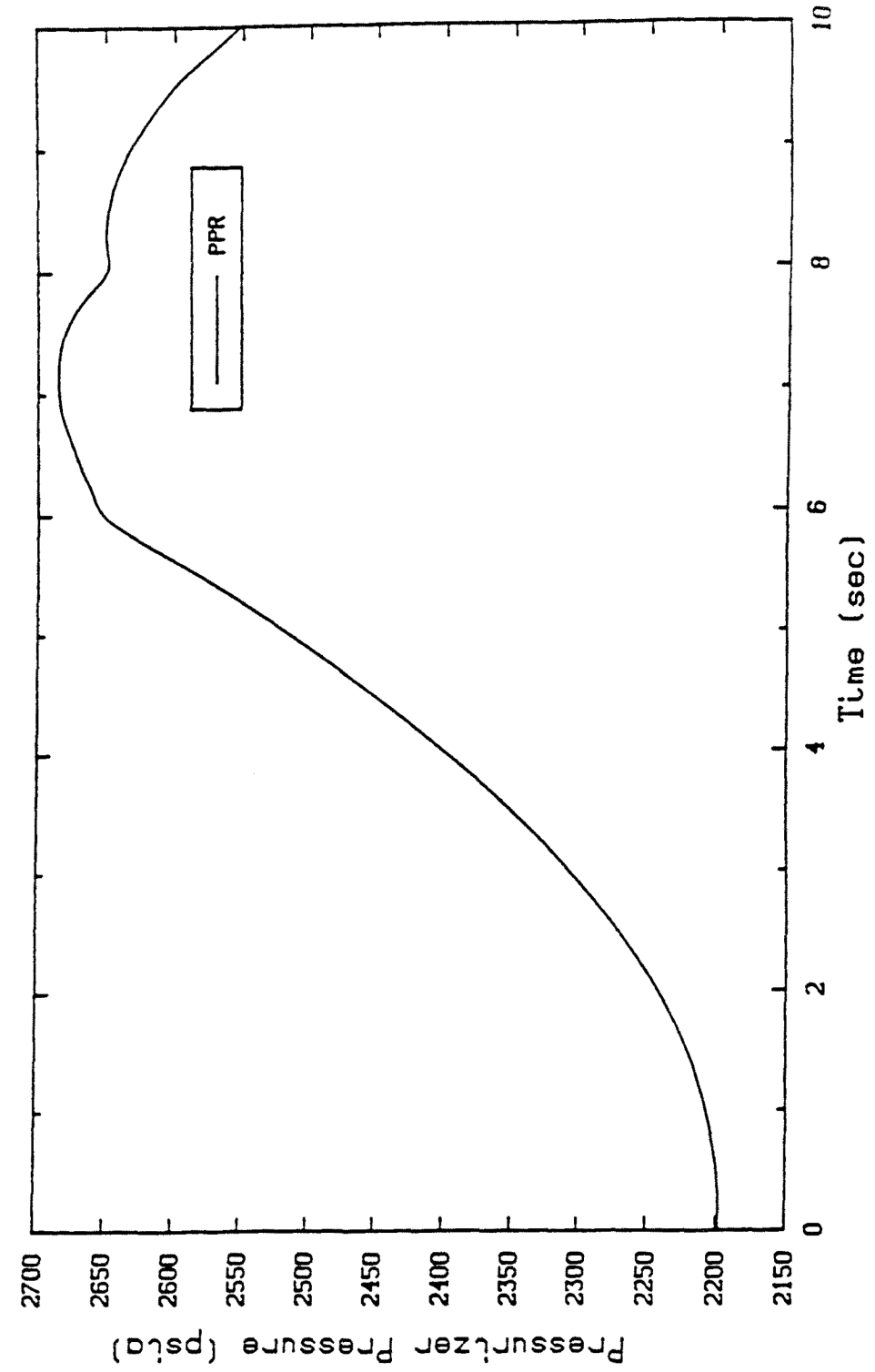


FIGURE 14.4.8–10 PRESSURIZER PRESSURE FOR A CEA EJECTION (OVERPRESSURE)

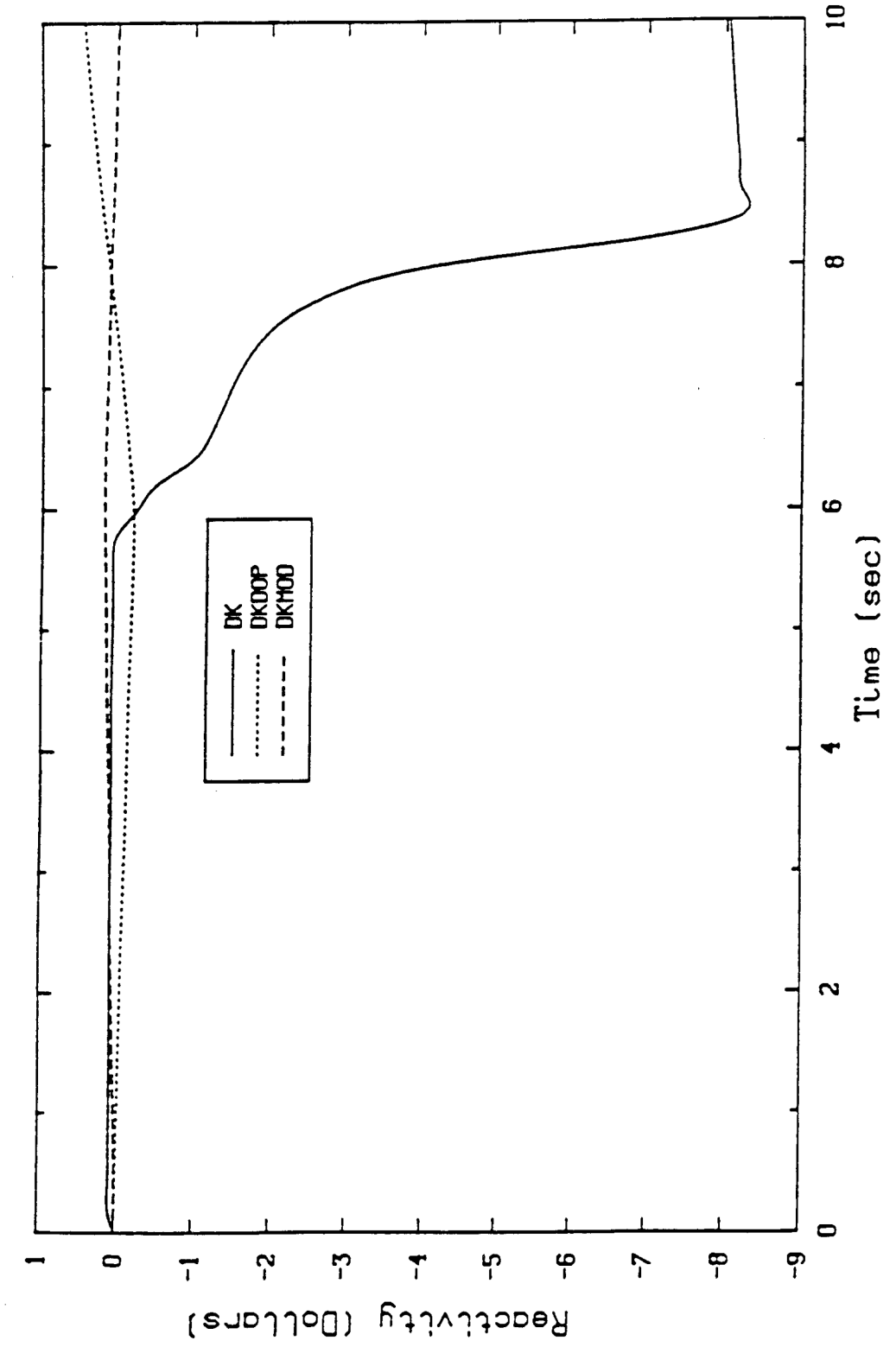


FIGURE 14.4.8–11 REACTIVITIES FOR A CEA EJECTION (OVERPRESSURE)

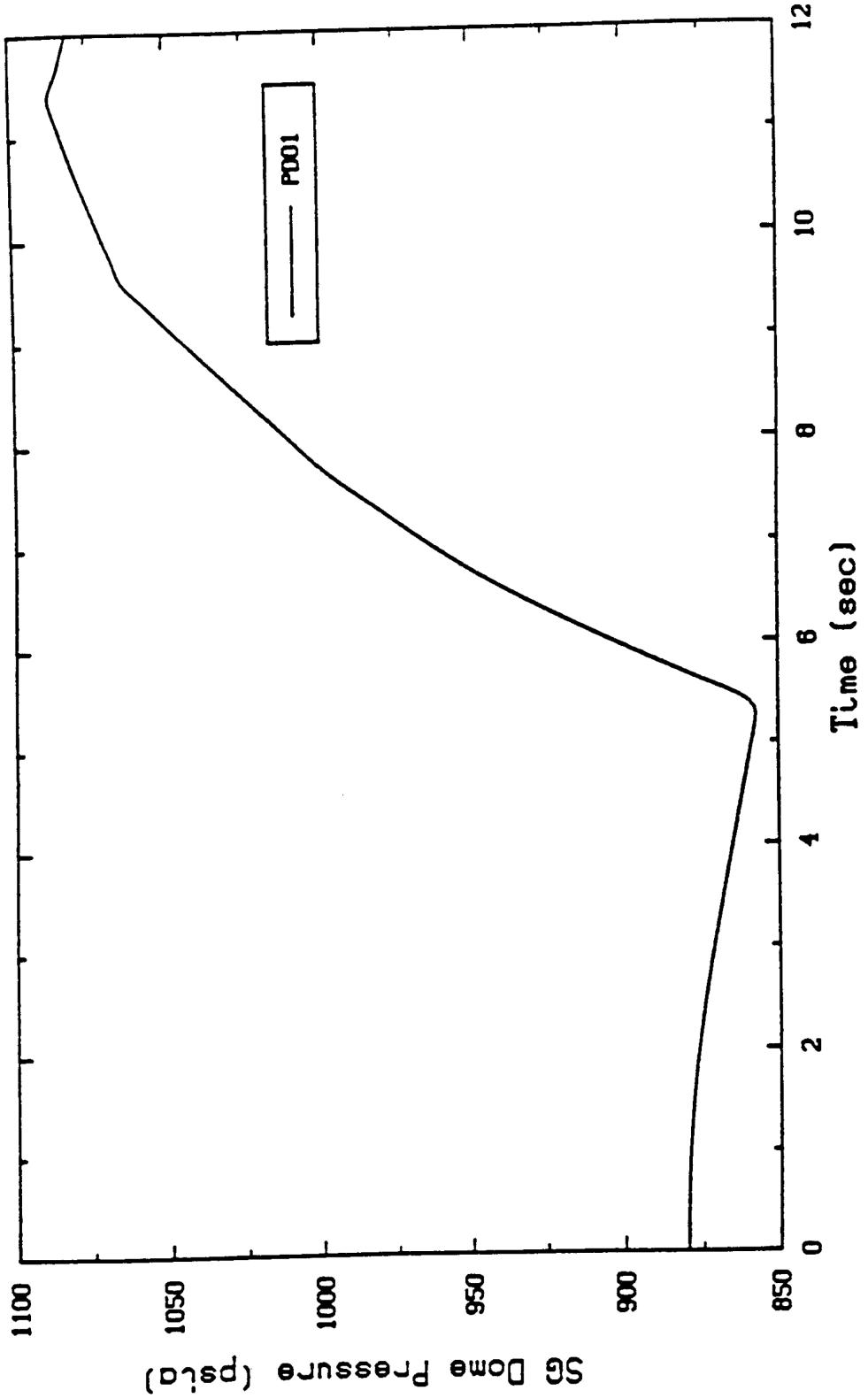


FIGURE 14.4.8–12 SECONDARY PRESSURE FOR A CEA EJECTION (OVERPRESSURE)

14.5 INCREASES IN REACTOR COOLANT SYSTEM INVENTORY

14.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM THAT INCREASES REACTOR COOLANT INVENTORY

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.5.2 CHEMICAL VOLUME AND CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. The potential consequences of diluting the primary system boron concentration are addressed in Event 14.4.6.

14.6 DECREASES IN REACTOR COOLANT INVENTORY

14.6.1 INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE

14.6.1.1 Event Initiator

The event is postulated to occur as a result of the inadvertent opening of one or more pressurizer pressure relief or safety valves due to an electrical or mechanical failure. The limiting event is obtained by assuming the inadvertent opening of a pressurizer safety valve which bounds the capacity of two pressurizer power-operated relief valves (PORVs).

14.6.1.2 Event Description

The opening of the pressurizer pressure relief valve or safety valve results in a blowdown of primary coolant as steam through the faulted valves. Primary system pressure drops rapidly until the pressurizer liquid is depleted, and then quite rapidly to a pressure determined by the saturation curve at the temperature of the coolant in the upper vessel head. Reactor scram will occur on thermal margin/low pressure (TM/LP) before the pressurizer liquid is depleted, terminating the challenge to Specified Acceptable Fuel Design Limits (SAFDLs). In this initial stage, pressurizer heaters would actuate in an attempt to maintain pressure, but would be turned off on a low-level signal before the heater elements were uncovered.

14.6.1.3 Reactor Protection

The TM/LP trip provides initial protection against loss of thermal margin and possible fuel damage. Reactor protection for the Inadvertent Opening of a Pressurized Water Reactor (PWR) Pressurizer Pressure Relief Valve event is summarized in Table 14.6.1-1.

14.6.1.4 Disposition and Justification

The event proceeds as a depressurization of the primary coolant system with a loss of inventory. The core power and primary loop temperatures are relatively unaffected by the pressure drop. Thus, a short term challenge to the SAFDLs exists due to the depressurization prior to scram. There is also a long term concern in that if primary inventory cannot be restored and maintained, core uncovery may result.

The greatest challenge to core uncovery exists at rated power conditions when the core power and primary coolant stored energy are maximized. The greatest challenge to the SAFDLs occurs for the event initiated at rated power where the margin to Departure from Nucleate Boiling (DNB) is minimized.

The disposition of events for the Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve event is summarized in Table 14.6.1-2. As shown, the analysis performed for Mode 1 bounds the event at lower modes.

14.6.1.5 Definition of Events Analyzed

As discussed above, this event is analyzed for Minimum Departure from Nucleate Boiling Ratio (MDNBR).

The system response was evaluated by using PTSPWR2 (Reference 14.6-1). The event MDNBR was calculated using XCOBRA-IIIC (Reference 14.6-2).

14.6.1.6 Analysis Results

The sequence of events for the analysis are given in Table 14.6.1-3. Figures 14.6.1–1 to 14.6.1–6 show the transient response for key system variables. The MDNBR for this event is above the CHF correlation limit. This event does not challenge the FCMLHR limit. Therefore, LHR is not evaluated.

The High Pressure Safety Injection (HPSI) system has been shown to have sufficient capacity to compensate for the loss of primary coolant mass through the inadvertent opening of the pressurizer pressure relief valves. Analysis has shown that core uncovery does not occur during this event.

14.6.1.7 Conclusions

The results of the analysis demonstrate that the event acceptance criteria are met since the MDNBR predicted is greater than the DNBR safety limit. The correlation limits assure with 95% probability and 95% confidence, that DNB is not expected to occur; therefore, no fuel is expected to fail. The FCMLHR limit is not violated in this event.

14.6.2 RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE OF CONTAINMENT

Millstone Unit 2 does not have any instrument lines connected to the reactor coolant system (RCS) which penetrate the containment. A break in either the letdown line or a RCS sample line is not in the current licensing basis for Millstone Unit 2. Therefore, this event is not analyzed.

14.6.3 RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE

14.6.3.1 Event Initiator

The event is initiated by a loss of integrity in a single tube in a steam generator, resulting in a flow of primary side reactor coolant water into the secondary side.

14.6.3.2 Event Description

Experience with nuclear steam generators indicates that the probability of complete severance of a tube is small. The more probable modes of failure are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet.

A leaking steam generator tube would allow transport of primary coolant into the main steam system. Radioactivity contained in the primary coolant would mix with shell side water in the affected steam generator. Some of this radioactivity would be transported by steam to the turbine and then to the condenser. Noncondensible radioactive materials would then be passed to the atmosphere through the condenser air ejector discharge via the Plant stack. The vent path is via the Millstone stack until actuation of an Enclosure Building Filtration Actuation Signal (EBFAS). Actuation of EBFAS automatically isolates the vent path to the Millstone stack after which the Operators manually align the vent path to the Unit 2 stack.

The radioactive products would be sensed by the condenser air ejector radiation monitor or the stack radiation monitor. These monitors have audible alarms that will be annunciated in the control room to alert the operator to abnormal activity levels so that corrective action could be taken.

The behavior of the systems will vary depending upon the size of the steam generator tube failure. For small leaks the chemical and volume control charging pumps will be able to maintain the necessary primary coolant inventory and an automatic reactor trip will not occur. The gaseous fission products will be released from the main steam system at the air ejector discharge and will be discharged via the Plant stack. Nonvolatile fission products will tend to concentrate in the water of the steam generators.

For leaks larger than the capacity of the charging pumps, the pressurizer water level and pressure will decrease and a reactor trip will occur. Upon reactor trip, the turbine will trip and the steam system atmospheric dump valves, steam generator safety valves and the turbine bypass valves will open. In this case it is possible that in addition to the noble fission gases a substantial amount of the radioiodines contained in the secondary system may also be released to the atmosphere through the steam generator safety valves and atmospheric dump valves.

The amount of radioactivity released increases with break size. For this analysis, a double-ended break of one tube was assumed. The selection of one double-ended break as an upper limit is conservatively based upon the experience obtained with other steam generators.

14.6.3.3 Reactor Protection

The leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps. Therefore, the Primary Coolant system pressure will decrease and a low pressurizer pressure trip or TM/LP trip will occur. The thermal margin trip has a low pressure floor below which trip will always occur. Following the reactor trip the Primary Coolant System is cooled by exhausting steam through the atmospheric dump valves, steam generator safety valves, and turbine bypass valves. The radioactivity exhausted through the atmospheric dump valves and steam generator safety valves passes directly to the atmosphere. The radioactivity exhausted through the turbine bypass valves flows to the condenser where the gaseous products remaining are vented to the atmosphere through the condenser air ejector and Plant stack. Due to loss of offsite power, a release pathway via the condenser is not credited.

Reactor protection for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 14.6.3-1.

14.6.3.4 Disposition and Justification

The radiological consequences of a steam generator tube rupture (SGTR) accident are maximized at rated power operation due to the stored energy in the primary coolant which must be removed by the steam generators in order to bring the primary and secondary systems into pressure equilibrium, thereby terminating the primary to secondary leak.

The challenge to the SAFDLs exists due to the depressurization prior to scram. As such, this challenge is very similar to that which exists due to the inadvertent opening of a pressurizer relief valve (Event 14.6.1). Since the depressurization rates associated with Event 14.6.1 are substantially larger than those encountered for this event, the corresponding pressure undershoot will also be greater. Event 14.6.1 will thus be characterized by lower pressures at the time of MDNBR than those obtained for this event. Therefore, the DNB aspects of this event will be bounded by those of Event 14.6.1.

The disposition of events for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 14.6.3-2.

14.6.3.5 Definition of Events Analyzed

The analysis of the SGTR event was performed with assumptions regarding system operation that were chosen to maximize the radiological doses. The analysis assumed that a loss of offsite power and a reactor trip occur at the initiation of a double ended rupture of a steam generator tube. This causes a loss of forced circulation which results in a higher hotleg temperature and a larger portion of the break flow flashing. The loss of offsite power also results in the loss of the ability to steam via the condenser. Plant cooldown occurs via atmospheric dump valves or main steam safety valves, which is the release pathway for primary and secondary activity. This results in a slower cooldown and RCS depressurization, and a reduced capability to cool down the plant via the unaffected steam generator. All of these effects result in higher doses.

The plant simulation includes modeling of the RCS, the steam generators, the main steam and feedwater systems, the charging and letdown systems, and the HPSI System. The pressurizer was modeled as a non-equilibrium volume. Single failure is not postulated in conjunction with the SGTR event. The following assumptions are made to ensure a conservative estimate of the radiological consequences:

1. The initial core power is 2754 MWt;

2. The initial reactor pressure is 2300 psia including instrument uncertainty;

3. The initial main steam pressure is 933 psia including instrument uncertainty;

4. The initial inlet temperature is 551.25°F including instrument uncertainty;

5. A double-ended rupture of one steam generator tube occurs instantaneously;

6. On reactor trip and turbine trip, loss of offsite power is assumed along with loss of instrument air and the condenser. The ADVs may be operated by local manual action due to loss of instrument air;

7. Following the reactor trip, the MSSVs lift for removal of decay heat from the RCS;

8. The analysis assumed the lowest allowed opening setpoint (-3% drift) for the ruptured steam generator MSSVs and the highest allowed opening setpoint (+3% drift) for the intact steam generator MSSVs;

9. The reseat pressure of the MSSVs on the ruptured steam generator is 12% below the opening pressure, Reference 14.6-5, while the reseat pressure of the MSSVs on the intact steam generator is nominal 6% below the opening pressure. This maximizes the releases to the atmosphere from the ruptured steam generator;

10. All three charging pumps are assumed to be operable, which will lead to a larger primary to secondary break flow. Letdown is conservatively isolated at the time of tube rupture;

11. SIAS is initiated on low pressurizer pressure which starts two HPSI pumps to deliver maximum flow;

12. AFW auto-initiates, accounting for system delay, and delivers a minimum flow.

The operator actions assumed in this analysis are consistent with the EOPs. The major post-trip analysis assumptions regarding operator actions are:

1. Commence Cooldown to Hotleg Temperature Less Than 515°F

Once the event is diagnosed, the operators will cool the RCS at a maximum controllable rate until the hotleg temperature of both loops reaches 515°F, for ruptured steam generator isolation. This temperature assumed in the analysis conservatively includes instrument uncertainties to delay the time till the ruptured steam generator can be isolated. The analysis assumes a loss of offsite power leading to a loss of the condenser. Therefore, the cooldown is performed using the ADVs. Since the analysis assumes a loss of offsite power, a loss of instrument air is postulated, requiring a local manual control of the ADVs for this cooldown. The analysis assumes that, to account for local manual operator action, the cooldown starts 30 minutes from the time of reactor trip.

2. Reduce and Control RCS Pressure

   The analysis conservatively does not depressurize the RCS till after the hotleg temperature is less than 515°F and the ruptured steam generator is isolated. In the EOPs, the RCS depressurization begins just after the cooldown to 515°F commences. It is more conservative for dose consequences to delay the RCS depressurization since this will provide a larger primary to secondary break flow rate.

3. Determine and Isolate the Most Affected Steam Generator

   The operator isolates the most affected steam generator once the hotleg temperature of the loops have reached the isolation temperature of hotleg less than 515°F.

4. Cooldown and Depressurize RCS to SDC Entry Condition

   Cooldown and depressurization to SDC entry would minimize the primary to secondary break flow. The analysis assumes that cooldown to SDC entry is achieved by steaming just the intact steam generator per the EOPs. This is performed for 16 hours from the time of the tube rupture. The analysis conservatively assumes that the hotleg temperatures of the two loops fail to stay coupled, impeding the depressurization to SDC condition. There are five options available in the EOPs to cool and depressurize the isolated steam generator: 1) if RCPs are operating, use at least one RCP and perform a backflow into the RCS; 2) if time permits, allow ambient cooling; 3) if the condenser is available, steam to the condenser; 4) feed and bleed via steam generator blowdown; 5) steam to the atmosphere using ADV and feeding. Since the last option would lead to a larger offsite dose, it is the method modeled. Given a loss of offsite power/loss of instrument air condition, the feed and bleed via the steam generator blowdown or ambient cooling may be chosen to limit the offsite dose as well as dose to the operator for performing a local manual operation of the ruptured steam generator ADV.

5. Maintain Isolated Steam Generator Level Less Than 90%

The EOPs prevent the ruptured steam generator from overfill by maintaining the pressurizer pressure within 50 psi of the isolated steam generator pressure or backflow into the RCS, in order to minimize the primary to secondary break flow. Alternatively, the steam generator blowdown may be used to restore level less than 90% narrow range level. For offsite dose purposes, this is not explicitly modeled. However, the model assumes primary side depressurization, facilitated by steaming of the isolated steam generator. Therefore, the pressurizer pressure can be maintained within 50 psi of the isolated steam generator, avoiding ruptured steam generator overfill.

14.6.3.6 Analysis Results

14.6.3.6.1 Thermal-Hydraulic Calculation

The portion of the SGTR analysis, till the time the hotleg temperatures reach less than 515°F, was performed using RETRAN-02 MOD 3 (Reference 14.6-4) computer code. The sequence of results for this transient is presented in Table 14.6.3-3. Figures 14.6.3–1 through 14.6.3–9 present the dynamic behavior of important NSSS parameters during this event.

Following a double-ended break of a steam generator tube rupture, reactor coolant flows from the primary side into the secondary side of the ruptured steam generator (see Figure 14.6.3–5). A portion of this break flow is released as flashed steam (see Figure 14.6.3–6). The model has the reactor tripping at the time of tube break. This is conservative, since any pre-trip mass releases would be via the condenser air ejector where a partition factor would greatly reduce the iodine releases (this is further discussed in Section 14.6.3.6.2). Therefore, to conservatively maximize the direct atmospheric releases, the earliest possible trip is limiting. A loss of offsite power at the time of trip leads to a loss of forced flow and a momentary spike in the coldleg temperature as shown in Figure 14.6.3–1. The pressurizer level decreases as the reactor coolant shrinks post-trip. Also, the break flow is greater than the capacity of the charging pumps. As a result, the pressurizer level decreases as shown in Figure 14.6.3–2. The pressurizer pressure also drops as shown in Figure 14.6.3–3. While all three charging pumps and pressurizer heaters attempt to maintain level and pressure, letdown is conservatively isolated at the time of tube break. The pressurizer heaters are turned off as the pressurizer level decreases towards heater uncovery.

As the steam bypass to the condenser is assumed to be unavailable, the post-trip steaming is accomplished via the ADVs and the MSSVs. However, the ADVs require instrument air, which is postulated to be lost with the loss of offsite power. Therefore, no releases from the ADVs are modeled till 1,800 seconds from trip, when local manual operator action can be credited (see Figure 14.6.3–7). Hence, the post-trip steaming to remove decay heat is accomplished, during the initial 30 minutes, by the MSSVs (see Figure 14.6.3–8). The turbine valve closure, due to reactor trip, causes the steam generator pressure to rise, as shown in Figure 14.6.3–4, till the MSSV lift pressure is reached. The main feedwater flow is terminated at the time of trip and the AFW initiates on low steam generator level at 277 seconds accounting for system response time. As the colder AFW is delivered, a hotter volume of feedwater is swept in first. Two AFW pumps deliver a minimum flow rate (see Figure 14.6.3–9).

The pressurizer level and pressure continue to decrease as the energy transfer to the secondary side shrinks the reactor coolant and the tube break flow continues to deplete the primary inventory. The decrease in pressure results in actuation of SIAS at 496 seconds. Once RCS pressure decreases below the HPSI shutoff head pressure, two HPSI pumps deliver maximum flow to slow the decrease in pressurizer pressure. The pressurizer pressure approaches an equilibrium pressure as the combined HPSI and charging flow rate matches the break flow rate. The hotleg temperature of 515°F is reached in 3,637 seconds post-trip.

Specific analyses of the potential for fuel failure is not performed for the steam generator tube accident. The potential for fuel failure is bounded by the analysis for the inadvertent opening of the pressurizer relief valve (Event 14.6.1). The analyses for Event 14.6.1 show that fuel failure does not occur for that event, therefore, fuel failure does not occur following a steam generator tube rupture.

14.6.3.6.2 Radiological Calculation

The intent of this radiological consequences analysis is to verify that offsite and control room doses do not exceed the guidelines of 10 CFR 50.67 and Regulatory Guide 1.183.

The mass releases following a SGTR were determined for use in evaluating the offsite and control room radiation exposure. Figures 14.6.3–5 through 14.6.3–8 show the break mass flow rate and the steam mass flow rate predicted by the thermal hydraulic analysis. This includes the flashing of the break flow as it enters the secondary side of the steam generator. Table 14.6.3-4 summarizes the mass releases for the SGTR event. This includes 103,225 lbm additional mass releases from the ruptured steam generator after it is identified and isolated associated with facilitating cooldown and depressurization for SDC entry, as well as 2,771,000 lbm released from the intact steam generator for cooldown to SDC entry.

The SGTR accident is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant is transported directly to the atmosphere via the ADVs and MSSVs.

The effects of iodine spiking were accounted for in the analysis. Two different spiking models were evaluated. The first assumes that the primary coolant concentration of Dose Equivalent (DE) I-131 is at the Technical Specification limit of 0.5 μCi/gm and the resulting tube rupture causes the iodine appearance rate to increase by a factor of 335 over the equilibrium appearance rate corresponding to the 0.5 μCi/gm (DE) I-131 coolant concentration. The duration of the spike is assumed to be 8 hours. The second spiking model assumes that a pre-accident iodine spike causes the primary coolant to reach a DE I-131 concentration of 30 μCi/gm at the time of the tube rupture. This concentration is assumed to last for the duration of the accident.

The RADTRAD-NAI computer program is used to calculate the TEDE dose due to RCS and secondary side activity. RADTRAD-NAI is a multiple compartment activity transport code with the dose model consistent with the Regulatory Guideline 1.183 model.

The offsite and control room doses calculated are presented in Table 14.6.3-6. The results are bounding for the assumptions on Table 14.6.3-5 and the thermal-hydraulic results presented in Table 14.6.3-4.

14.6.3.7 Conclusion

The radiological release criterion for this analysis, as well as the calculated results, are presented in Table 14.6.3-6. The calculated results are less than the NRC criteria for both the cases evaluated.

14.6.4 RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT

This event is only applicable to Boiling Water Reactors (BWR). As such, this event is not applicable to Millstone Unit 2.

14.6.5 LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

This event is initiated by a breach in the primary coolant system pressure boundary. Basically, a range of break sizes from small leaks up to a complete double-ended severance of a primary coolant system pipe must be considered. Typically, these breaks are classified as large breaks or small breaks. Large-break loss-of-coolant accidents (LBLOCA) are discussed in Section 14.6.5.1 for the AREVA Standard CE14 HTP fuel with M5 clad fuel rods. Small-break loss-of-coolant accidents (SBLOCA) are discussed in Section 14.6.5.2.

14.6.5.1 Large Break Loss of Coolant Accidents for M5 Clad Fuel

14.6.5.1.1 Event Initiator

This event is initiated by a large break in the primary coolant system pressure boundary. The size of breaks typically considered to be large breaks are from 0.5 ft^2 up to a double-ended severance of a primary coolant system pipe.

14.6.5.1.2 Event Description

The LBLOCA events are characterized by four sequential phases. They are:

1. blowdown
2. refill
3. reflood
4. long term cooling

The blowdown phase immediately follows the initiation of a large break. Primary system water is discharged through the break into containment. The system pressure decreases rapidly during the initial subcooled blowdown. As the saturation pressure is approached, local boiling and flashing takes place in the core and the reactor goes subcritical via the negative moderator reactivity feedback. The blowdown flow becomes a water-vapor mixture. The depressurization rate is reduced when core pressure falls below the saturation pressure. The water level continues to decrease until a large amount of water from the safety injection tanks reaches the lower plenum.

The refill phase starts when the safety injection tank water begins to fill the lower plenum. At this time, the core is uncovered by water and the fuel rods are cooled primarily by thermal radiation.

The reflood phase begins when the water level reaches the bottom of the core.

The long term cooling phase starts after the core has quenched to the point where the metal - water reaction is suppressed, or the water level covers the active fuel. During this phase, the water inventory is controlled by the safety injection pumps. The continuous operation of these pumps ensures the long term dissipation of the decay heat.

14.6.5.1.3 Reactor Protection

No credit is taken for a reactor trip by the reactor protection system (RPS) in the RLBLOCA analysis. The RPS is not necessary due to the rapid depletion of the moderator which shuts down the reactor core almost immediately, followed by ECCS injection which contains sufficient boron to maintain the reactor core in a subcritical configuration. Technical specification limits on hot rod power serve to limit the peak cladding temperature (PCT).

Available Reactor Protection for the Large Break Loss of Coolant Accidents is summarized in Table 14.6.5.1-1.

14.6.5.1.4 Disposition and Justification

Section 15.6.5 of Reference 14.6-11 indicates that the primary acceptance criteria for this event are to limit offsite doses, to limit fuel clad oxidation, and to keep PCT below 2200°F. Offsite doses are maximized in the RLBLOCA analysis by assuming full-power operation and conservative treatment of peaking factors and decay heat. The RLBLOCA analysis initiates the event at full power conditions and core peaking factors (i.e., axial power, radial peaking, LHGR) that are set or ranged over plant technical specifications or plant limiting values. In addition, the decay heat calculations for each case in the sample set are based on the ANSI/ANS 5.1-1979 standard, which assures conservative treatment. Fuel pin cladding temperatures and oxidation rates are highly influenced by these effects, which are maximized at full power conditions. Thus, the most limiting results for this event are obtained with the plant operating at full power in Mode 1. These results bound those from Modes 2-6.

Disposition of Events for the Large Break Loss of Coolant Accidents is summarized in Table 14.6.5.1-2.

14.6.5.1.5 Definition of Events Analyzed

The purpose of the analysis is to verify the adequacy of the Emergency Core Cooling System (ECCS) by demonstrating compliance with the following 10 CFR 50.46(b) criteria:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume were to react.

The NRC staff imposed a limitation on the Realistic Large Break LOCA Methodology (Reference 14.6-17) specifying that the equivalent cladding reacted (ECR) results calculated using the Cathcart-Pawel correlation are considered acceptable in conformance with 10 CFR 50.46(b)(2) if the ECR value is less than 13%.

14.6.5.1.5.1 Description of Large Break Loss of Coolant Accident Transient

A LBLOCA is initiated by a postulated rupture of the Reactor Coolant System (RCS) primary piping. The most challenging break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop. The plant is assumed to be operating normally at full power prior to the accident and the break is assumed to open instantaneously. A worst case single-failure is also assumed to occur during the accident. The limiting single-failure for this analysis is the loss of one ECCS pumped injection train without the loss of containment spray.

The LBLOCA event is typically described in the short-term by three phases: blowdown, refill, and reflood. Following the initiation of the break, the blowdown phase is characterized by a sudden depressurization from operating pressure down to the saturation pressure of the hot leg fluid. For larger cold leg breaks, an immediate flow reversal and stagnation occurs in the core due to flow out the break, which causes the fuel rods to pass through critical heat flux (CHF), usually within 1 second following the break. Following this initial rapid depressurization, the RCS depressurizes at a more gradual rate. Reactor trip and emergency injection signals occur when either the low pressure setpoint or the containment high-pressure setpoint are reached. However, for LBLOCA, reactor trip and scram are essentially inconsequential, as reactor shutdown is accomplished by moderator feedback. During blowdown, core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow.

When the system pressure falls below the Safety Injection Tanks (SIT) pressure, flow from the SIT is injected into the cold legs ending the blowdown period and initiating the refill period. Once the system pressure falls below the respective shutoff heads of the safety injection systems and the

system startup time delays are met, flow from the safety injection systems is injected into the RCS. While some of the ECCS flow bypasses the core and goes directly out of the break, the downcomer and lower plenum gradually refill until the mixture in the lower head and lower plenum regions reaches the bottom of the active core and the reflood period begins. Core cooling is supported by the natural evolution of the RCS flow pattern as driven by the break flow and condensation on the emergency coolant being injected. Towards the end of the refill period, heat transfer from the fuel rods is relatively low, steam cooling and rod-to-rod radiation being the primary mechanisms.

Once the lower plenum is refilled to the bottom of the fuel rod heated length, refill ends and the reflood phase begins. Substantial ECCS fluid is retained in the downcomer during refill. This provides the driving head to move coolant into the core. As the mixture level moves up the core, steam is generated and liquid is entrained, providing cooling in the upper core regions. The two-phase mixture expands into the upper plenum and some liquid may de-entrain and flow downward back into the cooler core regions. The remaining entrained liquid passes into the steam generators where it vaporizes, adding to the steam that must be discharged through the break and out of the system. The difficulty of venting steam is, in general, referred to as steam binding. It acts to impede core reflood rates. With the initiation of reflood, a quench front starts to progress up the core. With the advancement of the quench front, the cooling in the upper regions of the core increases, eventually arresting the rise in fuel rod surface temperatures. Later the core is quenched and a core cooling process is established that can maintain the cladding temperature near saturation, so long as the ECCS makes up for the core boil off.

14.6.5.1.5.2 Description of Analytical Models

The RLBLOCA methodology is documented in EMF-2103 Realistic Large Break LOCA Methodology for Pressurized Water Reactors (Reference 14.6-17). The RLBLOCA methodology uses a non-parametric statistical approach to determine that the first three criteria of 10 CFR 50.46 are met with a probability higher than 95 percent with 95 percent confidence.

The methodology follows the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology (Reference 14.6-18) and the requirements of the Evaluation Model Development and Assessment Process (EMDAP) documented in Reference 14.6-19. The CSAU method outlines an approach for defining and qualifying a best estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis.

The RLBLOCA methodology employs the following computer codes:

- COPERNIC for computation of the initial fuel stored energy, fission gas release, and the transient fuel cladding gap conductance.

- S-RELAP5 for the thermal-hydraulic system calculations (includes ICECON for containment response).

The governing two-fluid (plus non-condensable) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat

generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heat. The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on the other are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomena expected during the LBLOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the Reactor Coolant Pumps (RCPs) or the steam generator (SG) separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

The analysis considers blockage effects due to clad swelling and rupture as well as increased heat load due to fuel relocation in the ballooned region of the cladding in the prediction of the hot fuel rod PCT.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant technical specifications or to match measured data. Additionally, the COPERNIC code provides initial conditions for the S-RELAP5 fuel model. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Containment pressure is calculated by the ICECON module within S-RELAP5.

14.6.5.1.5.3 Plant Description and Summary of Analysis Parameters

The plant analyzed is the Millstone Unit 2, CE designed PWR, which has 2x4-loop arrangement. There are two hot legs each with an U-tube steam generator and four cold legs each with a RCP. The RCS includes one Pressurizer connected to a hot leg. The ECCS comprises four SITs, one per loop/cold leg, and one full train of Low Pressure Safety Injection (LPSI) and High Pressure Safety Injection (HPSI) (after applying the single failure assumption). The HPSI and LPSI feed into common headers (cross connected) that are connected to the SIT lines. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection does not need to be considered.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/HPSI path per RCS loop. The HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary-side steam generator that is instantaneously isolated (closed main steam isolation valve and feedwater trip) at the time of the break. The analysis includes AREVA fuel with M5 cladding and utilizes the COPERNIC code for fuel calculations within S-RELAP5.

As described in the AREVA RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. The values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in this analysis are given in Table 14.6.5.1-3. The uncertainties used in the analysis are summarized in Table 14.6.5.1-4. Two parameters (refueling water storage tank temperature and diesel start time) are set at conservative bounding values for all calculations. The passive heat sinks and material properties used in the containment input model are provided in Table 14.6.5.1-5.

14.6.5.1.5.4 Base Calculations

The application of the AREVA RLBLOCA methodology involves developing input decks, executing the simulations that comprise the uncertainty analysis, retrieving PCT, maximum local oxidation, and core-wide oxidation information and determining the simultaneous Upper Tolerance Limit (UTL) results for the criteria. The analysis assumes full-power operation at 2754 MWt (includes power calorimetric uncertainty), a tube plugging level of 5.87 percent per steam generator, a peak linear heat generation rate (LHGR) of 15.1 kW/ft, and a radial peaking factor of 1.854 (includes uncertainty). The analysis supports operation of a full core of AREVA Standard CE14 HTP fuel design with M5 cladding of fresh and once-burned fuel assemblies. Blockage effects due to clad swelling and rupture are also considered.

The analysis also addresses typical operational ranges or technical specification limits (whichever is applicable) with regard to pressurizer pressure and level; safety injection tank (SIT) pressure, temperature (containment temperature), and level; core inlet temperature; core flow; containment pressure and temperature; and refueling water storage tank temperature.

The UTL results are provided in Table 14.6.5.1-6. For a simultaneous coverage/confidence level of 95/95, the UTL results are shown in Section 14.6.5.1.6. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total core wide percent oxidation, which is well below the 1 percent limit. Table 14.6.5.1-7 provides a summary of the major input parameters for the demonstration case. The sequence of event times for the demonstration case is provided in Table 14.6.5.1-8. The analysis plots for the case set are shown in Figure 14.6.5.1–1 through Figure 14.6.5.1–5. Figure 14.6.5.1–1 shows linear scatter plots of the key parameters sampled for all cases. Parameter labels appear to the left of each individual plot. Figure 14.6.5.1–2 and Figure 14.6.5.1–3 show PCT scatter plots versus the time of PCT and versus break size, respectively. The scatter plots for the maximum local oxidation and total core-wide oxidation are shown in Figure 14.6.5.1–4 and Figure 14.6.5.1–5, respectively.

Figure 14.6.5.1–6 through Figure 14.6.5.1–17 show key results from the S-RELAP5 calculations for the demonstration case.

14.6.5.1.5.5 Reduced Primary Temperature Operation at End of Cycle

An assessment to support an end of cycle full power coastdown to an indicated RCS cold leg temperature of 537°F has been performed. The result of the assessment confirmed that the

RLBLOCA analysis bounds operation during an end of cycle full power coastdown to an indicated RCS cold leg temperature of 537°F.

14.6.5.1.6 Summary of Results

The UTL results providing 95/95 simultaneous coverage from this evaluation meet the 10 CFR 50.46(b) criteria with a PCT of 1615°F, a maximum local oxidation of 2.01 percent and a total core-wide oxidation of 0.025 percent. The maximum local oxidation UTL is less than 13% limit imposed by the NRC on the RLBLOCA EM (Reference 14.6-17).

14.6.5.1.7 Post Analysis of Record Evaluations

In addition to the analyses presented in this section, evaluations and reanalysis may be performed as needed to address ECCS Evaluation Model errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the peak cladding temperature (PCT) is determined. The resultant increases or decreases in PCT are applied to the analysis of record PCT.

There are no PCT penalties associated with the RLBLOCA analyses presented in this section.

14.6.5.1.8 Conclusions

The UTL results providing a 95/95 simultaneous coverage/confidence level from this evaluation meet the 10 CFR 50.46(b) criteria as seen in Section 14.6.5.1.6. The analysis also shows that the maximum local oxidation UTL is less than the 13% oxidation limit imposed by the NRC on the RLBLOCA EM (Reference 14.6-17). The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-3) criteria (Reference 14.6-14) given in Section 14.6.5.1.5.

14.6.5.2 Small Break Loss of Coolant Accident

The following subsections present the ECCS Small Break Loss of Coolant Accidents (SBLOCA) performance analysis supporting AREVA Standard CE14 HTP Fuel Assembly with M5® fuel rod cladding.

14.6.5.2.1 Event Initiator

This event is initiated by a small break in the primary coolant system pressure boundary. The size of breaks typically considered to be small breaks are from a leak exceeding the makeup capacity of the charging system up to approximately 0.5 ft^2. The most limiting break location is in the RCS cold leg pipe on the discharge side of the reactor coolant pump.

14.6.5.2.2 Event Description

The principal PWR design feature for mitigating the consequences of an SBLOCA is the Emergency Core Cooling System (ECCS) which maintains the water inventory. Its major

subsystems for restoring water inventory are the HPSI system, and the LPSI system and the safety injection tanks.

An SBLOCA is characterized by slow RCS depressurization rates and mass transfer rates within the RCS relative to similar parameters calculated for LBLOCA. If the break area is large enough that the charging pumps cannot maintain the reactor coolant inventory and allow RCS pressure control, the RCS will depressurize. The depressurization produces a low pressurizer pressure (TM/LP) reactor trip and an SIAS. The rate of RCS depressurization following SIAS depends on the break area and the HPSI shutoff head. With a combination of a very small break and a sufficiently high HPSI shutoff head, the depressurization may be arrested.

If the break area is sufficiently large to allow continued depressurization and net loss of coolant inventory even with the HPSI pumps in operation, the coolant level in the reactor vessel may recede below the top of the reactor core. If sufficient steam is produced in the RCS, natural circulation (the RCPs will have been tripped by this time to reduce coolant loss out the break) around the RCS loops will cease. Eventually, loss of reactor coolant inventory is arrested by ECCS flow exceeding flow out the break. In either case, the coolant level within the reactor vessel will rise, and the RCS will eventually be refilled (although leaking) to the cold leg elevation.

14.6.5.2.3 Reactor Protection

Primary reactor protection for this event is provided by the low pressurizer pressure (TM/LP) trip and the SIAS on a low pressurizer pressure signal.

Available Reactor Protection for the Small Break Loss of Coolant Accidents resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary event is summarized in Table 14.6.5.2-1.

14.6.5.2.4 Disposition and Justification

Section 15.6.5 of Reference 14.6-11 indicates that the primary acceptance criteria for this event are to limit offsite doses, to limit fuel clad oxidation, and to keep PCT below 2200°F. Offsite doses are maximized by assuming the highest concentration of radionuclides contained within the fuel pins at event initiation. This is accomplished by assuming steady state radionuclide concentrations characteristic of long term operation of the plant at full power. Fuel pin cladding temperatures and oxidation rates are maximized by initiating the event with the highest cladding temperatures and LHGR. Thus, the most limiting results for this event are obtained with the plant operating at full power in Mode 1. These results will bound those from Modes 2-6.

Disposition of Events for the Small Break Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary event is summarized in Table 14.6.5.2-2.

14.6.5.2.5 Definition of Events Analyzed

The purpose of the SBLOCA analysis is to demonstrate that the criteria stated in 10 CFR 50.46(b) are met. The criteria are:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. The calculated total local oxidation of the cladding shall not exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

14.6.5.2.5.1 Description of Small Break Loss of Coolant Accident Transient

The SBLOCA is generally defined as a break in the PWR pressure boundary which has an area of 0.5 ft^2 (approximately 10% of cold leg pipe area) or less. This range of break areas encompasses small lines which penetrate the primary pressure boundary. Small breaks could involve pressurizer relief and safety valves, charging and letdown lines, drain lines, and instrumentation lines. The limiting break size is generally in the neighborhood of 2% of the cold leg pipe area. The most limiting break location is in the cold leg pipe at the discharge side of the pumps, particularly with primary pumps tripped early to conservatively model a possible loss-of-offsite power on reactor trip. This break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid ejected out the break. This produces the greatest degree of core uncovery and the longest fuel rod heatup time.

The SBLOCA transient is characterized by a slow depressurization of the primary system with a reactor trip occurring at a low primary pressure of 1700 psia (conservatively bounding the actual value of 1865 psia) in the Millstone Unit 2 plant. The SIAS occurs when the system has depressurized to 1500 psia. The capacity and shutoff head of the HPSI pumps are important parameters in the SBLOCA transient. The single failure criterion is satisfied by the loss of one diesel generator. In the Millstone Unit 2 SBLOCA analysis, only one train of safety injection is available.

The SBLOCA transient can be categorized into three ranges of break sizes. The scenario is different for each range of break sizes. Very small breaks are characterized by inventory losses that are less than the makeup capacity of the HPSI pumps such that core uncovery is limited. The core level is eventually recovered and hot rod heatup is limited. Breaks with a large flow area are characterized by a sufficiently large primary system depressurization rate such that the safety

injection tank pressure is reached in sufficient time to limit the core uncovery and hot rod heatup. The HPSI pumps have limited influence on those transients. Breaks with a flow area between the two extremes are generally the most limiting. In those transients, the rate of inventory loss from the primary system is large enough that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is very slow, extending the time required to reach the safety injection tank pressure. This tends to maximize the heatup time of the hot rod and produces the maximum PCT. It also results in the longest time-at-temperature, which maximizes the local cladding oxidation. The limiting break is usually one with a large enough break area that there is a prolonged period of core uncovery.

The time allowed for the operators to manually trip the primary coolant pumps following the SIAS is an important parameter. Allowing the pumps to continue to operate during an SBLOCA would delay break uncovery (transition to mostly steam flow), resulting in additional inventory loss from the system. Subsequent loss of the primary pumps (due to cavitation or an eventual manual trip) would result in more core uncovery, and higher PCTs. The base calculations for this analysis tripped the primary coolant pumps at the reactor scram signal, when a loss of offsite power is assumed to occur. A sensitivity study was also performed to determine an acceptable primary coolant pump trip delay time (Section 14.6.5.2.5.5).

14.6.5.2.5.2 Analytical Models

The approved AREVA SBLOCA evaluation model (References 14.6-12 and 14.6-13) consists of two computer codes. The appropriate conservatisms prescribed by Appendix K of 10 CFR 50 are incorporated.

1. The RODEX2-2A code was used to determine the burnup dependent initial fuel conditions for the system calculations.

2. The S-RELAP5 code was used to model the transient thermal-hydraulic response of the reactor coolant and main steam systems. The S-RELAP5 code also models the thermal-hydraulic response of the hot rod during the transient. The governing conservation equations for mass, energy, and momentum transfer are used along with appropriate correlations consistent with Appendix K of 10 CFR 50.

14.6.5.2.5.3 Plant Description and Summary of Analysis Parameters

The Millstone Unit 2 nuclear power plant is a CE designed PWR which has two hot leg pipes, two U-tube steam generators, and four cold leg pipes with one RCP in each cold leg. The plant utilizes a large dry containment. The RCS is nodalized in the S-RELAP5 model into control volumes representing reasonably homogeneous regions, interconnected by flow paths or “junctions.” Each of the cold legs is modeled separately. The model considers four safety injection tanks, a pressurizer, and two steam generators with both primary and secondary sides of the steam generators modeled. The HPSI pumps were modeled as sources injecting directly into all RCS cold legs, with conservative flows given as a function of system backpressure. The LPSI pumps were modeled as sources injecting directly into two of the RCS cold legs, with conservative flows given as a function of system backpressure. The pumped ECCS flow rates as a function of cold

leg pressure are based on a detailed Safety Injection System flow calculation which includes pump degradation. No charging pump flow was credited in the analysis.

The primary coolant pump performance curves are characteristic of the Millstone Unit 2 pumps. Symmetrical steam generator tube plugging of 500 tubes per steam generator was assumed.

The analysis assumed loss of offsite power concurrent with reactor scram. The single failure criterion required by Appendix K was satisfied by assuming the loss of one Emergency Diesel Generator (EDG), which resulted in the disabling of one HPSI pump, one LPSI pump and one motor-driven auxiliary feedwater (AFW) pump. The swing HPSI pump was not credited, leaving only a single HPSI pump in operation. Initiation of the HPSI and LPSI systems were delayed by 25 and 45 seconds, respectively, beyond the time of SIAS representing the maximum Technical Specification delay time required for EDG startup, switching, and pump startup. The disabling of a motor-driven AFW pump would leave one motor-driven pump and the turbine-driven pump available. The initiation of the motor-driven pump was delayed 240 seconds beyond the time of the Auxiliary Feedwater Actuation Signal (AFAS) indicating low steam generator level (0% narrow range). The operator startup of the turbine-driven AFW pump was not credited in the analysis. The analysis assumes a constant minimum AFW flow rate of 72 gpm per SG. The sweep out of fluid at the main feedwater temperature in the piping between the AFW injection location and the steam generators is also accounted for.

The analysis supports a total unrodded integrated radial peaking factor (F_r^T) of 1.854 (1.69 plus uncertainties), and a maximum Linear Heat Rate (LHR) of 15.1 kW/ft.

All four RCPs were assumed to trip coincident with reactor scram, consistent with a loss of offsite power at the time of trip.

Values for system parameters used in the analysis are given in Table 14.6.5.2-3.

14.6.5.2.5.4 Break Spectrum

The Millstone Nuclear Plant Unit 2 break spectrum analysis for SBLOCA includes breaks of varying diameter up to 10% of the flow area for the cold leg. The spectrum includes a wide enough range of break sizes from 2.0 inch diameter to 9.49 inch diameters to establish a PCT trend. Additional break sizes are performed with a smaller break interval fine enough to identify the limiting break size and to capture different recovery phenomena. The limiting break size was determined to be 3.78-inch diameter (0.07793 ft2), resulting in a PCT of 1707°F. The PCT results for the break spectrum are presented in Table 14.6.5.2-6. Figure 14.6.5.2–1 shows the calculated PCTs for the break spectrum. The predicted event times for the break spectrum are provided in Table 14.6.5.2-5. For the break spectrum analysis, RCP trip is assumed to occur on reactor trip.

The break opens at 0 seconds and initiates a subcooled depressurization of the primary system. The low pressurizer pressure trip setpoint is reached at 19 seconds and within 2 seconds the reactor is tripped. Offsite power is lost, coincident with the turbine trip, RCP trip, and MFW pump trip (Figure 14.6.5.2–2, Figure 14.6.5.2–11, Figure 14.6.5.2–12 and Table 14.6.5.2-5). The SIAS is issued at 27 seconds on low pressurizer pressure. As MFW to the SGs is ramped down,

the pressure in the SGs increase for approximately 30 seconds until the MSSV inlet reaches the lowest opening pressure setpoint. This provides core heat removal in the early stages of the transient.

The primary system depressurization continues at a relatively fast rate for the first 125 seconds as fluid rushes out of the break (Figure 14.6.5.2–3). The broken leg loop seal clears at 356 seconds (Figure 14.6.5.2–6), as demonstrated in the horizontal loop seals and break void fractions in Figure 14.6.5.2–6 and Figure 14.6.5.2–5, respectively.

Prior to loop seal clearing in the broken leg, the core uncovers about 4 feet below the top of the active fuel (Figure 14.6.5.2–9, Figure 14.6.5.2–20 and Figure 14.6.5.2–21). As there is no loop flow, a large amount of steam is generated and accumulated in the core by the decay heat until enough pressure is built to blow the upflow leg of the loop seal in the broken leg around 356 seconds into the transient. This causes an abrupt level drop in the downcomer region (Figure 14.6.5.2–8) with a simultaneous core recovery (Figure 14.6.5.2–9). As the broken leg clears, the plant then enters a fairly slow boil-off phase where mass is lost out the break, and the primary system continues to empty. All intact loops remained plugged for the duration of the transient.

As liquid drains out of the loop piping, the break flow transitions from liquid to two phase flow, and then to steam. The break flow becomes primarily steam around 366 seconds resulting in a reduced mass flow rate out of the break (Figure 14.6.5.2–4) and an increase in the depressurization rate of the primary system (Figure 14.6.5.2–3). The liquid level in the reactor vessel continues to drop until the reactor vessel reaches a minimum level at 1386 seconds (Figure 14.6.5.2–10).

Although HPSI flow to the primary system cold legs began at approximately 62 seconds into the transient (Figure 14.6.5.2–16), it does not provide sufficient inventory at this time to offset the large amounts lost out the break at this time. As effective cooling is lost in the core, the fuel rods begin to heat up at approximately 800 seconds (Figure 14.6.5.2–22). The fuel continues to heat up until the maximum PCT of 1707°F is reached at 1824 seconds. Fuel rod rupture does occur for the hot rod, the calculated blockage factor indicates that the channel around the hot rod is not completely blocked and that all other channels in the core are also not completely blocked. Therefore, the hot rod and all other channels in the core are amenable to cooling.

For this break size, the HPSI flow is eventually sufficient to compensate for the rate of inventory loss out of the break. At the time of the PCT, the primary system has depressurized to a pressure slightly above the SIT pressure and the LPSI shut-off head. SIT injection begins at 4580 seconds (Figure 14.6.5.2–18), followed by LPSI injection 56 seconds later (Figure 14.6.5.2–17), resulting in no influence on PCT turnaround.

The downcomer level (Figure 14.6.5.2–8) and the reactor vessel inventory (Figure 14.6.5.2–10) start slowly increasing at approximately 1400 seconds. The onset of SIT and LPSI injection helps the reactor vessel levels to step up, ensuring core recovery and long term core cooling.

In conclusion, the limiting PCT break spectrum case is a 3.78-inch diameter cold leg break. The PCT of this case is 1707°F. The transient maximum local oxidation (MLO) is 3.6% and the maximum corewide oxidation (CWO) is less than 0.04%. The total maximum local oxidation is less than 6%, including a pre-transient oxidation of 2.3%. The hot rod resulted in rupture, but remained amenable to cooling. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 14.6-14).

14.6.5.2.5.5 Pump Trip Delay Results

For plants such as Millstone Nuclear Plant Unit 2 that do not have an automatic RCP trip, a delayed RCP trip can potentially result in a more limiting condition than tripping the RCPs at reactor trip. Continued operation of the RCPs may result in earlier loop seal clearing with associated two-phase flow out the break, which would result in less inventory loss out the break early in the transient, but in the longer term could result in more overall inventory loss out the break. It has been postulated that tripping the pumps when the minimum RCS inventory occurs could cause a collapse of voids in the core, thus depressing the core level and provoking a deeper core uncovery, and a potentially higher PCT. Therefore, an RCP trip sensitivity for both the cold and hot leg breaks was performed with a delay time following the loss of subcooling margin in the cold leg to demonstrate 10 CFR 50.46(b)(1-4) criteria. This manual RCP trip study was performed consistent with the Combustion Engineering Owners Group guidelines described in Generic Letter 86-06 (Reference 14.6-10) where compliance with 10 CFR 50.46 is demonstrated when operator action to trip the RCPs is taken within 2 minutes after the RCP trip criterion is reached using the 10 CFR 50 Appendix K method.

Also, consistent with Generic Letter 86-06, additional delayed RCP trip sensitivity studies were performed to determine the maximum delay time for operator action under a more realistic scenario. Best-estimate assumptions were applied using the same model as the Appendix K RCP analysis with relaxation in two areas: decay heat multiplier reduction from 1.2 to 1.0 and critical break flow model change from Moody to the Homogeneous Equilibrium Model. A range of RCP trip delay times was examined for both hot and cold leg break locations.

The results of the delayed RCP trip sensitivity, with Appendix K assumptions demonstrated that all four RCPs can be tripped by the operator within 2 minutes after subcooling margin is lost in the cold leg pump suction in order to meet the 10 CFR 50.46(b)(1-4) criteria. In addition, relaxation of Appendix K assumptions demonstrated that longer delay times of up to 10 minutes could be accommodated and still meet the 10 CFR 50.46(b)(1-4) criteria.

14.6.5.2.5.6 Asymmetric Steam Generator Tube Plugging

Small break LOCA analyses have no significant sensitivity to steam generator tube plugging. The analysis was performed using the maximum allowed total tube plugging of 1000 tubes applied symmetrically (500 tubes per steam generator), and supports the maximum allowed asymmetry between the generators of 500 tubes. By applying the maximum allowed tube plugging, the analysis is conservative because the initial primary system mass inventory is minimized.

14.6.5.2.5.7 Reduced Primary Temperature Operation

An assessment to support an end of cycle full power coastdown to an indicated RCS cold leg temperature of 537°F has been performed. The result of the assessment confirmed that break spectrum analysis bounds operation during an end of cycle full power coastdown to an indicated RCS cold leg temperature of 537°F.

14.6.5.2.5.8 Attached Piping Break Sensitivity Study

Although breaks in the attached piping are not typically PCT limiting, they do result in reduced ECCS flows available to mitigate the event. Therefore, an analysis of the limiting break size and location in attached piping was performed. For Millstone Nuclear Plant Unit 2, the limiting break location and size for an attached piping break is a double-ended guillotine break of a SIT line. The break is located in the SIT line connected to Loop 2B.

For the double-ended guillotine break in the SIT line, the calculated PCT is 1239°F, which is bounded by the limiting PCT of the RCS break spectrum. The minimal HPSI and LPSI flow rates modeled were sufficient to prevent a subsequent heatup after the initial quench from the SIT discharge.

14.6.5.2.5.9 Safety Injection Low Fluid Temperature Sensitivity Study

A sensitivity study was performed on which SIT and safety injection (SI) fluid temperatures were reduced to approximate nominal temperatures as opposed to the maximum temperatures used in the spectrum analysis. The result of the sensitivity study confirmed that the SIT and SI fluid temperatures used in the break spectrum are conservative.

14.6.5.2.6 Analysis Results

The limiting PCT break spectrum case is a 3.78-inch diameter cold leg break. The PCT of this case is 1707°F. The transient maximum local oxidation (MLO) is 3.6% and the maximum core-wide oxidation (CWO) is less than 0.04%. The total maximum local oxidation is less than 6%, including a pretransient oxidation of 2.3%. The hot rod resulted in rupture, but remained amenable to cooling. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 14.6-14).

The results of the delayed RCP trip sensitivity with Appendix K assumptions demonstrated that all four RCPs can be tripped by the operator within 2 minutes after subcooling margin is lost in the cold leg pump suction in order to meet the 10 CFR 50.46(b)(1-4) criteria. In addition, relaxation of Appendix K assumptions demonstrated that longer delay times of up to 10 minutes could be accommodated and still meet the 10 CFR 50.46(b)(1-4) criteria.

The attached pipe break and the ECCS temperature sensitivity results are bounded by the RCS break spectrum. The sensitivity studies justify the applicability of the break spectrum as the licensing basis.

Margin between the calculated PCT and the 2200°F limit of 10 CFR 50.46 is available to accommodate other permanent adjustments due to 10 CFR 50.59 Safety Evaluations and LOCA model assessments. These adjustments are summarized in the 30 day and annual 10 CFR 50.46 reporting of PCT Margin Utilization. The reporting process and attention to PCT margin assure that the PCT remains below the 2200°F limit of 10 CFR 50.46.

The analysis supports full power operation at 2754 MWt (2700 MWt plus 2% uncertainty). A maximum LHR of 15.1 kW/ft and a radial peaking factor of 1.69 are supported by this analysis. The analysis demonstrates that the 10 CFR 50.46(b) criteria are satisfied for the Millstone Unit 2 reactor.

14.6.5.2.7 Post Analysis of Record Evaluations

In addition to the analyses presented in this section, evaluations and reanalysis may be performed as needed to address ECCS Evaluation Model errors and emergent issues, or to support plant changes. The issues or changes are evaluated, and the impact on the peak cladding temperature (PCT) is determined. The resultant increases or decreases in PCT are applied to the analysis of record PCT.

The PCT, including all penalties, is presented in Table 14.6.5.2-7 for the Small Break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

14.6.5.2.8 Conclusions

The limiting PCT break spectrum case is a 3.78-inch diameter cold leg break. The PCT of this case is 1707°F. The transient maximum local oxidation (MLO) is 3.6% and the maximum core-wide oxidation (CWO) is less than 0.04%. The total maximum local oxidation is less than 6%, including a pretransient oxidation of 2.3%. The hot rod resulted in rupture, but remained amenable to cooling. The results of the analysis demonstrate the adequacy of the ECCS to support the 10 CFR 50.46(b) (1-4) criteria (Reference 14.6-14) given in Section 14.6.5.2.5.

14.6.5.3 Post-LOCA Long Term Cooling

Following the short term mitigation actions for a large or small break LOCA (as discussed in Sections 14.6.5.4 and 14.6.5.2, respectively), long term cooling will continue to maintain the core at an acceptably low temperature. LOCA mitigation in the long term will be accomplished by the methods referred to as the Long Term Cooling (LTC) Plan. The LTC Plan (shown in Figure 14.6.5.3–1) consists of the events and actions that will assure acceptable long term core cooling and prevention of boric acid precipitation in the core region. The Post-LOCA Long Term Cooling Analysis, demonstrates that Post-LOCA Long Term core cooling and boric acid precipitation prevention can be accomplished for all LOCAs.

14.6.5.3.1 The Post-LOCA Long Term Cooling Plan

Figure 14.6.5.3–1 shows the basic sequence of events for the initial automatic actions and the subsequent operator actions of the LTC Plan. The operator’s first action is to initiate a plant

cooldown within 1 hour post-LOCA by releasing steam from the steam generators. The steam is released either through the turbine bypass system, if it is available, or through the atmospheric dump valves (ADVs).

When pressurizer pressure is less than 600 psia and stable with a controlled cooldown in progress, the Safety Injection Tanks (SITs) are isolated to avoid injecting nitrogen, a non-condensable gas, into the reactor coolant system (RCS).

At 8 to 10 hours post-LOCA, the operator will determine if the RCS is filled. If the RCS is filled, then natural circulation will prevent boric acid precipitation and simultaneous hot and cold leg injection will not be necessary. The operators will attempt to establish shutdown cooling (SDC) if entry conditions exist or can be established. If SDC cannot be established (whether due to single failure, SDC pressure/temperature limits unsatisfied, or RCS activity beyond appropriate limits), then steam generator (SG) cooling will be continued.

If the RCS is not filled at 8 to 10 hours post-LOCA, then simultaneous hot and cold leg injection will be established to provide core flushing. The preferred method of simultaneous hot and cold leg injection is low pressure safety injection (LPSI) to the hot side (one LPSI pump to the SDC warm up line to the SDC suction line to a hot leg.) Cold side injection will be via a high pressure safety injection (HPSI) pump and the running LPSI pump. If the preferred method cannot be established, then the alternative method of simultaneous hot and cold leg injection, which is HPSI injection to the hot side (1 HPSI pump to a charging line to the pressurizer auxiliary spray line to a hot leg), will be established. Cold side injection will be via a LPSI pump. Either of these simultaneous hot and cold leg injection methods will provide flow that is adequate to cool the core and prevent boric acid precipitation.

14.6.5.3.2 Post-LOCA Long Term Cooling Equipment and Operator Actions

The following discussion elaborates on equipment and operator actions that support the Long Term Cooling Plan.

As stated above, simultaneous hot and cold leg injection will be required if the RCS is not filled at 8 to 10 hours post-LOCA.

If simultaneous hot and cold leg injection is required, then either the preferred method (LPSI hot leg injection) or the alternative method (HPSI hot leg injection) can be established and operated despite various single failures to simultaneous hot and cold leg injection equipment. Either simultaneous hot and cold leg injection configuration will provide adequate delivery flows while ensuring acceptable HPSI and LPSI pump operation. Operator actions outside the control room will be required to realign manually operated valves. Additional operator actions outside the control room will be required in the event of either a facility Z1 or Z2 loss of power (when offsite power is unavailable). These additional operator actions are described as follows.

For a failure of Facility Z1, the position of the LPSI injection valves 2-SI-615, 2-SI-625 must be known, whereas a failure of Facility Z2 requires the position of LPSI injection valves 2-SI-635

and 2-SI-645 to be known. This is required to correctly align for simultaneous hot and cold leg injection. These positions would be determined by operator actions outside the control room.

For the failure of the emergency Facility Z1 AC power to the safety injection system (SIS), the operator actions to establish LPSI hot leg injection would include aligning an alternate power source to SDC suction line valve 2-SI-651. This manual aligning would require operator action outside the control room.

For the failure of the emergency Facility Z2 DC power for the SIS train 2, the operator actions to establish HPSI hot leg injection would include aligning an alternate power source to charging valves 2-CH-517 and 2-CH-519. This manual aligning would require operator action outside the control room.

In addition to the above operator actions directly concerned with the simultaneous hot and cold leg injection realignment and operation, the following operator actions outside the control room may be required to support the Long Term Cooling Plan.

If the condenser is unavailable, the auxiliary feedwater (AFW) system and the ADVs will be used to cooldown the RCS. The cooldown will be initiated within 1 hour after the start of the LOCA. If, when initiating the cooldown, the ADVs are in a closed position, then they will be manually opened by operator action outside the control room. If the primary source of AFW to the steam generator—the condensate storage tank (CST)—becomes depleted beyond the capability of the Ecolochem system to replenish, then operator action outside the control room will be required to manually realign the Fire Protection System to supply AFW.

14.6.5.3.3 Assumptions Used in the Long Term Cooling Analysis

The major assumptions used in performing the Long Term Cooling analysis are listed below:

1. No offsite power is available.

2. The worst single failure is the failure of an emergency diesel generator. As a consequence of the failure, one ECCS train, one containment spray pump and one motor-driven auxiliary feedwater pump are unavailable.

3. Plant cooldown begins at two hours post-LOCA. (The EOPs conservatively initiate the cooldown within one hour post-LOCA.)

4. The analysis assumes that a cooldown rate of 40°F/hr is maintained until the ADVs are fully open (i.e., until flow limiting of the ADVs causes the cooldown rate to decrease from 40°F/hr).

5. The SITs are isolated prior to establishing shutdown cooling.

6. The pressurizer is included in the mass that is cooled down in establishing shutdown cooling entry conditions.

7. A continuous supply of auxiliary feedwater is available for the duration of steam generator cooling. One turbine-driven and one motor-driven auxiliary feedwater pump are assumed to be in operation.

8. Initial boric acid concentrations and inventories and pump flow rates used in the boric acid precipitation analysis are selected to maximize the boric acid concentration in the core.

9. A boric acid precipitation limit of 27.6 wt% is used in the large break LOCA boric acid precipitation analysis. This is the precipitation limit in saturated water at 14.7 psia.

14.6.5.3.4 Method of Analysis

The objective of the post-LOCA Long Term Cooling analysis is to demonstrate that the Long Term Cooling Plan provides conformance to 10 CFR 50.46 Criterion 5, Long Term Cooling, of the ECCS acceptance criteria (Reference 14.6-14). Conformance is demonstrated by showing that under the Long Term Cooling Plan the calculated core temperature is maintained at an acceptably low value and that the boric acid concentration in the core is maintained below its solubility limit.

The Millstone 2 post-LOCA Long Term Cooling analysis was performed using the NRC accepted computer codes described in Reference 14.6-15. As described in Reference 14.6-15, the CELDA computer code is used to analyze the post-LOCA thermal-hydraulic response of the RCS for a spectrum of break sizes. The NATFLOW computer code is used to calculate RCS temperatures for the purpose of determining when the shutdown cooling entry temperature is achieved. The steam generator cooldown transient that is used as a boundary condition in CELDA and NATFLOW is calculated using the CEPAC computer code. The BORON computer code is used to calculate the boric acid concentration in the core following the LOCA.

The Millstone 2 post-LOCA Long Term Cooling Plan was developed using the NRC accepted methods described in Reference 14.6-15 with the following modification. In Reference 14.6-15, RCS pressure is used as the basis for determining whether to branch to shut down cooling (or continue steam generator cooling if SDC is inoperable) or to branch to simultaneous hot and cold side injection. In the Millstone 2 Long Term Cooling Plan, RCS status (i.e., the RCS is or is not filled) is used as the basis. This approach is consistent with the ABB CE emergency procedure guidelines (Reference 14.6-16).

14.6.5.3.5 Parameters Used in the Long Term Cooling Analysis

Significant core and system parameters used in the Long Term Cooling analysis are presented in Table 14.6.5.3-1.

14.6.5.3.6 Results of the Long Term Cooling Analysis

Figure 14.6.5.3–1 shows the Millstone 2 Long Term Cooling Plan. At 8 to 10 hours post-LOCA, the operator determines whether or not the RCS is filled. Eight to 10 hours is used as the decision

time because it provides the operator with ample time to initiate simultaneous hot and cold side injection prior to the earliest time that boric acid precipitation would occur.

The left branch of the Long Term Cooling Plan applies to those break sizes for which the RCS is filled at 8 to 10 hours. For these breaks, SDC operation or continued steam generator cooling will provide heat removal and natural circulation will prevent further boric acid buildup in the core.

The right branch of the Long Term Cooling Plan applies to those break sizes for which the RCS is not filled at 8 to 10 hours. For those break sizes, simultaneous hot and cold side injection is used to maintain core cooling and to provide for boric acid precipitation control.

A double-ended guillotine break in the cold leg is the limiting break for boric acid precipitation control. A double-ended guillotine break is limiting because the low RCS pressure associated with such a large break minimizes the boric acid solubility limit in the core. The cold leg is the limiting break location because it requires the initiation of hot side injection in order to create a core flushing flow to control boric acid precipitation. For a cold leg break, the core flushing flow is the difference between the hot side injection flow rate and the core boiloff flow rate.

As shown in Figure 14.6.5.3–3, the initiation of a hot side injection flow rate of at least 180 gpm at 13 hours post-LOCA provides a substantial and time-increasing core flushing. Figure 14.6.5.3–4 shows that with no core flushing flow, boric acid would begin to precipitate at approximately 15 hours post-LOCA. However, with a hot side injection flow rate of 180 gpm, initiated at 13 hours post-LOCA, the maximum boric acid concentration in the core is 25.4 wt% as compared to the precipitation limit of 27.6 wt%. The margin provided for the prevention of boric acid precipitation by a constant core flushing flow of 20 gpm is also shown in Figure 14.6.5.3–4.

The time by which the entrainment of hot side injection by the steam flowing in the hot leg would cease was calculated to be less than 2 hours post-LOCA. Therefore, the initiation of hot side injection at 13 hours is well after the potential for the entrainment of the hot side injection has ended.

In order for the most limiting configuration of simultaneous hot and cold leg injection to provide the required hot side injection flow rate of 180 gpm, the RCS pressure must be 86 psia or less. As shown in Figure 14.6.5.3–2, the RCS will not be filled at 8 hours for a break area as small as 0.01 ft^2. Consequently, this is the smallest break for which simultaneous hot and cold side injection would be required. The 0.01 ft^2 break was calculated to achieve a RCS pressure of 86 psia prior to 13 hours post-LOCA. Larger breaks will also reach 86 psia prior to 13 hours post-LOCA. This demonstrates that the simultaneous hot and cold leg injection configurations will provide sufficient hot side injection for all breaks for which it may be required.

14.6.5.3.7 Conclusions of the Long Term Cooling Analysis

The Millstone 2 post-LOCA Long Term Cooling analysis demonstrates conformance to 10 CFR 50.46 Criterion 5 of the ECCS acceptance criteria (Reference 14.6-14) for a complete spectrum of break sizes and locations. For breaks that are small enough for the RCS to refill at 8

to 10 hours post-LOCA, shutdown cooling or steam generator cooling provides core cooling and boric acid precipitation control. For breaks that are too large for the RCS to refill at 8 to 10 hours, initiating simultaneous hot and cold side injection provides core cooling and boric acid precipitation control. A simultaneous hot and cold side injection flow rate of 180 gpm (i.e., a flow rate of 180 gpm to both the hot side and cold side of the RCS) initiated by 13 hours post-LOCA maintains the boric acid concentration in the core below the solubility limit.

14.6.5.4 Deleted

14.6.6 REFERENCES

14.6-1 "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR), XN-NF-74-5(A), Rev. 2 and Supplements 3-6, Exxon Nuclear Company, Richland, WA 99352, October 1986.

14.6-2 "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(A), Revision 2, Exxon Nuclear Company.

14.6-3 Deleted

14.6-4 "RETRAN-02 A Program For Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CNN, dated October 1984.

14.6-5 W. G. Counsil letter to J. R. Miller, Docket No. 50-336, dated December 12, 1983.

14.6-6 Technical Specifications for Millstone Unit 2, Docket No. 50-336, Updated through Amendment Number 116.

14.6-7 Deleted

14.6-8 Deleted

14.6-9 Deleted

14.6-10 Nuclear Regulatory Commission Generic Letter 86-06, "Subject: Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' (Generic Letter No. 86-06)," May 29, 1986

14.6-11 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

14.6-12 EMF-2328(P)(A), Rev. 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based", March 2001.

14.6-13 EMF-2328(P)(A), Rev. 0, Supplement 1 (P)(A), Rev. 0, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” September 2015.

14.6-14 Code of Federal Regulations, Title 10, Part 50, Section 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors”.

14.6-15 CENPD-254-P-A, “Post-LOCA Long Term Cooling Evaluation Model”, June 1980, (Proprietary).

14.6-16 CEN-152, Rev. 04, “Combustion Engineering Emergency Procedure Guidelines”, October, 1996.

14.6-17 EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” June 2016.

14.6-18 NUREG/CR-5249, “Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large Break, Loss-of-Coolant Accident,” U.S. NRC, December 1989.

14.6-19 Regulatory Guide 1.203, “Transient and Accident Analysis Methods” U.S. NRC, December 2005.

TABLE 14.6.1-1 AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE EVENT

Reactor Operational Mode	Reactor Protection
1, 2 (≥ 1E-4% RTP)	Thermal Margin/Low Pressure Trip Safety Injection Actuation Signal
2 (< 1E-4% RTP), 3	Safety Injection Actuation Signal Available Thermal Margin *
4	Available Thermal Margin *
5, 6	No Significant Consequences for these Reactor Operating Conditions

* Defense in depth. Significantly low power levels do not present a challenge to SAFDLs.

TABLE 14.6.1-2 DISPOSITION OF EVENTS FOR THE INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER RELIEF VALVE EVENT

Reactor Operational Mode	Disposition
1	Analyze for DNBR
2-6	Bounded by the above

TABLE 14.6.1-3 EVENT SUMMARY FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE

Event	Time (seconds)
Letdown Valve Open	0.00
Pressurizer Relief Valve Opens	0.01
Reactor Scram Signal	8.21
Turbine Stop Valve Closed	9.13
Peak Power	9.58
MDNBR	9.69
Peak Core Average Temperature	9.72
Steam Line Safety Valves Open	12.35
Peak Steam Dome Pressure	13.60

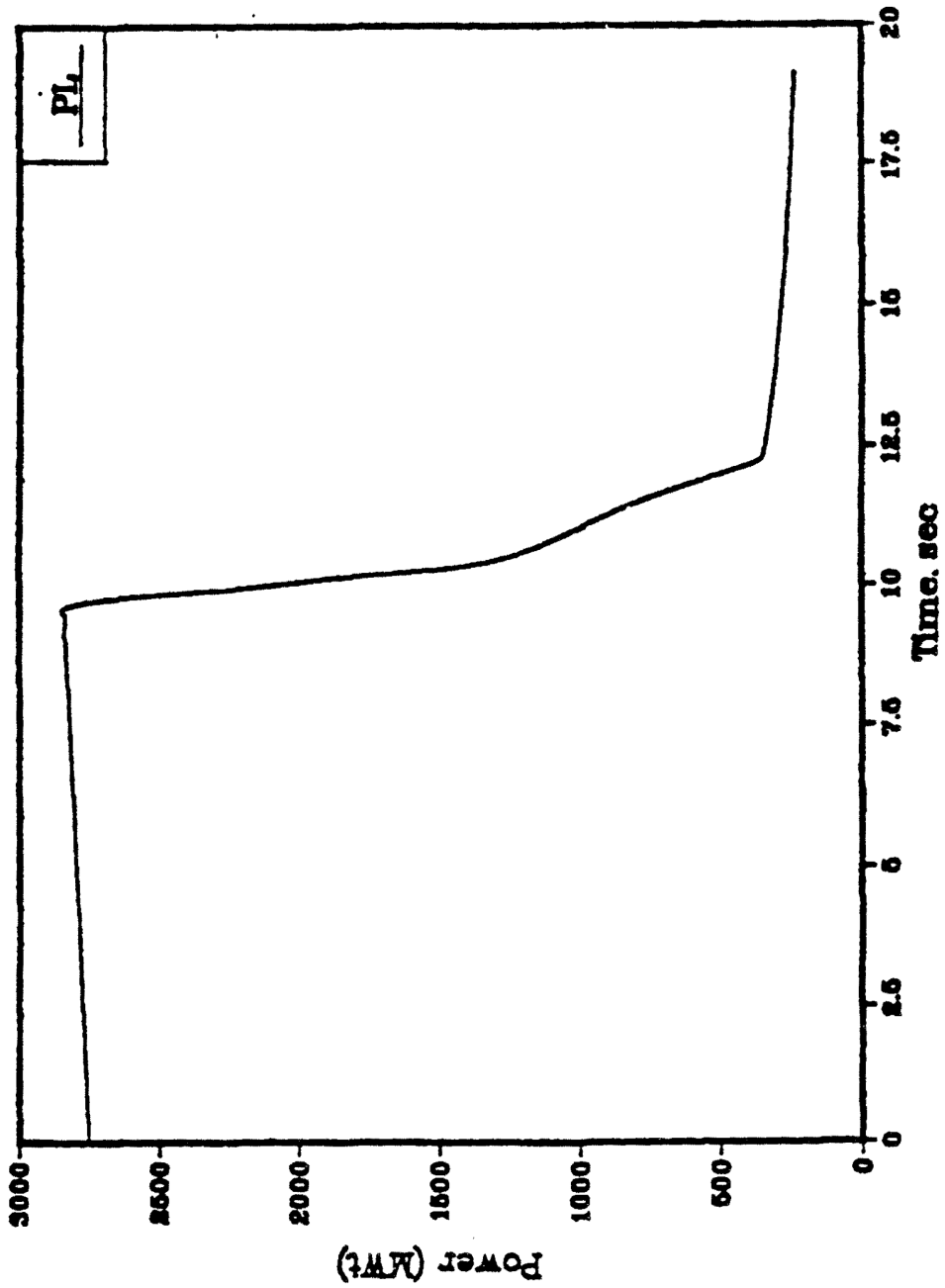


FIGURE 14.6.1–1 REACTOR POWER LEVEL FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

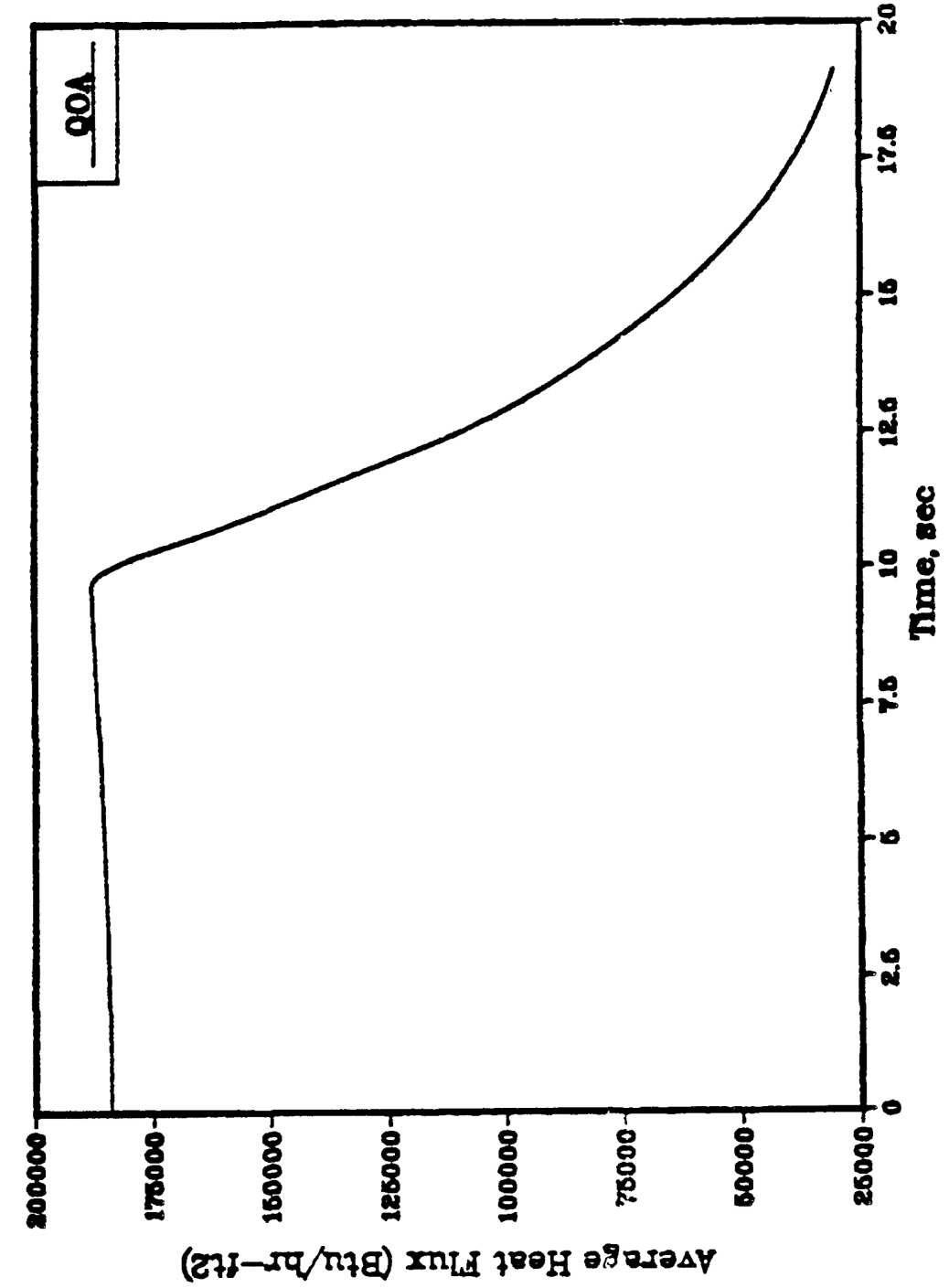


FIGURE 14.6.1–2 CORE AVERAGE HEAT FLUX FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

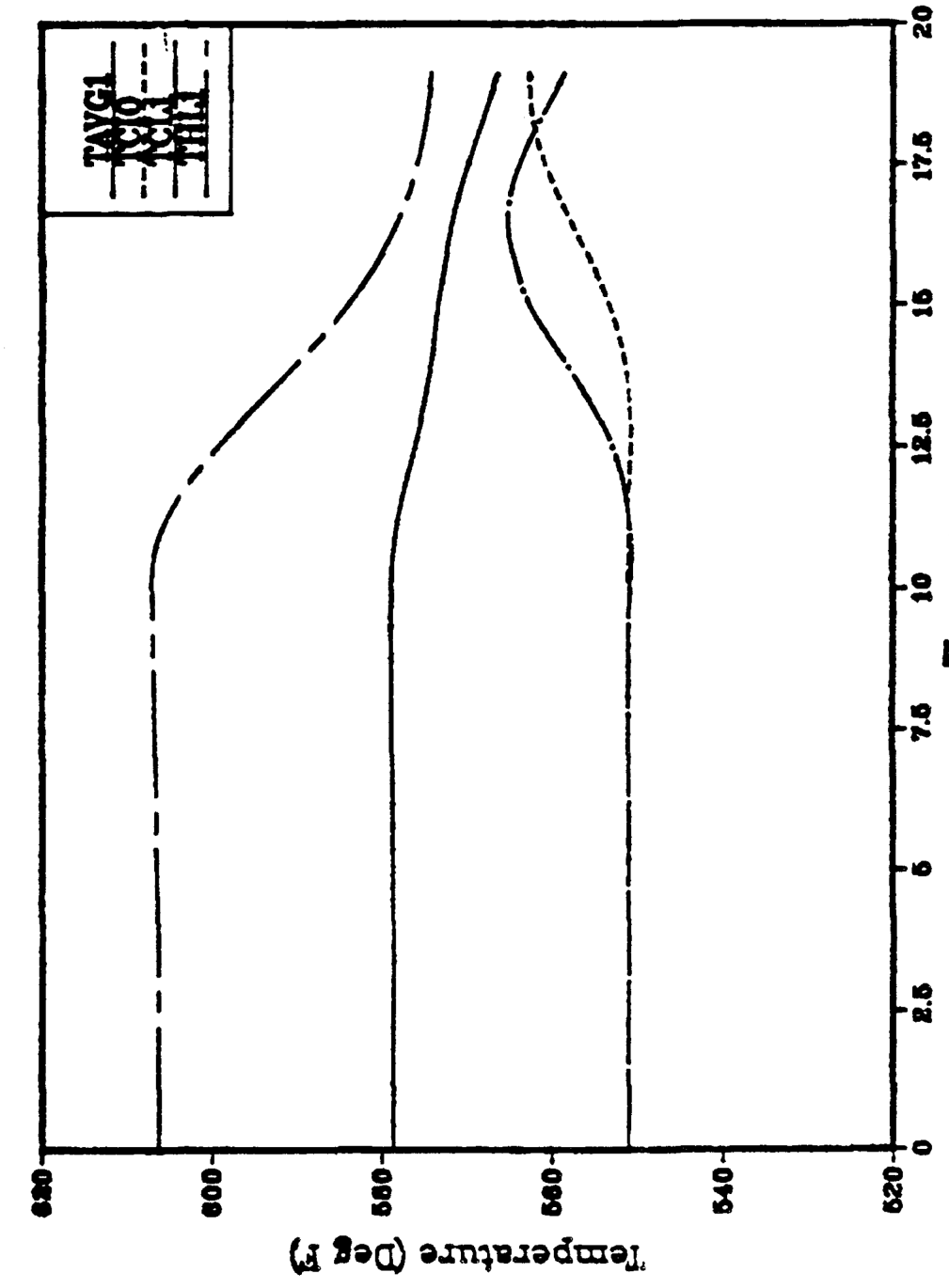


FIGURE 14.6.1–3 REACTOR COOLANT SYSTEM TEMPERATURES FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

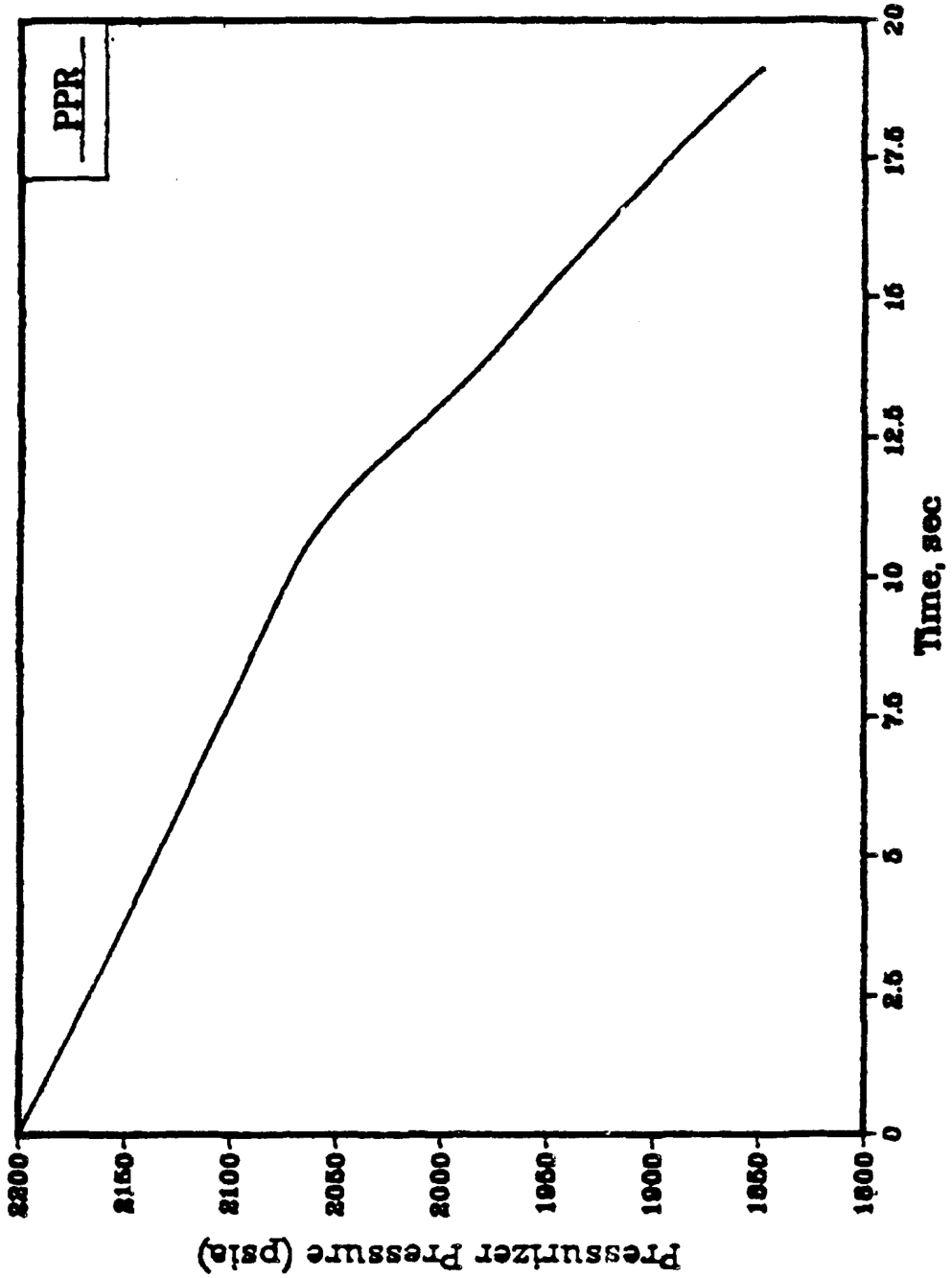


FIGURE 14.6.1–4 PRESSURIZER PRESSURE FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

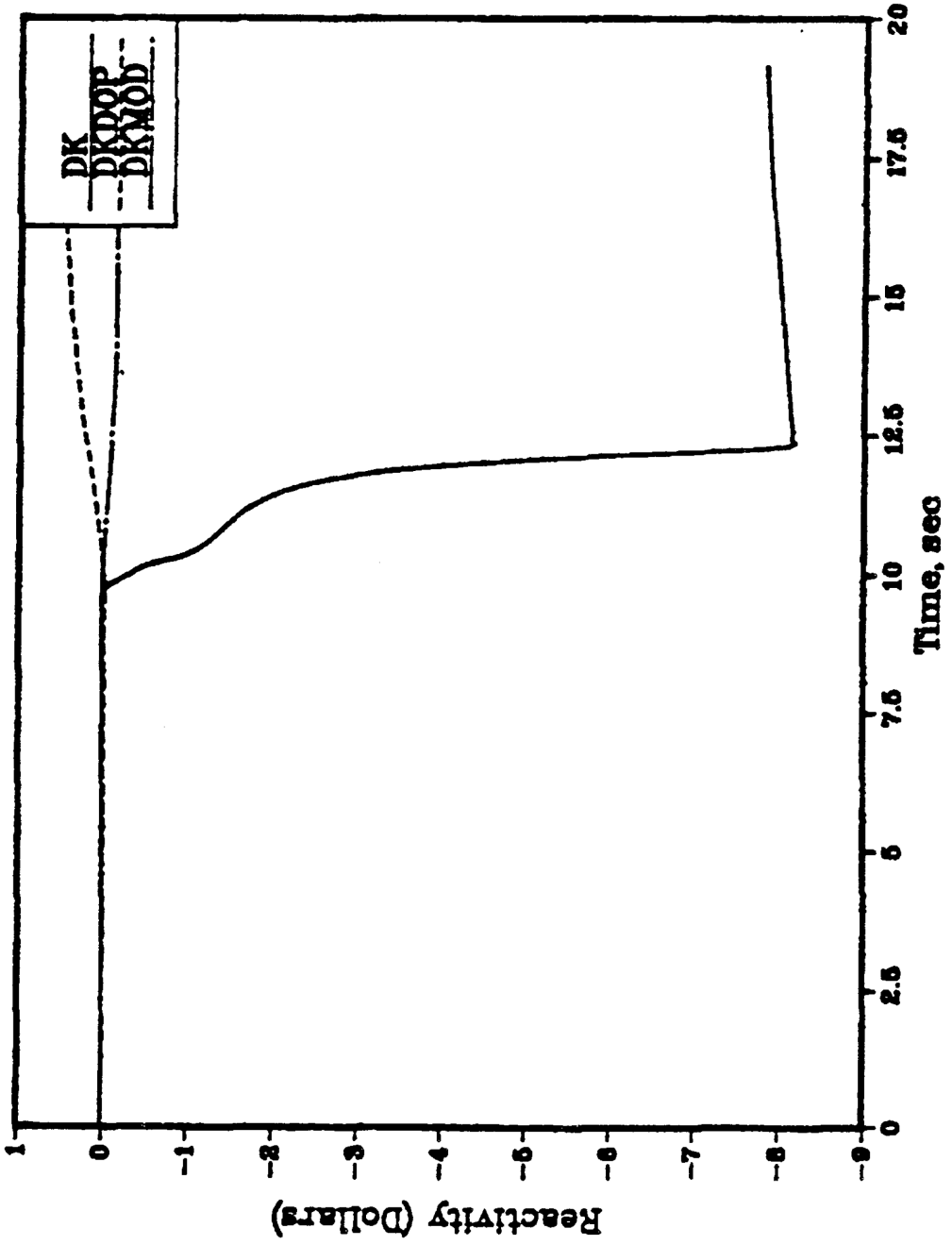


FIGURE 14.6.1–5 REACTIVITIES FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

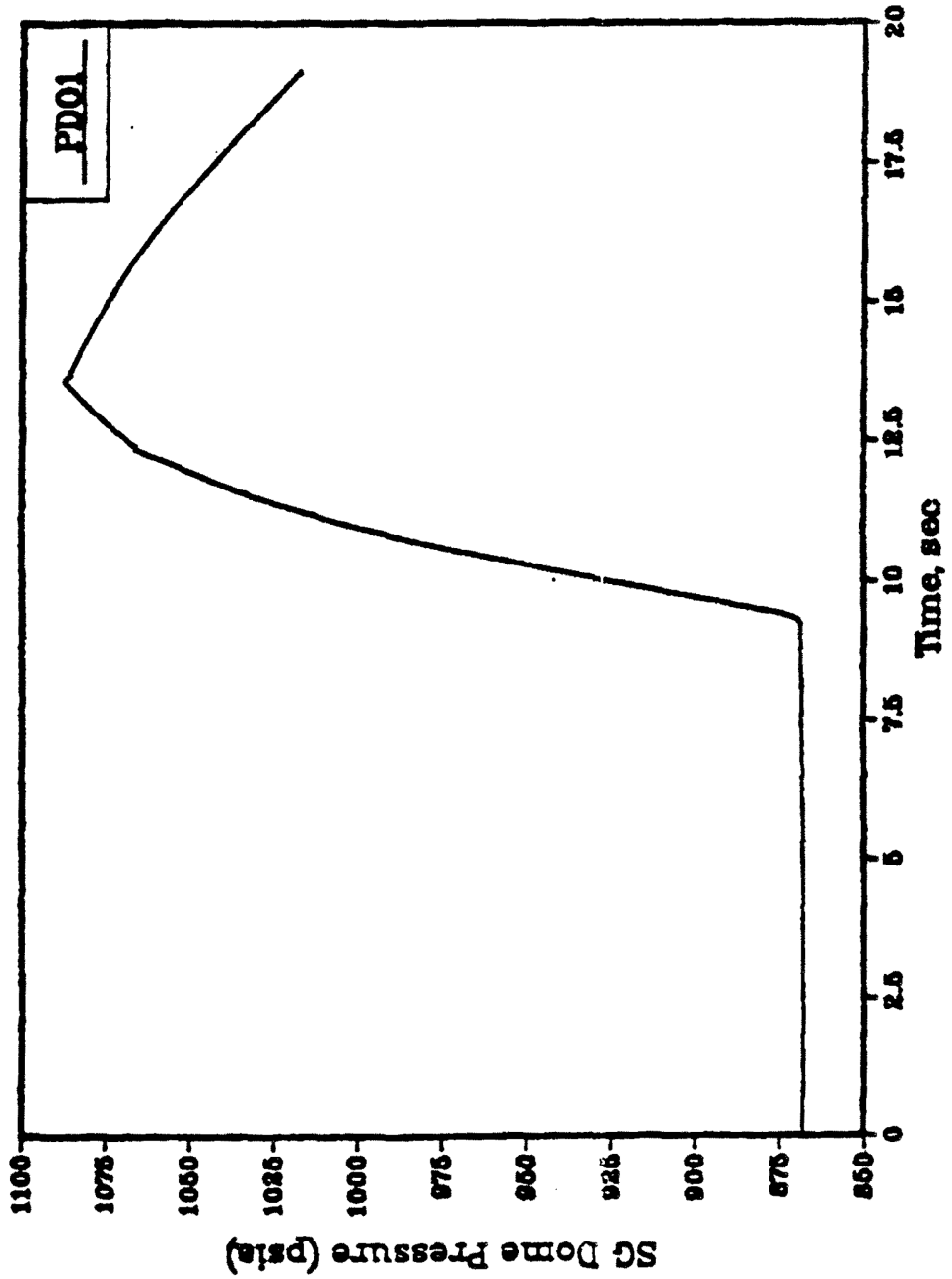


FIGURE 14.6.1–6 SECONDARY PRESSURE FOR AN INADVERTENT OPENING OF A PRESSURIZED WATER REACTOR PRESSURIZER PRESSURE RELIEF VALVE (RATED POWER)

TABLE 14.6.3-1 AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Reactor Operational Modes	Reactor Protection
1	Thermal Margin/Low Pressure Trip
	Safety Injection Actuation Signal
2, 3	Safety Injection Actuation Signal
4-6	No Significant Consequences for These Reactor Operational Modes

TABLE 14.6.3-2 DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Reactor Operational Modes	Disposition
1	Analyze radiological consequences
	Fuel performance bounded by Event 14.6.1
2-6	Bounded by the above

TABLE 14.6.3-3 SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME [1] (seconds)	EVENT	SETPOINT OR VALUE
0	Tube Rupture Occurs	
0	Reactor Trip	
0	Concurrent Loss of Offsite Power; Loss of Forced Circulation	
0	Loss of Instrument Air; Loss of ADV Auto-Actuation	
1	Ruptured SG MSSVs Begin to Lift (MSSVs Cycle - See Figure 14.6.3–8)	970 psia
3	Intact SG MSSVs Begin to Lift (MSSVs Cycle - See Figure 14.6.3–8)	1030 psia
9	Maximum SG Pressure Reached	
	- Ruptured SG	1054 psia
	- Intact SG	1054 psia
37	AFW Actuation Condition Reached on Low SG Level	10% Narrow Range
277	AFW Delivery Starts	
530	Pressurizer Empties	
1200	Intact SG MSSVs Close for Final Time	940 psia
1220	SI Flow Begins to Enter RCS	
1548	Ruptured SG MSSVs Close for Final Time	880 psia
1800 [2]	RCS Cooldown to T_{hot} < 515°F Initiated Using ADVs (Manual Local Action)	
3090	Pressurizer Begins to Refill	
3637	T_{hot} < 515°F Achieved; Ruptured Steam Generator Isolated	

1. Time values are rounded to the nearest second.
2. This is an assumed analytical time and is not a required operator action time.

TABLE 14.6.3-4 MASS RELEASES FOR THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

Affected Steam Generator - Break Flow

Time Period, hours		Total Break Flow	Flashed Break Flow	Liquid Break Flow
From	To	lbm	lbm	lbm
0	1	165,000	5,500	159,500
After 1 hour (1)		57,895	1,346	56,549

Affected Steam Generator - Total Steam Flow Through MSSVs and ADVs

Time Period, hours		Mass Flow
From	To	lbm
0	1	1.870E+05
1	(1)	1.032E+05

Intact Steam Generator - Total Steam Flow Through MSSVs and ADVs

Time Period, hours		Steam Flow Rate, lbm/minute
From	To	
0.00	1.00	2.200E+03
1.00	1.11	8.224E+03
1.11	1.71	5.775E+03
1.71	2.33	4.712E+03
2.33	2.74	4.308E+03
2.74	3.18	4.275E+03
3.18	3.72	4.241E+03
3.72	6.50	3.078E+03
6.50	17.61	2.413E+03

(1) until RCS is on shutdown cooling - release stops

TABLE 14.6.3-5 ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

Primary to Secondary Leak Rate for Intact Steam Generator	150 gpd
Primary Coolant Iodine Concentration	0.5μCi/gm DE I-131
Secondary Coolant Iodine Concentration	0.05 μCi/gm DE I-131
Primary Coolant Noble Gas Concentration	550 μCi/gm DE Xe-133
Pre-accident Spike Iodine Concentration	30 μCi/gm DEQ I-131

Concurrent Iodine Spike equivalent to 335 times iodine appearance rate at 0.5 μCi/gm DEQ I-131

Loss of Offsite Power is assumed at time of tube rupture.

Steam Generator Partition Factors

Iodine	0.01
Noble Gases	1
Particulates [Note 1]	0.01

Reactor Coolant Minimum Mass	423,000 lbm
Steam Generator Minimum	80,000 lbm

Site Boundary Breathing Rate (m^3/sec)

EAB:	0 - 720 hour	3.5 E -04
LPZ:	0 - 8 hours	3.5 E -04
	8 - 24 hours	1.8 E -04
	24 - 720 hours	2.3 E -04

Site Boundary Dispersion Factors (sec/m^3)

EAB:	0 - 2 hours	3.66 E -04
LPZ:	0 - 4 hours	4.80 E -05
	4 - 8 hours	2.31 E -05
	8 - 24 hours	1.60 E -05
	24 - 96 hours	7.25 E -06
	96 - 720 hours	2.32 E -06

Control Room Breathing Rate	3.5 E -04 m^3/sec

Note 1: This constant value for Steam Generator Partition Factor for Particulates is a conservative representation of the moisture carryover anticipated during the event.

TABLE 14.6.3-5 ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE STEAM GENERATOR TUBE RUPTURE EVENT (CONTINUED)

Control Room Isolation Time after Event Initiation

pre-accident spike - 20 seconds

(includes time for damper closure and radiation monitor response)

concurrent spike - 620 seconds

(includes time for damper closure and radiation monitor response)

Control Room Intake Prior to Isolation	800 cfm
Control Room Inleakage During Isolation	250 cfm
Control Room Filtered Recirculation Rate (t=1 hour, 20 sec)	2,250 cfm

Control Room Intake Dispersion Factors (sec/m^3)

Atmospheric Dump Valves (ADVs)	
0 - 2 hours	7.40 E -3
2 - 8 hours	5.71 E -3
8 - 24 hours	2.13 E -3
24 - 96 hours	1.74 E -3
96 - 720 hours	1.43 E -3

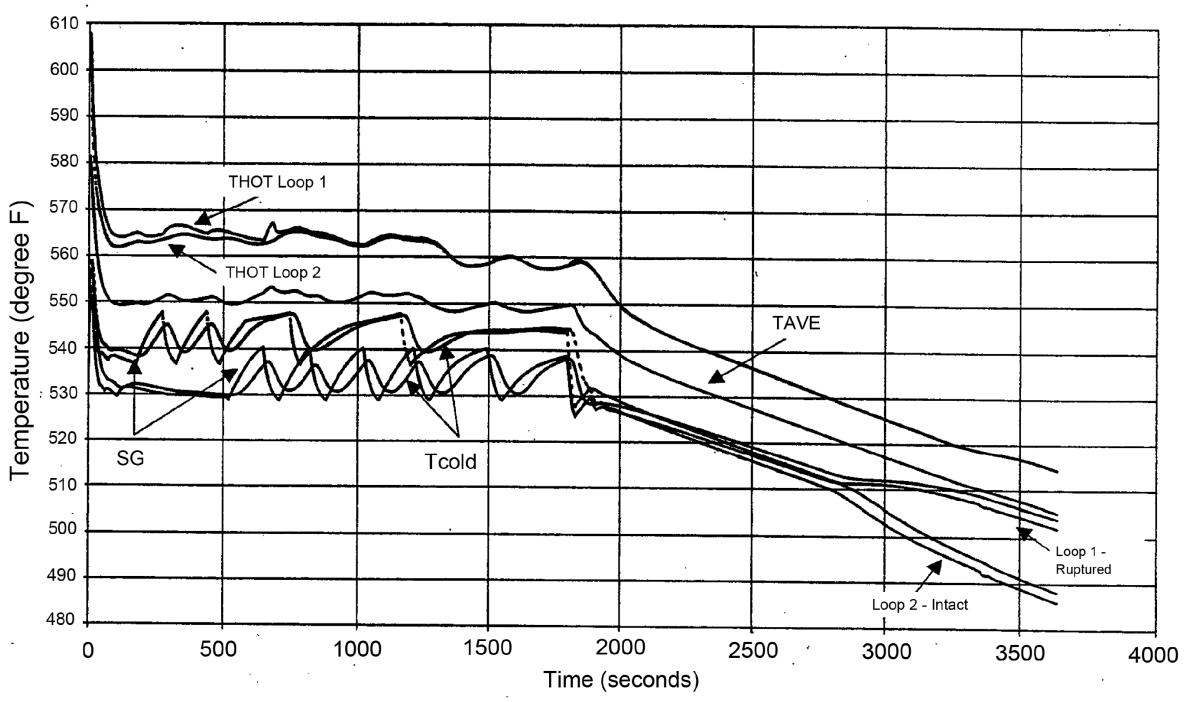
Control Room Free Volume	35,650 ft^3
Control Room Filter Efficiency (particulate/elemental/organic)	90 / 90 / 70 % (1)
Dose Conversion Factors	Federal Guidance Reports 11 and 12

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

TABLE 14.6.3-6 SUMMARY - RADIOLOGICAL CONSEQUENCES OF THE STEAM GENERATOR TUBE RUPTURE EVENT

SGTR	EAB, rem-TEDE	LPZ, rem-TEDE	Control Room, rem-TEDE
concurrent spike	0.8	0.2	3.1
pre-accident spike	1.0	0.2	3.4

FIGURE 14.6.3–1 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER RCS TEMPERATURE VERSUS TIME



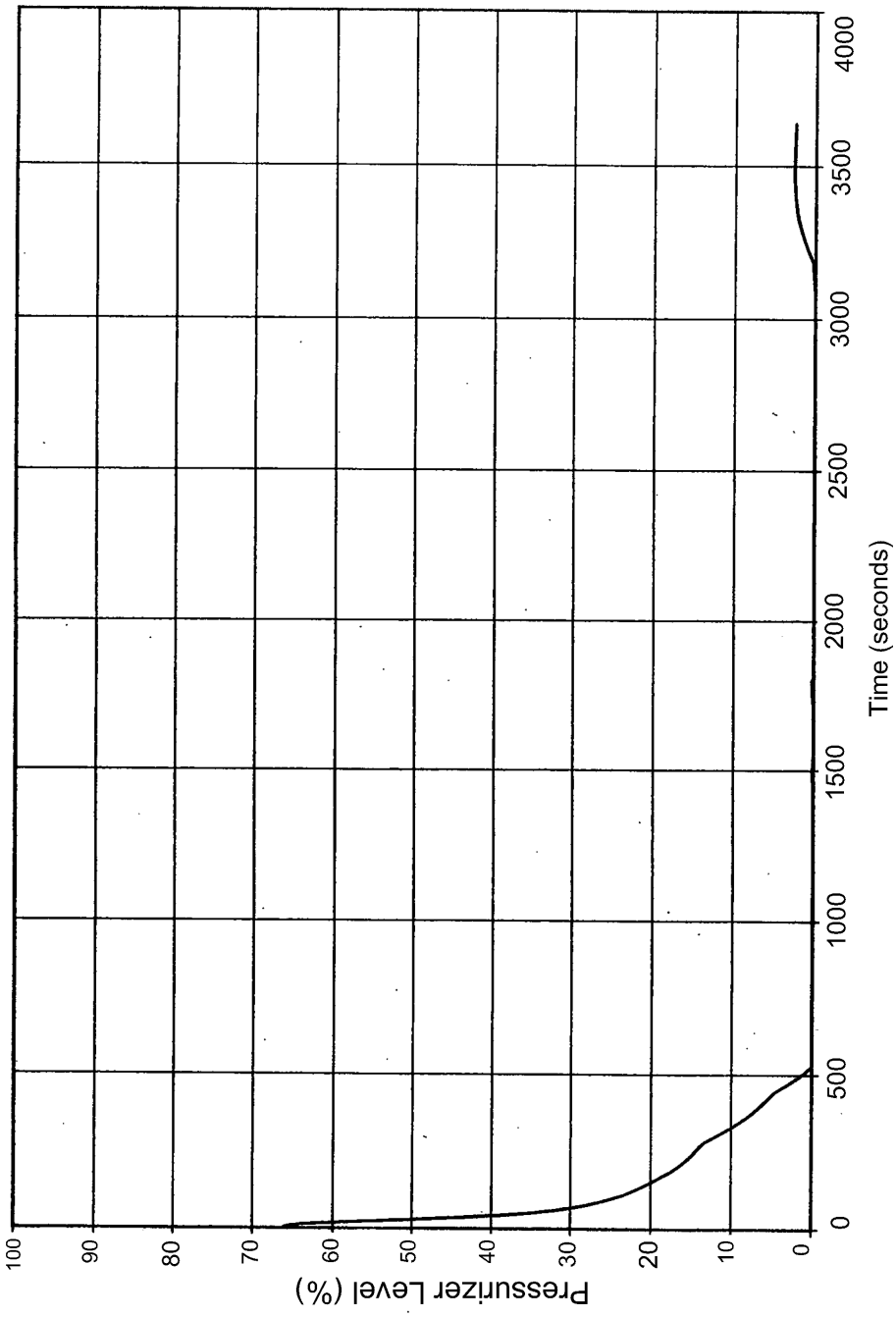


FIGURE 14.6.3–2 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER PRESSURIZER LEVEL VERSUS TIME

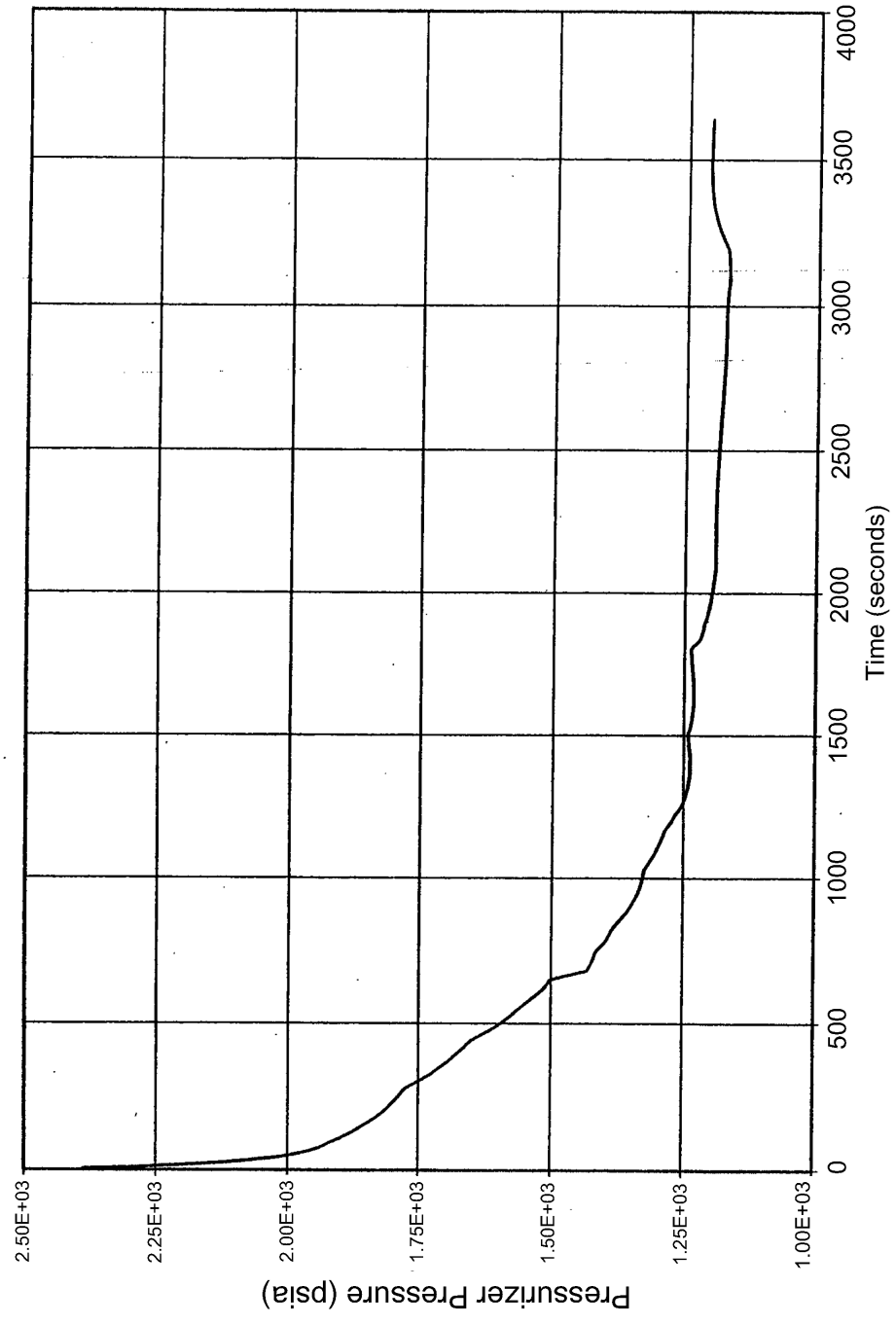


FIGURE 14.6.3–3 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER PRESSURIZER PRESSURE VERSUS TIME

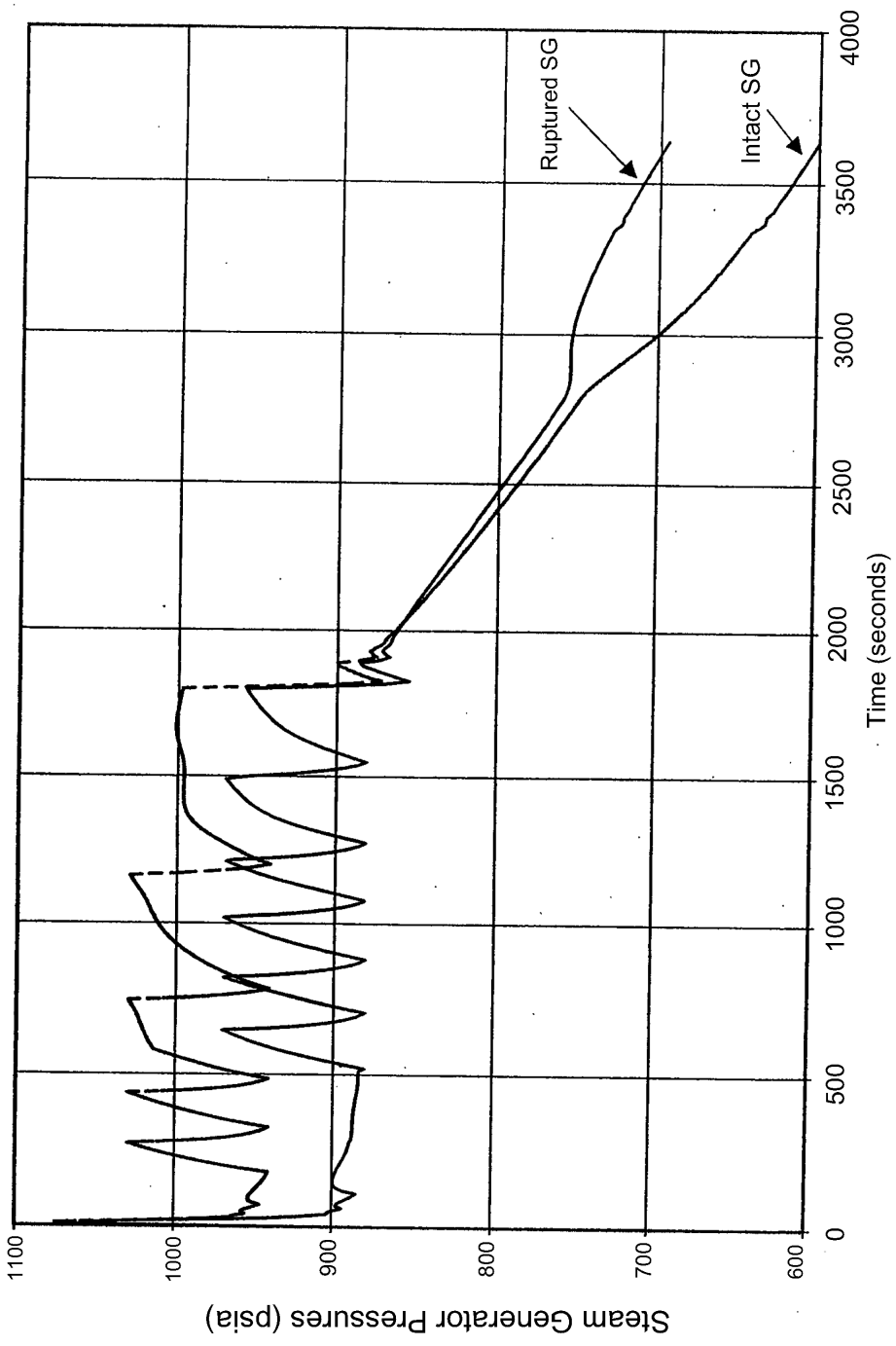


FIGURE 14.6.3–4 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER STEAM GENERATOR PRESSURE VERSUS TIME

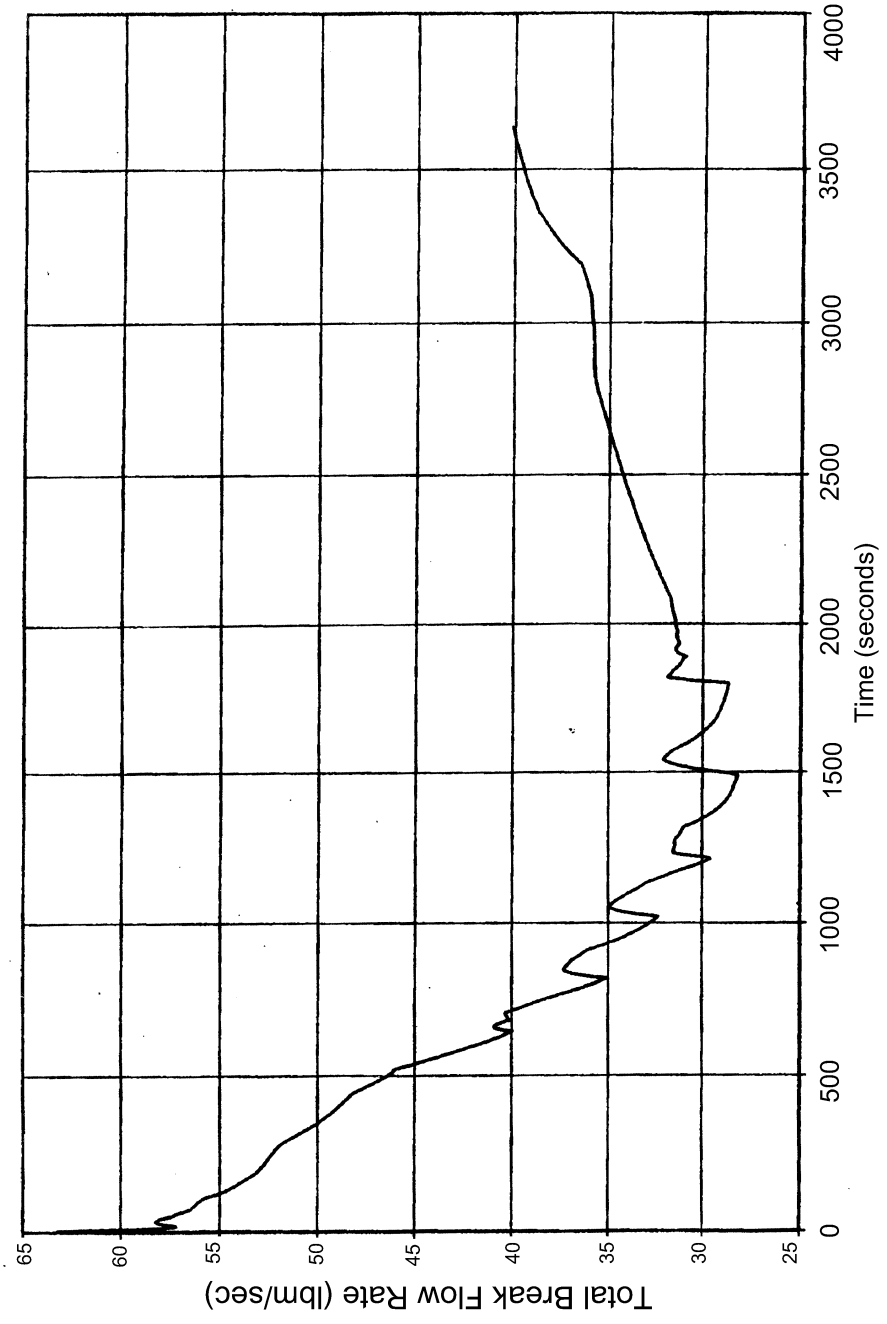


FIGURE 14.6.3–5 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER TOTAL BREAK FLOW RATE VERSUS TIME

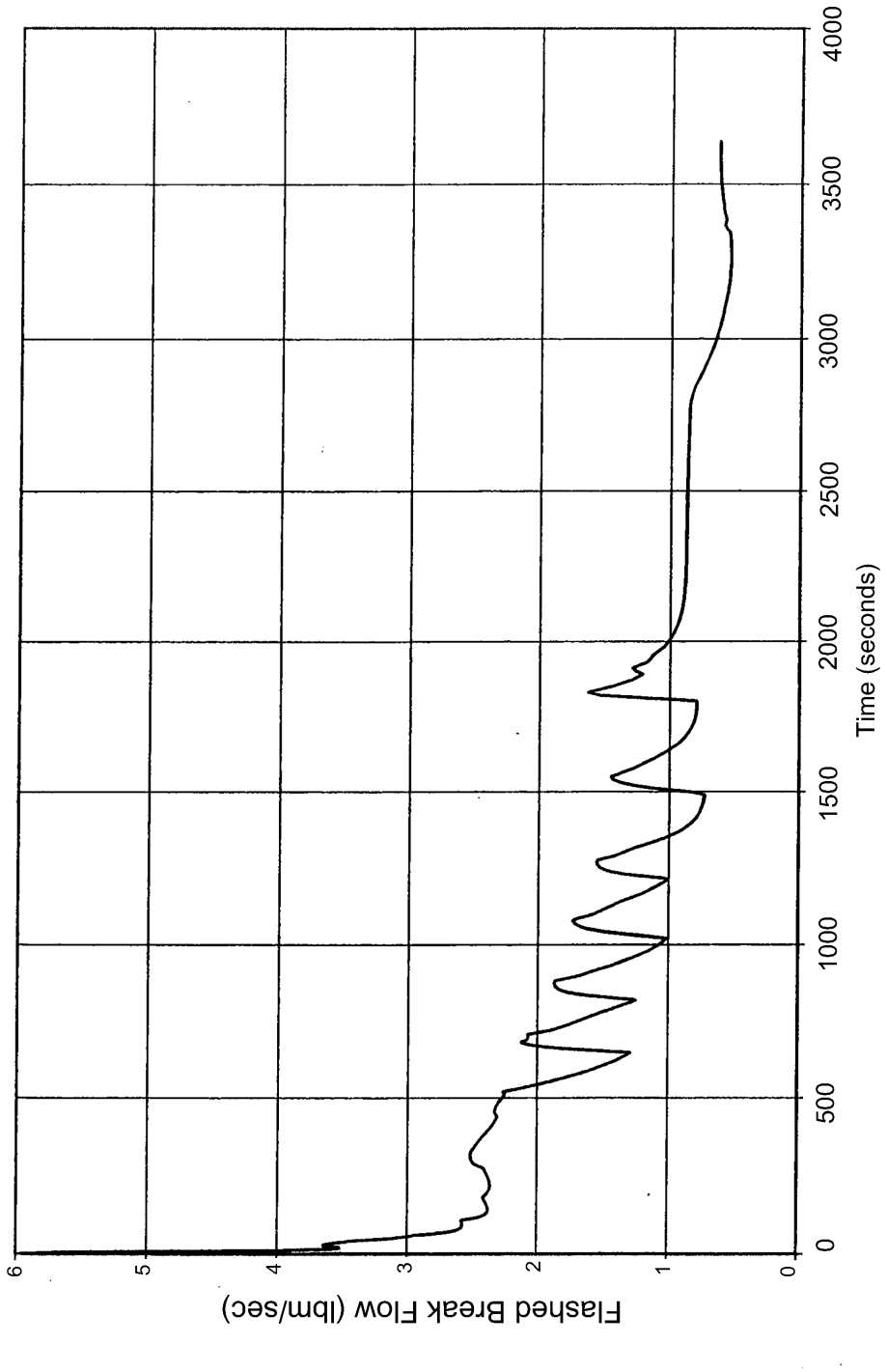


FIGURE 14.6.3–6 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER FLASHED BREAK FLOW RATE VERSUS TIME

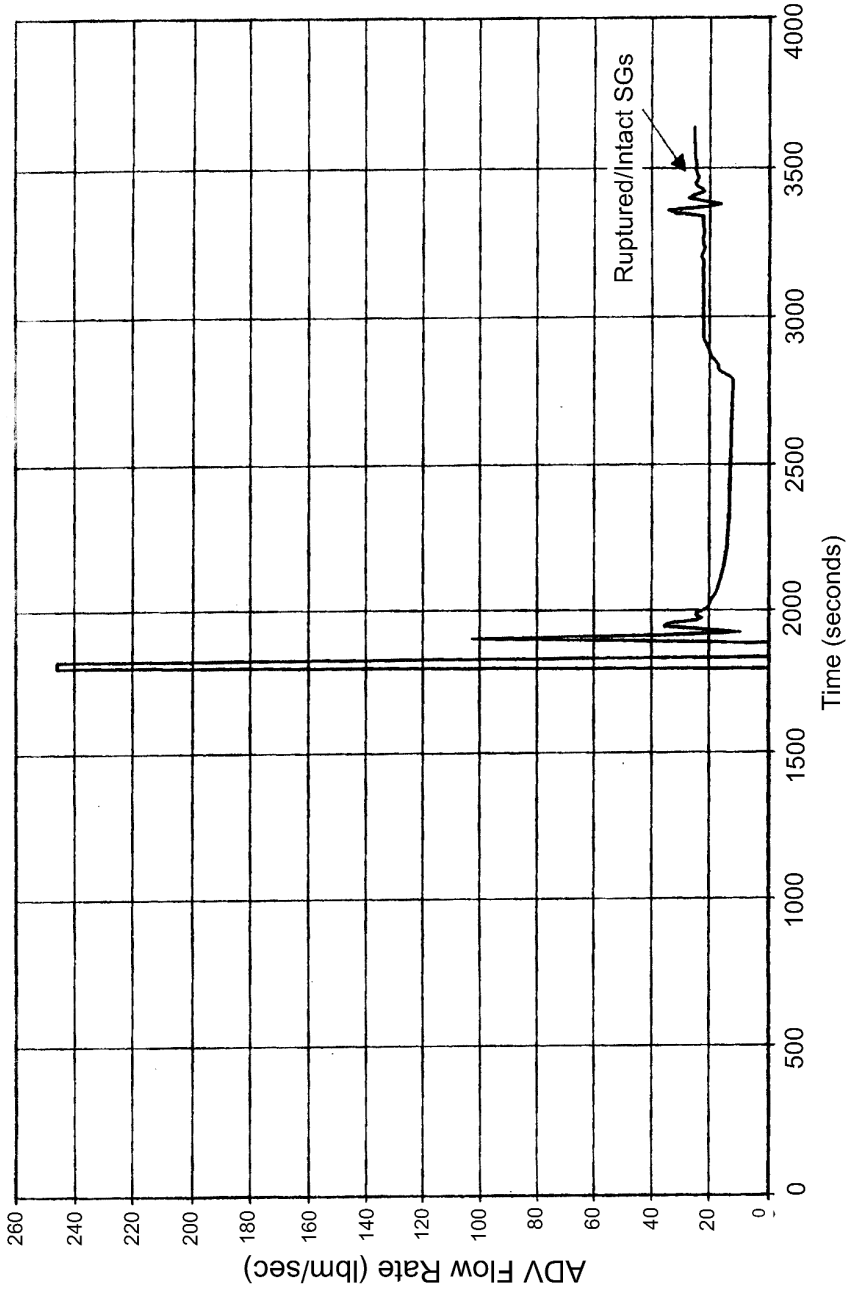


FIGURE 14.6.3–7 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER ATMOSPHERIC DUMP VALVE FLOW RATE PER STEAM GENERATOR VERSUS TIME

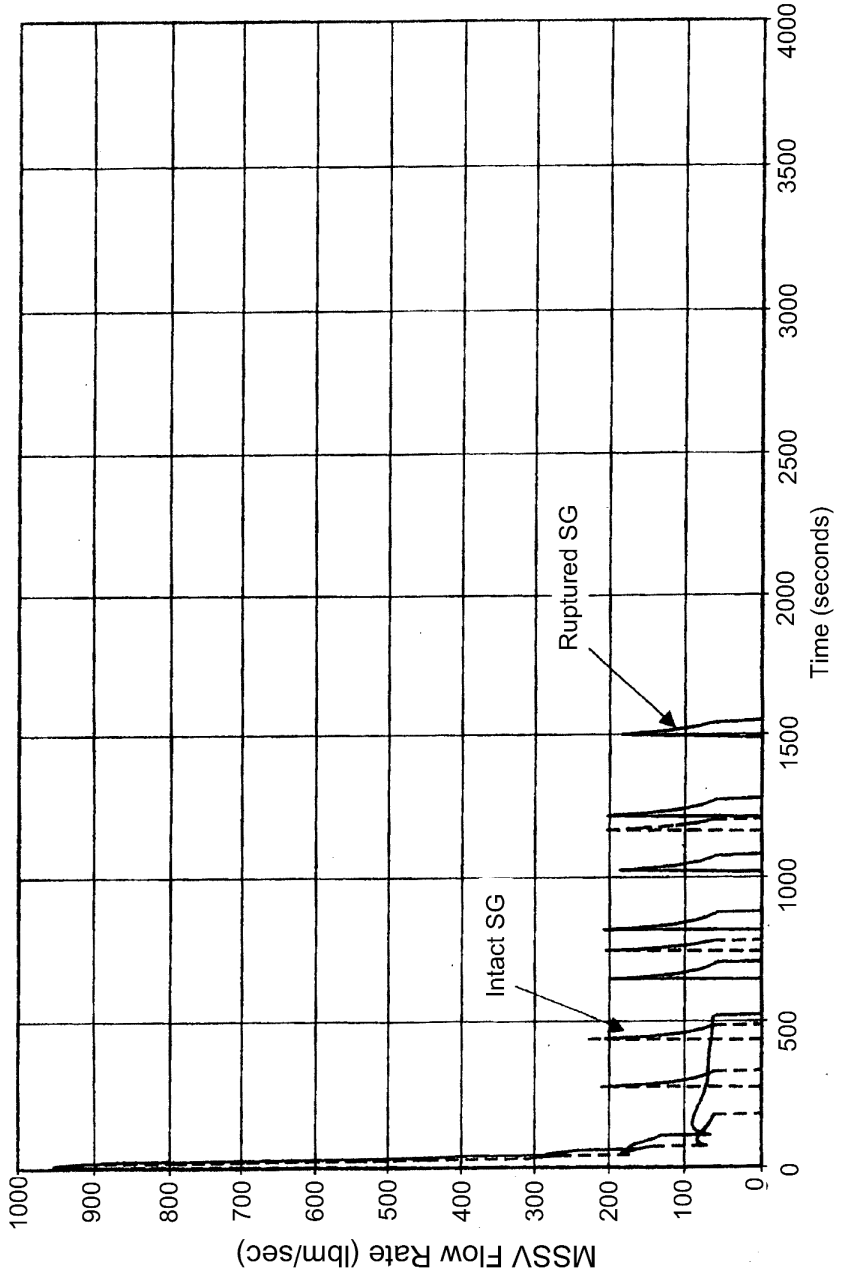


FIGURE 14.6.3–8 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER MAIN STEAM SAFETY VALVE FLOW RATES PER STEAM GENERATOR VERSUS TIME

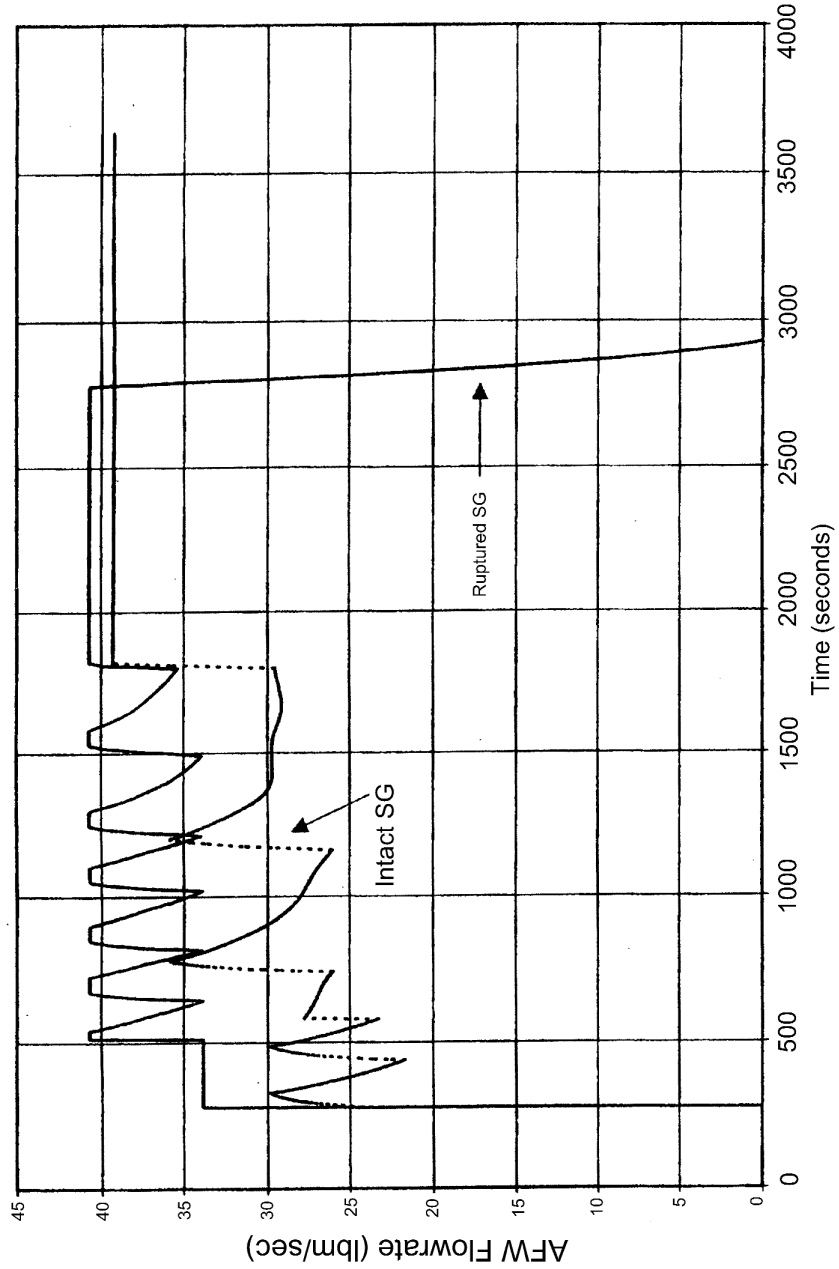


FIGURE 14.6.3–9 STEAM GENERATOR TUBE RUPTURE WITH THE LOSS OF OFFSITE POWER AUXILIARY FEEDWATER FLOW VERSUS TIME

TABLE 14.6.5.1-1 AVAILABLE REACTOR PROTECTION FOR THE LARGE BREAK LOSS OF COOLANT ACCIDENT

Reactor Operating Conditions	Reactor Protection
1, 2	No credit taken for reactor trip by the Reactor Protection System (RPS)
	ECCS - short and long-term cooling
3-6	No significant consequences for these reactor operating conditions

TABLE 14.6.5.1-2 DISPOSITION OF EVENTS FOR THE LARGE BREAK LOSS OF COOLANT ACCIDENT

Reactor Operating Conditions	Disposition
1	Analyze
2-6	Bounded by the event initiated from Mode 1

TABLE 14.6.5.1-3 MILLSTONE UNIT 2 REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - PLANT PARAMETER VALUES AND RANGES

	Plant Parameter	Parameter Value
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.440 in.
	b) Cladding inside diameter	0.387 in.
	c) Cladding thickness	0.0265 in.
	d) Pellet outside diameter	0.3805 in.
	e) Initial Pellet density	96 percent of theoretical
	f) Active fuel length	136.7 in.
	g) Gd_2O_3 concentrations	2, 4, 6, 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressurizer location	-
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	14 x 14
	e) SG tube plugging	5.87 percent
2.0	**2.0 Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Analyzed reactor power	2754 MWt
	b) F_Q	2.385[1,2]
	c) F_r	1.854[2]
	2.2 Fluid Conditions	
	a) Loop flow	132.2 Mlbm/hr $\leq$ M $\leq$ 160.0 Mlbm/hr
	b) RCS cold leg temperature	536°F $\leq$ T $\leq$ 554°F

1. The value used for F_Q is derived from the LHGR Technical Specification value
2. Includes measurement uncertainty.
3. Upper head temperature will change based on sampling of RCS temperature.

TABLE 14.6.5.1-3 MILLSTONE UNIT 2 REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - PLANT PARAMETER VALUES AND RANGES

	Plant Parameter	Parameter Value
	c) Upper head temperature	~Thot Temperature[3]
	d) Pressurizer pressure	2190 psia $\leq$ P $\leq$ 2310 psia
	e) Pressurizer level	35 percent $\leq$ L $\leq$ 75 percent
	f) SIT pressure	195 psia $\leq$ P $\leq$ 280 psia
	g) SIT liquid volume	1015 $ft^3 \leq$ V $\leq$ 1255 ft^3
	h) SIT temperature	60°F $\leq$ T $\leq$ 125°F (coupled with containment temperature)
	i) SIT resistance fL/D	As-built piping configuration
	j) SIT boron	1720 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	0.05 $\leq$ A $\leq$ 1.0 full pipe area (split) 0.05 $\leq$ A $\leq$ 1.0 full pipe area (guillotine)
	d) ECCS pumped injection temperature	140°F
	e) HPSI pump delay	10 s (No-LOOP) 25 s (LOOP)
	f) LPSI pump delay	30 s (No-LOOP) 45 s (LOOP)
	g) Initial containment pressure	14.27 psia
	h) Initial containment temperature	60°F $\leq$ T $\leq$ 125°F
	i) Containment sprays delay	0 s
	j) Containment spray water temperature	35°F

1. The value used for F_Q is derived from the LHGR Technical Specification value
2. Includes measurement uncertainty.
3. Upper head temperature will change based on sampling of RCS temperature.

TABLE 14.6.5.1-3 MILLSTONE UNIT 2 REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - PLANT PARAMETER VALUES AND RANGES

Plant Parameter	Parameter Value

k) LPSI Flow

RCS Cold Leg Pressure (psia)	Broken Loop Flow 1A (gpm)	Intact Loop Flow 1B (gpm)	Intact Loop Flow 2A (gpm)	Intact Loop Flow 2B (gpm)
14.7	1369	1314	0	0
50	1214	1164	0	0
100	945	904	0	0
150	546	519	0	0
200	0	0	0	0
300	0	0	0	0
500	0	0	0	0
700	0	0	0	0
900	0	0	0	0
1000	0	0	0	0
1050	0	0	0	0
1100	0	0	0	0
1144.34	0	0	0	0

l) HPSI Flow

RCS Cold Leg Pressure (psia)	Broken Loop Flow 1A (gpm)	Intact Loop Flow 1B (gpm)	Intact Loop Flow 2A (gpm)	Intact Loop Flow 2B (gpm)
14.7	138	140	139	140
50	136	138	137	138
100	133	135	134	135
150	131	131	131	131
200	128	128	128	128
300	121	121	121	122
500	106	106	106	106
700	89	89	89	89
900	68	68	68	68
1000	54	54	54	54
1050	43	43	43	43
1100	30	30	30	30
1144.34	0	00	0	0

1. The value used for F_Q is derived from the LHGR Technical Specification value
2. Includes measurement uncertainty.
3. Upper head temperature will change based on sampling of RCS temperature.

TABLE 14.6.5.1-4 MILLSTONE UNIT 2 LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - STATISTICAL DISTRIBUTION USED FOR PROCESS PARAMETERS

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution	Standard Deviation
Pressurizer Pressure (psia)	Uniform	2190 - 2310	Normal	0
Pressurizer Level (%)	Uniform	35 - 75	Normal	0
SIT Volume (ft^3)	Uniform	1015 - 1255	N/A	N/A
SIT Pressure (psia)	Uniform	195 - 280	N/A	N/A
Containment/SIT Temperature (°F)	Uniform	60 - 125	N/A	N/A
Containment Volume ($x10^6$ ft^3)	Uniform	1.899 - 2.125	N/A	N/A
Initial Flow Rate (Mlbm/hr)	Uniform	132.2 – 160.0	N/A	N/A
Initial Operating Temperature (°F)	Uniform	536 - 554	N/A	N/A

TABLE 14.6.5.1-5 MILLSTONE UNIT 2 LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - PASSIVE HEAT SINKS AND MATERIAL PROPERTIES IN CONTAINMENT GEOMETRY

Heat Sink	Surface Area, ft^2	Thickness, ft	Material
Containment Shell and Dome	71870	0.02083	Carbon Steel
		3.02083	Concrete
Unlined Concrete	62800	2.0	Concrete
Galvanized Steel	120100	0.0003	Galvanized Steel
		0.01697	Carbon Steel
Painted Thin Steel	61850	0.01667	Carbon Steel
Painted Steel	32600	0.021667	Carbon Steel
Painted Steel	25425	0.071667	Carbon Steel
Painted Thick Steel	4630	0.245	Carbon Steel
Containment Penetration	3000	0.0625	Carbon Steel
		3.8125	Concrete
Stainless Lined Concrete	8340	0.020833	Stainless Steel
		2.020833	Concrete
Base Slab	11130	8.0	Concrete
Neutron Shield	16270	0.0154	Stainless Steel
CEDM Cable Support	1380	0.1094	Stainless Steel
Painted Steel	2891	0.031327	Carbon Steel
Painted Steel	1856	0.02083	Carbon Steel
Painted Steel	624.4	0.0374	Carbon Steel
Non Galvanized Carbon Steel	23564	0.02167	Carbon Steel
Galvanized Carbon Steel	5423.4	0.003	Galvanized Steel
		0.0112	Carbon Steel
		0.0115	Galvanized Steel
Stainless Steel	6948	0.02167	Stainless Steel
Aluminum	2300	0.01333	Aluminum
Lead	7100	0.0325	Lead

Heat Sink Material	Thermal Conductivity Btu/hr-ft-°F	Volumetric Heat Capacity Btu/ft^3-°F
Concrete	0.92	22.62
Carbon Steel	27.00	58.80
Stainless Steel	8.47	58.60
Galvanized Steel	65.0	41.00
Aluminum	118	35.2
Lead	19.6	21.2

TABLE 14.6.5.1-6 MILLSTONE UNIT 2 LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - COMPLIANCE WITH 10 CFR 50.46(B)

UTL for 95/95 Simultaneous Coverage/Confidence		
Parameter	**Value**	**Case Number**
PCT, °F	1615	123
Maximum local oxidation, %	2.01	174
Core-wide oxidation, %	0.025	96
Characteristics of Case Setting the PCT UTL		
PCT, °F		1615
PCT Rod Type		Fresh 4% Gad Rod
Time of PCT, s		7.44
Elevation within Core, ft		9.36
Maximum Local Oxidation, %		1.98
Total Core-Wide Oxidation, %		0.006
PCT Rod Rupture Time, s		No rod rupture
Rod Rupture Elevation within Core, ft		No rod rupture

TABLE 14.6.5.1-7 MILLSTONE UNIT 2 LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - SUMMARY OF MAJOR PARAMETERS FOR THE DEMONSTRATION CASE

Parameter	Value
Core Power (MWt)	2754
Time in Cycle (hrs)	11619
Limiting Rod Assembly Burnup (GWd/mtU)	18.4
Limiting Rod LHGR (kW/ft)	13.25
Limiting Rod Equivalent F_Q	2.09
Limiting Rod Radial Peak, F_r	1.71
Limiting Rod Axial Shape Index	-0.076
Break Type	Split
Break Size (ft^2/side)	3.6942
Offsite Power Availability	LOOP

TABLE 14.6.5.1-8 MILLSTONE UNIT 2 LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS - CALCULATED EVENT TIMES FOR THE DEMONSTRATION CASE*

Event	Time (sec)
Break Opens	0.0
RCP Trip	0.0
SIAS Issued	0.7
PCT Occurred	7.4
Start of Broken Loop SIT Injection	16.6
Start of Intact Loop SIT Injection (Loop 2,3 and 4 respectively)	17.3, 17.3 and 17.3
HPSI Available	25.7
Broken Loop HPSI Delivery Began	25.7
Intact Loops HPSI Delivery Began (Loop 2, 3 and 4 respectively)	25.7, 25.7 and 25.7
Beginning of Core Recovery (Beginning of Reflood)	26.8
LPSI Available	45.7
Broken Loop LPSI Delivery Began	45.7
Intact Loops LPSI Delivery Began	45.7, N/A and N/A
Intact Loop SIT Emptied (Loop 2, 3 and 4 respectively)	62.0, 62.1 and 61.5
Broken Loop SIT Emptied	63.3
Transient Calculation Terminated	900.0

* Loops 1, 2, 3 and 4 correspond to the plant Loops 1A, 1B, 2A and 2B, respectively.

FIGURE 14.6.5.1–1 SCATTER PLOT OPERATIONAL PARAMETERS

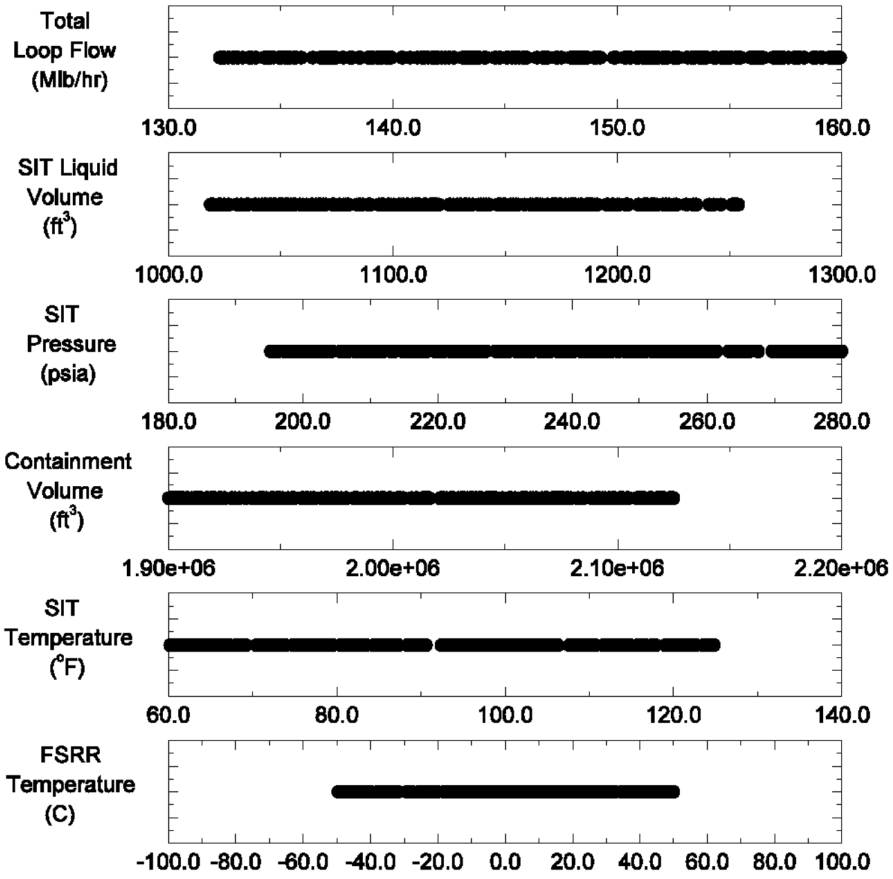


FIGURE 14.6.5.1–2 PCT VERSUS PCT TIME SCATTER PLOT

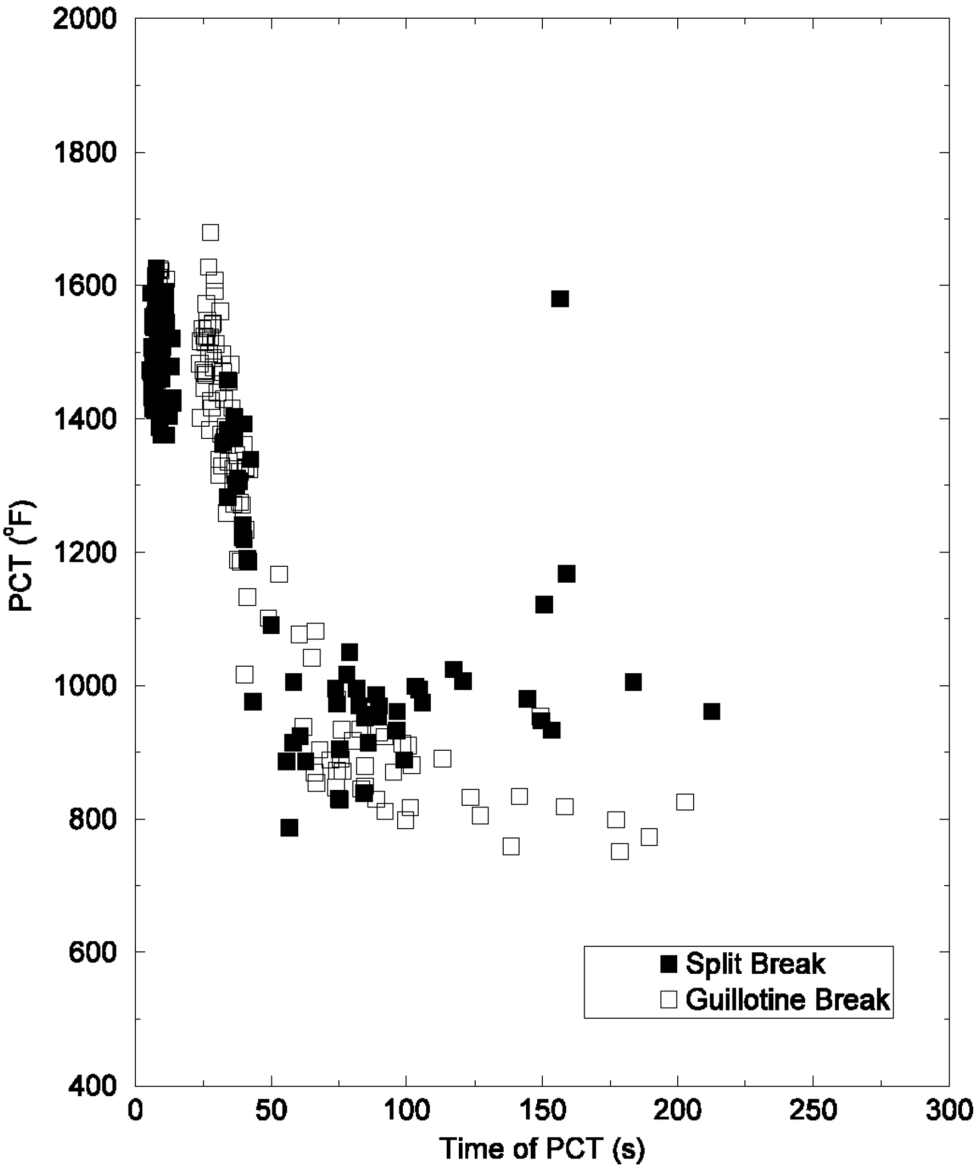


FIGURE 14.6.5.1–3 PCT VERSUS PCT TIME SCATTER PLOT

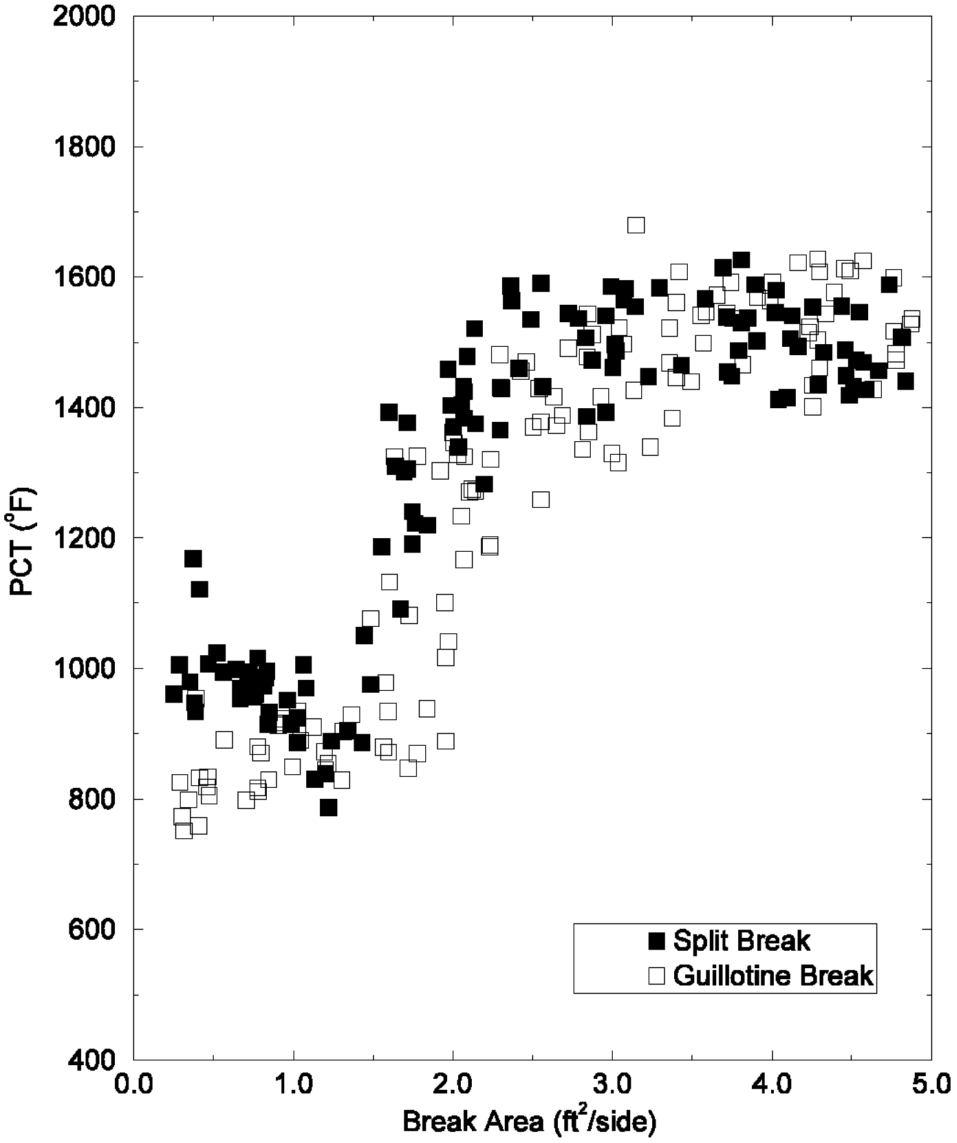


FIGURE 14.6.5.1–4 MAXIMUM LOCAL OXIDATION VERSUS PCT SCATTER PLOT

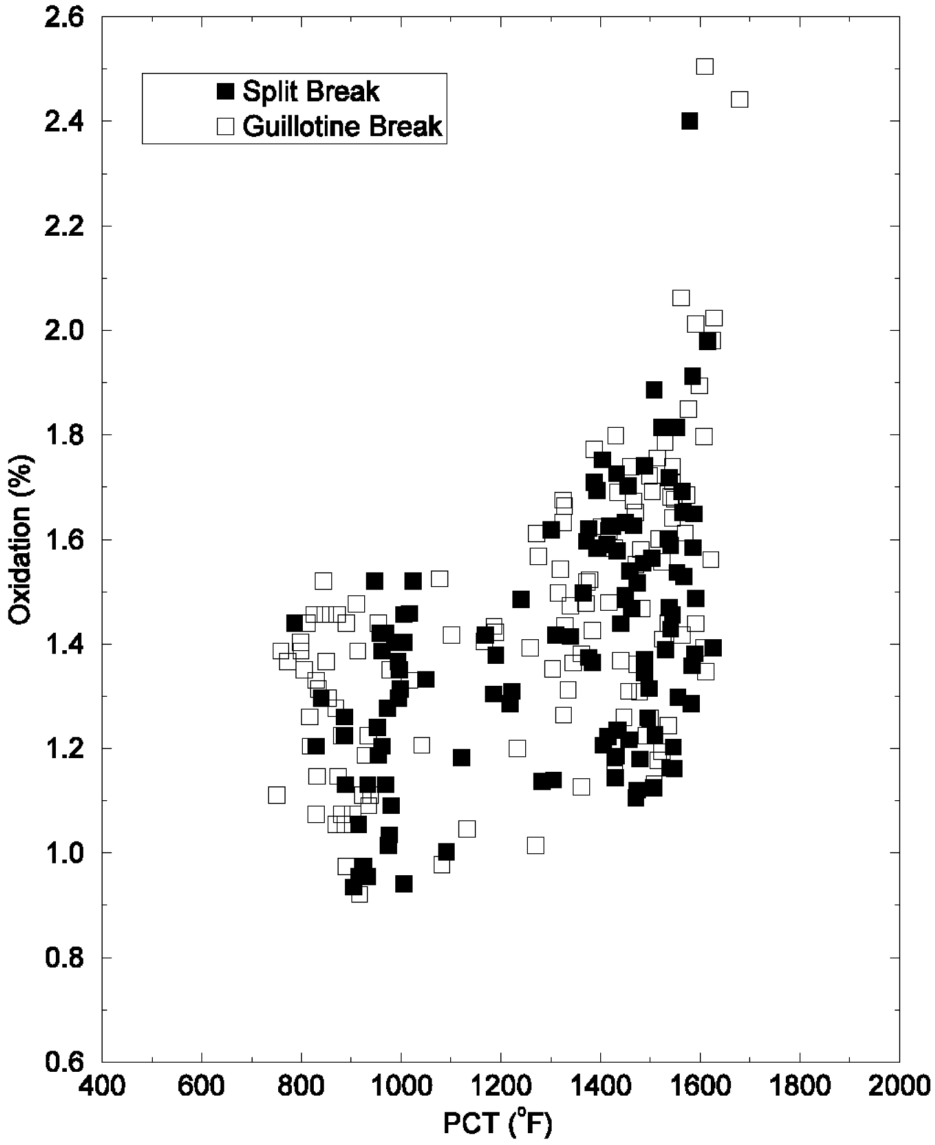


FIGURE 14.6.5.1–5 TOTAL CORE WIDE OXIDATION VERSUS PCT SCATTER PLOT

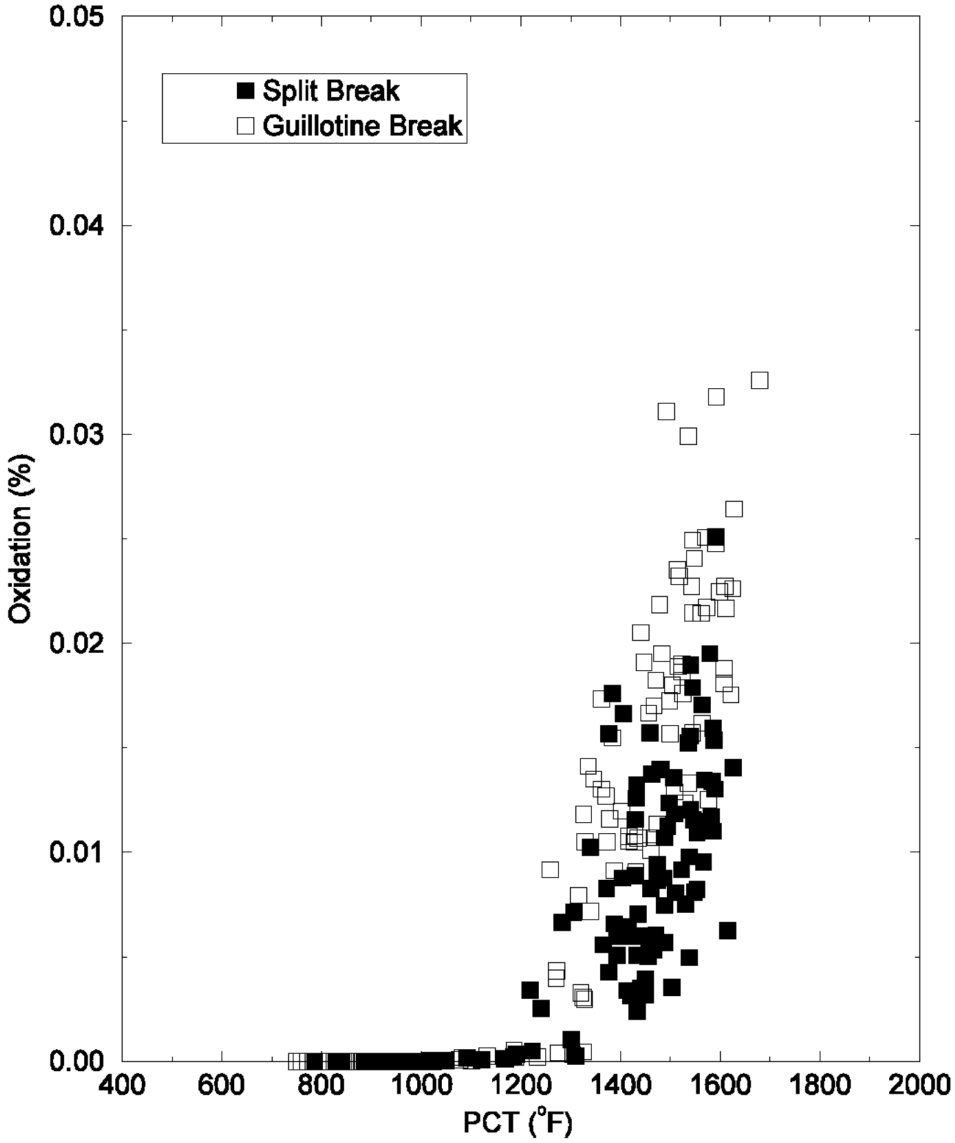


FIGURE 14.6.5.1–6 PEAK CLADDING TEMPERATURE (INDEPENDENT OF ELEVATION) FOR THE DEMONSTRATION CASE

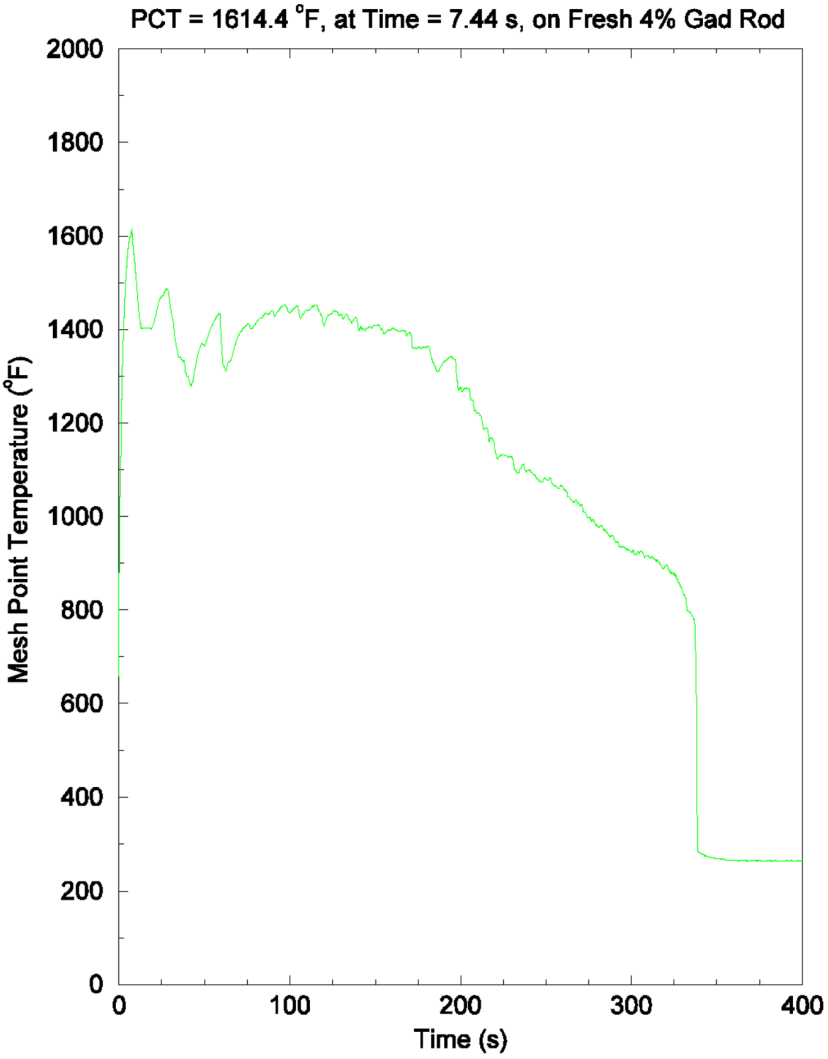


FIGURE 14.6.5.1–7 BREAK FLOW FOR THE DEMONSTRATION CASE

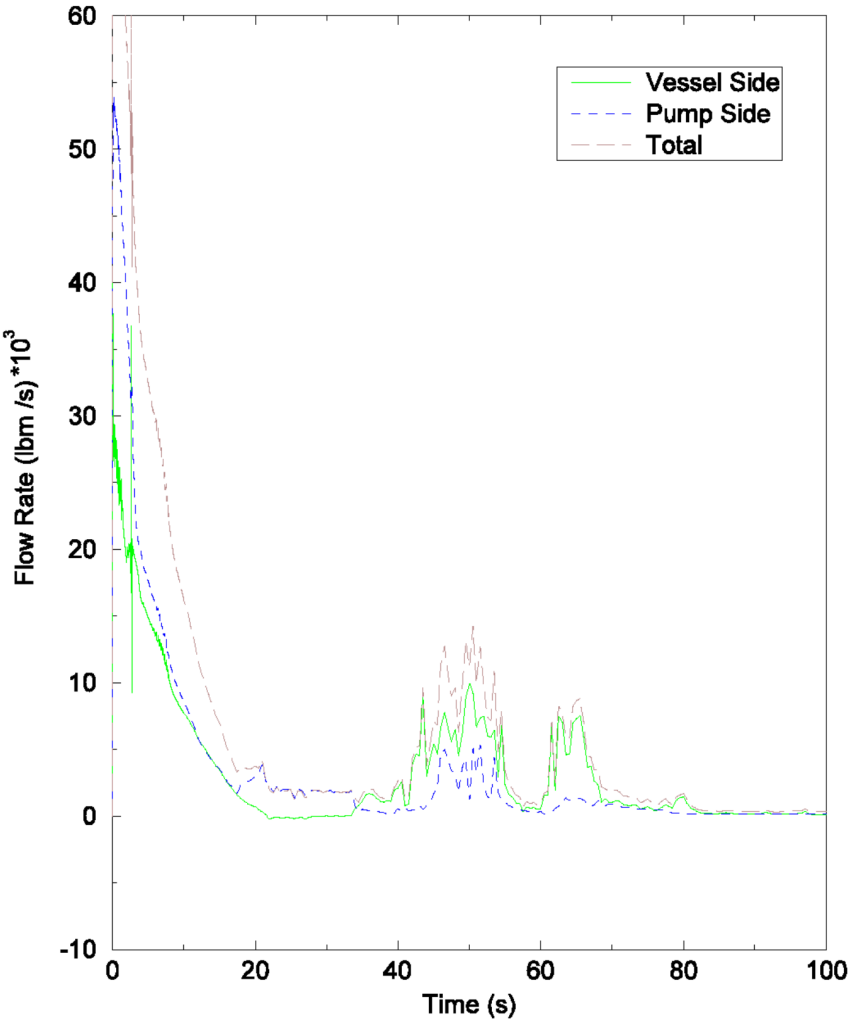


FIGURE 14.6.5.1–8 CORE INLET MASS FLUX FOR THE DEMONSTRATION CASE

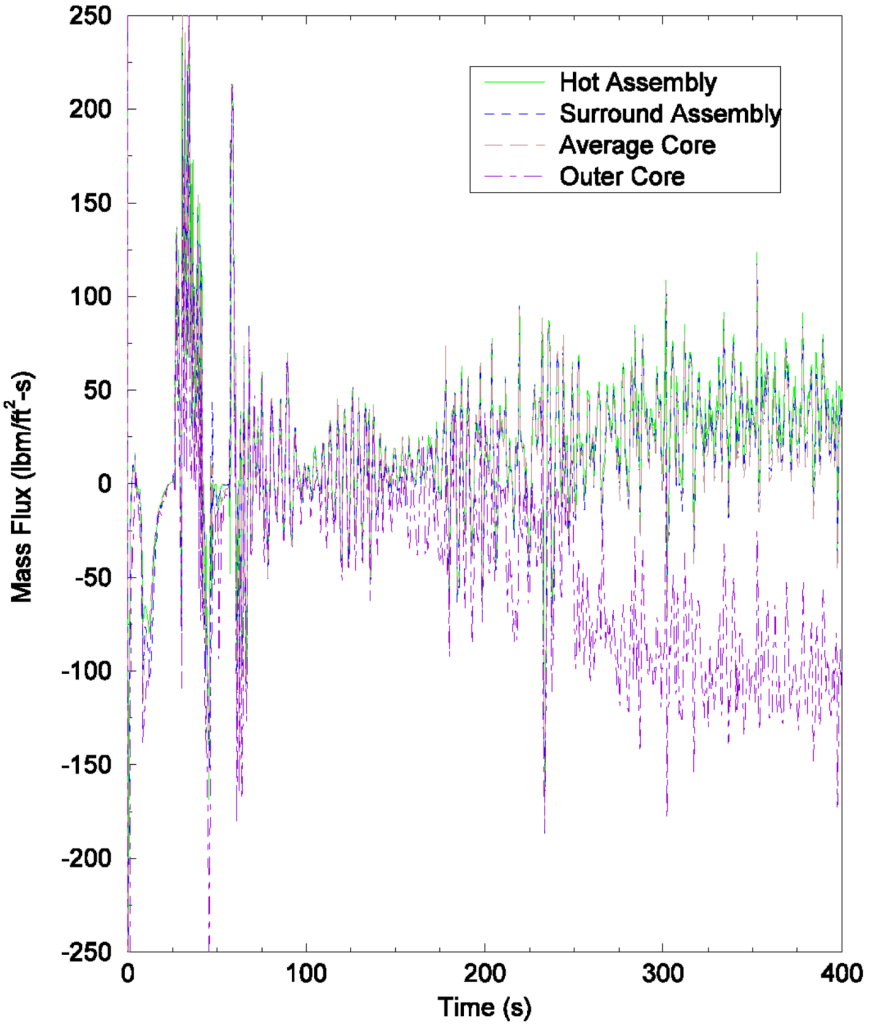


FIGURE 14.6.5.1–9 CORE OUTLET MASS FLUX FOR THE DEMONSTRATION CASE

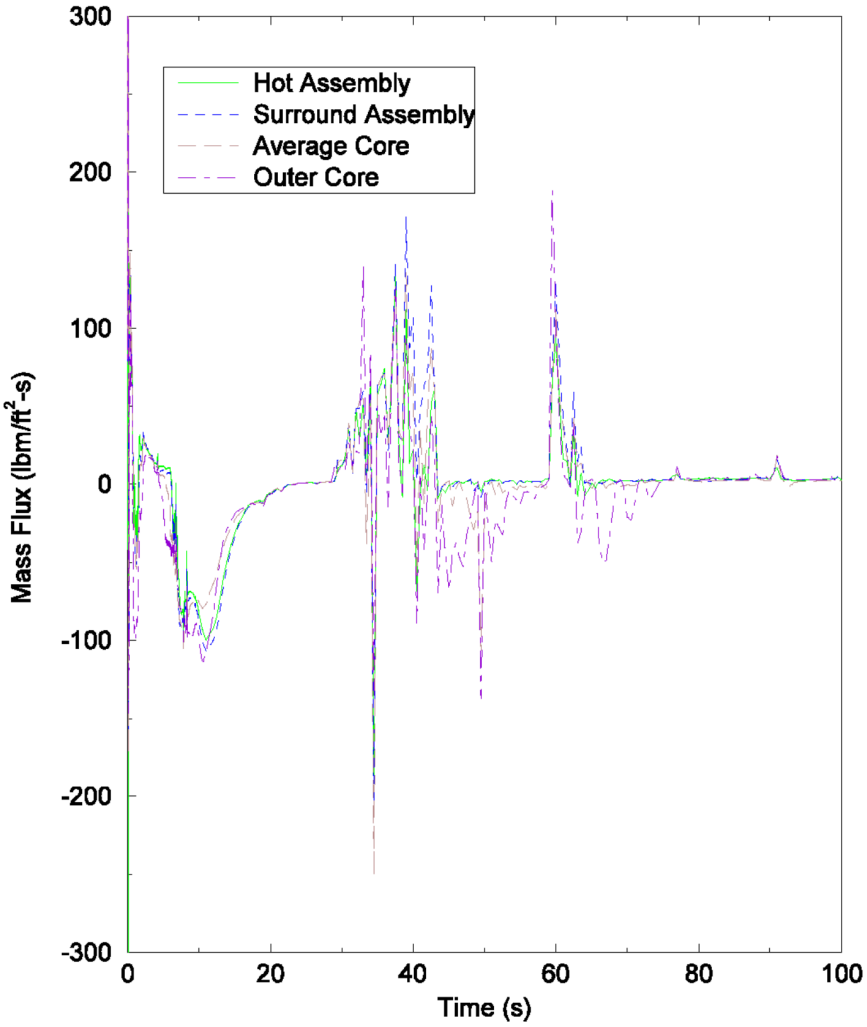


FIGURE 14.6.5.1–10 VOID FRACTION AT RCS PUMPS FOR THE DEMONSTRATION CASE

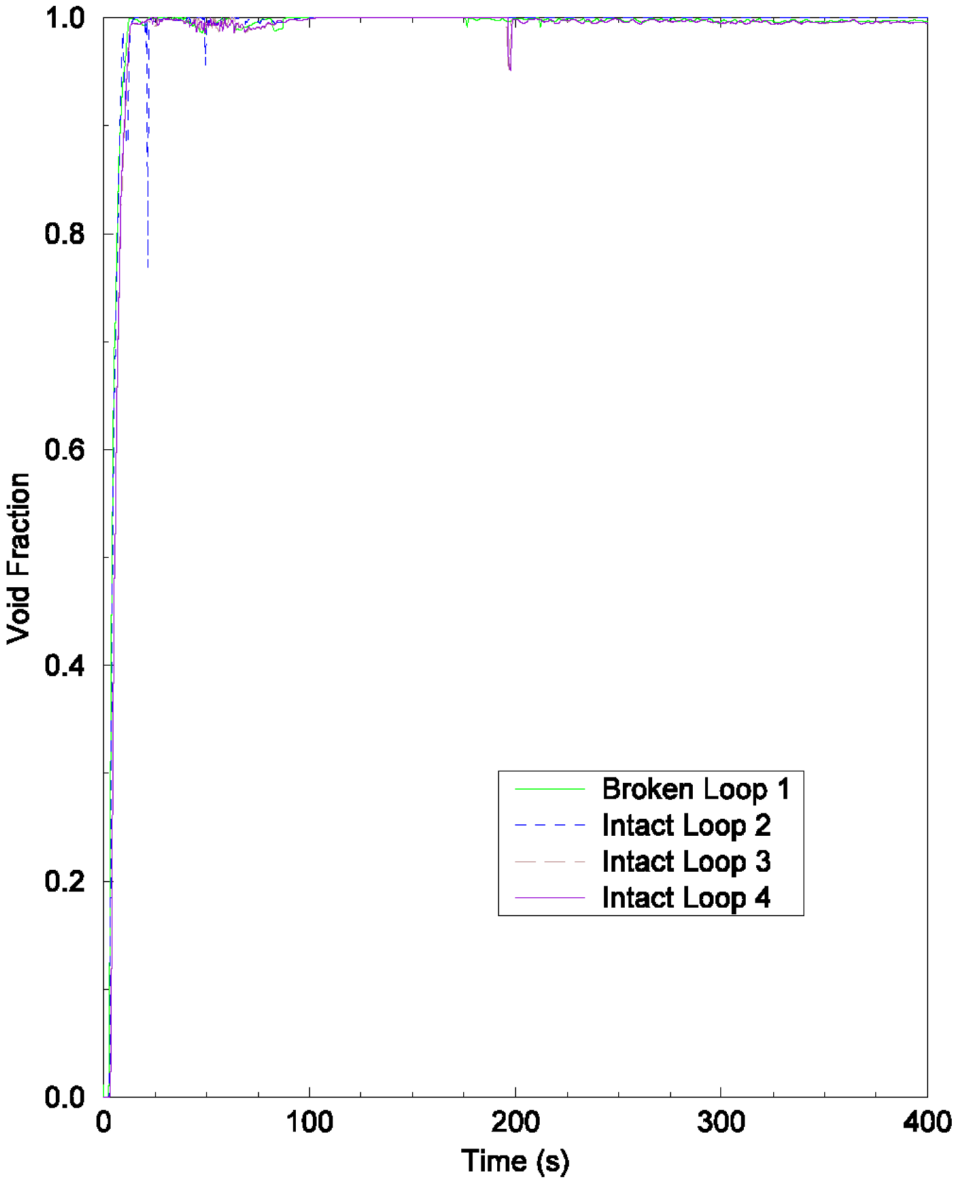


FIGURE 14.6.5.1–11 ECCS FLOWS (INCLUDES SIT, HPSI AND LPSI) FOR THE DEMONSTRATION CASE

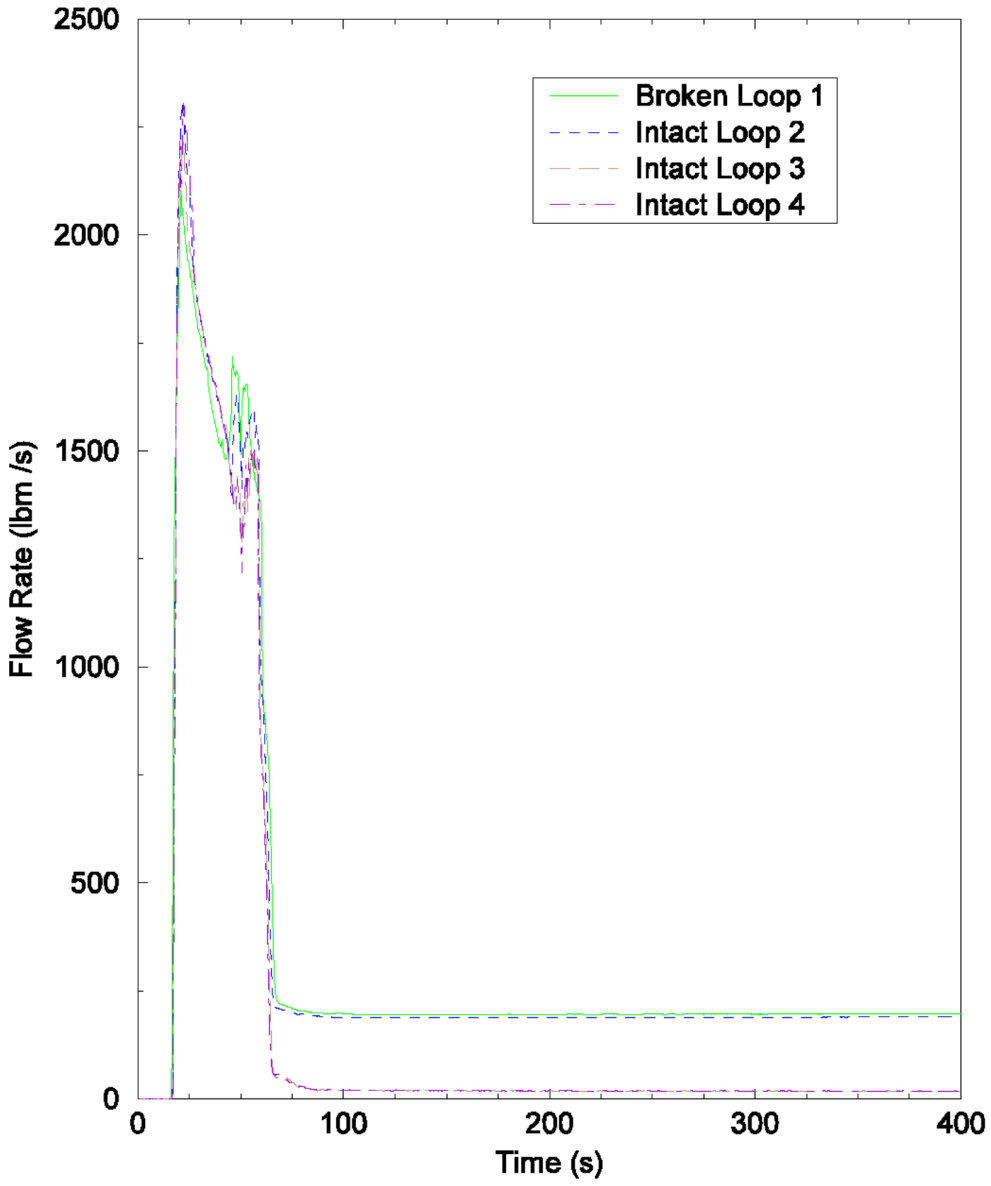


FIGURE 14.6.5.1–12 UPPER PLENUM PRESSURE FOR THE DEMONSTRATION CASE

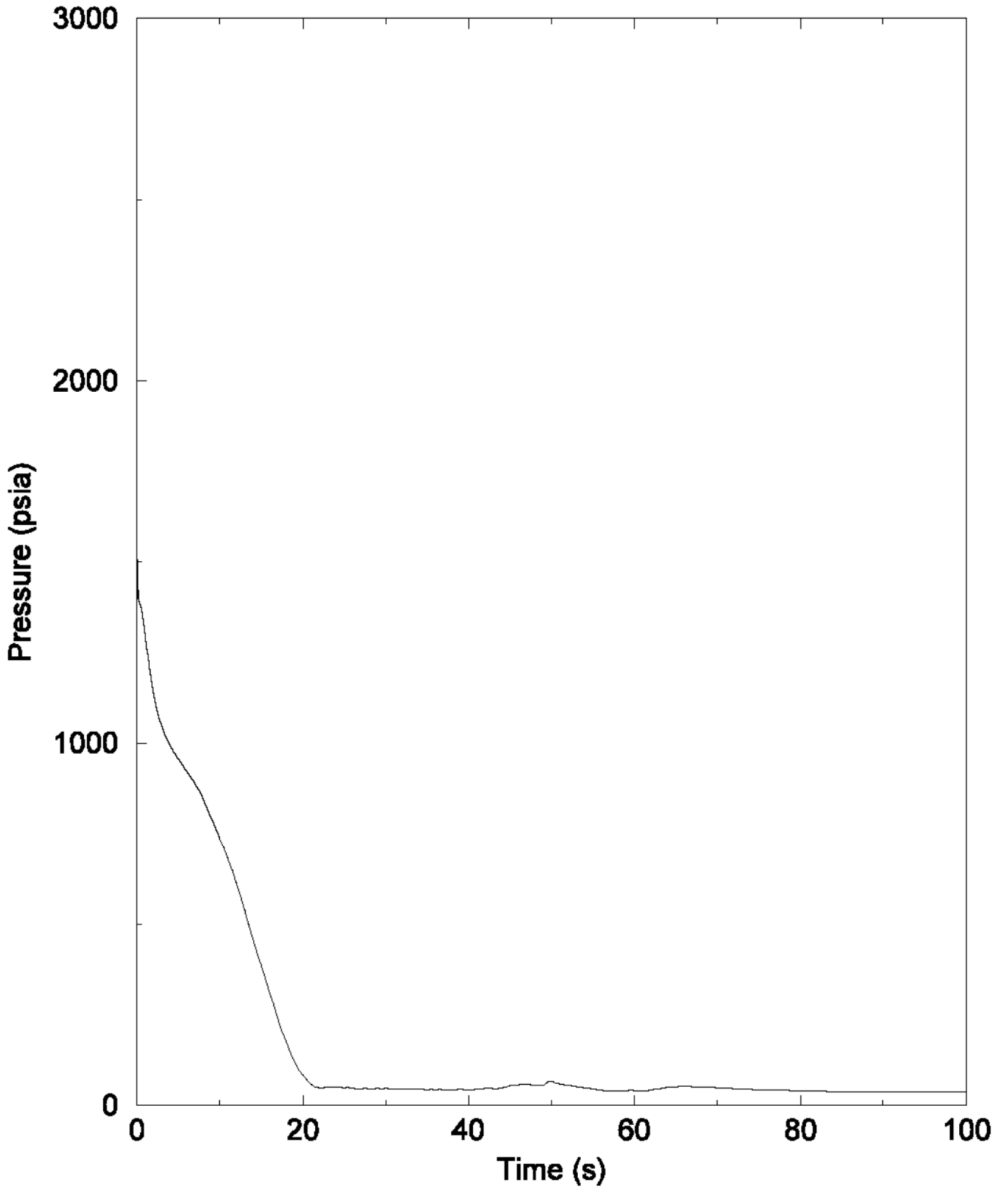


FIGURE 14.6.5.1–13 COLLAPSED LIQUID LEVEL IN THE DOWNCOMER FOR THE DEMONSTRATION CASE

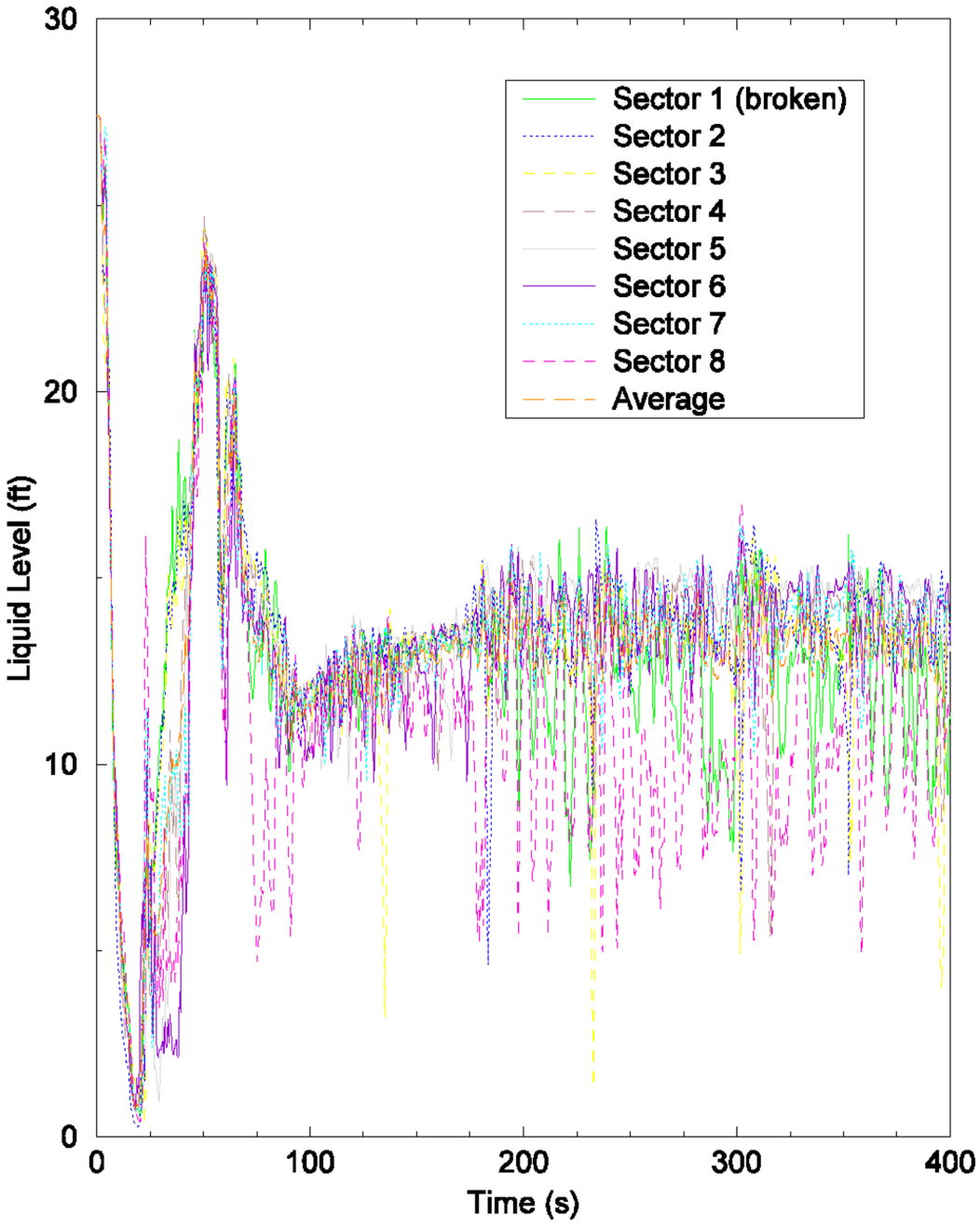


FIGURE 14.6.5.1–14 COLLAPSED LIQUID LEVEL IN THE LOWER PLENUM FOR THE DEMONSTRATION CASE

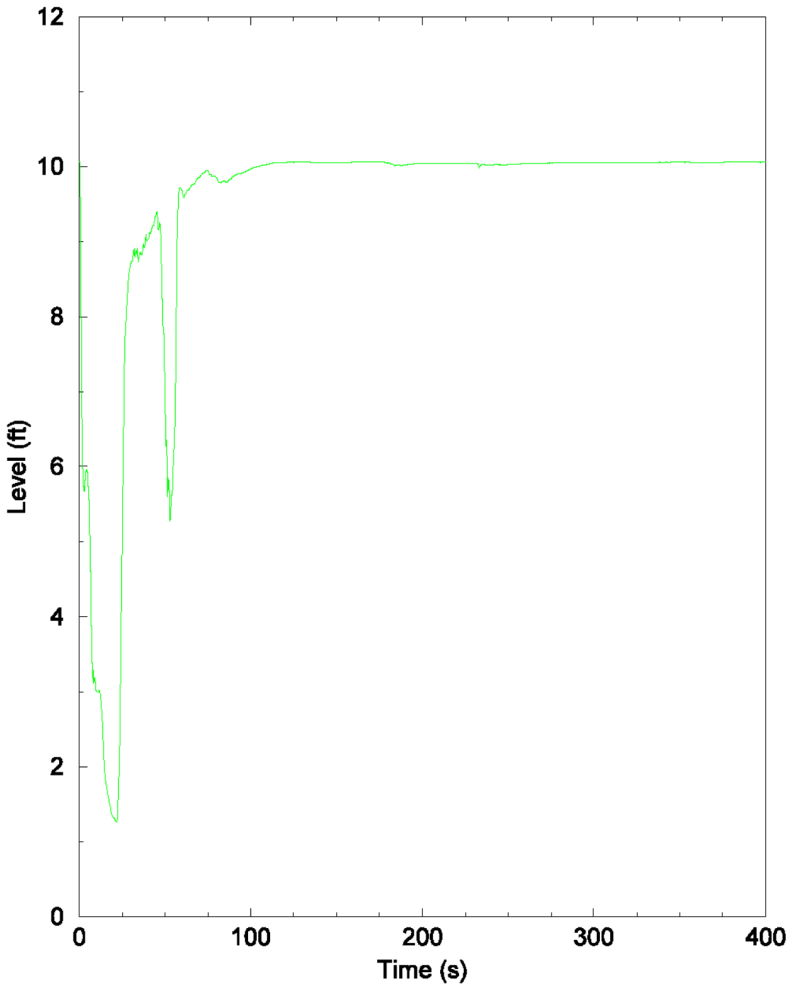


FIGURE 14.6.5.1–15 COLLAPSED LIQUID LEVEL IN THE CORE FOR THE DEMONSTRATION CASE

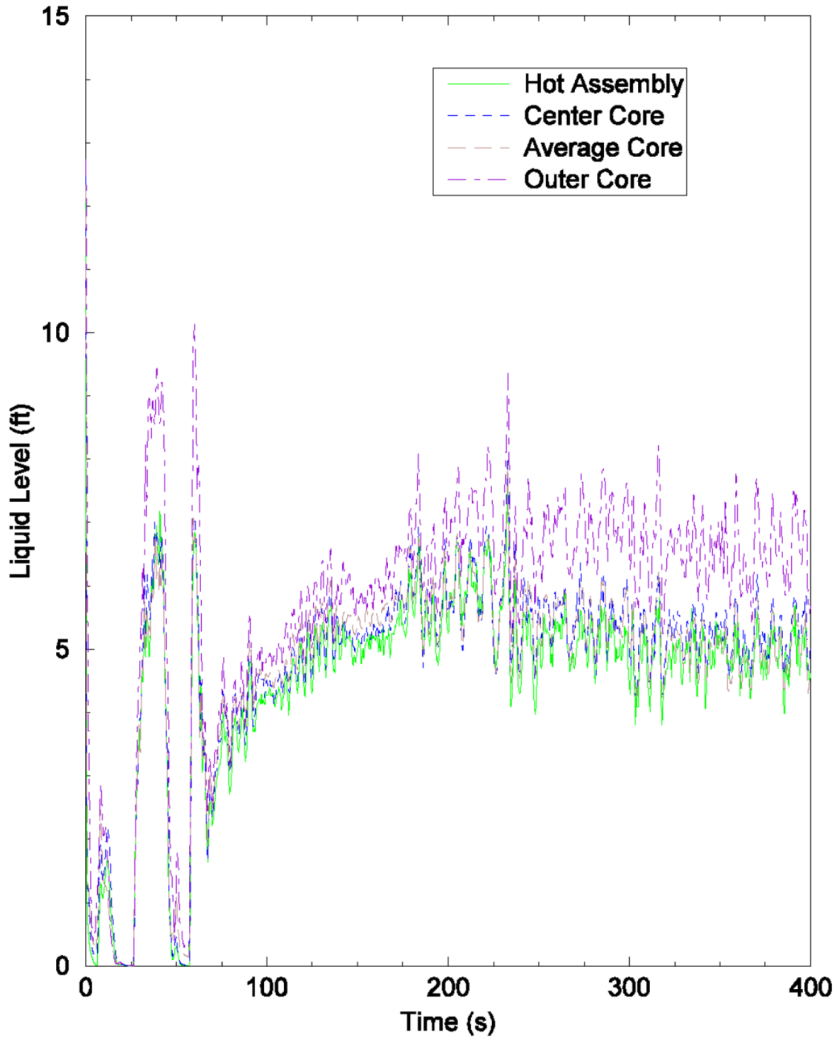


FIGURE 14.6.5.1–16 CONTAINMENT AND LOOP PRESSURES FOR THE DEMONSTRATION CASE

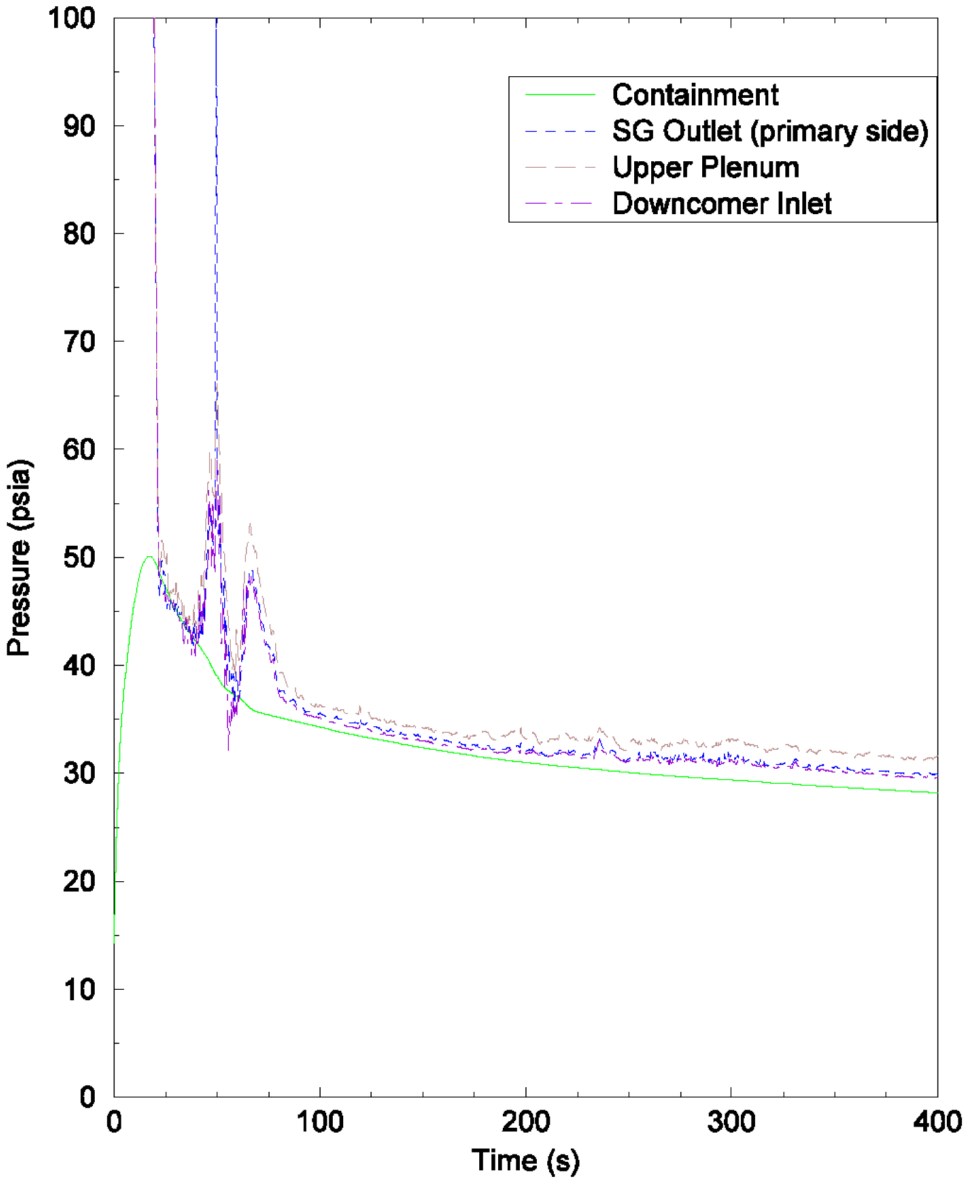


FIGURE 14.6.5.1–17 PRESSURE DIFFERENCES BETWEEN UPPER PLENUM AND DOWNCOMER FOR THE DEMONSTRATION CASE

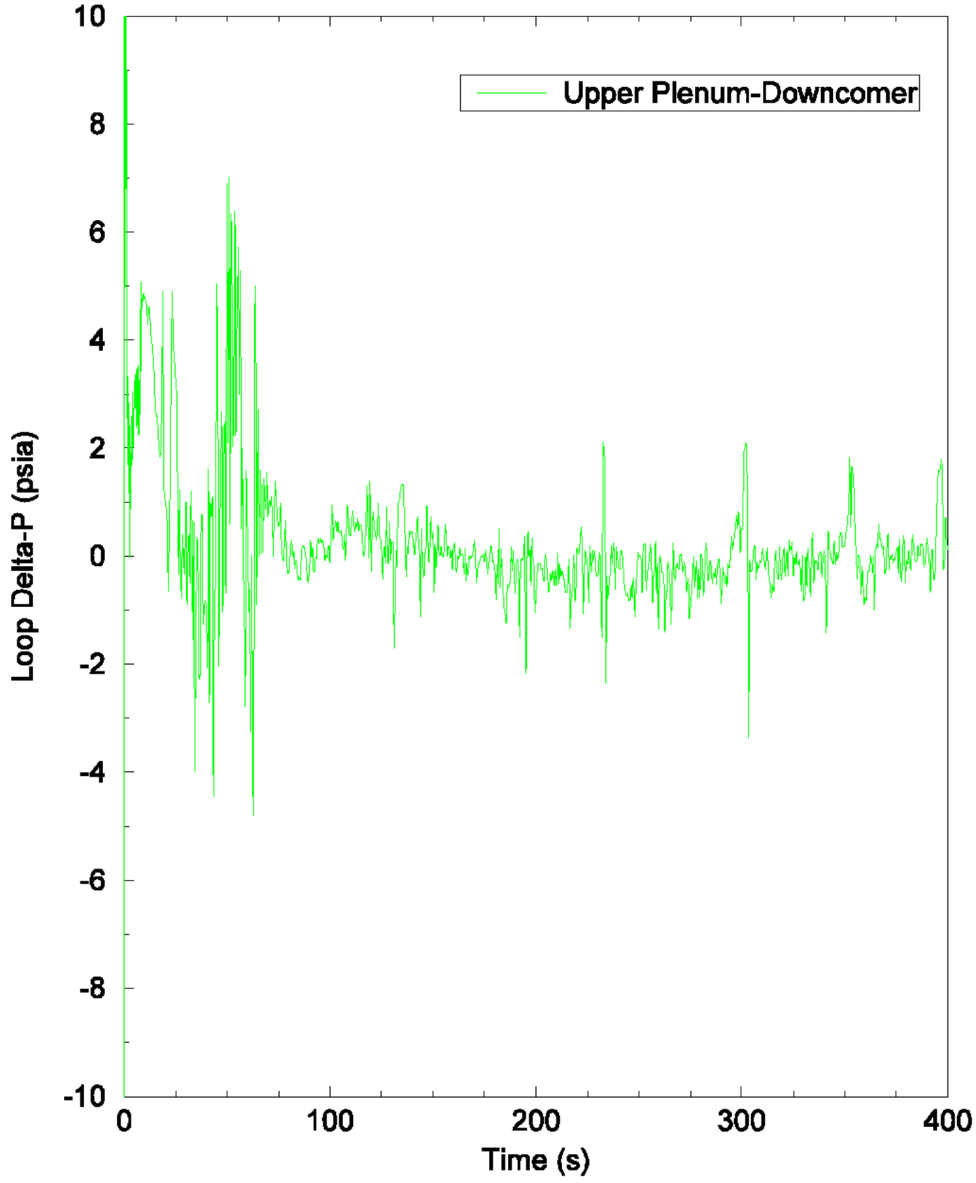


TABLE 14.6.5.2-1 AVAILABLE REACTOR PROTECTION FOR THE SMALL BREAK LOSS OF COOLANT ACCIDENT

Reactor Operational Mode	Reactor Protection
1	Thermal Margin/Low Pressure Trip
	Low Reactor Coolant Flow Trip
	Safety Injection Actuation Signal
2	Safety Injection Actuation Signal
3-6	No Significant Consequences for These Reactor Operating Conditions

TABLE 14.6.5.2-2 DISPOSITION OF EVENTS FOR THE SMALL BREAK LOSS OF COOLANT ACCIDENT

Reactor Operational Mode	Disposition
1	Analyze
2-6	Bounded by the Event Initiated from Mode 1

TABLE 14.6.5.2-3 MILLSTONE UNIT 2 SMALL BREAK LOSS OF COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS

Reactor Power, MWt	2754
Axial Power Shape	-
Peak LHR, kW/ft	15.1
Radial Peaking Factor (1.69 plus uncertainties)	1.854
RCS Flow Rate, gpm	360,000
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	549
SIT Pressure, psia	214.7
SIT Fluid Temperature, °F	120
SIT Water Volume, ft^3	1135
Maximum SG Tube Plugging Level per SG, %	5.87
SG Secondary Pressure, psia	880
MFW Temperature, °F	435
AFW Flow Rate per SG, gpm	72
AFW Temperature, °F	70
Low-Low SG Level Setpoint, % Narrow Range Span	0
AFW Delay, sec	240
HPSI and LPSI Fluid Temperature, °F	140
Pressurizer Pressure - Low Reactor Trip Setpoint (RPS), psia	1700
Reactor Trip Delay Time on Low Pressurizer Pressure, sec	0.9
CEA Holding Coil Release Delay Time, sec	0.5
SIAS Activation Pressurizer Pressure Setpoint (Harsh Environment Conditions), psia	1500
HPSI Pump Delay Time on SIAS, sec	25
LPSI Pump Delay Time on SIAS, sec	45
MSSV Lift Pressure and Tolerance	Nominal+ 3%

TABLE 14.6.5.2-3 MILLSTONE UNIT 2 SMALL BREAK LOSS OF COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS (CONTINUED)

High Pressure Safety Injection Flow Rate for Cold Leg Breaks

RCS Cold Leg Pressure (psia)	Loop 1A (gpm)	Loop 1B (gpm)	Loop 2A (gpm)	Loop 2B (gpm)
14.7	143	142	145	145
50	141	140	142	142
100	138	138	139	139
150	135	135	135	135
200	132	132	132	132
300	125	125	125	125
500	109	109	109	109
700	92	92	92	92
900	71	71	71	71
1000	59	59	59	59
1050	52	52	52	52
1100	43	43	43	43
1150	30	30	30	30
1190	15	15	15	15
1204	0	0	0	0

TABLE 14.6.5.2-3 MILLSTONE UNIT 2 SMALL BREAK LOSS OF COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS (CONTINUED)

Low Pressure Safety Injection Flow Rates for Cold Leg Breaks

RCS Cold Leg Pressure (psia)	Intact Loop 1A (gpm)	Intact Loop 1B (gpm)	Intact Loop 2A (gpm)	Intact Loop 2B (gpm)
14.7	1314	0	0	1369
50	1164	0	0	1214
100	904	0	0	945
150	519	0	0	546
200	0	0	0	0

TABLE 14.6.5.2-4 DELETED BY FSCR MPS-UCR-2016-016

TABLE 14.6.5.2-5 CALCULATED EVENT TIMES FOR SMALL BREAK LOSS-OF-COOLANT ACCIDENT

Break diameter (in)	PCT (°F)	Break opens	Low PZR Pressure Trip	Reactor Scram, RCP and Turbine Trip	SIAS issued	AFAS Available	HPSI available	LPSI available	HPSI Flow Begins	LPSI Flow Begins	Motor driven AFW on, Initial	Loop seal 1A clears	Loop seal 1B clears	Loop seal 2A clears	Loop seal 2B clears	Break uncovers	SIT injection begins	Minimum RV mass occurs	Hot Rod rupture occurs	PCT occurs	Time of core uncovery	Non-condensable gas at the break
2.00	1135	0	66	67	79	90	104	124	242	--	516	--	--	--	1088	1100	--	4116	--	4330	3030	--
3.00	1384	0	29	30	38	54	63	83	98	--	418	--	--	--	512	552	--	2082	--	2451	280	--
3.60	1472	0	20	22	29	46	54	74	68	--	414	--	--	--	384	394	5700	1638	--	2008	208	--
3.70	1501	0	19	21	27	46	52	72	64	--	412	--	--	--	364	374	5102	1572	--	1978	200	--
3.75	1599	0	19	20	27	46	52	72	64	--	422	--	--	--	360	370	4660	1458	1772	1892	190	--
3.76	1651	0	19	20	27	46	52	72	62	--	420	--	--	--	358	368	4638	1428	1654	1857	188	--
3.78	1707	0	19	20	27	44	52	72	62	--	408	--	--	--	356	366	4580	1386	1563	1824	186	--
3.785	1690	0	19	20	27	44	52	72	62	--	408	--	--	--	356	366	4534	1396	1591	1852	186	--
3.79	1476	0	18	20	26	44	51	71	62	--	408	944	--	1950	1330	374	--	1680	--	2035	186	--
3.90	1606	0	18	19	25	44	50	70	58	--	434	--	366	2340	1176	350	1944	1598	1852	1954	192	--
4.02	1596	0	17	18	24	42	49	69	56	--	430	700	--	1250	1278	340	1738	1452	1684	1746	184	--
4.40	1581	0	14	16	22	40	47	67	48	--	442	600	--	1230	954	292	1278	1230	1270	1284	154	--
4.60	1621	0	13	15	20	40	45	65	46	--	514	250	250	--	478	272	1068	1072	1047	1076	144	--
4.80	1637	0	13	14	19	40	44	64	46	956	--	--	480	--	574	260	934	936	907	939	136	--
5.00	1615	0	12	13	19	38	44	64	44	848	--	--	430	710	506	260	824	830	814	832	128	--
5.30	1591	0	11	13	18	38	43	63	44	714	--	--	290	250	224	232	692	698	691	699	118	--
5.50	1398	0	11	12	17	38	42	62	42	692	--	436	204	270	190	212	670	674	--	674	112	--
6.00	1535	0	10	12	16	38	41	61	42	514	--	--	186	180	180	188	496	500	--	501	102	--
7.00	1510	0	9	11	14	38	39	59	40	360	--	258	264	180	112	146	350	352	--	353	74	--
8.00	1540	0	9	10	13	218	38	58	40	264	--	--	98	116	86	114	254	256	--	258	56	--
9.00	1399	0	9	10	13	--	38	58	38	206	--	70	86	66	102	84	198	202	--	202	46	288
9.49	1437	0	9	10	12	--	37	57	38	184	--	74	76	60	84	78	178	180	--	180	42	--

TABLE 14.6.5.2-6 ANALYSIS RESULTS FOR SMALL BREAK LOSS-OF-COOLANT ACCIDENT

Break diameter (in)	2.00	3.00	3.60	3.70	3.75	3.76
Break Area (ft^2)	0.02182	0.04909	0.07069	0.07467	0.07670	0.07711
Peak Clad Temperature (°F)	1135	1384	1472	1501	1599	1651
Time of PCT (sec)	4330	2451	2008	1978	1892	1857
Time of Rupture (sec)	--	--	--	--	1772	1654
Transient MLO (%)	0.0864	0.4249	0.6647	0.7543	1.9966	2.6718
Total MLO (%)	2.337	2.676	2.915	3.005	4.247	4.922
Core Wide Oxidation (%)	0.0042	0.0111	0.0124	0.0139	0.0254	0.0314
PCT Elevation (ft)	10.52	10.52	10.77	10.77	11.02	11.02

Break diameter (in)	3.78	3.785	3.79	3.90	4.02	4.40
Break Area (ft^2)	0.07793	0.07814	0.07834	0.08296	0.08814	0.10559
Peak Clad Temperature (°F)	1707	1690	1476	1606	1596	1581
Time of PCT (sec)	1824	1852	2035	1954	1746	1284
Time of Rupture (sec)	1563	1591	--	1852	1684	1270
Transient MLO (%)	3.5273	3.2474	0.6689	1.3276	1.1046	0.6721
Total MLO (%)	5.778	5.498	2.920	3.578	3.355	2.923
Core Wide Oxidation (%)	0.0396	0.0368	0.0119	0.0175	0.0162	0.0110
PCT Elevation (ft)	11.02	11.02	10.77	11.02	11.02	10.52

Break diameter (in)	4.60	4.80	5.00	5.30	5.50	6.00
Break Area (ft^2)	0.11541	0.12566	0.13635	0.15321	0.16499	0.19635
Peak Clad Temperature (°F)	1621	1637	1615	1591	1398	1535
Time of PCT (sec)	1076	939	832	699	674	501
Time of Rupture (sec)	1047	907	814	691	--	--
Transient MLO (%)	0.7874	0.8477	0.6456	0.5072	0.1142	0.2316
Total MLO (%)	3.038	3.098	2.896	2.758	2.365	2.482
Core Wide Oxidation (%)	0.0108	0.0111	0.0091	0.0075	0.0020	0.0051
PCT Elevation (ft)	10.77	10.77	10.52	10.52	10.27	10. 27

TABLE 14.6.5.2-6 ANALYSIS RESULTS FOR SMALL BREAK LOSS-OF-COOLANT ACCIDENT (CONTINUED)

Break diameter (in)	7.00	8.00	9.00	9.49
Break Area (ft^2)	0.26725	0.34907	0.44179	0.49120
Peak Clad Temperature (°F)	1510	1540	1399	1437
Time of PCT (sec)	353	258	202	180
Time of Rupture (sec)	--	--	--	--
Transient MLO (%)	0.2133	0.2247	0.1097	0.1367
Total MLO (%)	2.464	2.475	2.360	2.387
Core Wide Oxidation (%)	0.0049	0.0046	0.0018	0.0024
PCT Elevation (ft)	10. 27	10. 27	10.02	10.02

TABLE 14.6.5.2-7 PEAK CLAD TEMPERATURE INCLUDING ALL PENALTIES AND BENEFITS - SMALL BREAK LOCA

LICENSING BASIS	**PCT (°F)**	**Delta PCT (°F)**
Analysis of Record Peak Clad Temperature (PCT)	1707	
PCT ASSESSMENTS (Delta PCT)		
1. Error in the S-RELAP5 Oxidation Calculation		3
LICENSING BASES PCT including all PCT ASSESSMENTS	**=**	**1710°F**

FIGURE 14.6.5.2–1 PEAK CLADDING TEMPERATURE VERSUS BREAK SIZE (SBLOCA BREAK SPECTRUM)

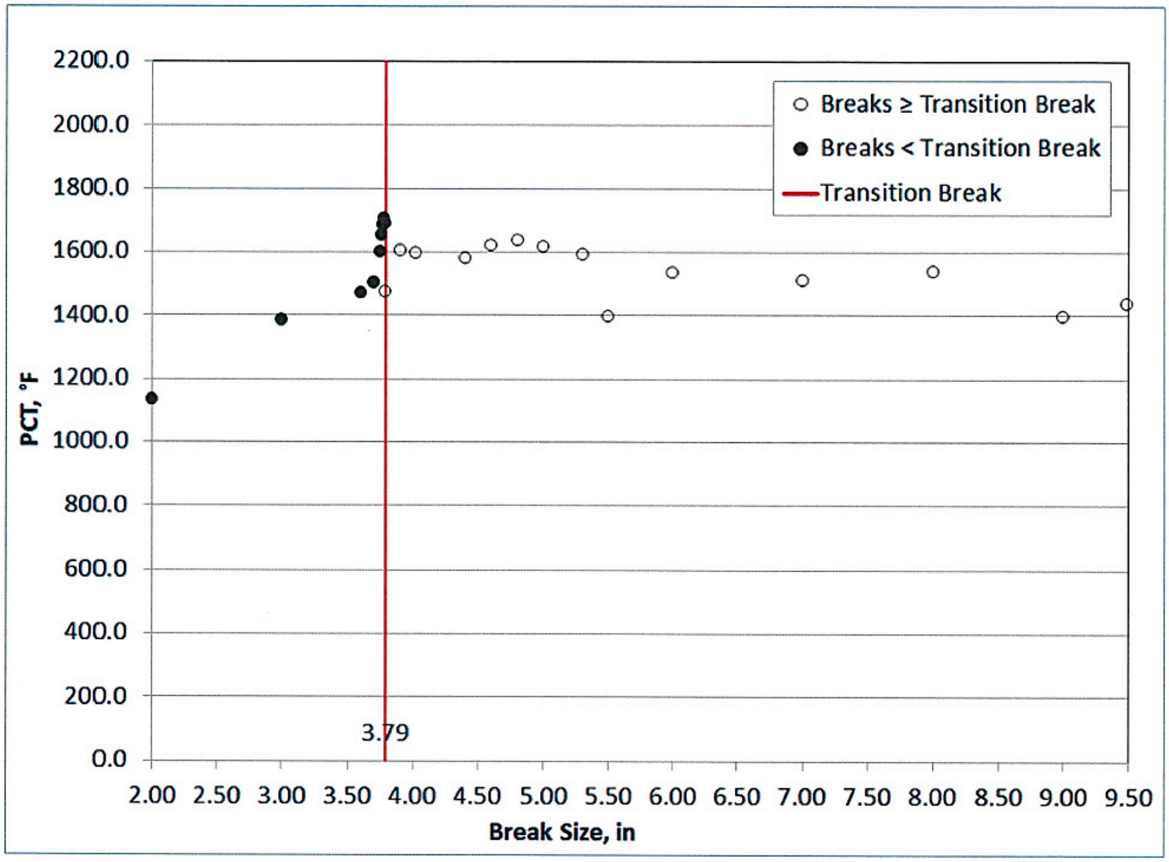


FIGURE 14.6.5.2–2 REACTOR POWER - 3.78-INCH BREAK

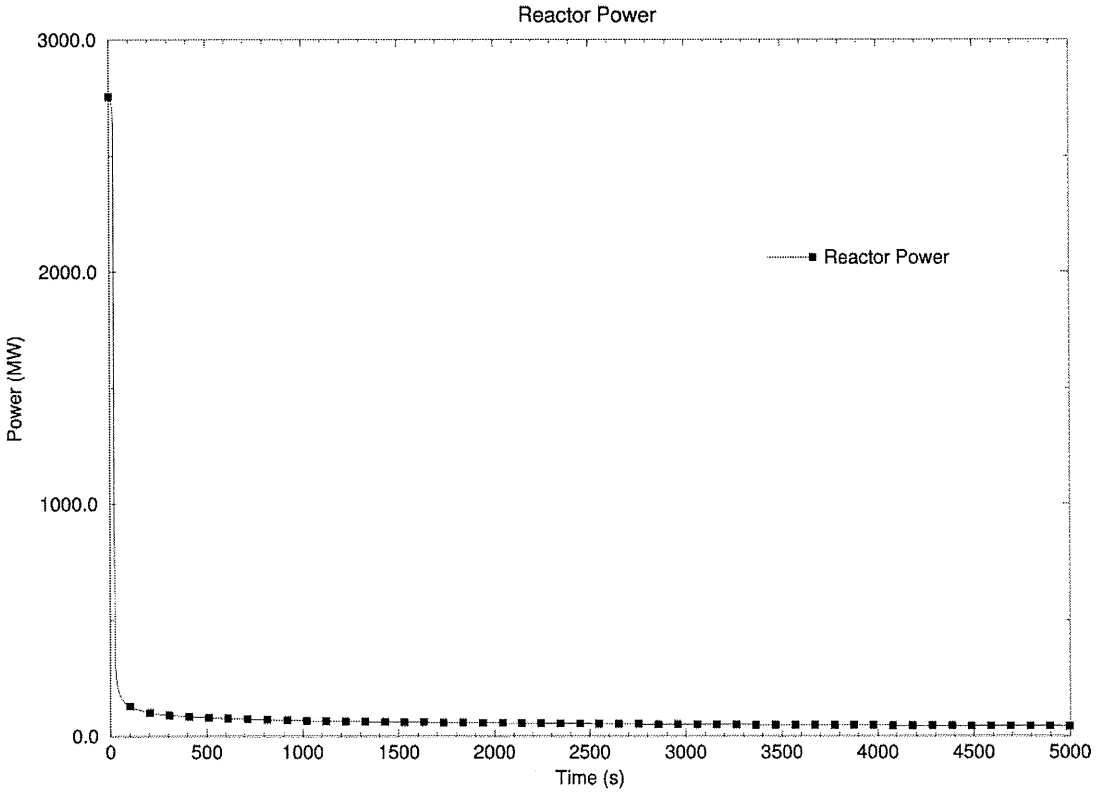


FIGURE 14.6.5.2–3 PRIMARY AND SECONDARY SYSTEM PRESSURES - 3.78-INCH BREAK

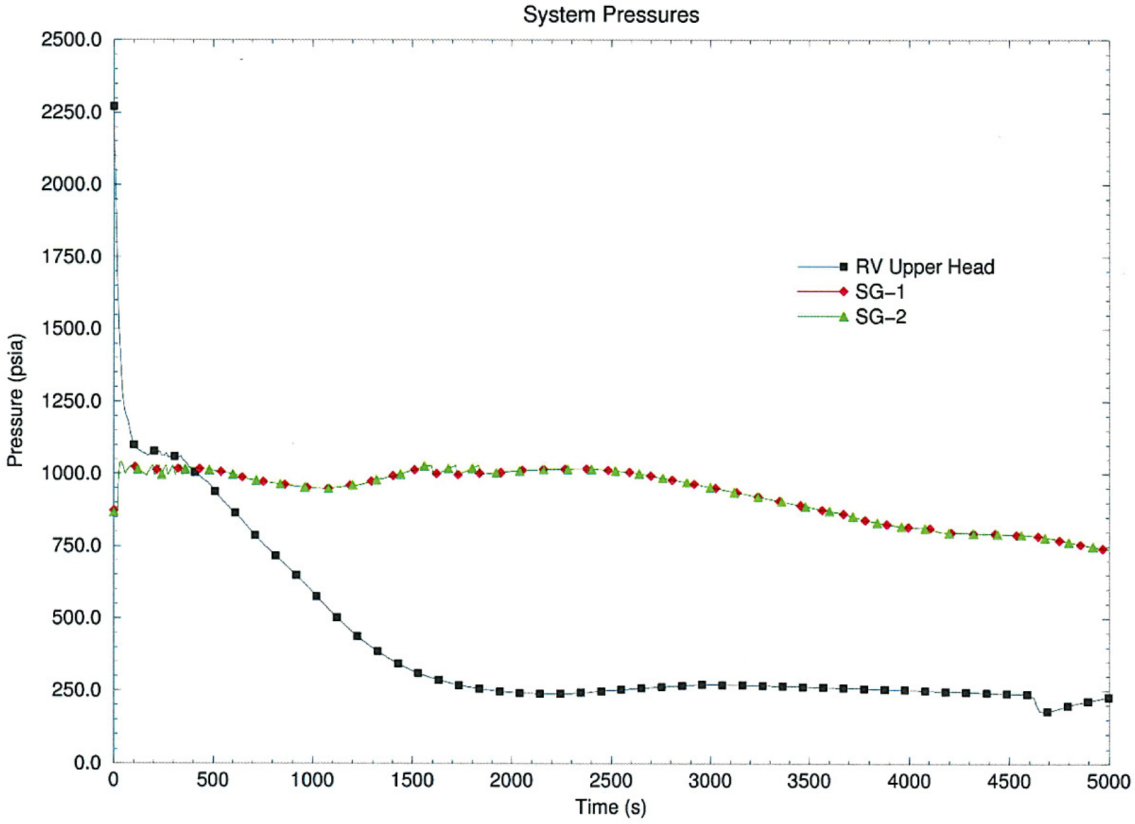


FIGURE 14.6.5.2–4 BREAK MASS FLOW RATE - 3.78-INCH BREAK

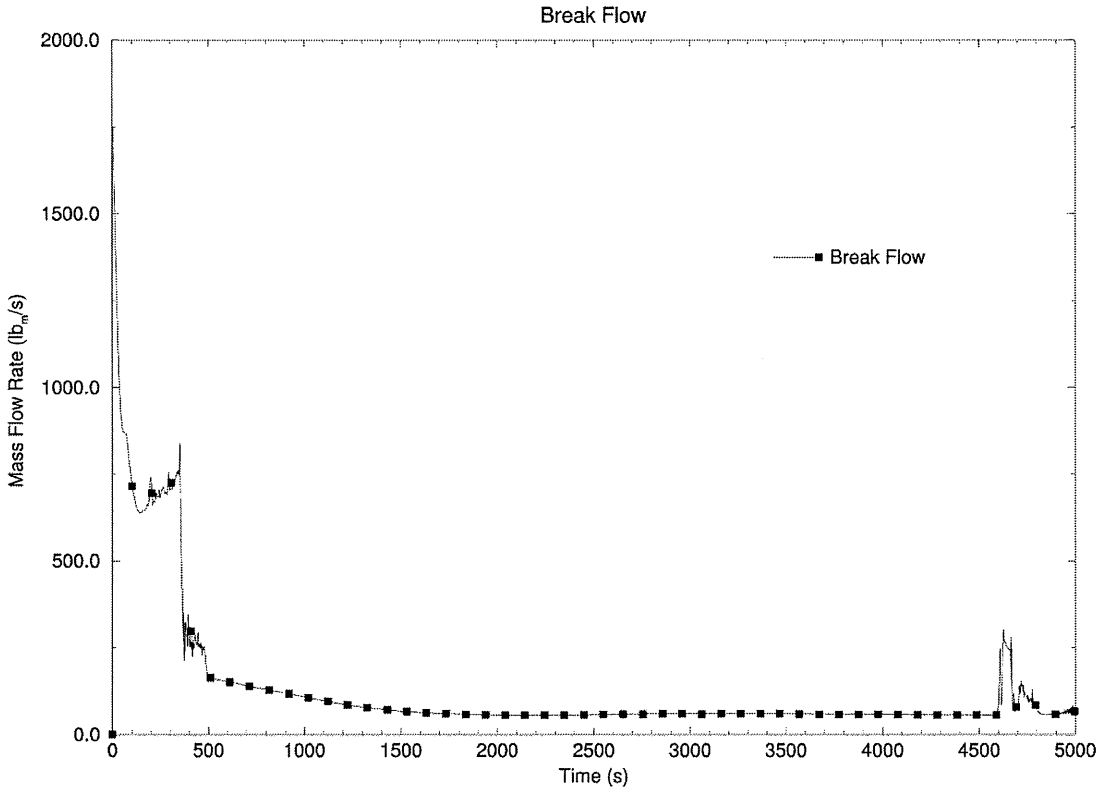


FIGURE 14.6.5.2–5 BREAK VAPOR VOID FRACTION - 3.78-INCH BREAK

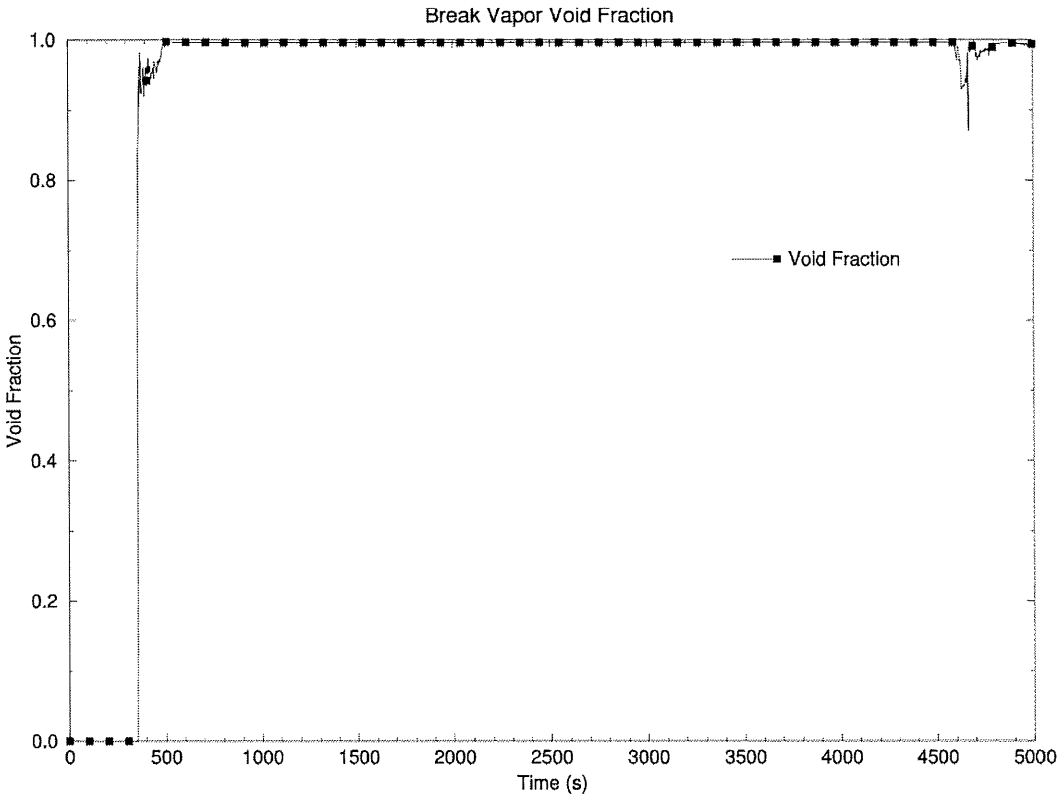


FIGURE 14.6.5.2–6 LOOP SEAL VOID FRACTION - 3.78-INCH BREAK

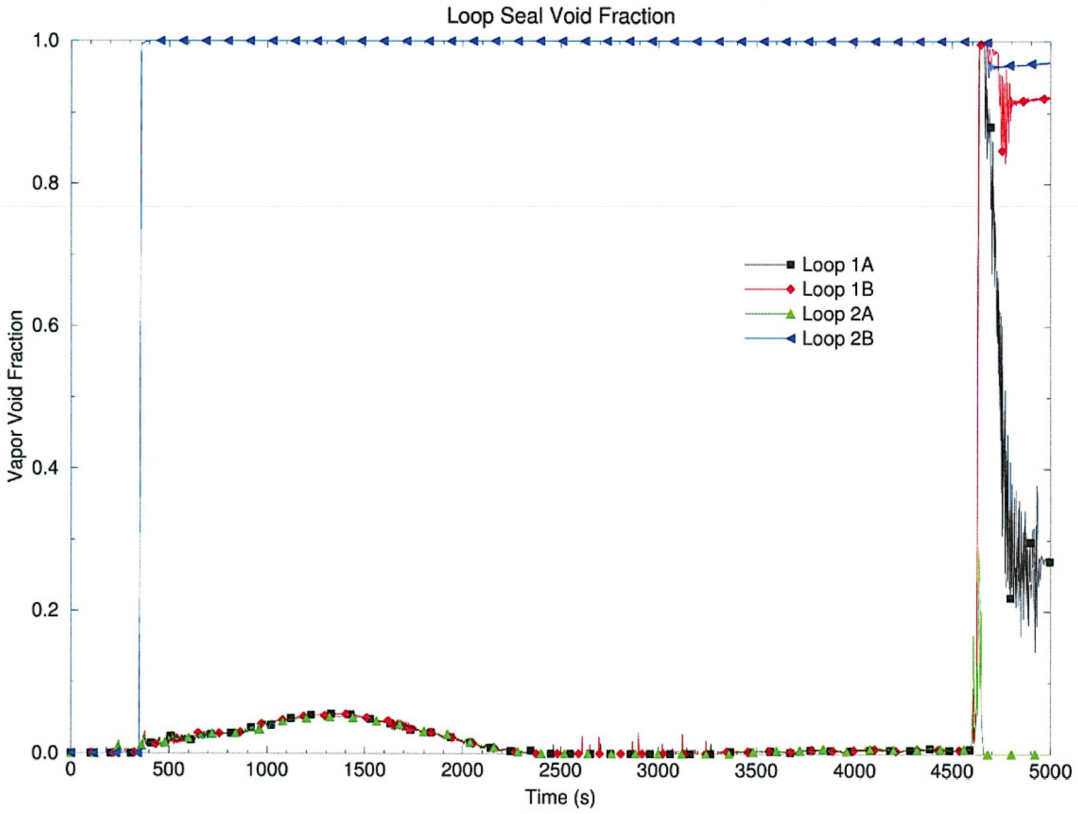


FIGURE 14.6.5.2–7 TOTAL CORE INLET MASS FLOW RATE - 3.78-INCH BREAK

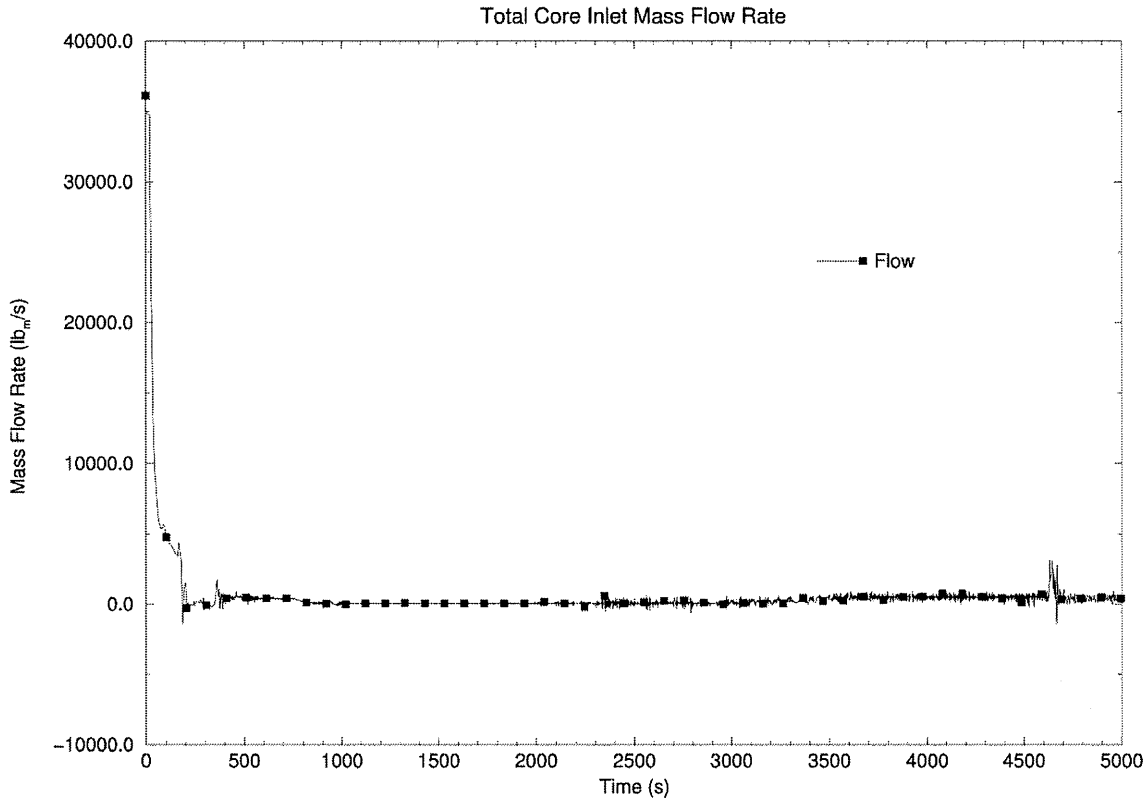


FIGURE 14.6.5.2–8 DOWNCOMER COLLAPSED LIQUID LEVEL - 3.78-INCH BREAK

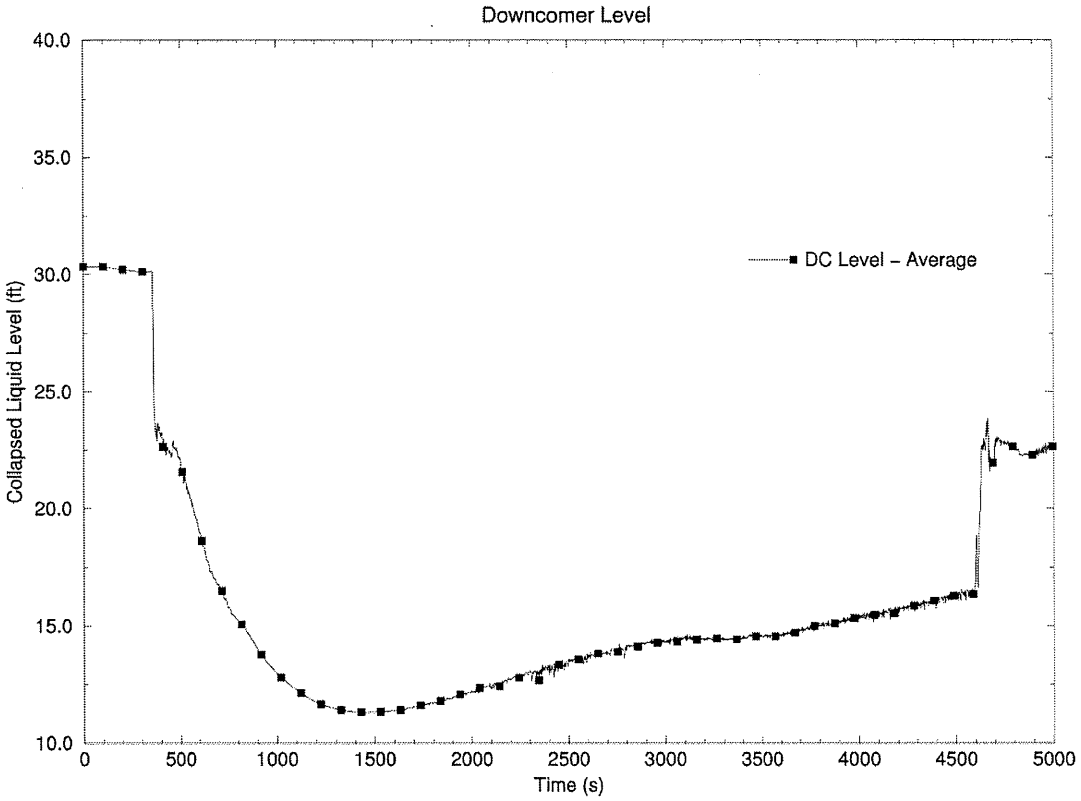


FIGURE 14.6.5.2–9 INNER AND OUTER CORE COLLAPSED LIQUID LEVEL - 3.78-INCH BREAK

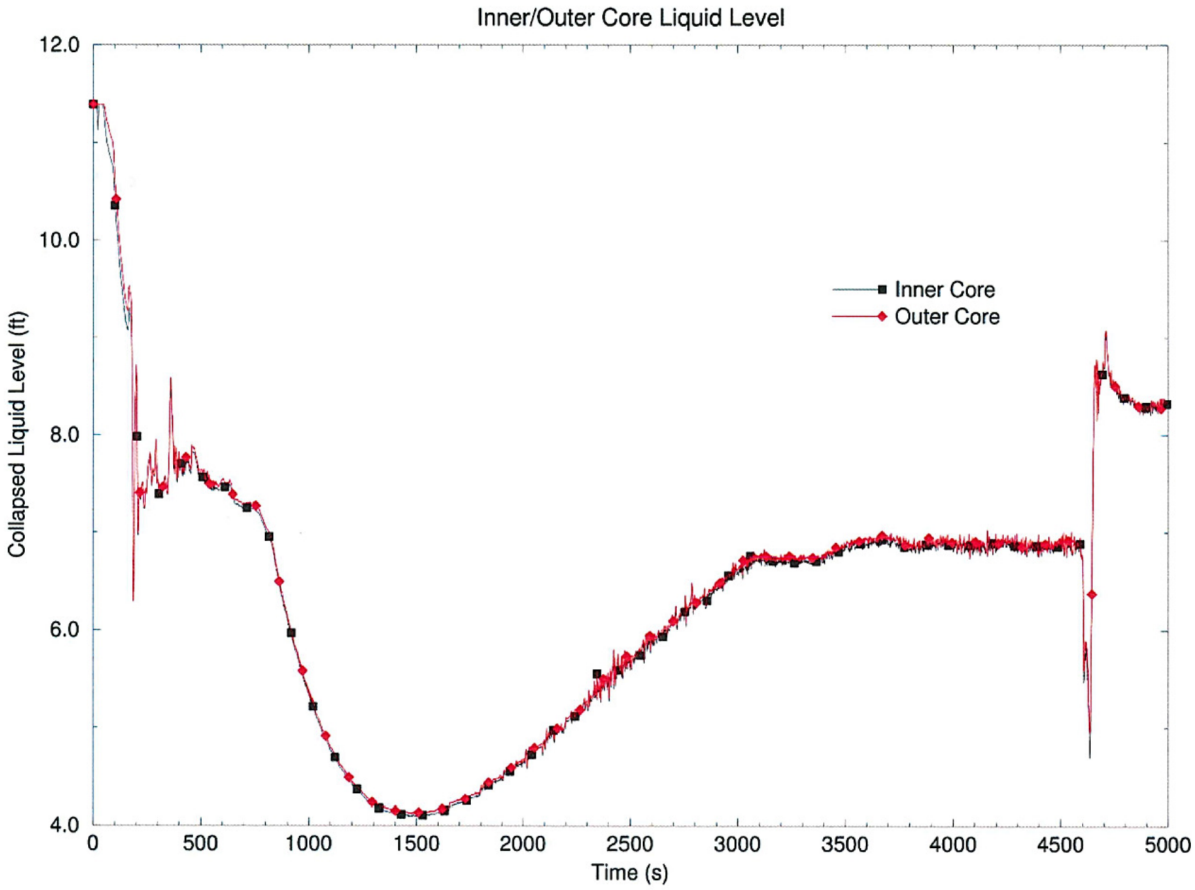


FIGURE 14.6.5.2–10 REACTOR VESSEL MASS - 3.78-INCH BREAK

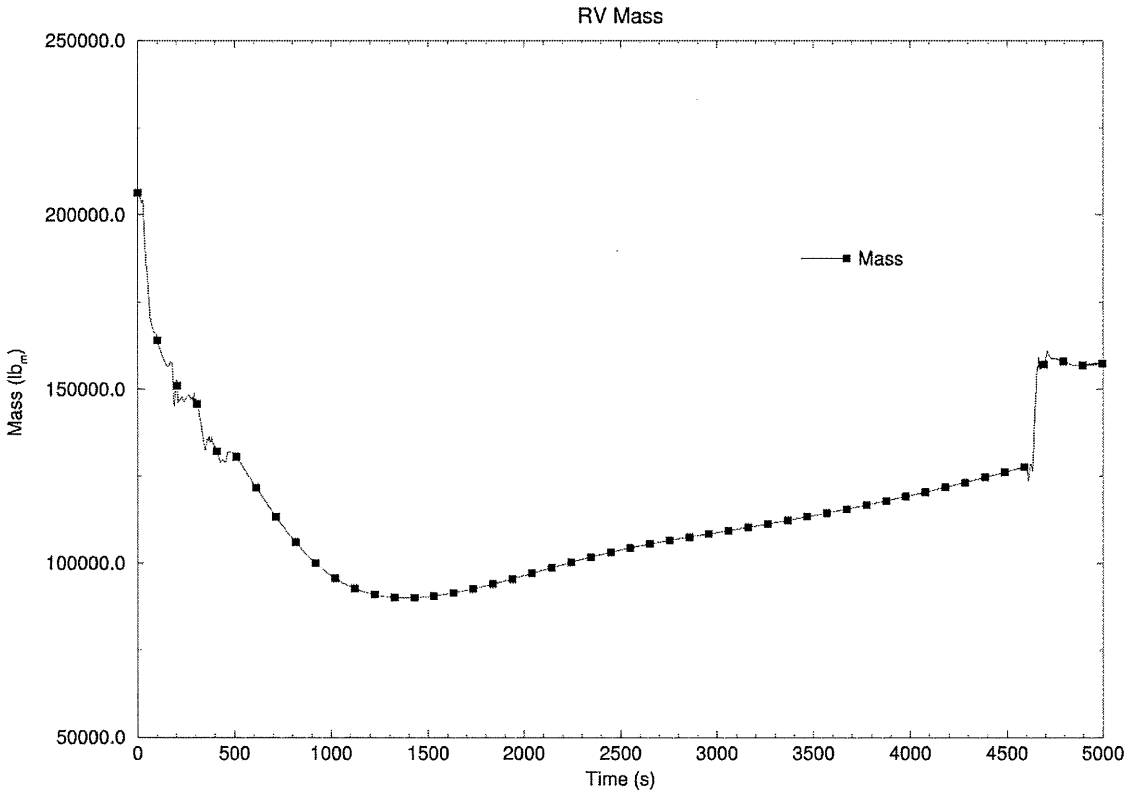


FIGURE 14.6.5.2–11 RCS LOOP MASS FLOW RATES - 3.78-INCH BREAK

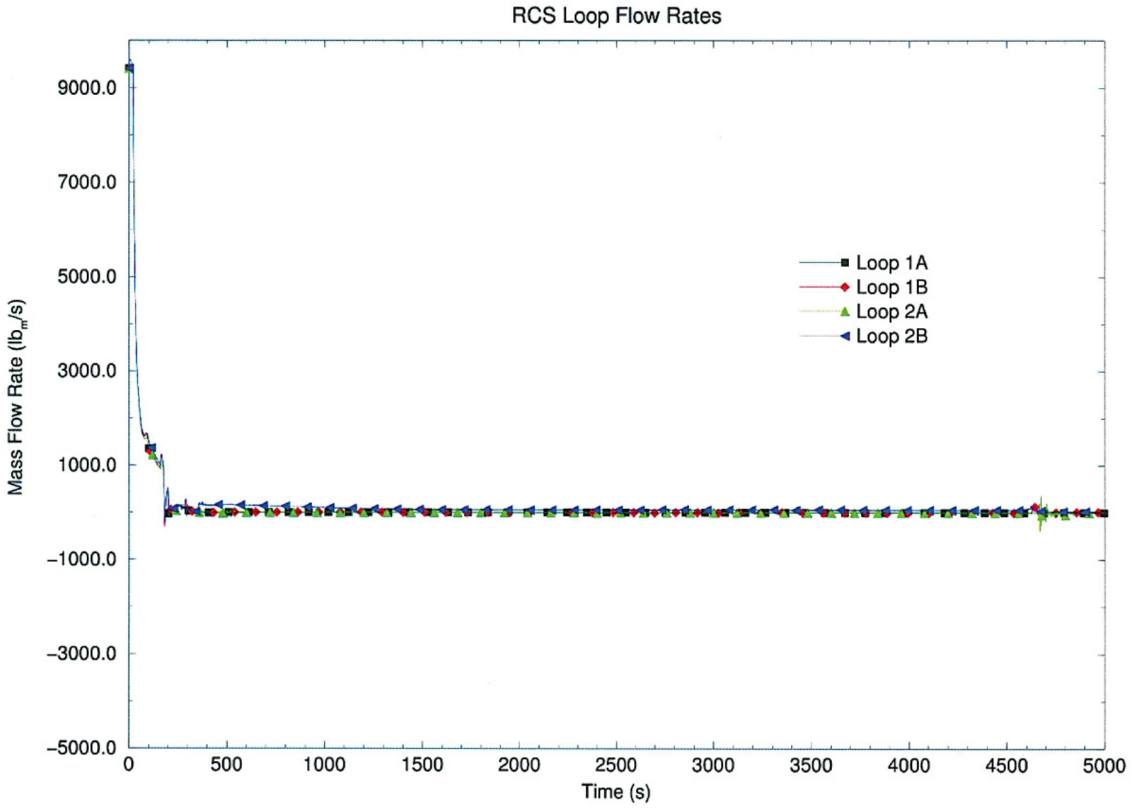


FIGURE 14.6.5.2–12 STEAM GENERATOR MAIN FEEDWATER MASS FLOW RATES - 3.78-INCH BREAK

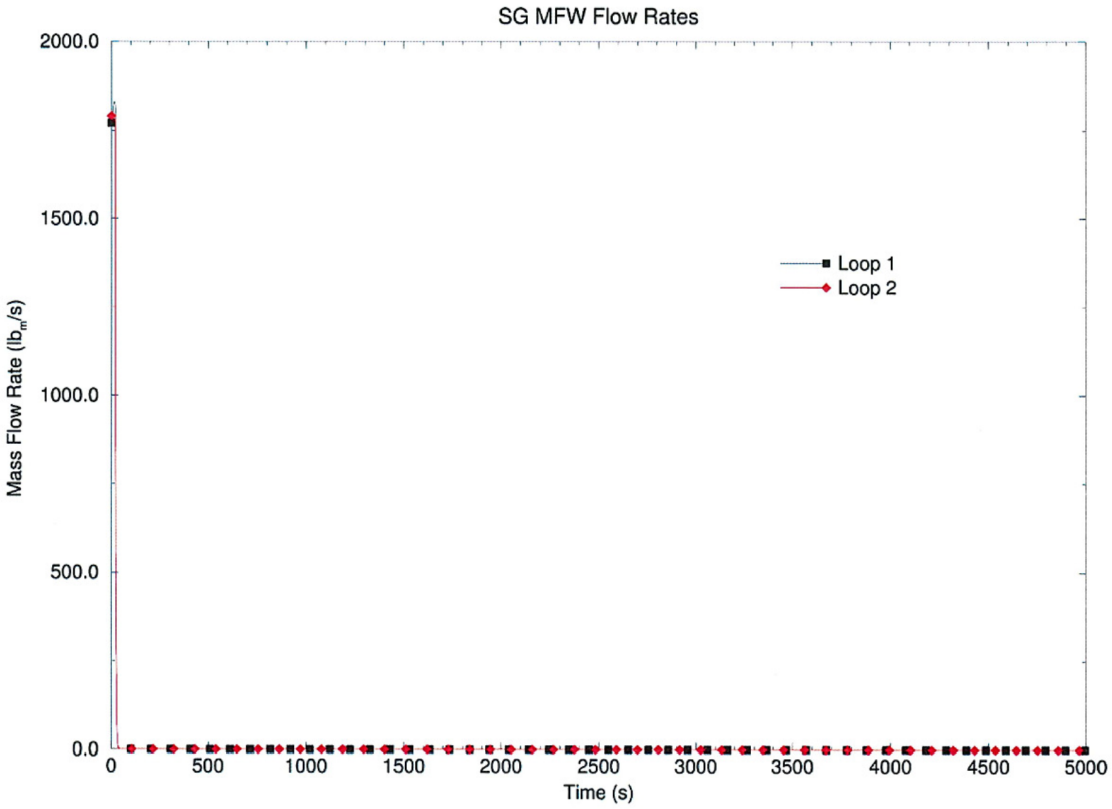


FIGURE 14.6.5.2–13 STEAM GENERATOR AUXILIARY FEEDWATER MASS FLOW RATES - 3.78-INCH BREAK

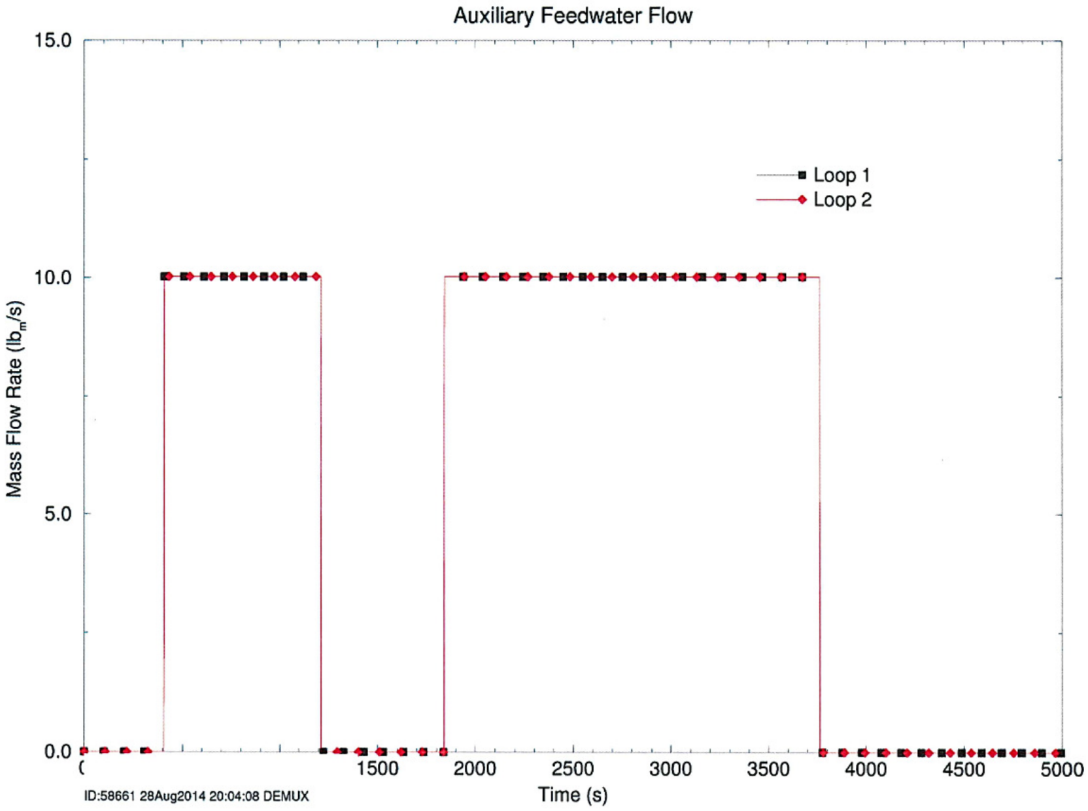


FIGURE 14.6.5.2–14 STEAM GENERATOR TOTAL MASS - 3.78-INCH BREAK

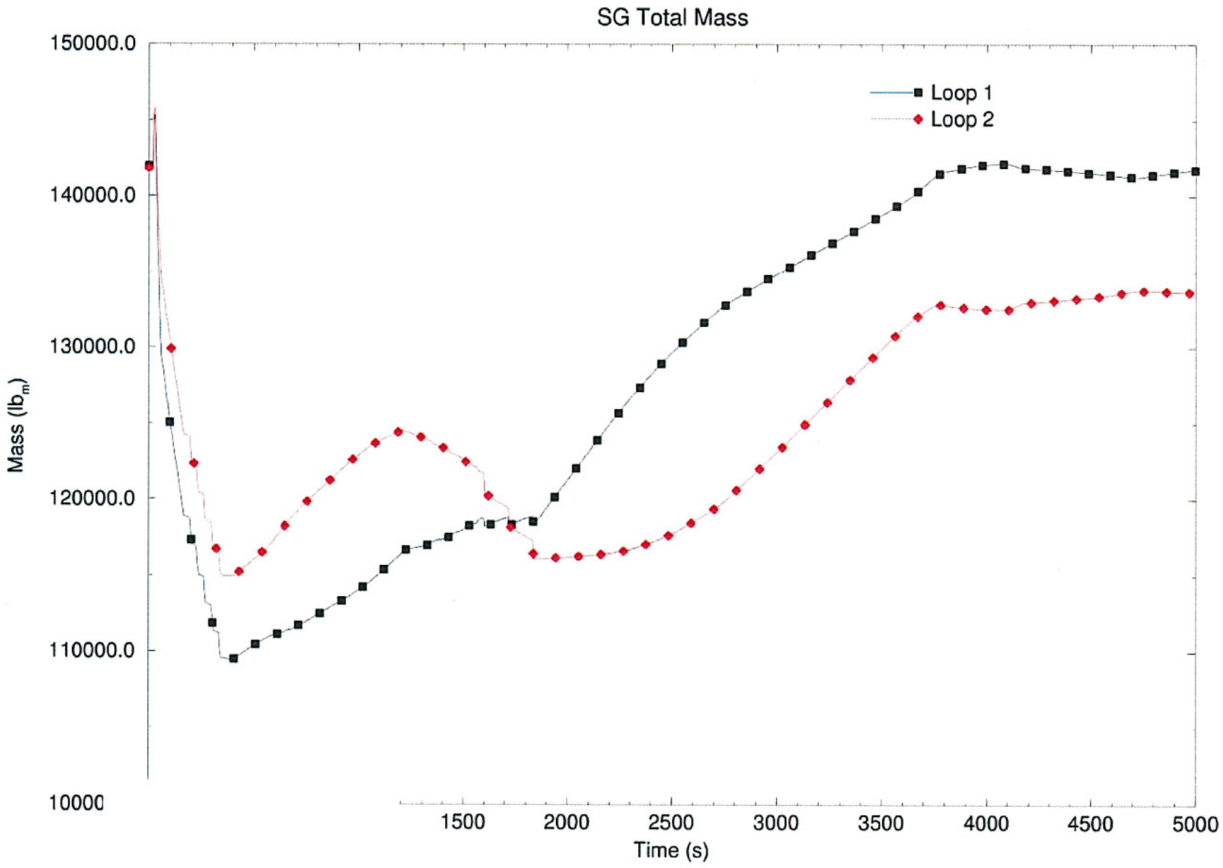


FIGURE 14.6.5.2–15 STEAM GENERATOR NARROW RANGE LEVEL % - 3.78-INCH BREAK

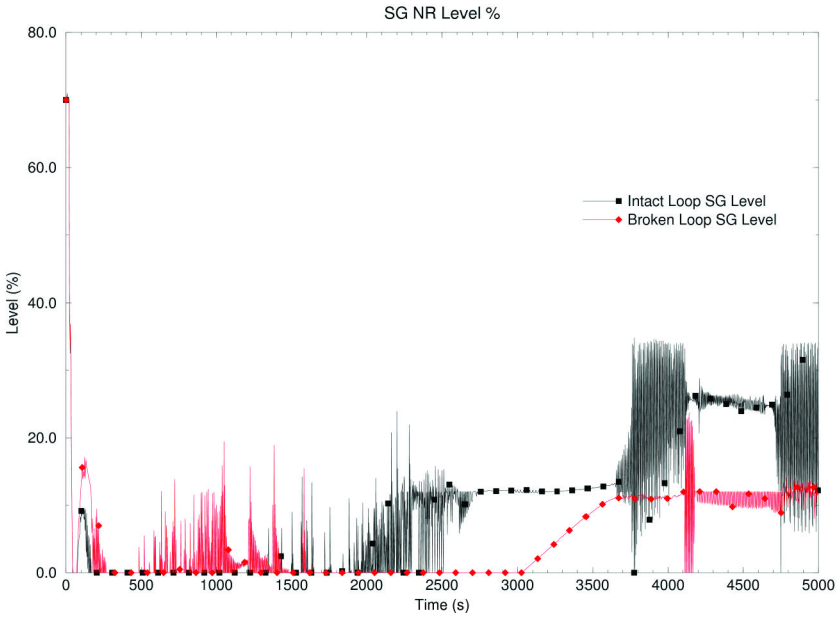


FIGURE 14.6.5.2–16 HIGH PRESSURE SAFETY INJECTION MASS FLOW RATES - 3.78-INCH BREAK

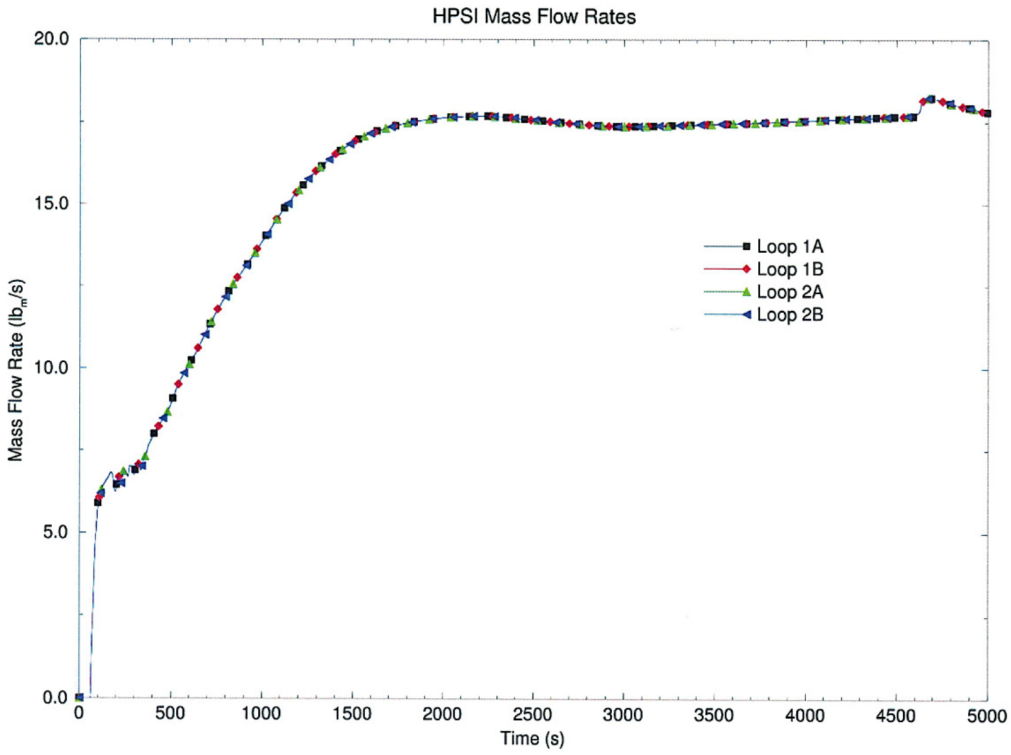


FIGURE 14.6.5.2–17 LOW PRESSURE SAFETY INJECTION MASS FLOW RATES - 3.78-INCH BREAK

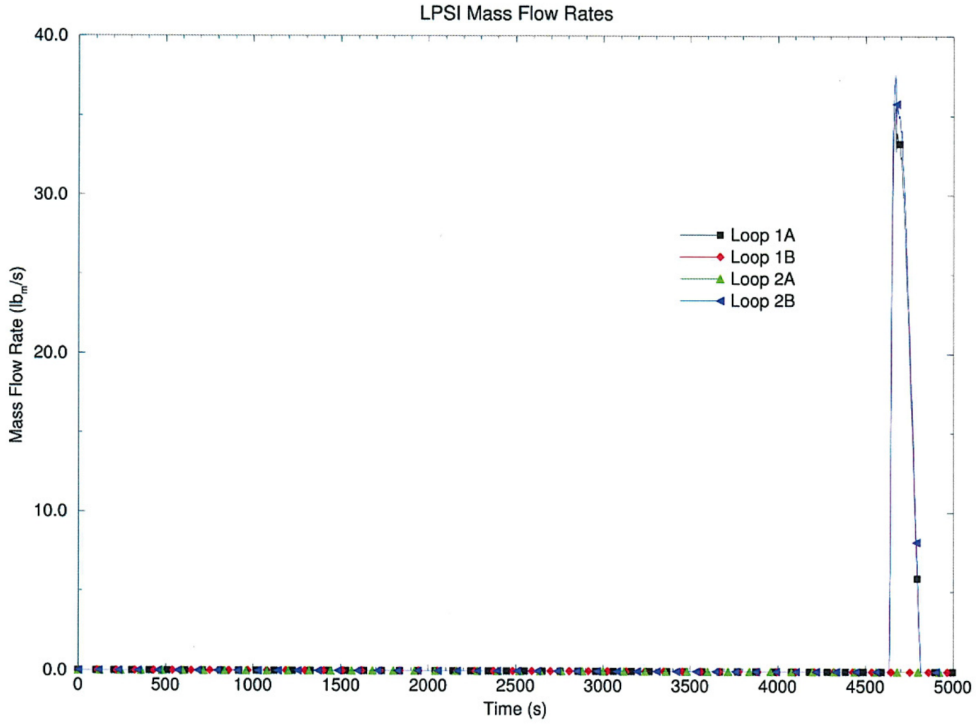


FIGURE 14.6.5.2–18 SAFETY INJECTION TANK MASS FLOW RATES - 3.78-INCH BREAK

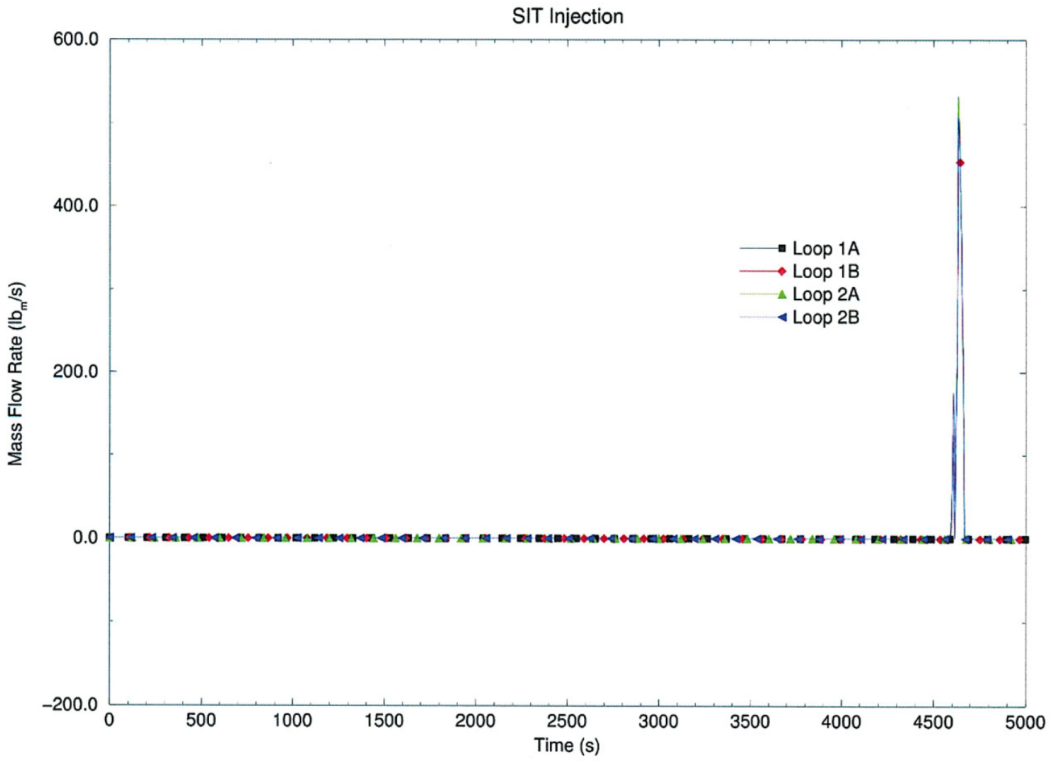


FIGURE 14.6.5.2–19 INTEGRATED BREAK FLOW AND ECCS FLOW - 3.78-INCH BREAK

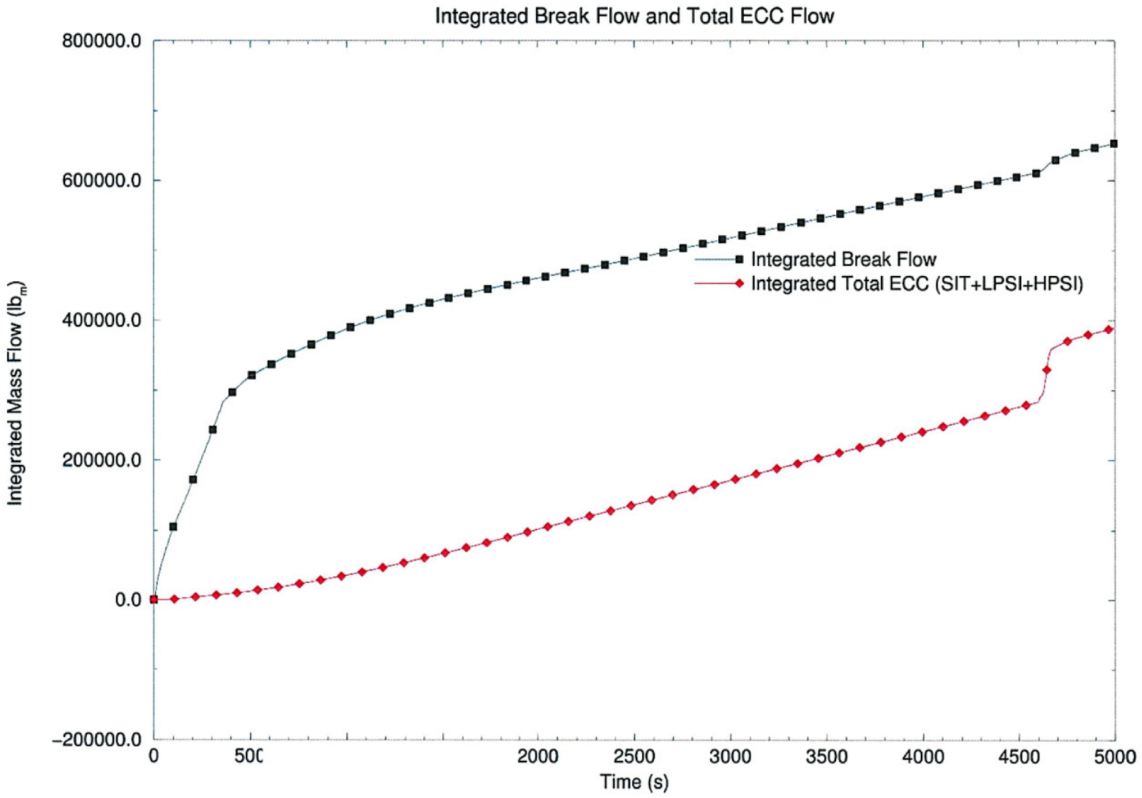


FIGURE 14.6.5.2–20 HOT ASSEMBLY COLLAPSED LIQUID LEVEL - 3.78-INCH BREAK

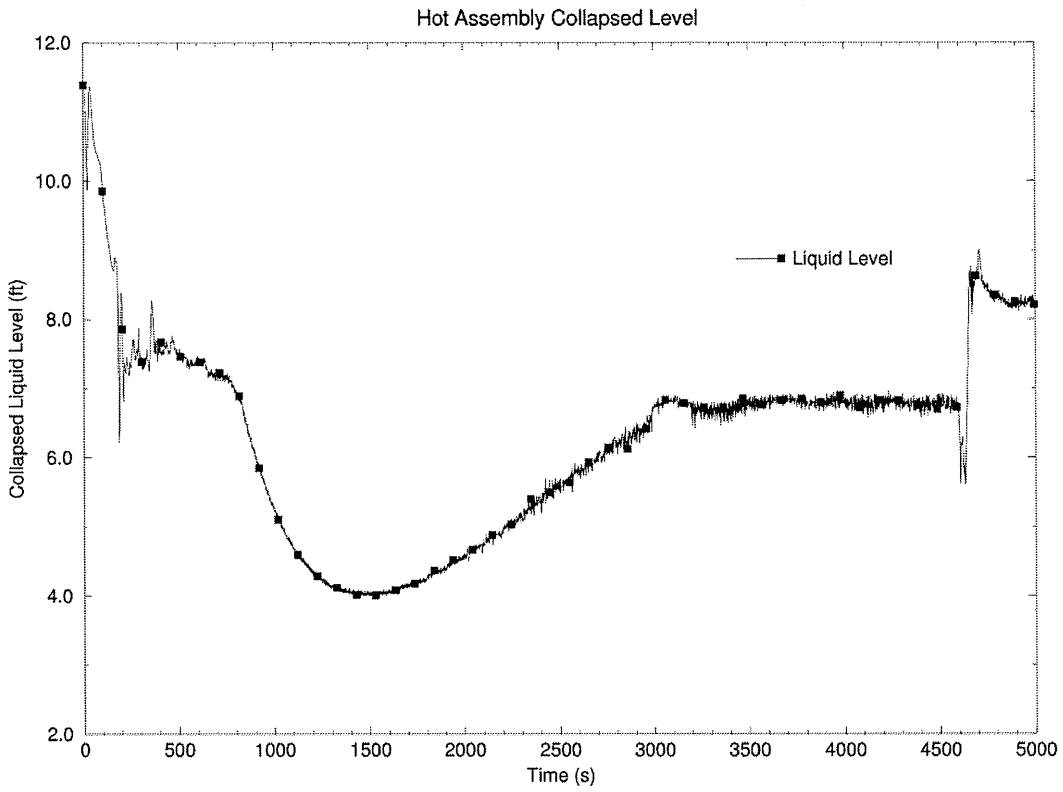


FIGURE 14.6.5.2–21 HOT ASSEMBLY MIXTURE LEVEL - 3.78-INCH BREAK

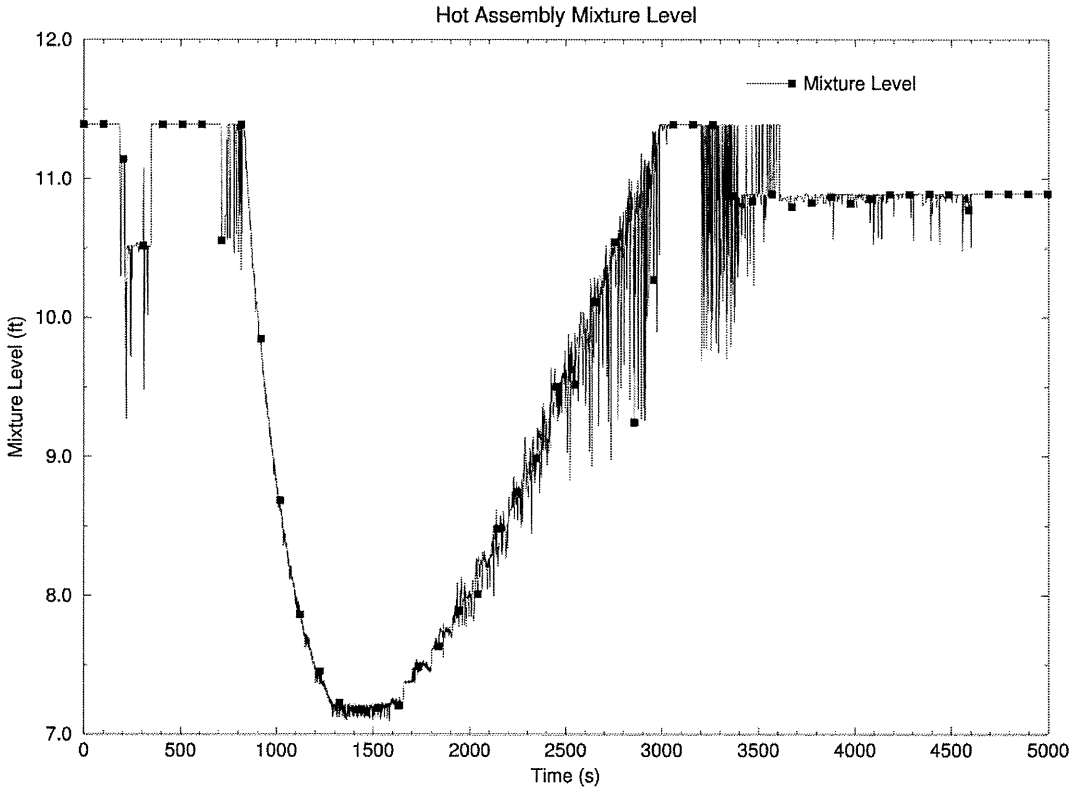


FIGURE 14.6.5.2–22 PEAK CLADDING TEMPERATURE AT PCT LOCATION (11.02 FT) - 3.78-INCH BREAK

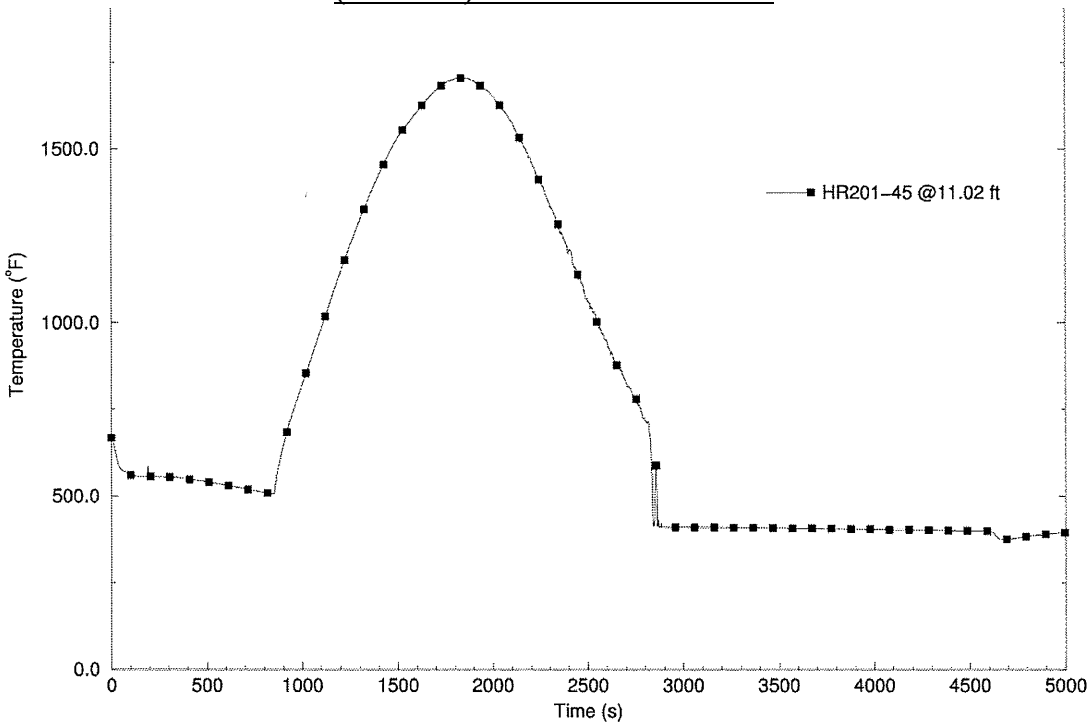


FIGURE 14.6.5.2–23 (DELETED BY FSARCR 00-MP2-023)

FIGURE 14.6.5.2–24 (DELETED BY FSARCR 00-MP2-023)

TABLE 14.6.5.3-1 CORE AND SYSTEM PARAMETERS USED IN THE LTC ANALYSIS

Parameter	Value
Reactor power level, MWt (102% of nominal)	2754
Number of plugged tubes per SG	1000
SG / RCS cooldown rate, °F/hr (maximum)	40
Atmospheric dump valve capacity at 1000 psia, lbm/hr/valve (minimum)	879,028
Boric acid concentration, ppm (maximum)	
Reactor coolant system	2640
Refueling water tank	2640
Safety injection tanks	2640
Boric acid storage tanks	6139
Initial inventory (maximum)	
Reactor coolant system, lbm	543,710
Refueling water tank, gal	448,520
Safety injection tanks, ft^3/tank	1190
Boric acid storage tank, ft^3/tank	865.7

FIGURE 14.6.5.3–1 LONG TERM COOLING PLAN

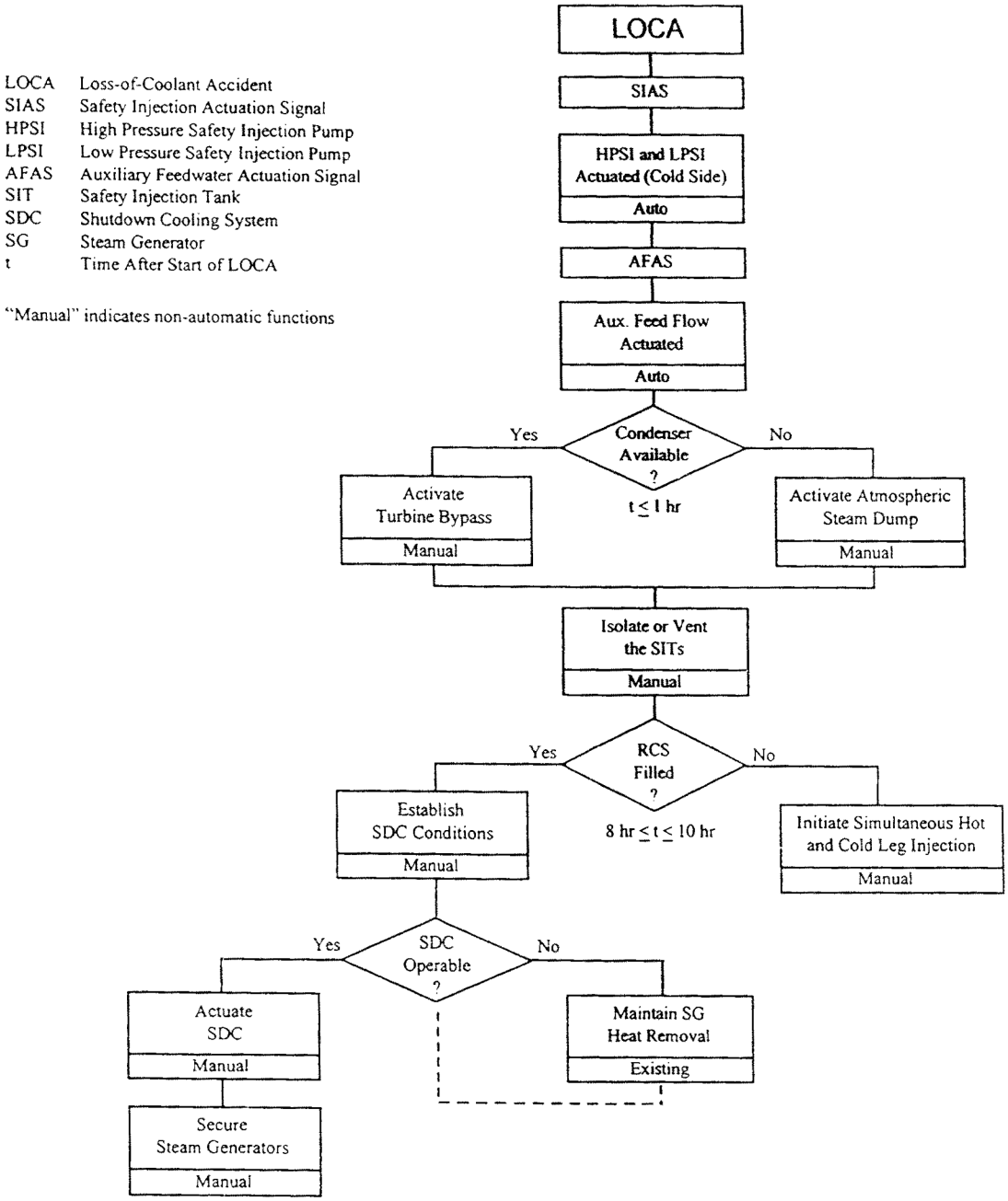


FIGURE 14.6.5.3–2 REACTOR COOLANT SYSTEM REFILL TIME VS. BREAK AREA

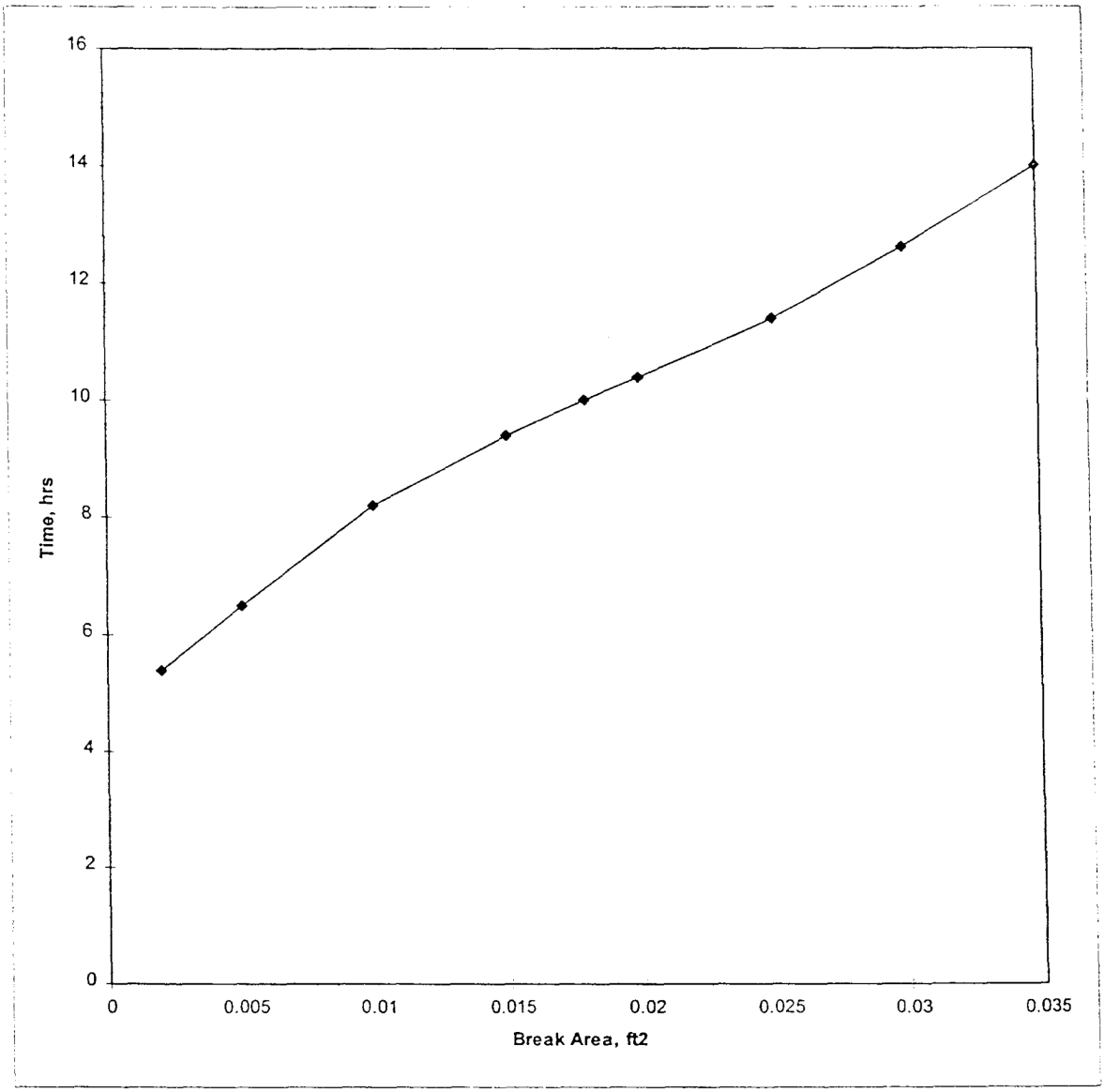


FIGURE 14.6.5.3–3 CORE FLUSH BY HOT SIDE INJECTION FOR A DOUBLE-ENDED GUILLOTINE COLD LEG BREAK

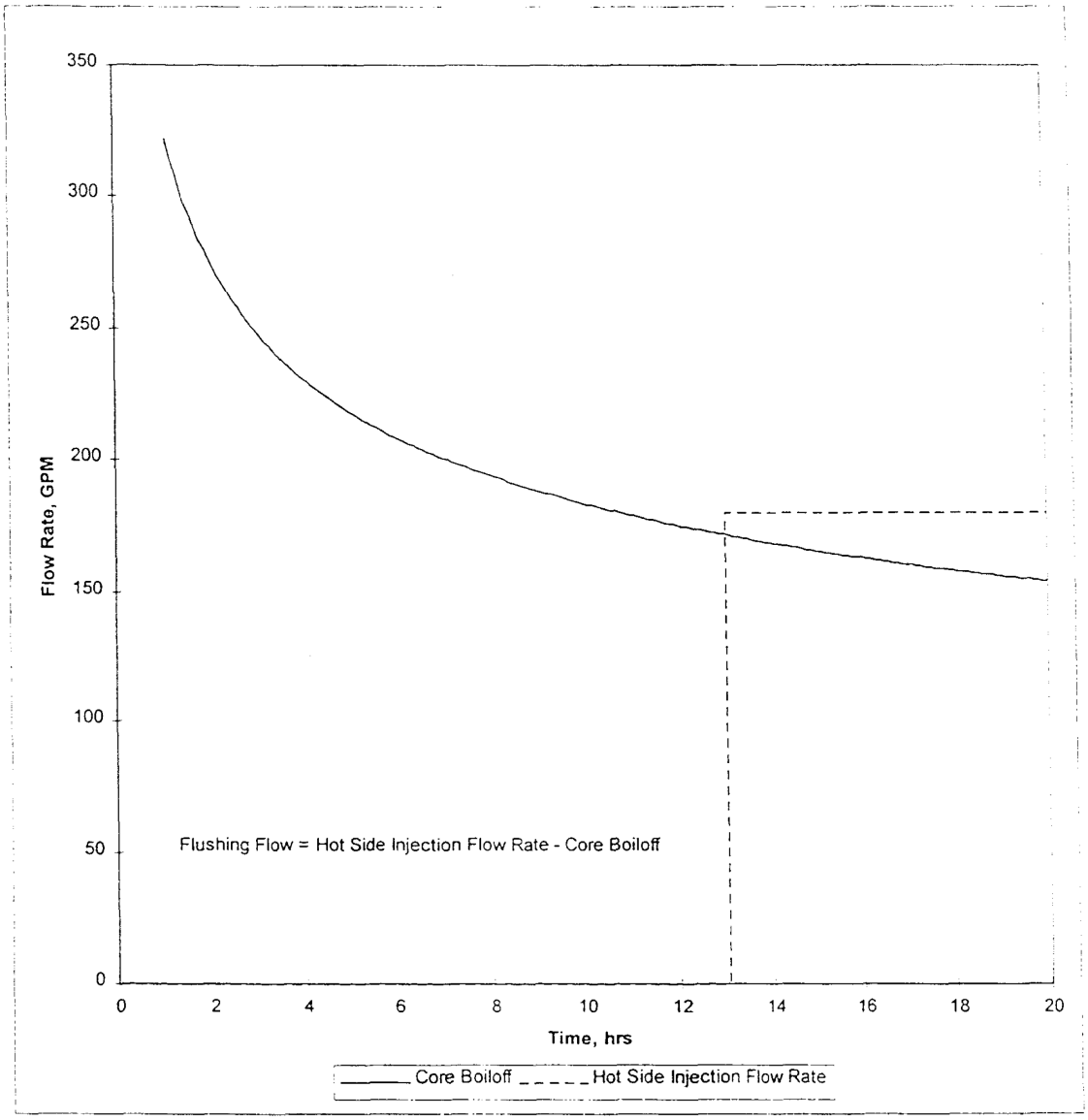


FIGURE 14.6.5.3–4 INNER VESSEL BORIC ACID CONCENTRATION VS. TIME FOR A DOUBLE-ENDED GUILLOTINE COLD LEG BREAK

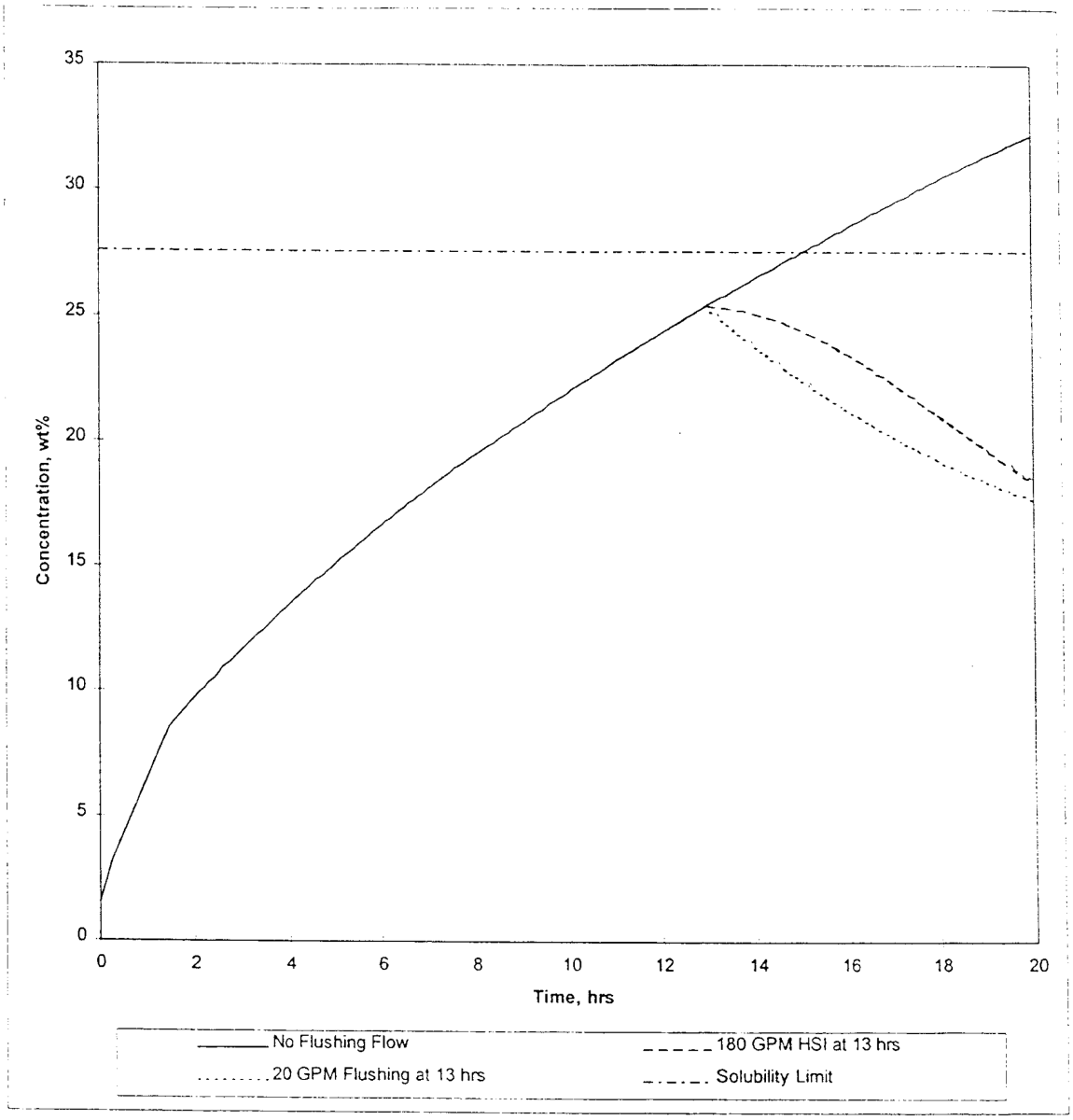


TABLE 14.6.5.4-1 DELETED

TABLE 14.6.5.4-2 DELETED

TABLE 14.6.5.4-3 DELETED

TABLE 14.6.5.4-4 DELETED

TABLE 14.6.5.4-5 DELETED

TABLE 14.6.5.4-6 DELETED

TABLE 14.6.5.4-7 DELETED

FIGURE 14.6.5.4–1 DELETED

FIGURE 14.6.5.4–2 DELETED

FIGURE 14.6.5.4–3 DELETED

FIGURE 14.6.5.4–4 DELETED

FIGURE 14.6.5.4–5 DELETED

FIGURE 14.6.5.4–6 DELETED

FIGURE 14.6.5.4–7 DELETED

FIGURE 14.6.5.4–8 DELETED

FIGURE 14.6.5.4–9 DELETED

FIGURE 14.6.5.4–10 DELETED

FIGURE 14.6.5.4–11 DELETED

FIGURE 14.6.5.4–12 DELETED

FIGURE 14.6.5.4–13 DELETED

FIGURE 14.6.5.4–14 DELETED

FIGURE 14.6.5.4–15 DELETED

FIGURE 14.6.5.4–16 DELETED

FIGURE 14.6.5.4–17 DELETED

FIGURE 14.6.5.4–18 DELETED

FIGURE 14.6.5.4–19 DELETED

14.7 RADIOACTIVE RELEASES FROM A SUBSYSTEM OR COMPONENT

14.7.1 WASTE GAS SYSTEM FAILURE

This section has been moved to Section 11.1.4.4.

14.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE)

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID CONTAINING TANK FAILURES

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed.

14.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENT

14.7.4.1 General

The likelihood of a fuel handling accident is minimized by administrative controls and physical limitations imposed upon fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a qualified supervisor. Also, before any refueling operations begin, verification of complete control element assembly (CEA) insertion is obtained by tripping each CEA individually to obtain indication of assembly drop and disengagement from the drive shaft. Boron concentration in the coolant is raised to the refueling concentration of 1720 ppm boron, or more per Technical Specifications and is verified by chemical analysis. At the required boron concentration, the core will be more than 5 percent subcritical, even with all CEA's withdrawn.

After the vessel head is removed, the CEA drive shafts are removed from their respective assemblies. A load cell is used to indicate that the drive shaft is free of the CEA as the lifting force is applied.

The maximum elevation to which the fuel assemblies can be raised is limited by the use of encoders and limit switches in the fuel handling hoists and manipulators to ensure that the minimum depth of water above the active fuel required for shielding is always present. This constraint applies in fuel handling areas inside containment and in the spent fuel pool area. Supplementing the physical limits on fuel withdrawal, radiation monitors located at the fuel handling areas provide both audible and visual warning of high radiation levels in the event of a low water level in the refueling cavity or fuel pool. Fuel pool structural integrity is assured by designing the pool and the spent fuel storage racks as Seismic Class I structures.

The design of the spent fuel storage racks and handling facilities in both the containment and fuel storage area is such that fuel will always be in a subcritical geometrical array. The spent fuel pool contains a minimum of 2100 ppm of boron, and the refueling pool water contains a minimum of

1720 ppm of boron. Natural convection of the surrounding water provides adequate cooling of fuel during handling and storage. Adequate cooling of the water is provided by forced circulation in the spent fuel pool cooling system. At no time during the transfer from the reactor core to the spent fuel storage rack is the spent fuel removed from the water.

Fuel failure during refueling as a result of inadvertent criticality or overheating is not possible. The possibility of damage to a fuel assembly as a consequence of mishandling is minimized by extensive personnel training, detailed procedures, and equipment design. Equipment design and administrative controls preclude the handling of heavy objects such as shipping casks over the spent fuel storage racks with the exception of the consolidated fuel storage box or any object bounded by the consolidated fuel storage box drop analysis and any single failure proof lift by the spent fuel cask crane in accordance with the guidelines of NUREG-0612. Inadvertent disengagement of a fuel assembly or consolidated fuel storage box from the fuel handling machine is prevented by mechanical interlocks. Consequently, the possibility of dropping either one and damaging of a fuel assembly is remote.

Should a fuel assembly be dropped or otherwise damaged during handling, radioactive release could occur in either the containment or the auxiliary building. If ventilation is available and boundary integrity is set the ventilation exhaust air from both of these areas is monitored before release to the atmosphere (see Section 7.5.6.3). The radiation monitors immediately indicate the increased activity level and alarm. The affected area would then be evacuated.

Forced ventilation is not required while handling irradiated fuel in containment or in the fuel building, nor is containment or fuel handling area boundary integrity required. This allows any penetration to the containment (including the equipment hatch and personnel access door) or fuel handling area boundaries (e.g., including roll-up doors) to be open during fuel movement. Suitable radiological monitoring is recommended per the Millstone Effluent Control Program when boundary integrity is not set to ensure releases to the environment are monitored.

There is no requirement for automatic isolation of containment purge to mitigate a release through the containment purge system during fuel movement.

14.7.4.2 Method of Analysis

For the purpose of defining the upper limit on fuel damage as the result of a fuel handling accident, it is assumed that the fuel assembly or consolidated fuel storage box is dropped during handling by the spent fuel platform crane. Interlocks, procedural and administrative controls make such events unlikely. However, if assemblies are damaged to the extent that a number of fuel rods fail, the accumulated fission gases and iodines in the fuel element gap could be released to the surrounding water. Release of the particulate fission products is considered negligible due to the surrounding water.

The fuel assemblies and consolidated fuel storage box are stored within the spent fuel rack at the bottom of the spent fuel pool. The top of the rack extends above the top of the stored fuel. A dropped fuel assembly or consolidated fuel storage box could not strike more than one fuel assembly in the storage rack. Impact can occur only between the ends of the involved

components, the bottom end fitting of the dropped components impacting against the top end fitting of the stored fuel assembly. The results of an analysis on the energy absorption capability of a fuel assembly indicate that a fuel assembly in the storage rack is capable of absorbing the kinetic energy of the fuel assembly, heavy dummy fuel assembly, or consolidated fuel storage box drop with no fuel rod failures. The worst fuel handling incident that could occur in the spent fuel pool is the dropping of a fuel assembly to the fuel pool floor. It is assumed all of the fuel rods within one fuel assembly will fail as a result of a fuel handling incident within containment or the spent fuel pool area.

All X/Q values have been chosen in the following manner: Site meteorological data has been examined for the years 1974 - 1981 for off site X/Qs and 1997 - 2001 for control room X/Qs. For each release point and dose calculation time period in question, the year with the largest (most conservative) 95% maximum off site X/Q value has been chosen. Control room X/Qs are developed consistent with Regulatory Guide 1.194 except for those from the Millstone Stack, based on Regulatory Guide 1.145.

For each accident, the results indicate that the radiological consequences are within the criteria identified by Regulatory Guide 1.183 and 10 CFR 50.67. The limiting criteria are 6.3 rem TEDE for EAB and LPZ and 5 rem TEDE for the control room.

14.7.4.2.1 Fuel Handling Accident in the Spent Fuel Pool

This accident has been re-analyzed using the methods and assumptions contained in Regulatory Guide 1.183. A complete list of assumptions is provided in Table 14.7.4-1. The results of this analysis are within the limits as defined by 10 CFR 50.67 and within the criteria identified in Regulatory Guide 1.183. This analysis does not require automatic initiation, isolation or re-alignment of main exhaust or AES ventilation system from radiation monitor response, nor does it require fuel handling area integrity.

14.7.4.2.2 Fuel Handling Accident in Containment

This accident has been re-analyzed using the methods and assumptions contained in Regulatory Guide 1.183. A complete list of assumptions is provided in Table 14.7.4-2. The results of this analysis are within the limits as defined by 10 CFR 50.67 and within the criteria identified in Regulatory Guide 1.183. This analysis does not require automatic isolation of purge from radiation monitor response, nor does it require containment integrity because it is assumed that containment penetrations such as the equipment hatch are open.

14.7.4.3 Results of Analysis

14.7.4.3.1 Fuel Handling Accident in the Spent Fuel Pool

TEDE, rem

EAB	1.5E+00
LPZ	2.0 E -01

Control Room	3.1E+00

14.7.4.3.2 Fuel Handling Accident in Containment

TEDE, rem

EAB	1.5E+00
LPZ	2.0 E -01
Control Room	3.1E+00

14.7.4.4 Conclusions

Doses at the exclusion area boundary (EAB), low population zone (LPZ) and the control room are within the requirements of 10 CFR 50.67 and within the guidelines identified in Regulatory Guide 1.183. Therefore, a fuel handling accident in the containment or spent fuel buildings will not present any undue hazard to the health and safety of the public, nor will it compromise control room operations.

14.7.5 SPENT FUEL CASK DROP ACCIDENTS

The spent fuel cask crane is designed with special features to the structure that will ensure a single failure does not result in the loss of the capability of the system to safely retain the load. The upgraded spent fuel cask crane is designed to meet the single failure proof requirements of NUREG-0554 and NUREG-0612. When a spent fuel cask is rigged to the crane and handled in accordance with single failure criteria, a cask drop is not credible and need not be postulated. Therefore, there will be no radiological consequences.

However, the cask drop accident that is postulated to have radiological consequences involves the unlikely scenario where the spent fuel cask is disengaged from the crane in the cask laydown area and tips over into the spent fuel pool damaging impacted fuel assemblies. This postulated cask drop accident is referred to as a cask tip accident to avoid confusion.

14.7.5.1 Spent Fuel Cask Tip Accident

The spent fuel cask is assumed to initially be in the cask laydown area, where it is postulated to tip into the spent fuel pool. All the fuel assemblies in the spent fuel pool that are within the distance L of the center of the spent fuel cask laydown area are assumed to fail, where the distance L is the major dimension of the spent fuel cask. As described below, fuel assemblies in the spent fuel pool within this distance L are assumed to have decayed for the minimum period specified in Technical Specification 3.9.16.

14.7.5.2 Method of Analysis

The spent fuel cask tip accident is based upon the assumptions listed in Table 14.7.5-1.

The analysis considered a full core discharge of 217 individual fuel assemblies decayed for 90 days and 1,376 fuel assemblies consolidated to 688 consolidated fuel canisters or fuel storage boxes. Each consolidated fuel canister or storage box contains the inventory of two individual fuel assemblies. The analysis conservatively included the inventory of 33 more fuel assemblies than actually fit in the potential impact area.

All X/Q values have been chosen in the following manner: Site meteorological data has been examined for years 1974 – 1981 for offsite X/Qs and 1997 – 2001 for control room X/Qs. For each release point and dose calculation time period in question, the year with the largest (i.e., most conservative) 95% maximum offsite X/Q value has been chosen. Control room X/Qs are developed consistent with Regulatory Guide 1.194 except for those from the Millstone Stack, based on Regulatory Guide 1.145.

The analysis considered two scenarios, an unisolated control room and an isolated control room:

- For an unisolated control room, the control room unfiltered ventilation flow consists of 800 cfm normal intake and 200 cfm inleakage for the entire accident.

- For an isolated control room, isolation occurs at 20 seconds, including radiation monitor and damper closure response times:

  0-20 seconds Prior to isolation, the control room unfiltered ventilation flow of 1,000 cfm, which consists of 800 cfm normal intake and 200 cfm inleakage.

  20 seconds – 1 hour 20 seconds During isolation but prior to recirculation, there is only 200 cfm unfiltered inleakage.

  1 hour 20 seconds – 720 hours During isolation and recirculation, 200 cfm unfiltered inleakage and 2,250 cfm filtered recirculation continue for the remainder of the accident.

The unisolated control room provides the limiting dose consequences.

14.7.5.3 Results of Analysis

TEDE, rem

EAB	5.0E-01
LPZ	1.0E-01
Control Room	8.0E-01

The results indicate, that the radiological consequences are within the criteria identified by Regulatory Guide 1.183 and 10 CFR 50.67. The limiting criteria are 6.3 rem TEDE for EAB and LPZ, and 5 rem TEDE for the control room.

14.7.5.4 Conclusions

The spent fuel cask tip accident has been analyzed using the methods and assumptions contained in Regulatory Guide 1.183. The dose consequences at the exclusion area boundary (EAB), low population zone (LPZ), and the control room are within the limits as defined by 10 CFR 50.67 and are within the criteria identified in Regulatory Guide 1.183. Therefore, a cask tip accident in the spent fuel pool will not present any undue hazard to the health and safety of the public, nor will it compromise control room operations.

TABLE 14.7.1-1 DELETED BY FSARCR PKG FSC 07-MP2-006

TABLE 14.7.4-1 ASSUMPTION FOR FUEL HANDLING ACCIDENT IN THE SPENT FUEL POOL

1)	Reactor Core Power Level:	2754 Mwt
2)	Iodine Pool Decontamination Factor:	200
3)	Activity Released from Rods	
	a) Iodines:	10%
	b) Noble Gases (Except Kr-85):	10%
	c) Kr-85:	30%
	c) I-131:	12%
4)	Chemical Form of Iodines Above Pool	
	a) organic:	43%
	b) elemental:	57%
5)	1 Assembly Assumed to Rupture	
6)	Peaking Factor:	1.83
7)	Decay time:	100 hours
8)	Duration of Release:	2 hours
9)	All Activity Bypasses EBFS Filters and is Released at Ground Level Using Worst Case X/Q's:	
10	Ground Level X/Q's (sec/m^3)	
	a) EAB:	3.66 E -04
	b) LPZ:	4.80 E -05
	c) Control Room:	3.00 E -03
11)	Dose Conversion Factors	Federal Guidance Report 11 & 12
12)	Breathing Rate (m^3/sec):	3.5 E -04
13)	Control Room Breathing Rate	3.5 E -04 m^3/sec
14)	Control Room Isolation Time post-accident (includes radiation monitor and damper closure response times)	20 seconds
15)	Control Room Intake Prior to Isolation	800 cfm
16)	Control Room Inleakage During Isolation	200 cfm

TABLE 14.7.4-1 ASSUMPTION FOR FUEL HANDLING ACCIDENT IN THE SPENT FUEL POOL (CONTINUED)

17)	Control Room Emergency Filtered Recirculation Rate (from 1 hour after isolation)	2,250 cfm
18)	Control Room Free Volume	35,656 ft^3
19)	Control Room Filter Efficiency (particulate/elemental/organic)	90 / 90 / 70 % (1)
20)	Dose Conversion Factors	Federal Guidance Reports 11 and 12

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

TABLE 14.7.4-2 ASSUMPTION FOR FUEL HANDLING ACCIDENT IN CONTAINMENT

1	Reactor Core Power Level:			2754 MWt
2	Iodine Pool Decontamination Factor:			200
3	Activity Released from Rods			
		a)	Iodines:	10%
		b)	Noble Gases (except Kr-85):	10%
		c)	Kr-85:	30%
		d)	I-131:	12%
4	Chemical Form of Iodines Above Pool			
		a)	organic:	43%
		b)	elemental:	57%
5	1 Assembly Assumed to Rupture			
6	Peaking Factor:			1.83
7	Decay Time:			100 hours
8	Duration of Release			2 hours
9	All Activity Bypasses EBFS Filters and is Released at Ground Level Using Worst Case X/Q's:			
10	Ground Level X/Q's (sec/m^3)			
		a)	EAB:	3.66 E -04
		b)	LPZ:	4.80 E -05
		c)	Control Room:	3.00 E -03
11	Dose Conversion Factors			Federal Guidance Report 11 & 12
12	Breathing Rate (m^3/sec):			3.5 E -04
13	Control Room Breathing Rate			3.5 E -04 m^3/sec
14	Control Room Isolation Time post-accident (includes radiation monitor and damper closure response times)			20 seconds
15	Control Room Intake Prior to Isolation			800 cfm
16	Control Room Inleakage During Isolation			200 cfm

TABLE 14.7.4-2 ASSUMPTION FOR FUEL HANDLING ACCIDENT IN CONTAINMENT (CONTINUED)

17	Control Room Emergency Filtered Recirculation Rate (from 1 hour after isolation)	2,250 cfm
18	Control Room Free Volume	35,656 ft^3
19	Control Room Filter Efficiency (particulate/elemental/organic)	90 / 90 / 70 % (1)
20	Dose Conversion Factors	Federal Guidance Reports 11 and 12

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

TABLE 14.7.4-3 DELETED BY FSARCR 02-MP2-015

Table deleted by FSARCR 02-MP2-015

TABLE 14.7.5-1 ASSUMPTIONS FOR SPENT FUEL CASK TIP ACCIDENT

PARAMETER	VALUE
Reactor Core Power Level	2,754 Mwt
Iodine Pool Decontamination Factor	200
Fraction of Activity Released from Rods	
Iodines (Except I-131)	10%
I-131	12%
Noble Gases (Except Kr-85)	10%
Kr-85	30%
Chemical Form of Iodines Above Pool	
Organic	43%
Elemental	57%
Ruptured Assemblies & Decay Time	217 assemblies with 90 days decay and 1,376 assemblies with 5 years decay
Peaking Factor	1
Duration of Release	2 hours
Ground Level X/Q's	
EAB	3.66E-04 sec/m^3
LPZ	4.80E-05 sec/m^3
Control Room	3.00E-03 sec/m^3
Dose Conversion Factors	Federal Guidance Reports 11 & 12
Breathing Rates	
EAB & LPZ (0-8 hours)	3.5E-04 m^3/sec
Control Room ((0-720 hours)	3.5E-04 m^3/sec
Control Room Isolation Time post-accident (includes radiation monitor and damper closure response times)	20 seconds
Control Room Unfiltered Normal Intake	800 cfm
Control Room Unfiltered Inleakage	200 cfm

TABLE 14.7.5-1 ASSUMPTIONS FOR SPENT FUEL CASK TIP ACCIDENT (CONTINUED)

PARAMETER	VALUE
Control Room Emergency Filtered Recirculation Rate (from 1 hour after isolation)	2250 cfm
Control Room Free Volume	35,656 ft^3
Control Room Filter Efficiency (Particulate/ Elemental/Organic)	90% / 90% / 70% (1)

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

14.8 SHMILLSTONE 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.

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)

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

14.8.2.1.3 Method of Analysis

A complete MSLB spectrum study has been performed to determine the limiting cases for peak containment pressure. The NRC approved methodology (References 14.8-2 and 14.8-3) associated with the Westinghouse SGN-III computer program was used to determine the mass and energy releases to containment. Using these mass and energy releases, the NRC approved Dominion GOTHIC methodology (Reference 14.8-4) was used to determine the containment pressure-temperature consequences of the MSLB. This methodology includes consideration for the following: (a) inclusion of the steam line and feed line volumes into the overall determination of blowdown volume available; (b) determination of temperature/pressure expansion factor for the SGs and RCS to maximize the volume available for blowdown; (c) increase in feedwater flow to the affected SG due to the increasing pressure imbalance between the affected and intact SG; (d) inclusion of SG shell metal heat transfer as part of the energy release; and lastly, (e) a complete determination of the effects of different component single failures during the accident.

14.8.2.1.4 Major Assumptions

The major assumptions are as follows:

1. Offsite power is assumed to be available for most of the cases. This increases the primary to secondary heat transfer since the reactor coolant pumps (RCPs) are operating. To verify this assumption, loss of offsite power cases were included as part of the single failure analysis.

2. For determination of peak containment pressure, the initial containment pressure/ temperature is conservatively assumed to be at the Technical Specification maximum of 15.7 psia and 120°F.

3. Consistent with the NRC Standard Review Plan (SRP) Section 6.2.1.4, break spectrum studies were used to address moisture carryover.

4. Preferential addition of feedwater to the affected SG is accounted for by conservatively doubling the initial feedwater flowrate as long as the MFW system is operating.

5. Credit is taken for the main steam non-return valves to prevent blowdown of the unaffected SG into the containment.

6. The maximum RCS flow rate was conservatively assumed to maximize the heat transfer from the primary to secondary side.

7. Cases initiated from 0% power assumed auxiliary feedwater (AFW) is the sole source of steam generator inventory control and AFW flow to the affected steam generator is maximized from the beginning (time = 0) of the analysis. All other cases initiate AFW at a conservative minimum time of 180 seconds.

8. Reactor coolant pump heat was included.

9. All actuation signals are redundant and safety grade. In some cases credit is taken for actuation of nonsafety grade components initiated by the safety grade signals.

10. Relative humidity is assumed at 25 percent.

11. A 1” steam line with a 0.75" diameter minimum flow area was previously located between the two steam generators and remained unisolated during the events. This caused the intact steam generator to continue to blowdown even after the MSIS. This line has been removed from the plant but remains accounted for (conservatively) in the MSLB containment pressurization analysis.

12. A cavitating venturi installed in each AFW discharge line will limit AFW flow to a steam generator to 550 gpm.

13. Operator action to isolate the AFW is assumed to occur no greater than 30 minutes following MSIS.

14. The feedwater flow rate for the cases that initiate with the main feedwater system operating conservatively account the following:

    - Feedwater flow increases as the affected steam generator pressure decreases.

    - Upon receipt of a MSIS, the feedwater pumps are assumed to coast down at a rate based on plant operating experience.

    - If the ruptured steam generator pressure decreases to the discharge pressure of the still running condensate and heater drain pumps, flow will again begin to increase.

- Flow is eventually stopped after the containment high pressure signal actuates the MSIS and the main feedwater is isolated.

14.8.2.1.5 Initial Conditions and Input Data

Initial conditions and input data are given in Tables 14.8.2-1, 14.8.2-2 and 14.8.2-5. Table 14.8.2-3 gives an accounting of the amounts of steel assumed to be inside of containment. Since this material acts as a heat sink to reduce containment temperature and pressure, minimum amounts are used.

14.8.2.1.6 Results

In order to determine the limiting conditions, four different spectrum studies were performed. These are as follows:

1. Power level and break size.

2. Feed system single failures.

3. Containment heat removal systems single failures.

4. Spectrum study for peak containment temperature.

14.8.2.1.6.1 Power Level and Break Size

A comprehensive sensitivity study was performed to determine the limiting break size for each power level. A sensitivity study was needed because of the interaction of power level with SG inventory and moisture carryover. The limiting break size at a given power level is the largest break size that would result in a pure steam blowdown, since a pure steam blowdown results in the greatest amount of energy being transferred to the containment atmosphere in a short period of time. The limiting results for each power level show that a maximum break size of 3.51 ft^2 is limiting for 25 percent, 50 percent, 75 percent, and 100 percent power. At 0 percent power the limiting size break is 1.89 ft^2.

14.8.2.1.6.2 Feed System Single Failures

A comprehensive feedwater system isolation single failure study was performed. For each single failure, a range of steady state initial power levels was analyzed, using the insights from the Power Level/Break Size sensitivity study.

1. Feed Pump Failure to Trip - The failure of a feed pump to trip on MSIS results in additional feed water being pumped preferentially into the affected SG until the Feedwater Regulating Valves (FRV) and isolation valves shut. For 25 percent and 50 percent power levels, only one feedwater pump was assumed to be running

when the accident commences. This event was not applicable to 0% cases (see assumption 7 of Section 14.8.2.1.4).

2. Inadvertent Initiation of AFW Feedwater to Affected Steam Generator - Maximum AFW flow was assumed to be inadvertently initiated at the start of the event for cases initiated from greater than 0% power.

3. Feedwater Bypass Valve Fails Open - This failure is only credible when the FW bypass valve is initially open (cases initiated from 25% power and below). The failed open feedwater bypass results in additional feedwater being pumped preferentially into the affected SG until the FW pump discharge valves shut. In addition, even with the feed pump discharge valves shut, flashing in the feedwater lines continues to add energy into the affected steam generator. This effect has been taken into account.

4. Failure of Vital Bus Cabinet VA-10 or VA-20 - This failure could prevent closure of the FRVs and results in the loss of one train of the Containment Heat Removal Systems. Feedwater addition to the affected SG will continue until closure of the main feed pump discharge valves.

14.8.2.1.6.3 Containment Heat Removal Systems Single Failures

A comprehensive containment heat removal systems single failure study was performed. For each single failure, a range of steady state initial power levels was analyzed, using the insights from the Power Level/Break Size sensitivity study.

1. Failure of Two CAR fans to start - This failure is bounded by Section 14.8.2.1.6.2, item 4 described above.

2. Failure of one spray train to start - This failure is bounded by Section 14.8.2.1.6.2, item 4 described above.

3. Failure of the Vital Bus Transfer Mechanism - This failure results in a loss of the normal off-site power supply for the vital buses. Thus initiation of the containment sprays and CAR fans is delayed until the EDGs are powering the vital buses and auto sequencing has occurred. Since the FRVs have a backup DC power source, they are unaffected by this failure and will isolate the affected SG. The RCPs and certain other nonvital loads are also unaffected by this failure, which contributes to the severity of this accident by providing more rapid heat transfer from the primary to the affected SG.

4. Loss of Offsite Power with a Loss of One EDG - A loss of offsite power will result in loss of power to the RCPs, the condensate pumps and feedwater heater drain pumps. While only one train of containment heat removal systems is available, the loss of power to these pumps results in a greatly degraded heat transfer in the affected SG and less limiting results. Feedwater isolation will be unaffected since the FRVs are powered by DC backup power supplies.

5. Loss of Offsite Power with a Loss of VA-10/20 - This case is similar to item 4 (the preceding paragraph), with the exception of the effect on feedwater isolation to the affected SG. With this failure, there is the potential for failure of the FRV and the other isolation valves to close. However, with the loss of the condensate and feedwater heater drain pumps, feedwater addition to the affected SG is terminated. The effect of continued energy addition to the affected SG from flashing in the feedwater lines has been taken into account.

14.8.2.1.6.4 Maximum Containment Temperature Spectrum Study

The limiting peak pressure cases were re-run with the following modified assumptions to maximize resultant containment temperature.

1. The initial containment pressure was reduced to 14.27 psia. This results in the maximum delay in containment spray actuation.

2. The relative humidity was increased to 100 percent.

3. The MSLB mass and energy releases model the Steam Generator steam super heating as it passes the uncovered portion of the Steam Generator tubes before exiting the break to address IE Information Notice 84-90. The containment wall re-evaporation is modeled using the GOTHIC built-in models for calculating the vaporization of the liquid in containment as described in Reference 14.8-4.

14.8.2.1.7 Conclusions

The results of a MSLB initiated from 102 percent reactor power with coincident loss of offsite power and the 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 325.1°F. With the loss of offsite power and this single failure, the condensate and feedwater heater drain pumps are lost, and pumped feedwater addition to the affected steam generator is quickly terminated. However, since the FRV and the other feedwater isolation valves fail to close on the MSIS, a significant volume of feedwater system remains connected to the affected steam generator. As the affected steam generator depressurizes, feedwater in this system flashes and adds significant mass and energy to the affected steam generator. The portion of the feedwater that flashes to steam is conservatively assumed to be directly added to the containment atmosphere separate from the mass and energy releases from the steam generator. The portion of the feedwater liquid that reaches the affected steam generator as it depressurizes increases the affected steam generator liquid mass, which increases the steam generator mass and energy

releases to containment. 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.

The results of MSLB initiated from 102 percent reactor power with offsite power available and the failure of Vital Bus Cabinet VA-10 or VA-20produces the limiting containment peak atmospheric temperature of 360.9°F.

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

Twenty four separate cases of Loss of Coolant Accidents (LOCAs) were analyzed with variations for break locations, break sizes, single failures and availability of offsite power. Break locations analyzed are the reactor coolant pump suction leg, pump discharge leg and hot leg. Break sizes include double-ended guillotine and slot breaks (9.82 sq. ft. area for the reactor coolant pump suction and discharge leg breaks, 19.24 sq. ft. area for a hot leg break) and smaller break sizes for hot leg breaks (10 sq. ft. and 2 sq. ft.) and reactor coolant pump suction and discharge leg breaks (5 sq. ft. and 2 sq. ft.). 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.

14.8.2.2.2 Method of Analysis

The NRC approved, Westinghouse containment analysis methodology was used for the development of the short term mass and energy releases following a LOCA (Reference 14.8-3). Mass and energy input are provided through the End-of-Blowdown (EOB) from CEFLASH-4A, and from the EOB to End-of-Post Reflood (EOPR) from FLOOD3. The long term boil-off phase mass and energy input was calculated using the Dominion GOTHIC code (Reference 14.8-4). The containment pressure and temperature response for the entire LOCA transient was calculated using the Dominion GOTHIC computer code.

14.8.2.2.3 Input and Assumptions

a. Containment input data, such as: heat sink area, spray flow rate, CAR fan cooler heat removal rate, spray water temperature, containment volume and initial containment temperature are the same as, or more conservative than, that used for the MSLB in Section 14.8.2.1.

b. Initial containment pressure is 15.7 psia for the peak pressure case and 14.27 psia for the peak temperature case.

c. Initial containment humidity is assumed to be 25 percent as in the MSLB for the peak pressure case and 100 percent for the peak temperature case.

d. The minimum usable Refueling Water Storage Tank (RWST) volume assumed for calculation of the time of Sump Recirculation Actuation Signal (SRAS) is 370,000 gallons.

e. Both the HPSI and the LPSI pumps operate prior to SRAS. Following SRAS, the LPSI are automatically stopped.

f. The Reactor Building Closed Cooling Water (RBCCW) is modeled with assumed flows before and after SRAS. The RBCCW is cooled by Service Water at 80°F.

g. The heat removal from the CAR fan cooler is modeled in the Dominion GOTHIC Code using a fan cooler model benchmarked to the post-LOCA specification data. The specification identifies that one CAR fan is capable of removing a minimum of 80 million BTU/hr based on a containment air inlet temperature of 289°F and a fan flowrate of 34,800 cfm, along with a cooling water inlet temperature of 130°F and a flowrate of 2000 gpm.

h. A minimum spray flow of 1300 gpm is credited prior to SRAS, and 1350 gpm following SRAS.

14.8.2.2.4 Results

The limiting LOCA with respect to maximum containment pressure was determined to be the 5 square foot discharge leg break with the LOP, the failure of two CAR fans and one spray train, and minimum ECCS. The maximum calculated containment pressure for this case is 52.5 psig. This pressure is rounded up to 53 psig to establish the P_a value stated in the containment leakage rate testing program Technical Specification. The limiting LOCA with respect to maximum containment temperature was determined to be the 10 square foot hot leg break with the LOP, the failure of two CAR fans and one spray train and minimum ECCS. The maximum calculated containment temperature for this case is 279.2°F. The maximum containment pressure and temperature of these limiting LOCAs are bounded by the MSLB results provided in Section 14.8.2.1.

14.8.2.2.5 Conclusion

The maximum containment pressure and temperature of the LOCA are less than the containment design pressure and temperature of 54 psig and 289°F.

14.8.3 DELETED

14.8.4 RADIOLOGICAL CONSEQUENCES OF THE DESIGN BASIS ACCIDENT

14.8.4.1 General

A LOCA would increase the pressure in the containment resulting in a containment isolation and initiation of the ECCS and containment spray systems. A SIAS signal automatically starts the Enclosure Building Filtration System (EBFS) which maintains a negative pressure within the enclosure building during accident conditions. The nuclide inventory assumed to be initially available for release is consistent with the requirements of Regulatory Guide 1.183 (Reference 14.8-5). A SIAS also isolates the control room by closing the fresh air dampers within 20 seconds. Within 1 hour after control room isolation, the control room emergency ventilation (CREV) is properly aligned. CREV recirculates air within the control room through a charcoal filter at 2,500 cfm (±10%) to remove iodines from the control room envelope.

The radiological consequences of a Design Basis LOCA at Millstone 2 were previously analyzed for a low and high wind speed condition based on guidance from Regulatory Guide 1.4 (Reference 14.8-6) and SRP 6.5.3 (Reference 14.8-7). The low wind speed case was found to bound the high wind speed case. Therefore, the low wind speed case is the design basis for a LOCA and the high wind speed case is no longer analyzed.

14.8.4.2 Release Pathways

The release pathways to the environment subsequent to a LOCA are leakages from containment and the enclosure building, which are collected and processed by EBFS and leakages from containment and the RWST which bypass EBFS.

Containment Leakage

The containment is assumed to leak at the design leak rate for 24 hours after the accident. After 24 hours, since the pressure has been decreased significantly, Regulatory Guide 1.183 allows for the leak rate to be reduced to one-half the design leakage rate.

All containment leakage for the first 110 seconds is assumed to bypass EBFS and is released directly out the MP-2 containment. This is due to the fact that it takes 110 seconds for EBFS to achieve the required negative pressure in the enclosure building, thereby ensuring that leakage will be into the enclosure building rather than out.

EBFS collects most of the containment leakage and processes it through HEPA and charcoal filters and releases it up the Millstone stack. All containment leakage is collected and filtered by EBFS except for the small amount that is assumed to bypass EBFS and is released directly out the MP-2 containment.

Credit is taken for iodine removal due to containment sprays. The sprays are effective after 75 seconds post-LOCA. The effectiveness of the sprays in removing elemental iodine ends at 3.03 hours and in removing particulate iodine at 3.23 hours. Credit is taken for iodine retention in the containment sump based on post-LOCA sump pH ≥ 7.0 as discussed in Section 6.2.4.1.

ESF System Leakage Pathway

Post-accident radioactive releases from the ESF system are derived from fluid leakages assumed during recirculation of the containment sump water through systems located outside containment. The nuclide inventory assumed to be available for release from this pathway consists of 40% of the core iodines. The quantity of leakage is based on the assumption that the ESF equipment leaks at twice the maximum expected operational leak rate and that 10 percent of the iodine nuclides contained in the leakage fluid become airborne in the enclosure building. The nuclides which become airborne are collected and released to the environment through EBFS to the Millstone stack.

RWST Backleakage Pathway

Post-accident radioactive releases from the ECCS system are a result of ECCS subsystems containing recirculated sump fluid backleaking to the RWST. The backflow rate to the RWST, as a result of isolation valve leakage, is predefined and time dependent. Due to this time dependency, the contaminated sump fluid from backleakage does not enter into the RWST until 6.45 hours post-LOCA. Since the RWST is vented to atmosphere, the release is a result of the breathing rate of the RWST due to solar heating.

14.8.4.3 Control Room Habitability

The radiation design objective of the control room is to limit the dose to personnel inside the control room to 5 rem TEDE, during a DBA. The potential radiation dose to a control room operator is evaluated for the LOCA. The analysis is based on the assumptions and meteorological parameters (X/Q values) given in a Tables 14.8.4-3 and 14.8.4-4.

The control room is designed to be continuously occupied for the duration of the accident, 30 days. Two basic sources of radiation have been evaluated: leakage of airborne activity into the control room from sources described in Section 14.8.2 and direct dose from sources outside the control room. The control room shielding serves to protect the operators from direct radiation due to the passing cloud of radioactive effluent assumed to have leaked from containment, enclosure building and the RWST. The control room walls also provide shielding protection for radiation emanating from the CREV filters and containment shine.

A SIAS from Millstone 2 initiates control room isolation within 20 seconds by securing the fresh air intake dampers. Within 1 hour, CREV is in operation recirculating air in the control room envelope through charcoal filters to remove radioactive iodines from the atmosphere. The calculated TEDE dose from a Millstone 2 LOCA is presented in Table 14.8.4-5 and is below the General Design Criteria 19 and 10 CFR 50.67 limits.

The calculated TEDE dose from a Millstone 3 LOCA to the Millstone 2 control room is below the General Design Criteria 19 limits and bounded by the dose consequences from the Millstone 2 LOCA. No credit has been taken for control room isolation or CREV operation.

14.8.4.4 Offsite Dose Computation

The radiological offsite dose consequences resulting from a postulated Millstone 2 LOCA are reported in Table 14.8.4-2. The offsite dose analysis shows that the consequences to the EAB (highest 2 hour) and LPZ (0-30 day) are less than the limit of 25 rem TEDE as specified in 10 CFR 50.67. The assumptions used to perform the radiological analysis are summarized in Table 14.8.4-1.

14.8.4.5 Conclusion

Analysis shows that the offsite and control room radiological consequences are within 10 CFR 50.67 criteria.

14.8.5 REFERENCES

14.8-1 Deleted.

14.8-2 Preliminary Safety Analysis Report (PSAR) to CESSAR, Appendix 6B, “Description of the SGN-111 Digital Computer Code Used In Developing Main Steam Line Break Mass/Energy Release Data For Containment Analysis.”

14.8-3 NRC Safety Evaluation Report - Standard Reference System CESSAR System 80, Combustion Engineering, Inc., December 1975.

14.8-4 Dominion Topical Report DOM-NAF-3, Revision 0.0-P-A, “GOTHIC Methodology For Analyzing the Response to Postulated Pipe Ruptures Inside Containment,” September 2006.

14.8-5 Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

14.8-6 Regulatory Guide 1.4, Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors Rev. 2, June 1974.

14.8-7 SRP 6.5.3, Fission Product Control Systems and Structures.

TABLE 14.8.2-1 CONTAINMENT DESIGN PARAMETERS

Internal dimensions (feet)	
Cylinder wall diameter	130.0
Cylinder wall height	132.4
Curved dome height	43.3
Net free internal volume	1,899,000 cubic feet

TABLE 14.8.2-2 INITIAL CONDITIONS FOR PRESSURE ANALYSES

Reactor Coolant System	
Core thermal power level (MWt/% of rated power)	2754/102
Reactor coolant Pump heat (MWt)	17.1
Coolant pressure (psig)	2300
Inlet coolant temperature (°F)	551.25
Internal coolant volume (cubic feet) (excludes the pressurizer)	10,104.4

Containment System	
Pressure (psia)	15.7
Relative humidity (%)	25
Inside temperature (°F)	120
Outside temperature (°F)	N/A *
Service Water Inlet temperature (°F)	80
Refueling Water Storage Tank (RWST) water temperature (°F)	100
Safety Injection Tank (SIT) water temperature (°F)	120

* No heat transfer credited from containment structure to outside environment.

TABLE 14.8.2-3 MINIMUM CONTAINMENT HEAT SINK DATA

Heat Sink	Exposed Surface Area (sq ft)
1. Containment Cylinder and Dome	
Cylinder	52,800
Dome	19,070
	71,870
2. Unlined Concrete	
Steam Generator Compartment Walls	26,114
Miscellaneous Slabs	4,488
Elevator Foundation	643
Pressurizer Wall and Roof	2,159
Refueling Canal (Outside)	10,043
Steam Generator Pedestals	2,860
Steam Generator Buttresses	3,840
Fuel Canal Buttresses	3,273
	53,420
3. Reactor Support	
Concrete (3 inches, exposed on one side to the containment atmosphere and on the other to a 150°F source to account for the higher reactor cavity operating temperature)	3,486
4. Galvanized Steel	116,497
5. Painted Steel Less than 0.12 in. Thick	5,605
6. Painted Steel 0.12 to 0.16 in. Thick	16,863
7. Painted Steel 0.16 to 0.24 in. Thick	36,713
8. Painted Steel 0.24 to 0.3 in. Thick	10,289
9. Painted Steel 0.3 to 0.4 in. Thick	19,366
10. Painted Steel 0.4 to 0.5 in. Thick	4,525

TABLE 14.8.2-3 MINIMUM CONTAINMENT HEAT SINK DATA (CONTINUED)

Heat Sink	Exposed Surface Area (sq ft)
11. Painted Steel 0.5 to 0.625 in. Thick	5,338
12. Painted Steel 0.625 to 0.75 in. Thick	2,243
13. Painted Steel 0.75 to 1.0 in. Thick	2,862
14. Painted Steel 1.0 to 1.5 in. Thick	4,322
15. Painted Steel Greater than 1.5 in. Thick	1,031
16. Unpainted Stainless Steel	18,464
17. Containment Floor	8,102
18. Safety Injection Tanks	2,541

TABLE 14.8.2-4 SEQUENCE OF EVENTS, MP2-MSLB: LOSS OF OFFSITE POWER AND THE FAILURE OF VITAL BUS VA-10 OR VA-20 FROM 102% POWER

TIME (seconds)	EVENT	SETPOINT/VALUE
0.0	MSLB occurs from 102% power, break size is 3.51 ft^2.	
0.0	Loss of offsite power.	
1.29	Low RCS flow reactor trip condition reached.	89.7% of initial RCS flow
1.94	Low RCS flow reactor trip signal generated.	0.65 second delay
2.50	Containment High Pressure Signal (CHPS) condition is reached. (Due to the assumed loss of offsite power and failure of VA-10 or VA-20, the affected steam generator FRV and main feedwater isolation valves do not close.)	5.83 psig with uncertainty
3.40	High Containment Pressure MSIS generated.	0.9 second delay
6.15	Containment High-High Pressure Signal (CHHPS) condition reached.	11.08 psig with uncertainty
28.5	Containment cooling fans energize. Time based on CHPS + 26 second delay.	
37.1	Peak Containment Pressure reached.	326.5°F
74.76	Containment spray flow commences. Time based on CHHPS + 68.6 seconds for pump start, valve stroke time, and header fill time.	See Note 1
180	Maximum AFW Flow to the Affected Steam Generator.	550 gpm, 100°F
552.1	Peak Containment Pressure reached.	53.8 psig
1000	Simulation ended.	

(1) The minimum containment spray flows used in this analysis are provided in Table 14.8.2-5.

TABLE 14.8.2-5 ENGINEERED SAFETY FEATURES PERFORMANCE FOR MSLB CONTAINMENT ANALYSIS

Safety Features	Value	Notes
1. Containment spray		
- water temperature	100°F	
- CSAS setpoint	11.08 psig	TS value plus uncertainty
- minimum flow rate	1361 gpm at 54 psig 1375 gpm at 51 psig 1394 gpm at 47 psig 1428 gpm at 40 psig	Values based on a minimum RWST level of 30 feet above tank bottom.
- delay time with normal AC power available	49 seconds	Includes 33 seconds for header fill time and 16 seconds for signal generation, pump start and valve stroke.
- delay time with the loss of normal AC power	68.6 seconds	Includes 33 seconds for header fill time and 35.6 seconds for signal generation, pump start and valve stroke.
2. Containment Air Recirculation (CAR) Cooling Fans		
- number of fans	4	2 for certain single failure cases
- activation setpoint	5.83 psig	TS value plus uncertainty
- delay time with normal AC power available	15 seconds	
- delay time with loss of normal AC power	26 seconds	
- heat removal capability of one CAR Fan	80 million BTU/hr based on air inlet temperature of 289°F and a fan flow rate of 34,800 cfm, along with a cooling water inlet temperature of 130°F and flow rate of 2000 gpm.	A GOTHIC Fan Cooler Model is used. This model is benchmarked to the cited specification data.
3. Safety Injection	None Assumed.	

FIGURE 14.8.2–1 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - CONTAINMENT PRESSURE VS. TIME

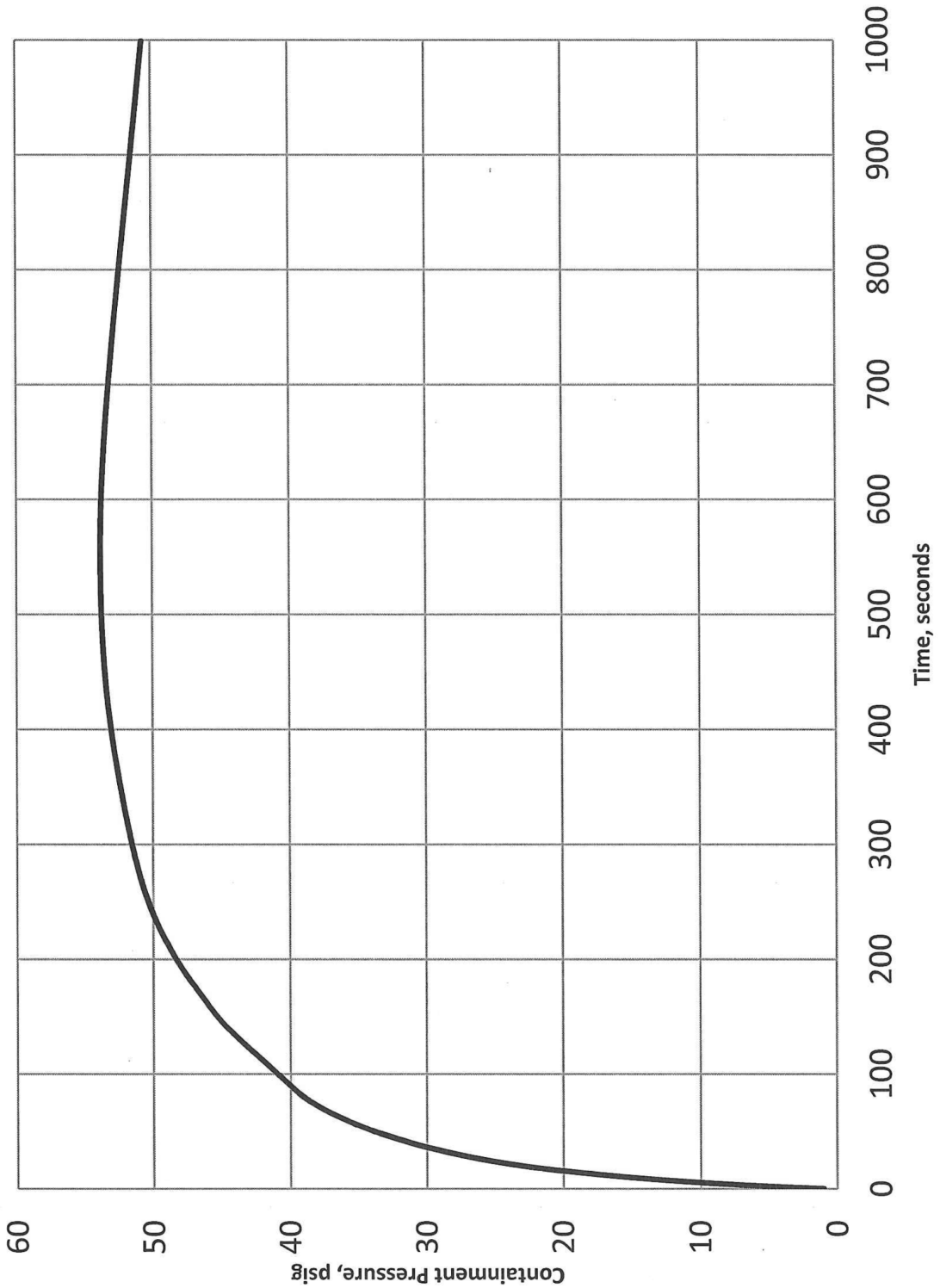


FIGURE 14.8.2–2 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - CONTAINMENT TEMPERATURE VS. TIME

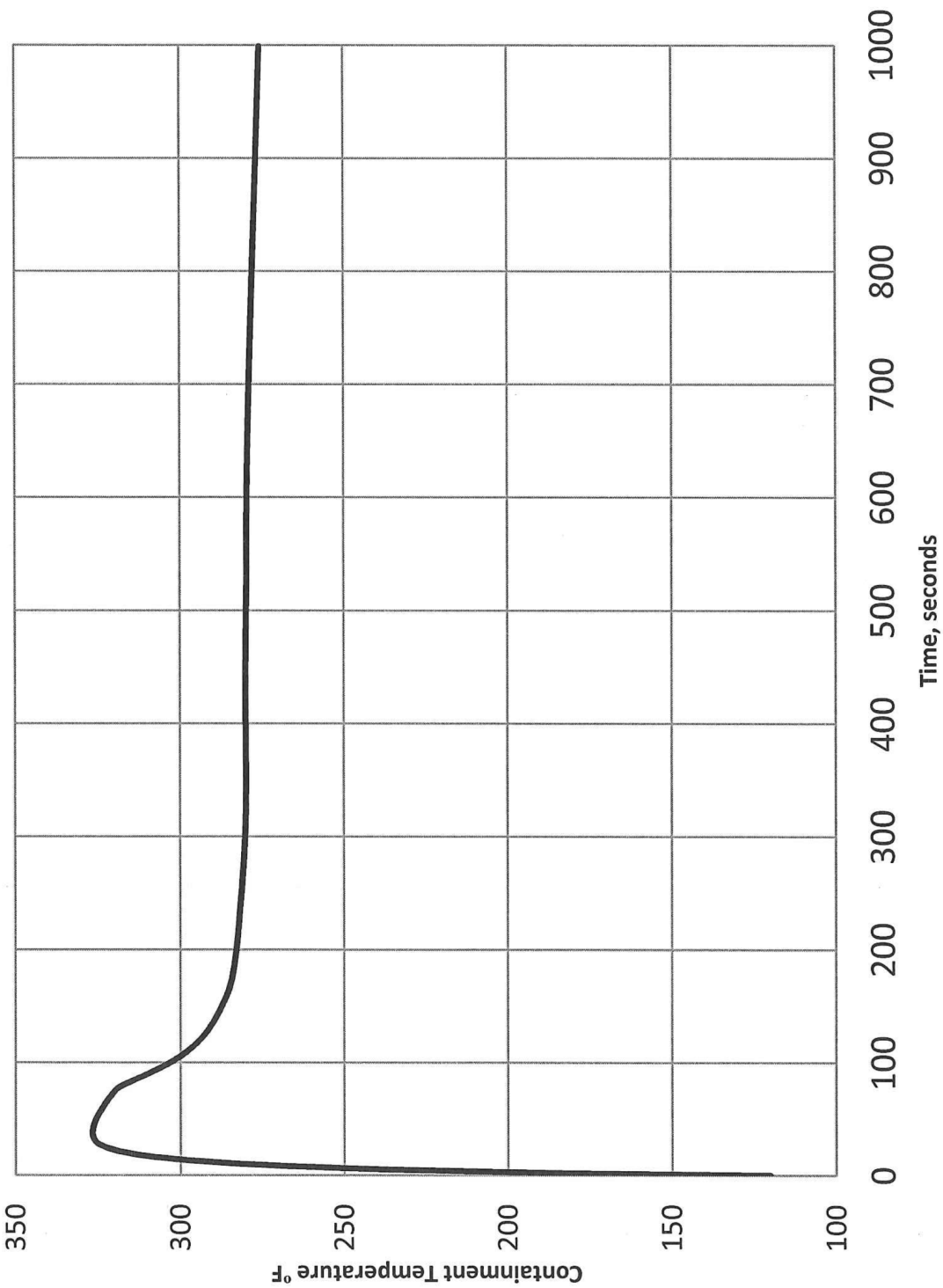


FIGURE 14.8.2–3 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - MASS FLOW RATE VS. TIME

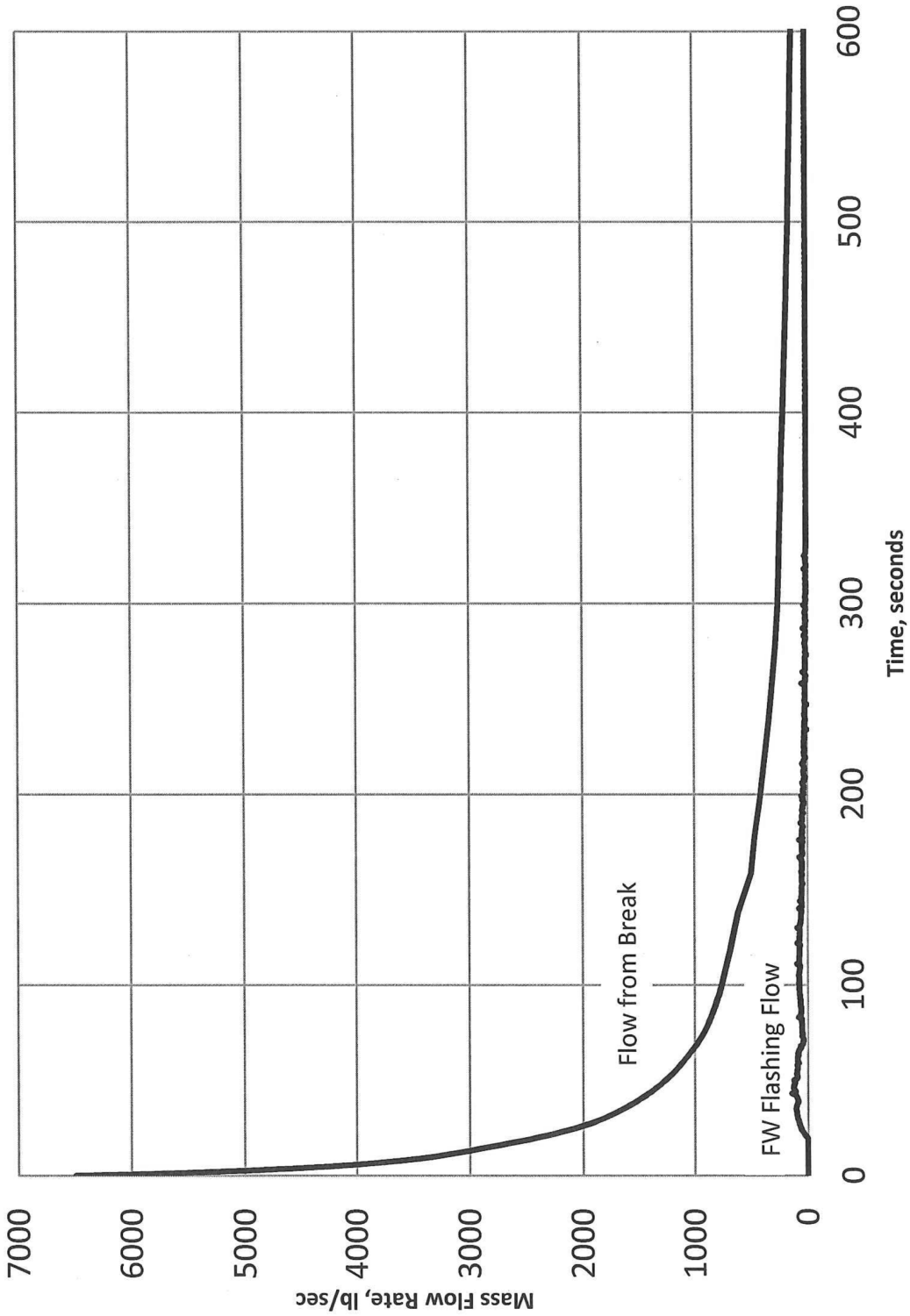
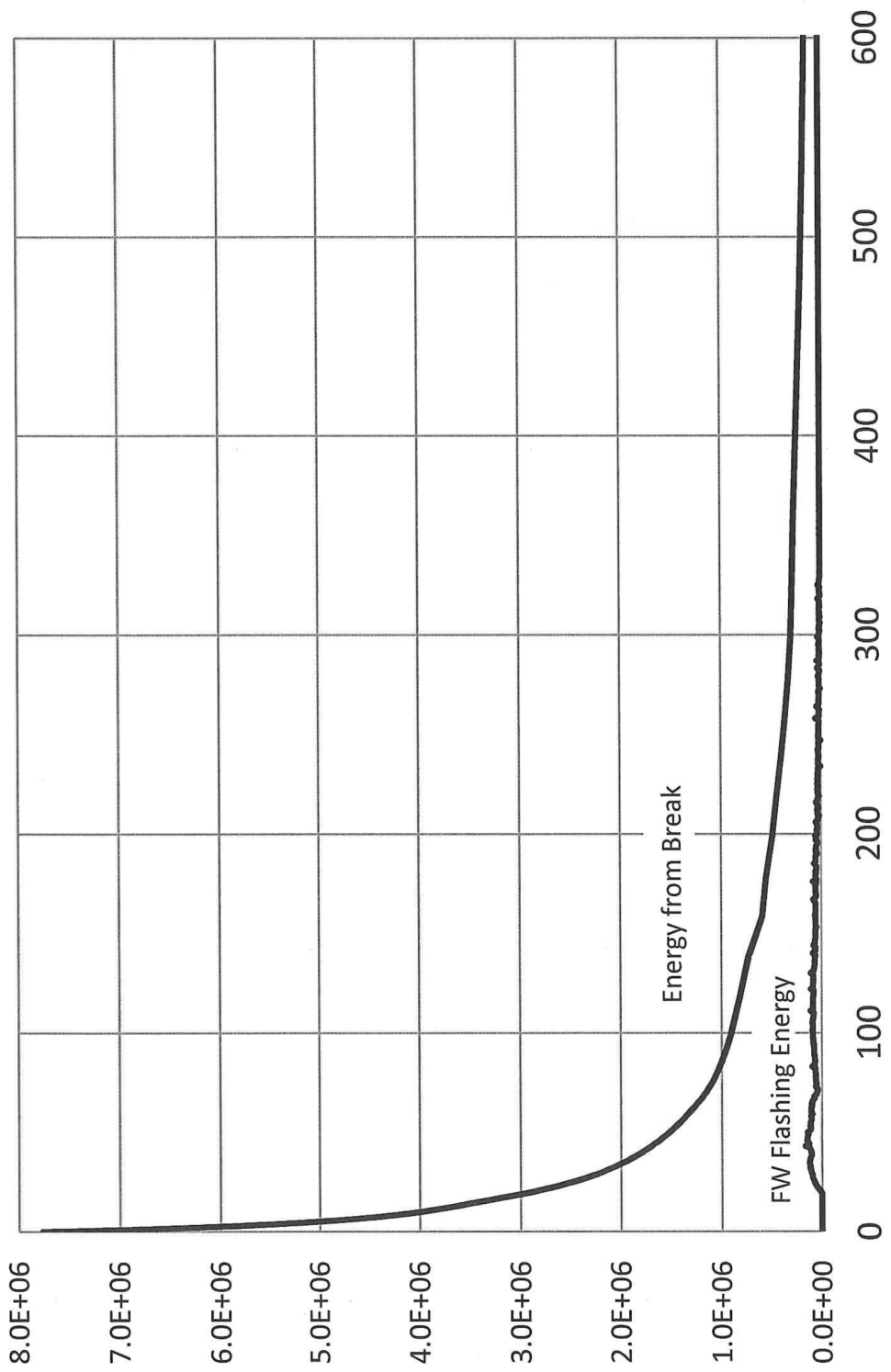


FIGURE 14.8.2–4 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - ENERGY RELEASE RATE VS. TIME



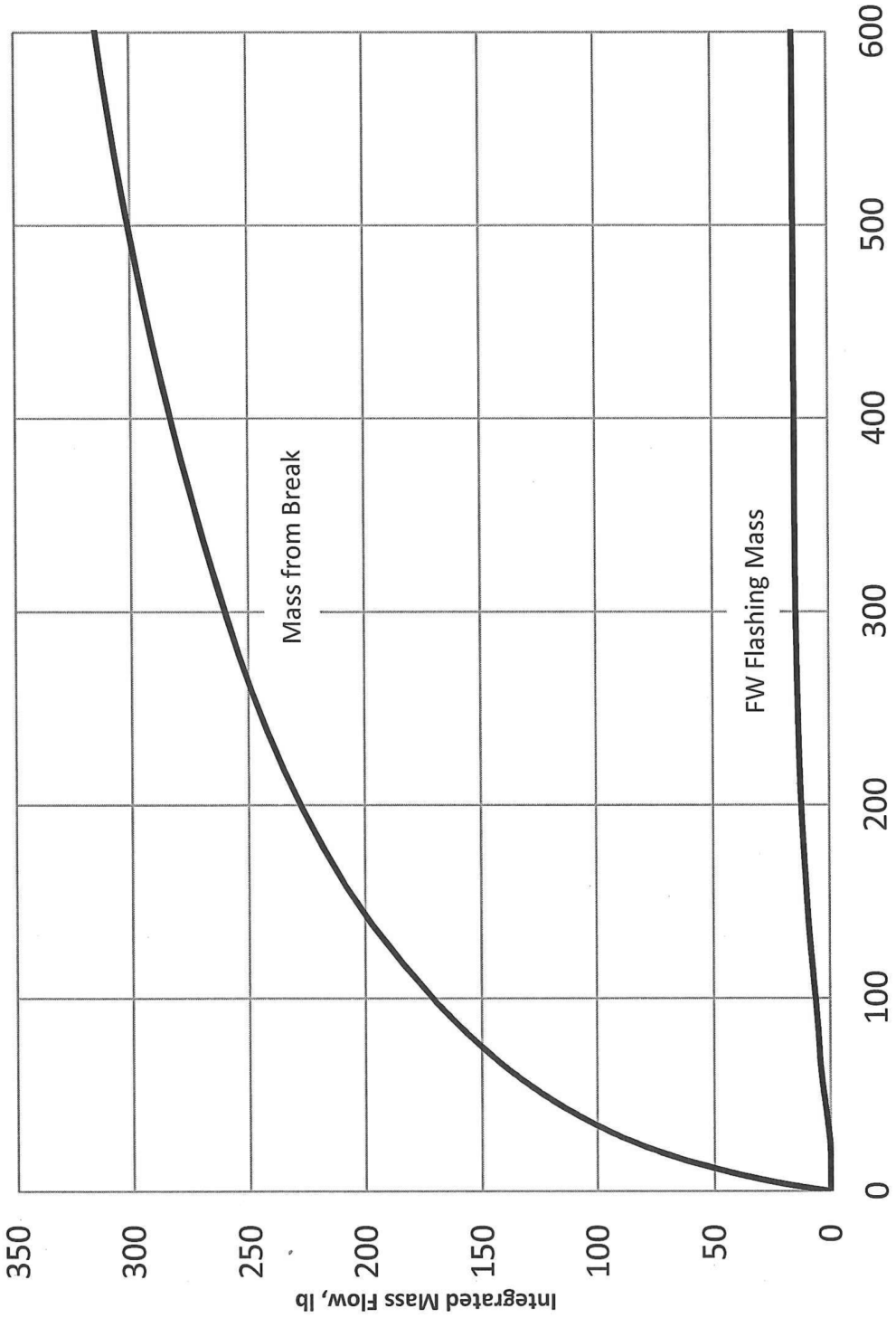


FIGURE 14.8.2–5 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - INTEGRATED MASS FLOW VS. TIME

FIGURE 14.8.2–6 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - INTEGRATED ENERGY RELEASE VS. TIME

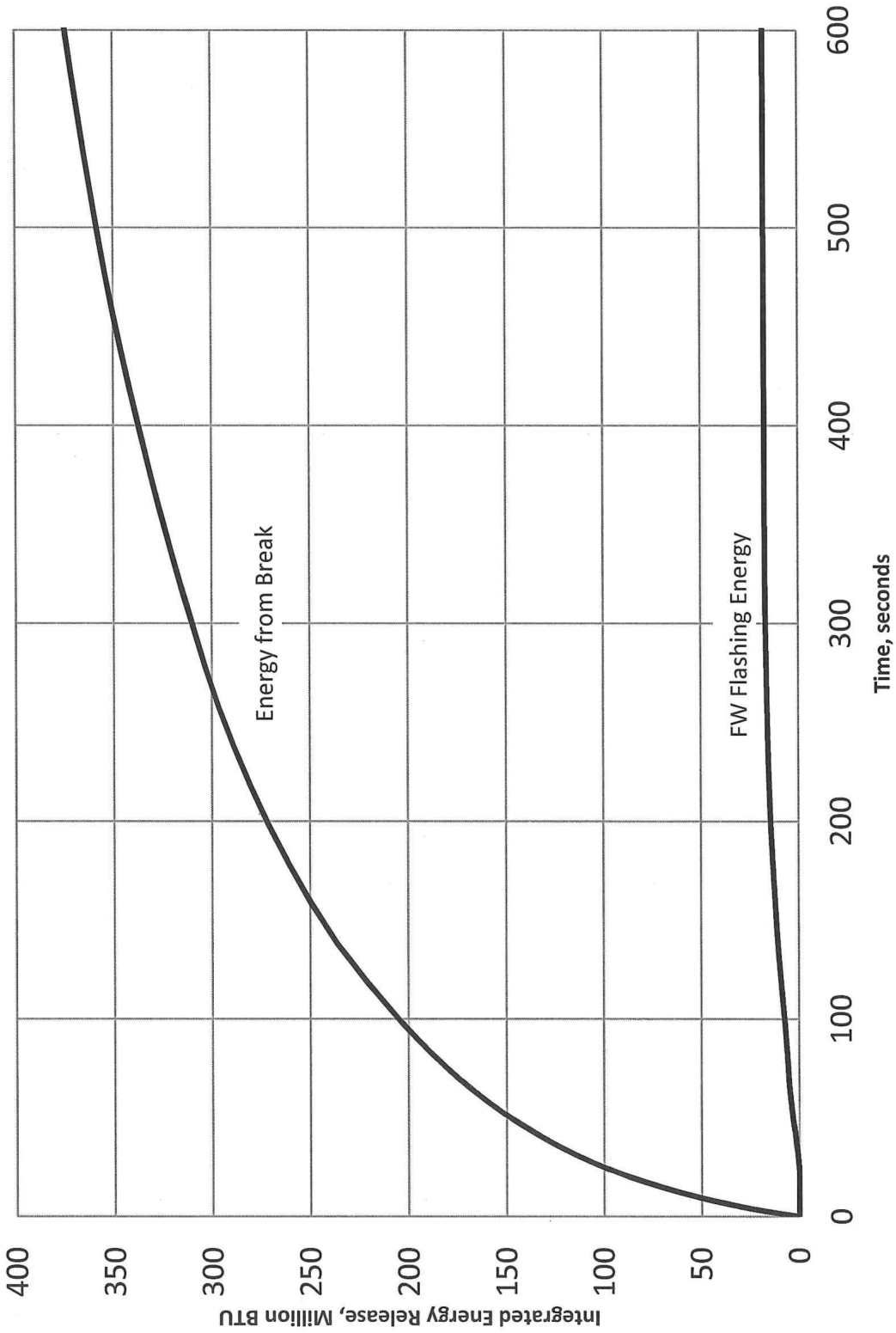


FIGURE 14.8.2–7 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - AFFECTED STEAM GENERATOR PRESSURE VS. TIME

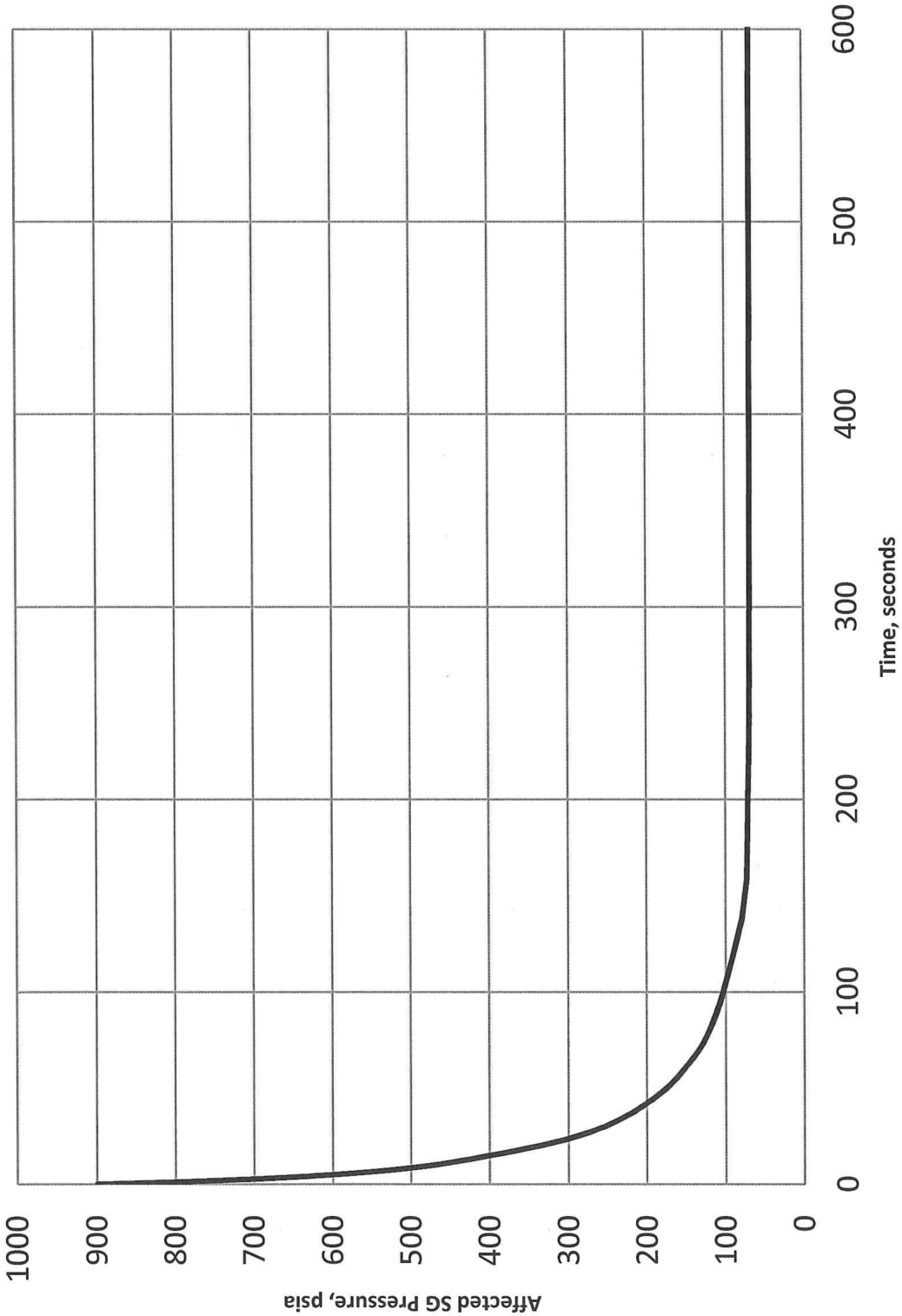


FIGURE 14.8.2–8 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - UNAFFECTED STEAM GENERATOR PRESSURE VS. TIME

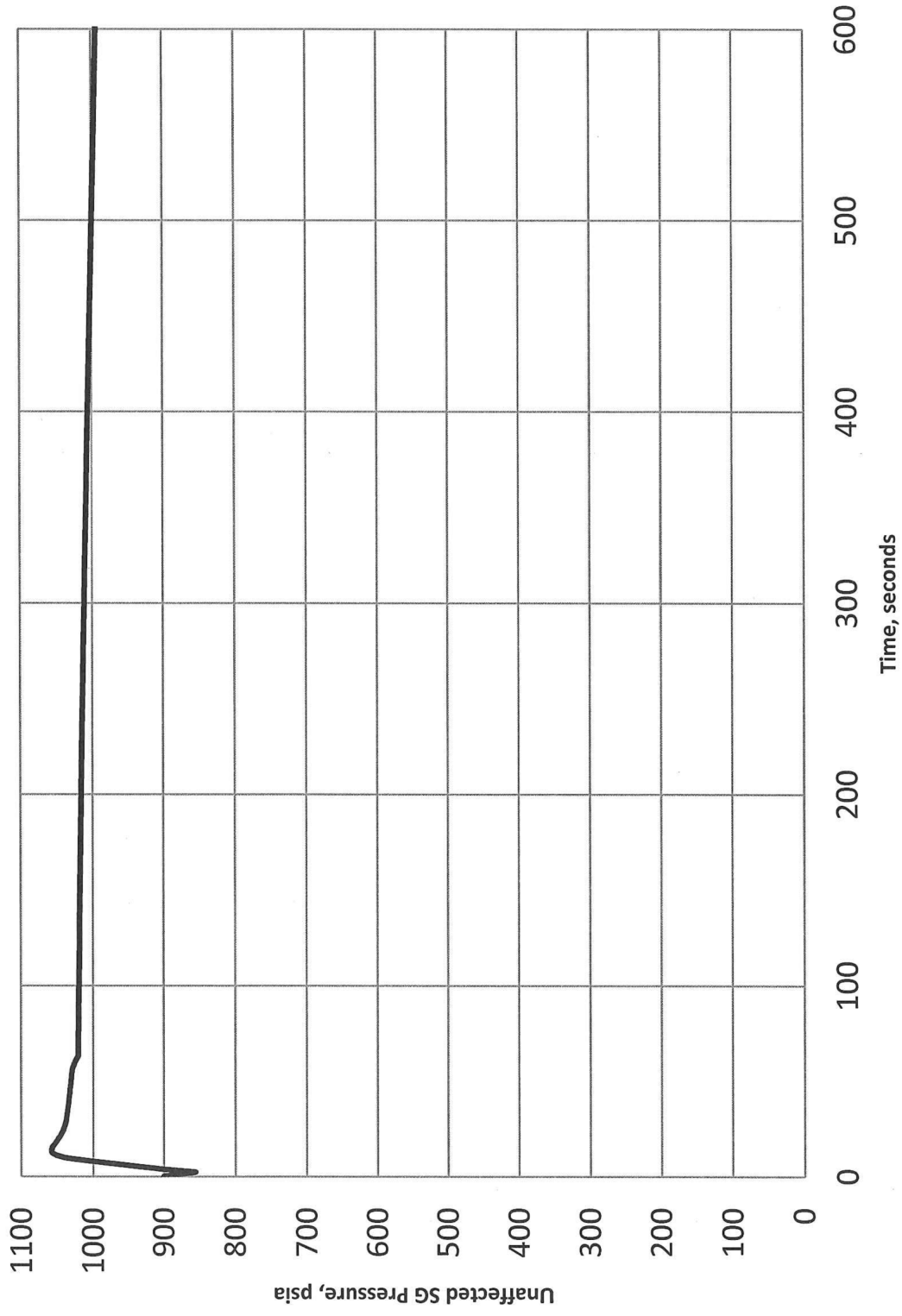


FIGURE 14.8.2–9 MAIN STEAM LINE BREAK ANALYSIS - 102% POWER WITH LOSS OF OFFSITE POWER AND FAILURE OF VITAL BUS CABINET VA-10 OR VA-20 - AFFECTED STEAM GENERATOR LIQUID MASS VS. TIME

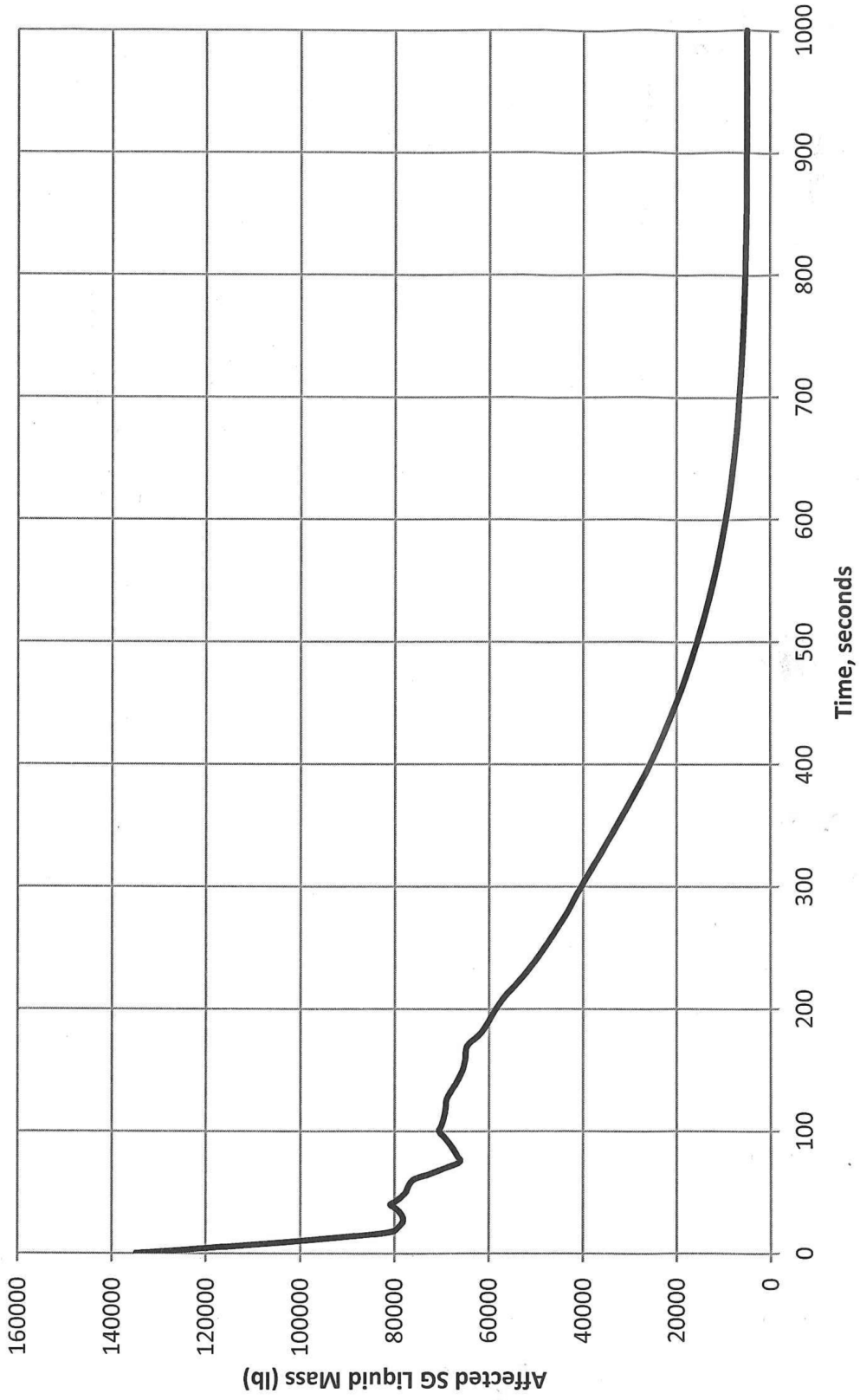


FIGURE 14.8.3–1 DELETED BY FSARCR 04-MP2-018

FIGURE 14.8.3–2 DELETED BY FSARCR 04-MP2-018

FIGURE 14.8.3–3 DELETED BY FSARCR 04-MP2-018

FIGURE 14.8.3–4 DELETED BY FSARCR 04-MP2-018

FIGURE 14.8.3–5 DELETED BY FSARCR 04-MP2-018

FIGURE 14.8.3–6 DELETED BY FSARCR 04-MP2-018

TABLE 14.8.4-1 LOSS OF COOLANT ACCIDENT (OFF SITE ASSUMPTIONS)

Assumption				
(1)	Core power level = 2754 MWt			
(2)	Core released fractions: Consistent with Table 2 of Regulatory Guide 1.183			
(3)	Iodine composition:	Containment	Sump	
	Particulate	95%	0%	
	Elemental	4.85%	97%	
	Organic	0.15%	3%	
(4)	Containment leak rate:	0.5%/day ≤ 24 hrs.		
		0.25%/day > 24 hrs.		
(5)	Enclosure Building Filtration System (EBFS) charcoal filter efficiencies:			
		particulate/elemental/organic		90 / 90 / 70% (1)
(6)	Time Before Enclosure Building Filtration System (EBFS) is Fully Functional			110 seconds
(7)	EBFS Bypass Leakage			
	(% by Weight of Containment Air per Day)			
	0 - 110 sec:	0.5%		
	110 sec - 24 hours:	0.007%		
	24 - 720 hours:	0.0035%		
	NOTE: Prior to EBFS draw down at 110 seconds, bypass leakage is the full containment leak rate of 0.5%. After 110 seconds, bypass leakage is based on 1.4% of containment leak rate and half that at 24 hours.			
(8)	X/Qs:			
	Location	Time Period	Elevated	Ground Release
	EAB	(0-2) hrs.	1.00 E-04	3.66 E-04
	LPZ	(0-4) hrs	2.69 E-05	4.80 E-05
		(4-8) hrs.	1.07E-05	2.31 E-05
		(8-24) hrs.	6.72E-06	1.60 E-05
		(24-96) hrs.	2.46E-06	7.25 E-06
		(96-720) hrs.	5.83E-07	2.32 E-06

TABLE 14.8.4-1 LOSS OF COOLANT ACCIDENT (OFF SITE ASSUMPTIONS) (CONTINUED)

Assumption				
(9)	Dose Conversion Factors			Federal Guidance Reports 11 and 12
(10)	Containment Free Air Volume = 1.899×10^6 ft^3			
(11)	Breathing Rates			
	(0 -8) hr. =	3.5×10^{-4} m^3/sec		
	(8-24) hr. =	1.8×10^{-4} m^3/sec		
	(24-720) hr. =	2.3×10^{-4} m^3/sec		
(12)	Release Points:Filtered - Millstone Stack			
	Bypass - MP-2 Containment			
(13)	Containment Sprayed Volume:			35.4%
(14)	Containment Spray Removal Coefficients:		elemental = 20 per hour	
			particulate = 6.42 per hour	
(15)	Containment Spray Effectiveness Time:		elemental: 75 seconds - 3.03 hours	
			particulate: 75 seconds - 3.23 hours	
(16)	ESF Leakage: 24 gallons per hour			
(17)	ESF Leakage begins at 27.5 minutes post LOCA			
(18)	Sump Volume:			3.773E+04 ft^3
(19)	RWST Back leakage begins at			6.45 hours
(20)	RWST Back leakage amount:			0.05 - 0.70 gpm
(21)	Iodine DF: 100			
(22)	Sump pH ≥ 7.0			

(1) 70% is a conservative analysis assumption for the organic iodine. Technical Specifications can support assumptions for efficiencies of 90% for all iodine species.

TABLE 14.8.4-2 SUMMARY OF DOSES FOR LOSS OF COOLANT ACCIDENT

DOSE (rems)

TEDE	
EAB	2.9E+00
LPZ	1.7E+00

TABLE 14.8.4-3 LOSS OF COOLANT ACCIDENT (CONTROL ROOM ASSUMPTIONS)

1. Control Room Volume = 3.565 E+04 ft^3.
2. Control Room Unfiltered Inleakage in Recirculation Mode
   200 cfm
3. Control Room Normal Makeup Air Flowrate
   800 cfm
4. Time from MP-2 LOCA Initiation to Time when Control Room Intake Dampers Close
   20 seconds
5. Time when Control Room Emergency Ventilation (Filtration) System Operating at Full Capacity
   1 hour
6. Control Room Emergency Ventilation (Filtration) System Flowrate
   2,250 cfm
7. CREV Filter Efficiency
   70% (1)
8. Control Room Shielding:

   North Wall: 2' concrete.

   West Wall: 1.5 feet concrete except 8 foot section which is 2 feet concrete.

   South Wall: 24.5 feet of 1 foot concrete except glass wall 86.75 feet long.

   East Wall: 2 feet concrete.

   Roof: 2 feet concrete.

   Floor: 2 foot concrete.

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(1) 70% is a conservative analysis assumption for particulate, elemental and organic iodine. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

TABLE 14.8.4-4 ATMOSPHERIC DISPERSION DATA FOR MILLSTONE UNIT 2 CONTROL ROOM

	Release Point		
	Ground X/Q, sec/m^3	**RWST X/Q, sec/m^3**	**Stack X/Q, sec/m^3**
0 - 2 hr	3.00 E-03	9.54 E-04	2.51 E-04
2 - 4 hr	1.87 E-03	7.56 E-04	2.51 E-04
4 - 8 hr	1.87 E-03	7.56 E-04	1.96 E-05
8 - 24 hr	6.64 E-04	2.72 E-04	5.46 E-06
24 - 96 hr	5.83 E-04	2.17 E-04	3.43 E-07
96 - 720 hr	4.97 E-04	1.51E-04	6.44 E-09

TABLE 14.8.4-5 DOSE TO MILLSTONE UNIT 2 CONTROL ROOM OPERATORS

Release	TEDE (1)
Millstone 2 (LOCA)	3.0E+00

Notes:

(1) Dose through wall and ceiling from external sources included.

Millstone Power Station Unit 2
Safety Analysis Report

Chapter 15: License Renewal

CHAPTER 15—LICENSE RENEWAL

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CHAPTER 15 – LICENSE RENEWAL

15.0 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix, which includes the following sections, comprises the FSAR supplement:

- Section 15.1 contains a listing of the aging management programs and the status of the program at the time the License Renewal Application was submitted. This section also contains a list of new aging management programs created after the issuance of the renewed operating license.

- Section 15.2 contains a description of the programs for managing the effects of aging.

- Section 15.3 contains the evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation.

- Section 15.4 contains a summarized description of the programs that support the TLAAs.

- Section 15.5 contains a summarized description of the plant-specific exemptions.

- Section 15.6 contains a matrix of the license renewal commitments.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the unit's previous 40 year operating license expiration date. Unless otherwise identified, references to the Operating License are considered a reference to the Renewed Operating License.

15.1 OVERVIEW OF AGING MANAGEMENT PROGRAMS

15.1.1 AGING MANAGEMENT PROGRAMS

The aging management programs for Millstone Unit 2 are described in the following sections. The programs are either consistent with generally accepted industry methods as discussed in NUREG-1801 (Reference 15.1-1), require enhancements to be consistent with generally accepted industry standards, or are site specific programs.

The following list reflects the status of these programs at the time this section was included in the FSAR and provides a historical perspective of their status at the completion of the NRC review of the License Renewal Application. The implementation status of the listed programs will change as new programs are developed and enhancements to existing programs are completed. Commitments for program additions and enhancements are identified in the appropriate sections.

1. Battery Rack Inspections (Section 15.2.1.1) (Existing - Requires Enhancement).

2. Boraflex Monitoring (Section 15.2.1.2) (Existing).

3. Boric Acid Corrosion (Section 15.2.1.3) (Existing).

4. Buried Pipe Inspection Program (Section 15.2.1.4) (Existing - Requires Enhancement).

5. Chemistry Control for Primary Systems Program (Section 15.2.1.5) (Existing).

6. Chemistry Control for Secondary Systems Program (Section 15.2.1.6) (Existing).

7. Closed-Cycle Cooling Water System (Section 15.2.1.7) (Existing - Requires Enhancement).

8. Electrical Cables and Connectors Not Subject to 10 CFR 50.49 (Section 15.2.1.8) (To Be Developed).

9. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements used in Instrumentation Circuits (Section 15.2.1.9) (Existing - Requires Enhancement).

10. Fire Protection Program (Section 15.2.1.10) (Existing - Requires Enhancement).

11. Flow-Accelerated Corrosion (Section 15.2.1.11) (Existing).

12. Fuel Oil Chemistry (Section 15.2.1.12) (Existing).

13. General Condition Monitoring (Section 15.2.1.13) (Existing - Requires Enhancement).

14. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section 15.2.1.14) (Existing - Requires Enhancement).

15. Infrequently Accessed Areas Inspection Program (Section 15.2.1.15) (To Be Developed).

16. Inservice Inspection Program: Containment Inspections (Section 15.2.1.16) (Existing).

17. Inservice Inspection Program: Reactor Vessel Internals (Section 15.2.1.17) (Existing - Requires Enhancement).

18. Inservice Inspection Program: Systems, Components and Supports (Section 15.2.1.18) (Existing - Enhancement complete).

19. Inspection Activities: Load Handling Cranes and Devices (Section 15.2.1.19) (Existing - Requires Enhancement).

20. Reactor Vessel Surveillance (Section 15.2.1.20) (Existing).

21. Service Water System (Open-Cycle Cooling) (Section 15.2.1.21) (Existing).

22. Steam Generator Structural Integrity (Section 15.2.1.22) (Existing).

23. Structures Monitoring Program (Section 15.2.1.23) (Existing - Requires Enhancement).

24. Tank Inspection Program (Section 15.2.1.24) (Existing - Requires Enhancement).

25. Work Control Process (Section 15.2.1.25) (Existing - Requires Enhancement).

26. Bolting Integrity Program (Section 15.2.1.26) (Existing).

15.1.2 TIME LIMITED AGING ANALYSES AGING MANAGEMENT PROGRAMS:

1. Electrical Equipment Qualification (Section 15.4.1) (Existing).

2. Metal Fatigue of Reactor Coolant Pressure Boundary (Section 15.4.2) (Existing).

15.1.3 AGING MANAGEMENT PROGRAMS CREATED AFTER ISSUANCE OF THE RENEWED OPERATING LICENSE

1. Alloy 600 Management Program (Section 15.2.1.27).

15.1.4 REFERENCES FOR SECTION 15.1

15.1-1 NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U. S. Nuclear Regulatory Commission, July 2001

15.2 PROGRAMS THAT MANAGE THE EFFECTS OF AGING ON STRUCTURES AND COMPONENTS WITHIN THE SCOPE OF LICENSE RENEWAL

This section provides summaries of the programs credited for managing the effects of aging on structures and components within the scope of license renewal.

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in NUREG-1800, Section A.2. The Quality Assurance program includes the elements of corrective action, confirmation process, and administrative controls and is applicable to the safety-related and non safety-related structures, and components that are within the scope of license renewal.

15.2.1 AGING MANAGEMENT PROGRAMS

15.2.1.1 Battery Rack Inspections

Program Description

Battery Rack Inspections is a plant-specific program that manages the aging effect of loss of material. The structural integrity of the support racks for the station batteries, within the scope of license renewal, is verified by visually inspecting for loss of material.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inclusion of In-Scope Battery Racks

  The existing inspection program has been modified to include those battery racks that require monitoring for license renewal, but were not originally included in the program. Inclusion of those battery racks completes the actions required for Commitment Item 1 in Table 15.6-1.

- Inspection Criteria

  Implementing procedures have been modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks for all in-scope batteries. Revision of those procedures completes the actions required for Commitment Item 2 in Table 15.6-1.

15.2.1.2 Boraflex Monitoring

Program Description

Boraflex is no longer credited as a neutron absorber. Thus, Boraflex monitoring is no longer performed.

15.2.1.3 Boric Acid Corrosion

Program Description

Boric Acid Corrosion corresponds to NUREG-1801, Section XI.M10 “Boric Acid Corrosion.” The program manages the aging effect of loss of material and ensures that systems, structures, and components susceptible to boric acid corrosion are properly monitored. The program uses visual inspections to detect the boric acid leakage source, path, and any targets of the leakage. It ensures that boric acid corrosion is consistently identified, documented, evaluated, trended, and effectively repaired. The Boric Acid Corrosion program provides both detection and analysis of leakage of borated water inside containment. The General Condition Monitoring program is the primary method for detecting borated water leakage outside containment. The analysis of the leakage is performed through the Boric Acid Corrosion program. Any necessary corrective actions are implemented through the Corrective Action Program.

Boric Acid Corrosion program implements the requirements of:

- NRC Bulletin 2001-01 (Reference 15.2-15)
- NRC Bulletin 2002-01 (Reference 15.2-16)
- NRC Bulletin 2002-02 (Reference 15.2-17)
- NRC Bulletin 2003-02 (Reference 15.2-18)
- NRC Order EA-03-009 (Reference 15.2-19)
- NRC Bulletin 2004-01 (Reference 15.2-20)

The acceptance criterion is the absence of any boric acid leakage or precipitation. If boric acid leakage or precipitation is found by any personnel, it is required to be reported using the

Corrective Action Program. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.4 Buried Pipe Inspection Program

Program Description

The Buried Pipe Inspection Program is an existing program that corresponds to NUREG-1801, Sections XI.M28, "Buried Piping and Tanks Surveillance" and XI.M34, "Buried Piping and Tanks Inspection." The program manages the aging effect of loss of material through the use of preventive measures and inspections. The inspections will be performed when the piping and components are excavated for maintenance or for any other reason.

There are no buried tanks within the scope of license renewal.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. In addition to visual inspections, the field inspections for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Baseline Inspection

  A baseline inspection of the in-scope buried piping located in a damp soil environment has been performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching has been performed by visual, and mechanical or other appropriate methods. These inspections complete the actions required for Commitment Item 3 in Table 15.6-1.

- Buried Piping Inspections

  The maintenance and work control procedures have been revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures include the inspection for the loss of material due to selective leaching, which will be performed by visual, and mechanical or other appropriate methods. These changes complete the actions required for Commitment Item 4 in Table 15.6-1.

15.2.1.5 Chemistry Control for Primary Systems Program

Program Description

Chemistry Control for Primary Systems Program corresponds to NUREG-1801, Section XI.M2, "Water Chemistry." The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, fluorides, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-105714 (Reference 15.2-1) for primary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.6 Chemistry Control for Secondary Systems Program

Program Description

Chemistry Control for Secondary Systems Program corresponds to NUREG-1801, Section XI.M2, "Water Chemistry." The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, sodium, dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or cracking. Water chemistry control is in accordance with the guidelines in EPRI TR-102134 (Reference 15.2-2) for secondary water chemistry.

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system-specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.7 Closed-Cycle Cooling Water System

Program Description

Closed-Cycle Cooling Water System corresponds to NUREG-1801, Section XI. M21, "Closed-Cycle Cooling Water System." The program manages the aging effect of loss of material through the maintenance of process fluid chemistry and performance monitoring of closed-cycle cooling water systems to ensure parameters remain within acceptable limits. The program is based directly on guidance contained in EPRI Report TR-107396 (Reference 15.2-3).

The acceptance criterion is that the maximum levels for the monitored contaminants are maintained below the system specific limits. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality

Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following commitment has been implemented prior to the period of extended operation:

- Heat Exchanger Baseline Inspection

  A baseline visual inspection has been performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one:

  - Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger,
  - Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and
  - Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.

The performance of these inspections complete the actions required for Commitment Item 29 in Table 15.6-1.

15.2.1.8 Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” as modified by NRC Interim Staff Guidance-05 (Reference 15.2-4). This program manages the aging effects of cracking and embrittlement to ensure that electrical cables, connectors, and fuse holders within the scope of license renewal that are exposed to an adverse localized environment (but not subject to the environmental qualification requirements of 10 CFR 50.49) are capable of performing their intended function. Adverse localized environments may be caused by heat, radiation or moisture.

The acceptance criterion for the visual inspections of accessible non-EQ cable jackets and connector coverings is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following actions will be implemented prior to the period of extended operation:

- Program Implementation

  The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program has been established.

  This completes the action required to complete Commitment Item 5 in Table 15.6-1, License Renewal Commitments.

- Inclusion of In-Scope Fuse Holders

  Fuse holders meeting the requirements have been evaluated. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The aging management review of fuse holders considered the aging stressors for metallic clips and concluded that there were no aging effects that require management.

  This completes the actions required to complete Commitment Item 6 in Table 15.6-1, License Renewal Commitments.

15.2.1.9 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Program Description

Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits corresponds to NUREG-1801, Section XI.E2, “Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” and the program as modified in draft NRC ISG-15 (Reference 15.2-5). This program manages the aging effects of cracking and embrittlement for electrical cables within the scope of license renewal that are used in circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation (but not subject to the environmental qualification requirements of 10 CFR 50.49), and are installed in adverse localized environments caused by heat, radiation or moisture.

The acceptance criterion for the calibration readings is the loop-specific tolerances established in Technical Specifications and surveillance procedures. Where calibration of the instrumentation is not performed in situ, the acceptance criteria for each test are defined by the specific type of test performed and the specific cable tested. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Testing of Cables for Instruments That Are Not Calibrated In Situ

  Procedures have been developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ. The first tests have been completed. The frequency of subsequent tests will be based on Engineering evaluation and will not exceed a 10 year interval. This completes the action required to complete commitment Item 7 in Table 15.6-1, License Renewal Commitments.

- Review of Surveillance Test Results for Cables Tested In Situ

  Calibration results for cables tested in situ have been reviewed to detect severe aging degradation of the cable insulation, and include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years. This completes the action required to complete commitment Item 32 in Table 15.6-1, License Renewal Commitments.

15.2.1.10 Fire Protection Program

Program Description

The Fire Protection Program is an existing program and corresponds to NUREG-1801, Sections XI.M26, “Fire Protection” and XI.M27, “Fire Water System” and to the revised XI.M27, “Fire Water System” program described in NRC Interim Staff Guidance (ISG)-04 (Reference 15.2-6). The program manages the aging effects of loss of material, cracking, and change of material properties for plant fire protection features and components. The program manages these aging effects through the use of periodic inspections and tests.

The program also manages the aging effects for the diesel-driven fire pump fuel supply line, the reactor coolant pump oil collection systems, and Appendix R support equipment.

Visual inspection of fire protection piping internal surfaces that are exposed to water is performed when the system is opened for maintenance and/or repair. The Work Control Process provides guidance for the performance of internal inspections of fire protection piping and components whenever the system is opened for maintenance or repair.

The acceptance criteria for the Fire Protection Program are:

- For visual inspections, the absence of anomalous indications that are signs of degradation.

- For fire barriers and fire doors, the sizes for breaks, holes, cracks, spalling gaps, and/or clearances are in accordance with the limits established in the inspection procedures.

- For fire protection equipment performance tests (i.e., flow and pressure tests), acceptance criteria are provided in the appropriate surveillance procedures.

Additionally, the fire protection water system pressure is continuously monitored to be above the minimum setpoint. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Baseline Fire Protection Inspections

  A baseline visual inspection has been performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation. These inspections complete the action required for Commitment Item 8 in Table 15.6-1.

The following program enhancement will be implemented prior to the sprinkler heads achieving 50 years of service life:

- Testing or Replacement of Sprinkler Heads

  Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years has been included in the Fire Protection Program. The first tests will be completed prior to the sprinkler heads achieving 50 years of service life. The frequency of subsequent tests will not exceed a 10 year interval. This change completes the action required for Commitment Item 9 in Table 15.6-1, License Renewal Commitments.

15.2.1.11 Flow-Accelerated Corrosion

Program Description

Flow-Accelerated Corrosion Program corresponds to NUREG-1801, Section XI.M17, “Flow-Accelerated Corrosion.” The program manages the aging effect of loss of material in accordance with the EPRI guidelines in NSAC-202L (Reference 15.2-7). It includes procedures or administrative controls to assure that the structural integrity of carbon steel and low-alloy steel piping and components, such as valves, steam traps, and feedwater heaters, is maintained.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.12 Fuel Oil Chemistry

Program Description

Fuel Oil Chemistry corresponds to NUREG-1801, Section XI.M30, "Fuel Oil Chemistry." The program manages the aging effect of loss of material by monitoring and controlling fuel oil quality to ensure that it is compatible with the materials of construction for in-scope components containing diesel fuel oil.

The Fuel Oil Chemistry program uses the following industry standards as the basis for the program:

ASTM Standard D 1796 (Reference 15.2-8),

ASTM Standard D 2276 (unmodified) (Reference 15.2-9), and

ASTM Standard D 4057 (Reference 15.2-10).

The acceptance criterion is adherence to the specific guidelines and limits defined in related plant procedures for parameters that have been shown to contribute to component degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.13 General Condition Monitoring

Program Description

General Condition Monitoring is a plant specific program that manages the aging effects of loss of material, change of material properties, and cracking on the external surfaces of components. It is performed in accessible plant areas for components and structures including those within the scope of license renewal and involves visual inspections for evidence of age related degradation. General Condition Monitoring is implemented by Health Physics technicians, System Engineers, and Plant Equipment Operators while performing their routine in-plant activities.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The

corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancement will be implemented prior to the period of extended operation:

- Procedure and Training Enhancements

  The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns have been enhanced to provide expectations that identify the requirements for the inspection of aging effects. These actions complete the requirements for Item 10 in Table 15.6-1.

15.2.1.14 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements corresponds to NUREG-1801, Section XI.E3, "Inaccessible Medium-Voltage Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements." This program manages the aging effect of formation of water trees and ensures that inaccessible medium-voltage (2 kV to 15 kV) electrical cables within the scope of license renewal (but not subject to the environmental qualification requirements of 10 CFR 50.49) that have been submerged, remain capable of performing their intended function. The program considers the combined effects of submergence, simultaneous with a significant voltage exposure. Significant voltage exposure is defined as being subjected to system voltage for more the twenty-five percent of the time.

The acceptance criterion for the inspections performed under the Structures Monitoring Program is to confirm that in-scope, medium-voltage cables have not become submerged. In-scope cable found to be submerged in standing water for an extended period of time will be subject to an engineering evaluation and corrective action. The evaluation will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The test will use a proven methodology for detecting deterioration of the insulation due to wetting. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Verification Testing

  In scope cable found to be submerged has been subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time was based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation also addresses the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test used a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Cables, or other appropriate testing. Testing has acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality was entered into the Corrective Actions Program. This completes the actions required to complete commitment Item 11 in Table 15.6-1, License Renewal Commitments.

- Testing of Inaccessible Medium Voltage Cables

  The in-scope cables in Unit 3 duct lines number 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and number 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C, 35D) have been tested to demonstrate that water treeing will not prevent the cables from performing their intended function. Subsequent testing has been scheduled to be performed on a frequency not to exceed a 10 year interval.

  This completes the actions required to complete commitment Item 33 in Table 15.6-1, License Renewal Commitments.

- Sample Testing of Inaccessible Medium Voltage Cables

  In addition to the testing specified in Commitment 33, a representative sample of in-scope medium voltage cables has been tested to demonstrate that water treeing will not prevent the cables from performing their intended function. Subsequent testing has been scheduled to be performed on a frequency not to exceed a 10 year interval.

15.2.1.15 Infrequently Accessed Areas Inspection Program

Program Description

Infrequently Accessed Areas Inspection Program is a plant-specific program that manages the aging effects of loss of material, change of material properties, and cracking. The program uses

visual inspections of the external surfaces of in-scope structures and components located in infrequently accessed areas of the plant.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements have been implemented prior to the period of extended operation:

- Program Implementation

  The Infrequently Accessed Inspection Program has been established.

  The establishment of the Infrequently Accessed Inspection Program completes the action required for Commitment Item 12 in Table 15.6-1.

15.2.1.16 Inservice Inspection Program: Containment Inspections

Program Description

Inservice Inspection Program: Containment Inspections corresponds to the following NUREG-1801 program descriptions:

- Section XI.S1, “ASME Section XI, Subsection IWE”,
- Section XI.S2, “ASME Section XI, Subsection IWL”, and
- Section XI.S4, “10 CFR Part 50, Appendix J.”

The program manages the aging effects of loss of material, change of material properties, and cracking. The program is consistent with ASME Section XI, Subsections IWE and IWL, and 10 CFR 50.55a(b)(2), which provide the criteria for ISI Containment inspections.

Appendix J Leakage Rate Testing is included as part of the Inservice Inspection Program: Containment Inspections. The Containment Appendix J Leakage Rate Test Program implements Type A and B tests to measure the overall primary Containment integrated leakage rate.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Containment Inspections are based on the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action

process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.17 Inservice Inspection Program: Reactor Vessel Internals

Program Description

Inservice Inspection Program: Reactor Vessel Internals corresponds to the following NUREG-1801 program descriptions:

- Section XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”.

- Section XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)”.

- Section XI.M16, “PWR Vessel Internals”.

The Inservice Inspection Program: Reactor Vessel Internals manages the effects of aging for those reactor internals that are susceptible to loss of material, cracking, loss of preload, change in dimension and loss of fracture toughness (which presents itself as cracking due to embrittlement).

Industry groups are in place whose objectives include the investigation of the aging effects applicable to reactor vessel internals regarding such items as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC and IASCC), and loss of preload for baffle and former-assembly bolts.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Reactor Vessel Internals are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be implemented at least two years prior to period of extended operation:

- Reactor Vessel Internals Inspections

  Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), and stress corrosion cracking (PWSCC and IASCC) and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, has been submitted to the NRC for approval.

The submission of the program description to the NRC completes the required actions for Commitment Item 13.

15.2.1.18 Inservice Inspection Program: Systems, Components and Supports

Program Description

Inservice Inspection Program: Systems, Components and Supports corresponds to the following NUREG-1801 program descriptions:

- Section XI.M1, “ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD”,
- Section XI.M3, “Reactor Head Closure Studs”,
- Section XI.M12, “Thermal Aging Embrittlement of Cast Austentic Stainless Steel (CASS)”, and
- Section XI.S3, “ASME Section XI, Subsection IWF.”

The Inservice Inspection Program: Systems, Components and Supports is an existing program that was developed to comply with the requirements of ASME Boiler and Pressure Vessel Code, Section XI (Reference 15.2-11). The ASME program provides the requirements for ISI, repair, and replacement for all Class 1, 2 and 3 components and the associated component supports. For license renewal, the Millstone program has been credited to manage the effects of aging for only Class 1 and specific Class 2 components (on the secondary side of the steam generators as determined through the aging management review process) and for Class 1, 2, and 3 components supports. Inservice Inspection Program: Systems, Components and Supports manages the aging effects of cracking, loss of fracture toughness, loss of material and loss of preload.

The acceptance criteria for examinations performed in accordance with the Inservice Inspection Program: Systems, Components and Supports are based on the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following action will be taken prior to the period of extended operation:

- Monitoring Fracture Toughness

For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3.

An engineering evaluation determined that CASS components susceptible to thermal aging embrittlement exist only in the pressurizer surge line. The evaluation determined that the surge line is highly flaw tolerant but recommended volumetric examinations to confirm that no flaws exist that are deeper than the allowable flaw depths. Successful inspections of the pressurizer surge line during the M2R22 refueling outage confirmed that no adverse flaws exist. Those inspections complete the actions required for Commitment Item 27, Table 15.6-1.

- Pressurizer Replacement

  Dominion has replaced the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC. Replacing the pressurizer completes the action required for Commitment Item 36.

15.2.1.19 Inspection Activities: Load Handling Cranes and Devices

Program Description

Inspection Activities: Load Handling Cranes and Devices corresponds to NUREG-1801, Section XI. M23, “Inspection of Overhead Heavy Load [Related to Refueling] Handling Systems.” The program manages the aging effect of loss of material for the load handling cranes and devices within the scope of license renewal. The in-scope load handling cranes and devices are either safety-related or seismically designed to ensure that they will not adversely impact safety-related components during or subsequent to a seismic event.

Inspection Activities: Load Handling Cranes and Devices addresses the overall condition of the crane or device, including checking the condition of the structural members (i.e., rails, girders, etc.) and fasteners on the crane or device, the runways along which the crane or device moves, and the base plates and anchorages for the runways and monorails.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements have been implemented prior to the period of extended operation:

- Inclusion of In-Scope Lifting Devices

  The existing inspection program has been modified to include all lifting devices that require monitoring for license renewal. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 15.

- Inspection Criteria

  Implementing procedures and documentation have been modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 16.

15.2.1.20 Reactor Vessel Surveillance

Program Description

Reactor Vessel Surveillance corresponds to NUREG-1801, Section XI.M31 “Reactor Vessel Surveillance”. The Reactor Vessel Surveillance program manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy subcomponents in the beltline region of the reactor vessel. Neutron dosimetry and material properties data derived from the reactor vessel materials’ irradiation surveillance program are used in calculations and evaluations that demonstrate compliance with applicable regulations. This program ensures compliance with Technical Requirements Manual requirements that surveillance specimens are removed and examined at predetermined intervals established in the Technical Specification to monitor the changes in the material properties and the results of the examinations used to update the Technical Specification operating limits.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.21 Service Water System (Open-Cycle Cooling)

Program Description

The Service Water System (Open-Cycle Cooling) program corresponds to NUREG-1801, Section XI.M20, “Open Cycle Cooling Water System.” The program manages the aging effects of loss of material and buildup of deposits. The program implements the NRC guidelines in Generic

Letter 89-13 (Reference 15.2-12), which includes (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion (including microbiologically influenced corrosion), erosion, protective coating failure, silting, and biofouling do not degrade the performance of safety-related systems serviced by Service Water System; (d) a system walkdown inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures. In lieu of thermal performance testing, Millstone Unit 2 relies on frequent, regular inspection and cleaning of heat exchangers to preclude fouling.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.22 Steam Generator Structural Integrity

Program Description

Steam Generator Structural Integrity corresponds to NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity Program.” This program manages the aging effects of loss of material and cracking and adopts the performance criteria and guidance for monitoring and maintaining steam generator tubes as defined in NEI 97-06 (Reference 15.2-13). The program incorporates performance criteria for structural integrity, accident-induced leakage, and operational leakage. The program includes preventive measures to mitigate degradation through the control of primary and secondary side water chemistry; assessment of degradation mechanisms; inservice inspection of the steam generator tubes to detect degradation; evaluation and plugging or repair, as needed; and leakage monitoring to ensure the structural and leakage integrity of the pressure boundary.

The acceptance criteria are established in the current licensing basis as compliance with the applicable regulations and acceptance standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.23 Structures Monitoring Program

Program Description

Structures Monitoring Program corresponds to the following NUREG-1801 program descriptions:

- Section XI.S5 “Masonry Wall Program”,
- Section XI.S6 “Structures Monitoring Program”, and

- Section XI.S7 “R. G. 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants”.

The Structures Monitoring Program manages the aging effects of loss of material, change of material properties, and cracking by the monitoring of structures and structural support systems that are in the scope of license renewal. The majority of these structures and structural support systems are monitored under 10 CFR 50.65 (Reference 15.2-14). Other structures in the scope of license renewal (such as non-safety related buildings and enclosures, duct banks, valve pits and trenches, HELB barriers, and flood gates) are also monitored to ensure there is no loss of intended function.

The scope includes all masonry walls and water-control structures identified as performing intended functions in accordance with 10 CFR 54.4.

The acceptance criterion for visual inspections is the absence of anomalous indications that are signs of degradation. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Modification of Structures Monitoring Program procedures

  NUREG-1801 recommends the use of ACI 349.3R-96 and ANSI/ASCE 11-90, as a reference for recommendations for the development of an evaluation procedure for nuclear safety-related concrete structures and existing buildings. These documents were not used or referenced as a standard for establishing the Structures Monitoring Program. The implementing procedure for the Structures Monitoring Program has been modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program. Modification of the implementing procedure completes the action required for Commitment Item 17 in Table 15.6-1.

- Addition of Structures to the Structures Monitoring Program

  The Structures Monitoring Program did not initially monitor all structures in-scope for license renewal. The Structures Monitoring Program and the implementing procedure have been modified to include all in-scope structures. Modification of the Program and the implementing procedure completes the actions required for Commitment Item 18 in Table 15.6-1.

- Sampling of Groundwater

Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade. The implementing procedure for the Structures Monitoring Program has been revised to direct groundwater sampling. Modification of the implementing procedure completes the action required for Commitment Item 19 in Table 15.6-1.

- Engineering Notification of Submerged Medium Voltage Cables

  The Structures Monitoring Program and implementing procedures have been modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged. Modification of the implementing procedure for the Structures Monitoring Program completes the action required for Commitment Item 20 in Table 15.6-1.

- Inspection of Normally Inaccessible Areas That Become Accessible

  Initially, the license renewal implementation plans stated that the maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason. The procedure revision task is accomplished by a modification of the Structures Monitoring implementing procedure for the situation of an inaccessible area becoming accessible by excavation or installation of shielding. Modification of the implementing procedure completes the action required for Commitment Item 21 in Table 15.6-1.

15.2.1.24 Tank Inspection Program

Program Description

Tank Inspection Program corresponds to NUREG-1801, Section XI.M29, “Aboveground Carbon Steel Tanks.” The program manages the aging effect of loss of material through periodic internal and external tank inspections. The program includes inspections of the sealant and caulking in and around the tank and the concrete foundation and evaluations to monitor the condition of coatings, linings, and structural elements, to prevent deterioration of the tanks to unacceptable levels. The program also includes volumetric examination of inaccessible locations, such as the external surfaces of tank bottoms.

The acceptance criterion for visual inspections of paint, coatings, sealant, caulking, and structural elements is the absence of anomalous indications that are signs of degradation. Thickness measurements of the tank walls and bottoms are evaluated against design thickness, established baseline values, or loss of material allowances. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Inspection of sealants and caulkings

  Appropriate inspections of sealants and caulkings used for moisture intrusion prevention in and around aboveground tanks have been performed. These inspections complete the actions required for Commitment Item 22 in Table 15.6-1.

- Non-destructive Volumetric Examination of Inaccessible Tank Bottoms

  Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, have been performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities. These examinations and scheduled subsequent inspections complete the actions required for Commitment Item 23 in Table 15.6-1.

- Tanks Being Added to Tank Inspection Program

  The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and have been included in the respective Tank Inspection Program inspection plan. These changes complete the actions required for Commitment Item 24 in Table 15.6-1.

15.2.1.25 Work Control Process

Program Description

Work Control Process is a plant specific program that integrates and coordinates the combined efforts of Maintenance, Engineering, Operations, and other support organizations to manage maintenance activities. The Work Control Process is utilized to manage the aging effects of loss of material, change of material properties, cracking, and buildup of deposits for components and plant commodities within the scope of license renewal. Performance testing and maintenance activities, both preventive and corrective, are planned and conducted in accordance with the Work Control Process. The Work Control Process also provides opportunities to collect oil and engine coolant fluid samples for subsequent analysis of contaminants and chemical properties, which could either indicate or affect aging.

In addition to visual inspections, the field inspection for loss of material due to selective leaching will include mechanical means, such as resonance when struck by another object, scraping, or chipping.

The acceptance criterion for visual inspections is the absence of anomalous signs of degradation. The acceptance criteria for testing or sampling are specified in the various station procedures and/or vendor technical manuals or recommendations. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Commitments

The following program enhancements will be implemented prior to the period of extended operation:

- Performance of Inspections During Maintenance Activities

  Changes have been made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities. These changes complete the actions required for Commitment Item 25 in Table 15.6-1.

- Selective Leaching Inspection

  Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching have been performed on a representative sample of buried pipe locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation. Susceptible aluminum-bronze valves have been replaced using a material (AL6XN stainless steel) that is more resistant to selective leaching. The buried pipe inspections and valve replacements complete the actions required for Commitment Item 30 in Table 15.6-1.

- Verification of Scope

  A review of the Work Control Process inspection opportunities for each material and environment group supplemental to the initial review conducted during the development of the LRA has been performed. No additional baseline inspections were needed to be performed for the material and environment combinations inspected as part of the Work Control Process. This action completes the requirements for Item 31 in Table 15.6-1.

15.2.1.26 Bolting Integrity Program

Program Description

The Bolting Integrity Program corresponds to NUREG-1801, Section XI.M18, “Bolting Integrity”. The program manages the aging effects of cracking, loss of material and loss of preload.

This is accomplished by establishing good bolting practices in accordance with EPRI NP-5067, Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual, and Volume 2: Small Bolts and Threaded Fasteners and EPRI TR-04213, Bolted Joint Maintenance and Application Guide. For ASME Class bolting, aging effects are additionally managed by the performance of inservice examinations in accordance with ASME Section XI, Subsections IWB, IWC, IWD, and IWF.

The engineering evaluations determine if a component needs to be repaired/replaced or is acceptable for continued operation until the next scheduled inspection. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.2.1.27 Alloy 600 Management Program

Program Description

The Alloy 600 Management Program manages the aging effect of primary water stress corrosion cracking (PWSCC) in Alloy 600 base metal and Alloy 82/182 dissimilar metal welds. The Alloy 600 Management Program is consistent with the Nickel-Alloy Nozzles and Penetrations aging management program described in Chapter XI of NUREG-1801, Rev. 0 (GALL Report). The program monitors susceptible components for indications of PWSCC before there is a loss of intended function. Since the program is based on more recent industry guidance, EPRI TR-1009561, MRP-126 (Reference 15.2-21), the program is enhanced from the program described in Rev. 0 of the GALL. The program scope has been increased according to industry guidelines to include all Alloy 600/82/182 components in the primary system. The inspection requirements have been updated to the requirements of ASME Section XI (Reference 15.2-11) Code Cases N-722-1, N-729-1 and N-770-1.

The Alloy 600 Management Program credits inspections performed under existing inspection processes. Inspection requirements for each type of component are listed in the Alloy 600 Management Plan. Defects found during inspections will be dispositioned in accordance with applicable regulatory and code requirements, utilizing the corrective action system.

Commitments

The following action will be implemented at least two years prior to the period of extended operation:

- PWSCC of Nickel Based Alloys

Millstone will follow the industry efforts investigating the aging effects applicable to nickel based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description was submitted two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 54.21(a)(3).

The submission of the Alloy 600 Management Program description to the NRC (Reference 15.6-3) completes the required actions for Commitment Item 14 in Table 15.6-1.

15.2.2 REFERENCES FOR SECTION 15.2

15.2-1 TR-105714, PWR Primary Water Chemistry Guidelines, Technical Report, Revision 3, Electric Power Research Institute.

15.2-2 TR-102134, PWR Secondary Water Chemistry Guidelines, Technical Report, Revision 3, Electrical Power Research Institute.

15.2-3 EPRI TR-107396, Closed Cooling Water Chemistry Guideline, Technical Report, Electrical Power Research Institute, Palo Alto, CA, November 1997.

15.2-4 NRC Interim Staff Guidance (ISG)-05, The Identification And Treatment Of Electrical Fuse Holders For License Renewal, U.S. Nuclear Regulatory Commission, March 10, 2003.

15.2-5 Letter from Pao-Tsin Kuo, Nuclear Regulatory Commission, to Alex Marion, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, Proposed Interim Staff Guidance (ISG)-15: Revision of Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) X1.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits”, August 12, 2003.

15.2-6 NRC Interim Staff Guidance (ISG)-04, “Aging Management of Fire Protection Systems for License Renewal”, U.S. Nuclear Regulatory Commission, December 3, 2002.

15.2-7 NSAC-202L-R4 Recommendations for an Effective Flow Accelerated Corrosion Program, Electric Power Research Institute, November, 2013.

15.2-8 ASTM D 1796, Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method, American Society for Testing Materials, West Conshohocken, PA.

15.2-9 ASTM D 2276, Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling, American Society for Testing Materials, West Conshohocken, PA.

15.2-10 ASTM D 4057, Standard Practice for Manual Sampling of Petroleum and Petroleum Products, American Society for Testing Materials, West Conshohocken, PA.

15.2-11 ASME Boiler and Pressure Vessel Code Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers.

15.2-12 Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment, Nuclear Regulatory Commission, July 18, 1989 (Supplement 1 dated 4/4/90).

15.2-13 NEI 97-06, Steam Generator Program Guidelines, Technical Report, Nuclear Energy Institute.

15.2-14 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.2-15 NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, U.S. Nuclear Regulatory Commission, August 3, 2001.

15.2-16 NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, March 18, 2002.

15.2-17 NRC Bulletin 2002-02, Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs, U.S. Nuclear Regulatory Commission, August 9, 2002.

15.2-18 NRC Bulletin 2003-02, Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity, U.S. Nuclear Regulatory Commission, 08/21/03.

15.2-19 NRC Order EA-03-009, Issuance Of Order Establishing Interim Inspection Requirements For Reactor Pressure Vessel Heads At Pressurized Water Reactors, U.S. Nuclear Regulatory Commission, February 11, 2003."

15.2-20 NRC Bulletin 2004-01, Inspection of Alloy 82/182/600 Materials used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors, May 28, 2004.

15.2-21 EPRI Report 1009561, Materials Reliability Program: Generic Guidance for Alloy 600 Management (MRP-126), Electric Power Research Institute, Palo Alto, CA November 2004.

15.3 TIME-LIMITED AGING ANALYSIS

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-limited Aging Analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

15.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in FSAR Section 4.3.1. Time-limited aging analyses (TLAAs) applicable to the reactor vessel are:

- Upper-shelf energy (USE)
- Pressurized thermal shock (PTS)
- Pressure-temperature limits

The Reactor Vessel Surveillance program manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time dependent parameters used in the aging analyses for pressurized thermal shock, upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

The reactor vessel neutron embrittlement evaluations have been based on 54 effective full power years of operation. 54 effective full power years would be reached at the end of the period of extended operation (60 years) assuming a capacity factor of 90% for the lifetime of the unit.

15.3.1.1 Upper Shelf Energy

10 CFR 50, Appendix G contains screening criteria that establish limits on how far the upper shelf energy values for a reactor pressure vessel material may be allowed to drop due to neutron irradiation exposure. The regulation requires the initial upper shelf energy value to be greater than 75 ft-lbs in the unirradiated condition and for the value to be greater than 50 ft-lbs in the fully irradiated condition as determined by Charpy V-notch specimen testing throughout the licensed life of the plant. Upper shelf energy values of less than 50 ft-lbs may be acceptable to the NRC if it can be demonstrated that these lower values will provide margins of safety against brittle fracture equivalent to those required by ASME Section XI, Appendix G.

Acceptable upper shelf energy values have been calculated in accordance with Regulatory Guide 1.99, Revision 2 to the end of the period of extended operation. Calculated upper shelf energy values for the most limiting reactor pressure vessel beltline plate and weld materials remain greater than 50 ft-lbs.

15.3.1.2 Pressurized Thermal Shock

Reactor pressure vessel beltline fluence is one of the factors used to determine the margin to reactor pressure vessel pressurized thermal shock as a result of radiation embrittlement. The

margin is the difference between the maximum nil ductility reference temperature in the limiting beltline material (RT_{PTS}) and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forging and axial weld materials, and 300°F for beltline circumferential weld materials.

Acceptable RT_{PTS} values have been calculated in accordance with Regulatory Guide 1.99, Revision 2, requirements to the end of the period of extended operation.

15.3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature limits. These limits identify the maximum allowable pressure as a function of reactor coolant temperature. As the pressure vessel becomes irradiated and its fracture toughness is reduced, the allowable pressure at low temperatures is reduced. Therefore, in order to heatup and cooldown the Reactor Coolant System, the reactor coolant temperature and pressure must be maintained within the limits of Appendix G as defined by the reactor vessel fluence.

Heatup and cooldown limit curves have been calculated using the adjusted reference temperature nil-ductility transition or adjusted reference temperature (adjusted RT_{NDT} or ART) corresponding to the limiting beltline material of the reactor pressure vessel for the current period of licensed operation. Current low temperature overpressure protection (LTOP) system heatup and cooldown limit curves were approved in License Amendment 218.

In accordance with 10 CFR 50, Appendix G, updated pressure-temperature limits for entering the period of extended operation have been developed and implemented prior to the period of extended operation. Low temperature overpressure protection system enable temperature requirements were found to remain valid and will ensure that the pressure-temperature limits will not be exceeded for postulated plant transients during the period of extended operation. Millstone Unit 2 will calculate USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation.

Commitments

The following actions will be taken prior to the period of extended operation.

Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by regulatory guidance changes, were required to be submitted to the NRC for review at least two years prior to the period of extended operation. This commitment is identified in Table 15.6-1, License Renewal Commitments, Item 37.

Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements were submitted to the NRC for review in 2005 (Reference 15.6-4) and approved in 2006 (Reference 15.6-5).

The submittal of the updated USE, RT_{PTS}, and P-T limits completes the required actions for Commitment Item 37 in Table 15.6-1.

15.3.2 METAL FATIGUE

Fatigue is defined as structural deterioration that can occur through repeated stress or strain cycles resulting from fluctuations in loads and/or temperatures. After repeated cyclic loading of sufficient magnitude, micro-structural damage can accumulate leading to microscopic crack initiation at the most highly affected locations. Fatigue cracks typically initiate at points of maximum local stress ranges and minimum local strength. Further cyclic mechanical and/or thermal loading can lead to crack growth.

Fatigue represents an aging mechanism. As such, fatigue evaluations represent a time-limited aging analysis even though the system, structure and component design limits are based upon the number of cycles and the associated fatigue (cumulative) usage factors rather than specific time limits.

15.3.2.1 Millstone Unit 2 Class 1 Components

Components within the Millstone Unit 2 nuclear steam supply system are subject to a wide variety of varying mechanical and thermal loads that contribute to fatigue accumulation. The Reactor Coolant System components are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III (Reference 15.3-1) and ANSI Standard B31.7 (Reference 15.3-2). Use of these codes requires that design analyses for Class A (Class 1) systems and components address fatigue and the establishment of load limits to preclude initiation of fatigue cracks.

The type and number of Reactor Coolant System design transients have been identified. In all instances, the number of Reactor Coolant System design transients assumed in the original design were found to be acceptable for the period of extended operation.

NRC Bulletin 88-08 identified a concern regarding potential temperature stratification or temperature oscillations in unisolable sections of piping attached to the Reactor Coolant System. Based upon the Millstone Unit 2 response (Reference 15.3-3) and supplemental communications, the NRC concluded that Millstone Unit 2 meets the requirements of Bulletin 88-08 (Reference 15.3-4).

Pressurizer surge line thermal stratification was a concern raised by the NRC in Bulletin 88-11. One of the requirements of this bulletin was to analyze the effects of thermal stratification on surge line integrity. These analyses were collectively performed as a Combustion Engineering Owners Group task (Reference 15.3-5) supplemented by additional unit specific inspections and activities. Based upon the Combustion Engineering Owners Group task, the NRC concluded that the owners group analysis CEN-387-P is bounding for thermal stratification and thermal striping

(Reference 15.3-7). Confirmation that this analysis is bounding for Millstone Unit 2 and confirmation that the actions required by Bulletin 88-11 have been completed was provided to the NRC (Reference 15.3-6).

Thermal aging refers to changes in the microstructure and properties of a susceptible material due to prolonged exposure to elevated temperatures above approximately 480 °F. Reactor Coolant System temperatures exceed this threshold. At these temperatures, the hardness of potentially susceptible Cast Austenetic Stainless Steel (CASS) materials increase while their ductility, impact strength and more importantly, their fracture toughness, decrease. Fracture toughness is one of the more important design inputs in a leak-before-break and a flaw tolerance evaluation, performed to ensure protection of the reactor coolant system against guillotine pipe breaks throughout plant life. The degree of change in fracture toughness (thermal embrittlement) is dependent on the time of exposure to these elevated temperatures.

Acceptable thermal and pressure transients, and operating cycles have been projected for ASME Section III Class A, Class 1 and ANSI Standard B31.7 Class 1 components though the period of extended operation.

15.3.2.2 Non-Class 1 Components

Non-Class 1 components can include ASME Section III Classes 2 and 3, ANSI Standard B31.7 Classes 2 and 3, and ANSI Standard B31.1 (Reference 15.3-8) piping and tubing. Piping systems designed to these requirements (e.g., sample lines) incorporate a stress range reduction factor to conservatively address the effects of thermal cycling on fatigue. For those sample lines projected to experience greater than 7,000 equivalent full-temperature thermal cycles, actual expansion stresses did not exceed allowable expansion stresses.

Acceptable numbers of thermal cycles and acceptable expansion stresses have been projected to the end of the period of extended operation.

15.3.2.3 Environmentally Assisted Fatigue

The effect of reactor coolant environment on fatigue is generally referred to as environmentally assisted fatigue. As part of an industry effort to address environmental effects on operating nuclear power plants during the current 40 year licensing term, Idaho National Engineering Laboratories evaluated fatigue-sensitive component locations at plants designed by all four domestic nuclear steam supply system vendors. These evaluations are presented and discussed in NUREG/CR-6260 (Reference 15.3-9). The evaluations associated with the newer-vintage Combustion Engineering plants are applicable, since the majority of the Millstone Unit 2 Class 1 systems and components were designed to ASME Section III/ANSI B31.7 requirements.

The influence of the reactor water environment on the cumulative usage factor was evaluated for the following representative components identified in NUREG/CR-6260 for the period of extended operation, using the most recent laboratory data and methods:

- Reactor vessel shell and lower head.

- Reactor vessel inlet and outlet nozzles.
- Surge line.
- Charging System nozzle.
- Safety Injection System nozzle.
- Shutdown Cooling line.

These six fatigue-sensitive locations have been evaluated using the methods identified in NUREG/CR-6583 (Reference 15.3-10), and NUREG/CR-5704 (Reference 15.3-11).

Utilizing Millstone Unit 2 cyclic and transient information, all six fatigue sensitive component locations were determined to be acceptable for the period of extended operation.

Commitments

The following actions will be implemented prior to the period of extended operation:

- Millstone follows industry efforts that provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will implement the appropriate recommendations resulting from this guidance. This action completes the requirements for Commitment Item 28 in Table 15.6-1.

15.3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Electrical Equipment Qualification (EEQ) program is an integral part of the design, construction and operation of nuclear power generating stations. A description of this program is provided in Section 15.4.1, Electrical Equipment Qualification.

10 CFR Part 50 requires that certain categories of systems, structures and components be designed to accommodate the effects of both normal and accident environmental conditions, and that design control measures be employed to ensure the adequacy of these designs. Specific requirements pertaining to the environmental qualification of these categories of electrical equipment are embodied within 10 CFR 50.49 (Reference 15.3-12). The categories include safety-related (Class 1E) electrical equipment, non-safety-related electrical equipment whose failure could prevent the satisfactory accomplishment of a safety function by safety-related equipment, and certain post-accident monitoring equipment. As required by 10 CFR 50.49, electrical equipment not qualified for the current license term is to be refurbished, replaced or have its qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical equipment that specify a qualification of 40 years or greater are considered to represent a time-limited aging analysis. Guidance relating to the methods and procedures for implementing the requirements of 10 CFR 50.49 is contained within Regulatory

Guide 1.89 (Reference 15.3-13). Further guidance for post-accident monitoring equipment is contained within Regulatory Guide 1.97 (Reference 15.3-14).

Environmental qualification of electrical equipment will be adequately managed for the period of extended operation.

15.3.4 CONCRETE CONTAINMENT TENDON PRESTRESS

The Millstone Unit 2 Containment consists of a pre-stressed, reinforced concrete cylinder and dome, and a flat, reinforced concrete mat foundation supported on unweathered bedrock. The cylindrical portion of the Containment is prestressed by a post-tensioning system composed of horizontal and vertical tendons, with the horizontal tendons placed in three 240 degree systems that use three buttresses as support for the anchorages. The dome has a three-way post tensioning system. Prestress on the containment tendons is expected to decrease over the life of the unit as a result of such factors as elastic deformation, creep and shrinkage of concrete, anchorage seating losses, tendon wire friction, stress relaxation and corrosion.

Containment tendon surveillance results has been performed and indicate acceptable losses in Containment tendon prestress have been projected through the period of extended operation. Inspection results, Table 15.3-1, were used to project acceptable dome, vertical and horizontal forces through the current and extended period of operation. Prestress values are projected to remain above the minimum value of 1308 kips (7.03 kips per wire) for each of the tendon groups. Since containment tendon examinations are performed on a 5 year interval, force projections will be refined and updated as necessary following each examination.

The evaluation of containment tendon examination and surveillance test results involves the use of time-limited assumptions such as corrosion rates, losses of tendon prestress, and changes in material properties. Regression analysis incorporating the most recent 25 years.

15.3.5 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

15.3.5.1 Containment Liner Plate

Millstone Unit 2 has a prestressed, post-tensioned concrete Containment surrounded by an enclosure building. A welded carbon steel liner plate is attached to the inside surface of the concrete. Both the liner plate and the penetration sleeves are designed to serve as the primary Containment leakage barrier. Components of the liner plate include penetration sleeves, access openings, piping penetrations, and electrical penetrations.

Evaluations of the Containment liner plate involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses. Acceptable Containment liner plate integrity has been projected to the end of the period of extended operation.

15.3.5.2 Containment Penetrations

All Millstone Unit 2 Containment penetrations are pressure resistant, leak-tight, welded assemblies, fabricated, installed, inspected, and tested in accordance with ASME Nuclear Vessel Code, Section III (Reference 15.3-1) and ANSI Nuclear Piping Code B31.7 (Reference 15.3-2).

Evaluations of Containment liner plate components involve the use of time-limited assumptions such as corrosion rates and thermal cycles. These evaluations meet the requirements of 10 CFR 54.3 and, as such, represent time-limited aging analyses.

Acceptable Containment penetration integrity has been projected to the period of extended operation.

15.3.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

15.3.6.1 Crane Load Cycle Limit

The containment polar crane, spent fuel crane, and monorails are examples of the types of cranes within the scope of license renewal. These cranes meet the guidance contained in NUREG-0612.

The evaluation of crane loads represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of a time-limited assumption, load cycles. The most frequently used crane is the spent fuel crane. Considering all uses, the spent fuel crane is expected to conservatively experience a total number of load cycles over a 60 year period, that is well below the number of cycles allowed in Crane Manufacturers Association of America, Inc. Specification Number 70.

Acceptable crane load cycles have been projected to the end of the period of extended operation.

15.3.6.2 Reactor Coolant Pump Flywheel

The reactor coolant pump motors are provided with flywheels to increase rotational inertia, thus prolonging pump coast-down and assuring a more gradual loss of primary coolant flow to the core in the event that pump power is lost. During normal operation, the reactor coolant pump flywheels develop sufficient kinetic energy to produce high energy missiles in the event of failure. Conditions that may result in overspeed of the pump increase both the potential for failure and the kinetic energy of the flywheel.

An evaluation was performed of the likelihood of flywheel failure over a 60 year period of operation and a justification was developed for relaxation of Regulatory Guide 1.14, Revision 1, Regulatory Position C.4.b(1), requirements to those identified in Regulatory Position C.4.b(2) (Reference 15.3-15). Using this evaluation, the NRC issued Amendment Number 264 to the unit Technical Specifications, consistent with Regulatory Guide 1.14, Revision 1, Regulatory Position C.4.b(2), to allow the examination of each reactor coolant pump flywheel at least once every 10 years, coinciding with the ASME Section XI inservice inspection program schedule.

The evaluation of reactor coolant pump flywheels represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as thermal cycles and crack growth rates. This evaluation, which indicates a low likelihood of flywheel fatigue failure over a 60 year period, along with implementation of the Inservice Inspection Program: Systems, Components and Supports, provides reasonable assurance that flywheel cracking will be adequately managed for the period of extended operation.

Reactor coolant pump flywheel fatigue cracking will be adequately managed for the period of extended operation.

15.3.6.3 Reactor Coolant Pump Code Case N-481

ASME Boiler and Pressure Vessel Code, Section XI, specifies that a volumetric inspection of the reactor coolant pump casing welds and a visual inspection of pump casing internal surfaces be performed on a reactor coolant pump within each ten year inspection period. These 10 year volumetric inspections are significant for a number of reasons, including; the reactor coolant pumps are welded to the piping, and the pumps must be disassembled in order to gain access to the inside surface of the cast austenitic stainless steel casings. In recognition of these difficulties, ASME Code Case N-481, “Alternative Examination Requirements for Cast Austenitic Pump Casings”, was developed to allow for the replacement of volumetric examinations with a fracture mechanics-based evaluation, supplemented by specific visual inspections.

The evaluation of reactor coolant pump casings represents a time-limited aging analysis per 10 CFR 54.3 since it involves the use of time limited assumptions such as thermal cycles and crack growth rates. This evaluation, which indicates a low likelihood of casing fatigue failure over a 60 year period, along with implementation of the Inservice Inspection Program: Systems, Components and Supports, provides reasonable assurance that cracking of cast stainless steel reactor coolant pump casing welds will be adequately managed throughout the period of extended operation.

15.3.6.4 Leak-Before-Break

The Leak-Before-Break (LBB) analyses were evaluated as time-limited aging analyses (TLAAs) to determine that the analyses remain valid for the period of extended operation. The systems and components that have been analyzed for LBB include the reactor coolant loop piping (hot leg, cold leg, and crossover piping), the pressurizer surge line, and portions of the safety injection and shutdown cooling systems.

The LBB analyses were determined to remain valid for the period of extended operation by evaluating their time-based inputs. Thermal aging of cast austenitic stainless steel (CASS) materials and fatigue crack growth calculations were determined to be time-based inputs as defined in 10 CFR 54.3 and required evaluation for the period of extended operation.

The metal fatigue TLAA evaluations described in FSAR Section 15.3.2.1 conclude that design basis limits are not exceeded for ASME Class 1 components (which envelopes the components evaluated for LBB) through the period of extended operation.

Thermal aging of CASS materials for components that have been evaluated for LBB has been evaluated as a TLAA since long term exposure of CASS materials to reactor coolant system operating temperatures results in an increase in material hardness while its ductility, impact strength and fracture toughness decrease. Fracture toughness represents one of the more important design inputs in a LBB evaluation. The degree of reduction in CASS fracture toughness is dependent on the time of thermal exposure. However, the change in material properties due to thermal aging reaches a saturation value, after which material property changes resulting from additional thermal exposure are not significant. The evaluation of the thermal aging of CASS material for the LBB evaluations consisted of a review to determine whether the fracture toughness value used in the analyses was conservative relative to the fully aged value for fracture toughness for the CASS components. The review concluded that the analysis values were either equal to or lower than the worst-case saturation (fully aged) values for fracture toughness in all cases. Therefore, since the CASS material property values used in current design basis LBB evaluations represent fully aged (saturation) values, and since these values would not change with further exposure time, the LBB evaluations remain valid for the thermal aging of CASS materials throughout the period of extended operation.

The LBB analyses have been projected to remain valid through the end of the period of extended operation.

15.3.7 REFERENCES FOR SECTION 15.3

15.3-1 ASME Section III, “Rules for Construction of Nuclear Vessels”, ASME Boiler and Vessel Pressure Code, American Society of Mechanical Engineers, 1971.

15.3-2 ANSI Standard B31.7, Nuclear Power Piping Code, American Society of Mechanical Engineers.

15.3-3 Letter from E. J. Mroczka to NRC, Response to NRC Bulletin No. 88-08, Thermal Stresses in Piping Connected to Reactor Coolant System, September 20, 1988.

15.3-4 Letter from G.S. Vissing to E. J. Mroczka, NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems (TAC 69652), September 30, 1991.

15.3-5 Letter from E. J. Mroczka to NRC, NRC Bulletin Number 88-11, Pressurizer Surge Line Thermal Stratification, February 28, 1989.

15.3-6 Letter from J. F. Opeka to NRC, Response to NRC Staff Request for Additional Information on Pressurizer Surge Line Thermal Stratification Evaluation, September 30, 1993.

15.3-7 Letter from G. S. Vissing to J. F. Opeka, Safety Evaluation for Combustion Engineering Owners Group Report CEN-387-P, Revision 1 - Pressurizer Surge Line Thermal Stratification Evaluation (Bulletin 88-11)(TAC No. M72144), July 6, 1993.

15.3-8 ANSI B31.1, Power Piping Code, American Society of Mechanical Engineers, 1967.

15.3-9 NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, U.S. Nuclear Regulatory Commission.

15.3-10 NUREG/CR-6583, Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels, U.S. Nuclear Regulatory Commission.

15.3-11 NUREG/CR-5704, Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steel, U.S. Nuclear Regulatory Commission.

15.3-12 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.3-13 Regulatory Guide 1.89, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants, U. S. Nuclear Regulatory Commission.

15.3-14 Regulatory Guide 1.97, Instrumentation of Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, U. S. Nuclear Regulatory Commission.

15.3-15 SIR-94-080-A, Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements, Structural Integrity Associates, Inc., September 1997.

TABLE 15.3-1 CONTAINMENT TENDON PRESTRESS

Inspection Year	Dome Tendon Projected (kips)	Dome Minimum Value (kips)	Vertical Tendon Projected (kips)	Vertical Minimum Value (kips)	Horizontal Tendon Projected (kips)	Horizontal Minimum Value (kips)
40	1453	1343	1521	1339	1467	1325
60	1435	1343	1509	1339	1449	1325

15.4 TLAA SUPPORT PROGRAMS

15.4.1 ELECTRICAL EQUIPMENT QUALIFICATION

The Electrical Equipment Qualification program corresponds to the Time-Limited Aging Analysis (TLAA) support program described in NUREG-1801, Section X.E1, "Environmental Qualification (EQ) of Electrical Components." The program applies to certain electrical components that are important to safety and could be exposed to post-accident environmental conditions, as defined in 10 CFR 50.49. The EEQ program ensures the continued qualification of this equipment during and following design basis accidents. The program determines the necessity for, and frequency of, component replacement or refurbishment in order to maintain the qualification of the equipment. Performance of preventive maintenance and surveillance activities, and monitoring of normal ambient conditions, ensure that components remain within the bounds of their original qualification and provide a basis for extending qualified life through re-analysis.

The acceptance criterion is that the equipment remains within the bounds of its qualified life such that after maximum normal service conditions, the equipment retains sufficient capacity to perform its required safety function during design basis accident conditions. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.4.2 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

The Metal Fatigue of Reactor Coolant Pressure Boundary program mitigates fatigue cracking caused by cyclic strains in metal components of the reactor coolant pressure boundary. This is accomplished by monitoring and tracking the number of critical thermal and pressure transients for selected Reactor Coolant System components to ensure that the number of design transient cycles is not exceeded during the plant operating life.

The acceptance criterion is the fatigue usage factors bounded by the design usage factors. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

15.5 EXEMPTIONS

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analyses as defined in 10 CFR 54.3 have been identified.

15.6 LICENSE RENEWAL COMMITMENTS

Table 15.6-1, License Renewal Commitments, provides a listing of the license renewal commitments.

15.6.1 REFERENCES FOR SECTION 15.6

15.6-1 Letter from Leslie N. Hartz to NRC, Millstone Power Station Units 2 and 3, Response to Request for Additional Information License Renewal Applications, August 13, 2004 (Serial Number.: 04-398).

15.6-2 Letter from Daniel G. Stoddard to NRC, Dominion Nuclear Connecticut Inc, Millstone Power Station Unit 2, Information in Support of License Renewal Commitment #13, Program Description for Reactor Vessel Internals Inspections, July 31, 2013 (Serial Number 13-398).

15.6-3 Letter from Daniel G. Stoddard to NRC, Dominion Nuclear Connecticut Inc, Millstone Power Station Unit 2, Information in Support of License Renewal Commitment #14, Program Description for Alloy 600 Aging Management Program, July 31, 2013 (Serial Number 13-399).

15.6-4 Letter S/N 05-307, Millstone Power Station Unit 2, Proposed Revision to Technical Specifications (LBDCR 05-MP2-003) Reactor Coolant System Heatup/Cooldown Limits, July 14, 2005.

15.6-5 Letter S/N 06-420, U.S. Nuclear Regulatory Commission, Millstone Power Station, Unit 2, - Issuance of Amendment Re: Revision to Technical Specifications Pertaining to the Reactor Coolant System Heatup and Cooldown Limits (TAC No. MC7593) May 3, 2006.

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS

Item	Commitment	Source	Schedule [a]
1	The existing inspection program will be modified to include those battery racks that require monitoring for license renewal, but are not already included in the program.	Battery Rack Inspections	Complete
2	Implementing procedures will be modified to include loss of material as a potential aging effect and to provide guidance on the inspection of items (such as anchorages, bracing and supports, side and end rails, and spacers), which contribute to battery rack integrity or seismic design of the battery racks.	Battery Rack Inspections	Complete
3	A baseline inspection of the in-scope buried piping located in a damp soil environment will be performed for a representative sample of each combination of material and protective measures. Inspection for the loss of material due to selective leaching will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Complete
4	The maintenance and work control procedures will be revised to ensure that inspections of buried piping are performed when the piping is excavated during maintenance or for any other reason. These procedures will include the inspection for the loss of material due to selective leaching which will be performed by visual, and mechanical or other appropriate methods.	Buried Pipe Inspection Program	Complete
5	The Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be established.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
6	Fuse holders meeting the requirements will be evaluated prior to the period of extended operation for possible aging effects requiring management. The fuse holder will either be replaced, modified to minimize the aging effects, or this program will manage the aging effects. The program (if needed for fuse holders) will consider the aging stressors for the metallic clips.	Electrical Cables and Connectors Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Complete
7	Procedures will be developed to employ an alternate testing methodology to confirm the condition of cables and connectors in circuits that have sensitive, low level signals and where the instrumentation is not calibrated in situ.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	"Prior to Period of Extended Operation." Complete. Subsequent testing will not exceed a 10 year frequency.
8	A baseline visual inspection will be performed on a representative sample of the buried fire protection piping and components, whose internal surfaces are exposed to raw water, to confirm there is no degradation.	Fire Protection Program	Complete
9	Testing a representative sample of fire protection sprinkler heads or replacing those that have been in service for 50 years has been included in Fire Protection Program.	Fire Protection Program	Prior to The Sprinkler Heads Achieving 50 Years Of Service Life Not to Exceed a 10 Year Interval Thereafter
10	The procedures and training for personnel performing General Condition Monitoring inspections and walkdowns will be enhanced to provide expectations that identify the requirements for the inspection of aging effects.	General Condition Monitoring	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
11	In-scope cable found to be submerged will be subject to an engineering evaluation and corrective action. The evaluation of cables having significant voltage found to be submerged in standing water for an extended period of time will be based on appropriate testing (using available technology consistent with NRC positions) of cables that are determined to be wetted for a significant period of time. The Engineering evaluation will also address the appropriate testing requirements for the corresponding ten-year intervals during the period of extended operation. The test will use a proven methodology for detecting deterioration of the insulation system due to wetting. Examples of such tests include power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables, or other appropriate testing. Testing will have acceptance criteria defined in accordance with the specific test identified. Occurrence of degradation that is adverse to quality is entered into the Corrective Action Program.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.
12	The Infrequently Accessed Areas Inspection Program will be established.	Infrequently Accessed Areas Inspection Program	Complete
13	Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), and stress corrosion cracking (PWSCC and IASCC) and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, will be submitted to the NRC for approval.	ISI Program: Reactor Vessel Internals (Reference 15.6-2)	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
14	Millstone will follow the industry efforts investigating the aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals) and identifying the appropriate aging management activities and will implement the appropriate recommendations resulting from this guidance. The revised program description will be submitted at least two years prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in nickel based components per 10 CFR 54.21(a)(3).	Alloy 600 Management Program	At Least Two Years Prior to the Period of Extended Operation Complete
15	The existing inspection program will be modified to include those lifting devices that require monitoring for license renewal, but are not already included in the program.	Inspection Activities: Load Handling Cranes and Devices	Complete
16	Implementing procedures and documentation will be modified to include visual inspections for the loss of material on the crane and trolley structural components and the rails in the scope of license renewal in Commitment 15.	Inspection Activities: Load Handling Cranes and Devices	Complete
17	The implementing procedures will be modified to include ACI 349.3R-96 and ANSI/ASCE 11-90 as references and as input documents for the inspection program.	Structures Monitoring Program	Complete
18	The Structures Monitoring Program and implementing procedures will be modified to include all in-scope structures.	Structures Monitoring Program	Complete
19	Groundwater samples will be taken on a periodic basis, considering seasonal variations, to ensure that the groundwater is not sufficiently aggressive to cause the below-grade concrete to degrade.	Structures Monitoring Program	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
20	The Structures Monitoring Program and implementing procedures will be modified to alert the appropriate engineering organization if the structures inspections identify that medium voltage cables in the scope of license renewal have been submerged.	Structures Monitoring Program	Complete
21	The maintenance and work control procedures will be revised to ensure that inspections of inaccessible areas are performed when the areas become accessible by such means as excavation or installation of shielding during maintenance or for any other reason.	Structures Monitoring Program	Complete
22	Appropriate inspections of sealants and caulkings used for moisture intrusion prevention in and around aboveground tanks will be performed.	Tank Inspection Program	Complete
23	Non-destructive volumetric examination of the in-scope inaccessible locations, such as the external surfaces of tank bottoms, will be performed prior to the period of extended operation. Subsequent inspections will be performed on a frequency consistent with scheduled tank internals inspection activities.	Tank Inspection Program	Applicable inspections have been performed. Subsequent inspections will be performed on a frequency consistent with scheduled tank internal inspection activities.
24	The security diesel fuel oil tank and diesel fire pump fuel oil tank are in-scope for license renewal and will be included on the respective Tank Inspection Program inspection plan.	Tank Inspection Program	Complete
25	Changes will be made to maintenance and work control procedures to ensure that inspections of plant components and plant commodities will be appropriately and consistently performed and documented for aging effects during maintenance activities.	Work Control Process	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
26	Dominion actively participates in a comprehensive industry initiative, in response to NRC Generic Issue 23 (GI-23), "Reactor Coolant Pump Seal Failure." Dominion is following the industry efforts on this issue and will implement the appropriate recommendations resulting from this guidance prior to the period of extended operation.	Environmental Report - SAMA Analysis	Complete
27	For potentially susceptible CASS materials, either enhanced volumetric examinations or a unit or component specific flaw tolerance evaluation (considering reduced fracture toughness and unit specific geometry and stress information) will be used to demonstrate that the thermally-embrittled material has adequate fracture toughness in accordance with NUREG-1801 Section XI.M12.3.	Inservice Inspection Program: Systems, Components and Supports	Complete
28	Millstone will follow industry efforts that will provide specific guidance to license renewal applicants for evaluating the environmental effects of fatigue on applicable locations, other than those identified in NUREG/CR-6260. Millstone will also implement the appropriate recommendations resulting from this guidance.	Environmentally Assisted Fatigue	Complete
29	A baseline visual inspection will be performed of the accessible areas of the shell side (including accessible portions of the exterior side of the tubes) of one: • Millstone Unit 2 Reactor Building Closed Cooling Water heat exchanger, • Millstone Unit 2 Emergency Diesel Generator Jacket Cooling Water heat exchanger, and • Millstone Unit 3 Emergency Diesel Generator Jacket Cooling Water heat exchanger.	Closed-Cycle Cooling Water System	Complete

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

Item	Commitment	Source	Schedule [a]
30	Using the Work Control Process, a baseline inspection for the loss of material due to selective leaching will be performed on a representative sample of locations for susceptible materials by visual, and mechanical or other appropriate methods prior to entering the period of extended operation.	Work Control Process	Complete
31	A review of the Work Control Process inspection opportunities for each material and environment group, supplemental to the initial review conducted during the development of the LRA, will be performed. Baseline inspections will be performed for the material and environment combinations that have not been inspected as part of the Work Control Process.	Work Control Process	Complete
32	Calibration results for cables tested in situ will be reviewed to detect severe aging degradation of the cable insulation. The initial review will be completed prior to entering the period of extended operation and will include at least 5 years of surveillance test data for each cable reviewed. Subsequent reviews will be performed on a period not to exceed 10 years.	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Prior to period of extended operation complete. Subsequent reviews will not exceed a 10 year frequency.
33	The in scope cables in Unit 3 duct lines number 929 (SBO Diesel to Unit 3 4.16kV Normal Switchgear) and number 973 (RSST 3RTX-XSR-B to 6.9kV Normal Switchgear Bus 35A, 35B, 35C and 35D) will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.
34	In addition to the testing specified in Commitment 33, a representative sample of in-scope medium voltage cables will be tested to demonstrate that water treeing will not prevent the cables from performing their intended function.	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Prior to Period of Extended Operation Not to Exceed a 10 Year Frequency Thereafter

TABLE 15.6-1 LICENSE RENEWAL COMMITMENTS (CONTINUED)

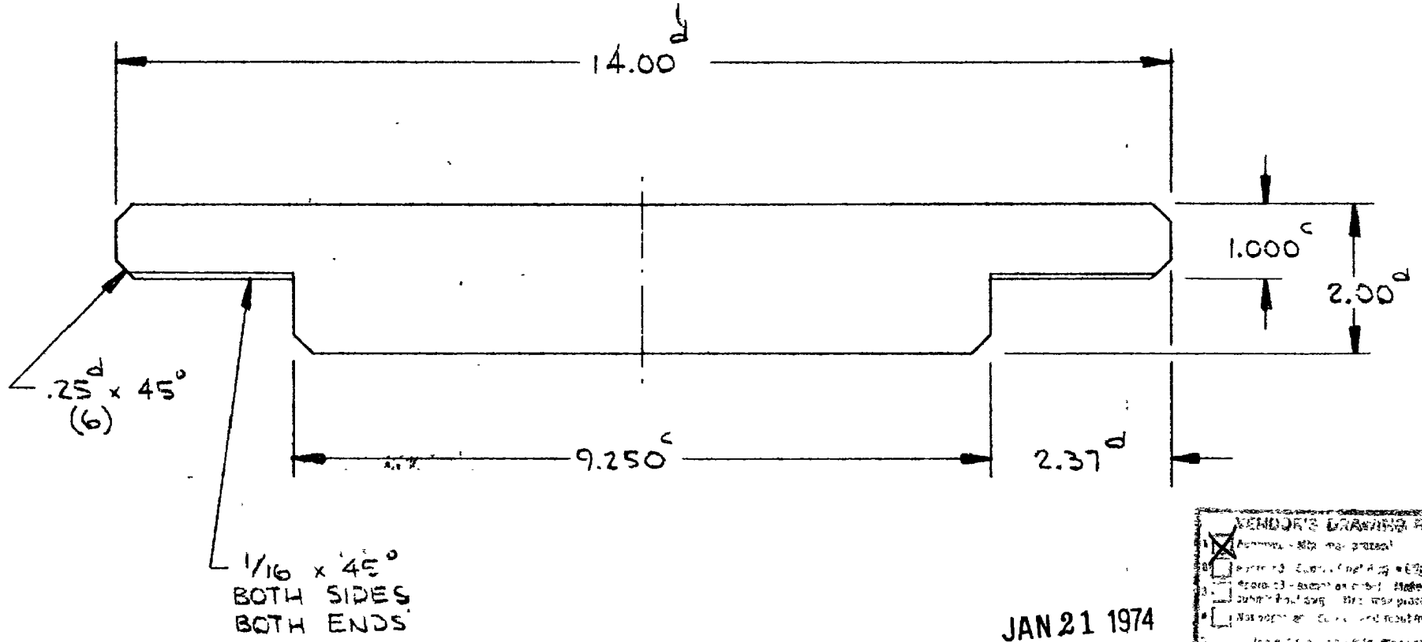
Item	Commitment	Source	Schedule [a]
35	Millstone Unit 2 will complete the SAMA evaluation of the capability to flash the Diesel Generator field in the event of extended loss of DC power with a loss of offsite power. If this SAMA is cost beneficial (i.e., can be accomplished without a hardware modification), a Severe Accident Management Guideline (SAMG) addressing this mitigation strategy will be developed.	Severe Accident Mitigation Alternatives (SAMA) Analysis (Reference 15.6-1)	Complete
36	Dominion will replace the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC.	Inservice Inspection Program: Systems, Components and Supports	Complete
37	Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, will be submitted to the NRC for review at least two years prior to the period of extended operation.	Reactor Vessel Neutron Embrittlement	"Prior to period of extended operation." Complete. Subsequent testing will not exceed a 10 year frequency.

a. The Period of Extended Operation is the period of 20 years beyond the expiration date of the unit's previous 40 year operating license.

Millstone Power Station Unit 2
Drawings Incorporated by Reference

25203-10003 SH-0000000
Contains Redacted Information

REVISIONS				
ECO NO	REV	DESCRIPTION	DATE	APPROVAL



JAN 21 1974

VENDOR'S DRAWING REVIEW
Approved - Skip mfr. process
Approved - Submit certified prints. Make changes and perform fabrication. Hold mfr. process for approval
Approved - Submit prints for approval. Make changes and perform fabrication. Hold mfr. process for approval
Not approved. Submit revised prints
By N. Jr. Date 3/26/74
BECHTEL
JOB NO. 7604 | BECHTEL CORPORATION POWER & INDUSTRIAL DIVISION P.O. BOX [illegible] SAN FRANCISCO, CALIF.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

7604-C84B-11-2

D-15526-B

NON Q.A.

	PO#							
	0203 1927	1	09/25 /01	INCORPORATE DCN'S DM2-00-0025-00	RJH	GHH	TP	CAM
MF	P.A. #	NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.

THE MILLSTONE POINT COMPANY
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BECHTEL-C-84B

PROGRAMMED AND REMOTE SYSTEMS CORP.
[illegible] WEST HIGHWAY 96 ST. PAUL, MINNESOTA 55112 TELEPHONE 484-7661 AREA CODE 61[illegible]
Amplifying the reach and strength of man

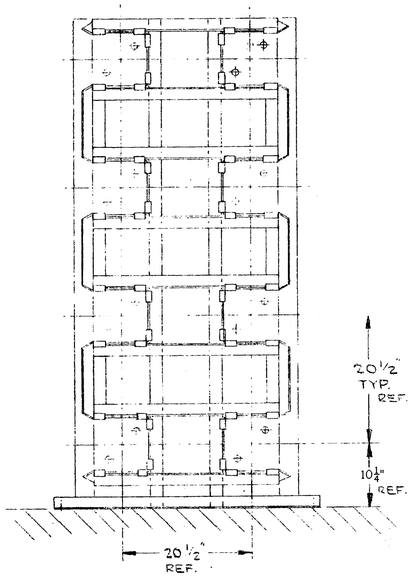
MATERIAL 1/4" FR 304 ST. STL. ASTM A-240	UNLESS OTHERWISE SPECIFIED TOLERANCES IN INCHES ARE: a = ± .005 d = ± .060 b = ± .010 e = ± .120 c = ± .030
HEAT TREATMENT	FINISH

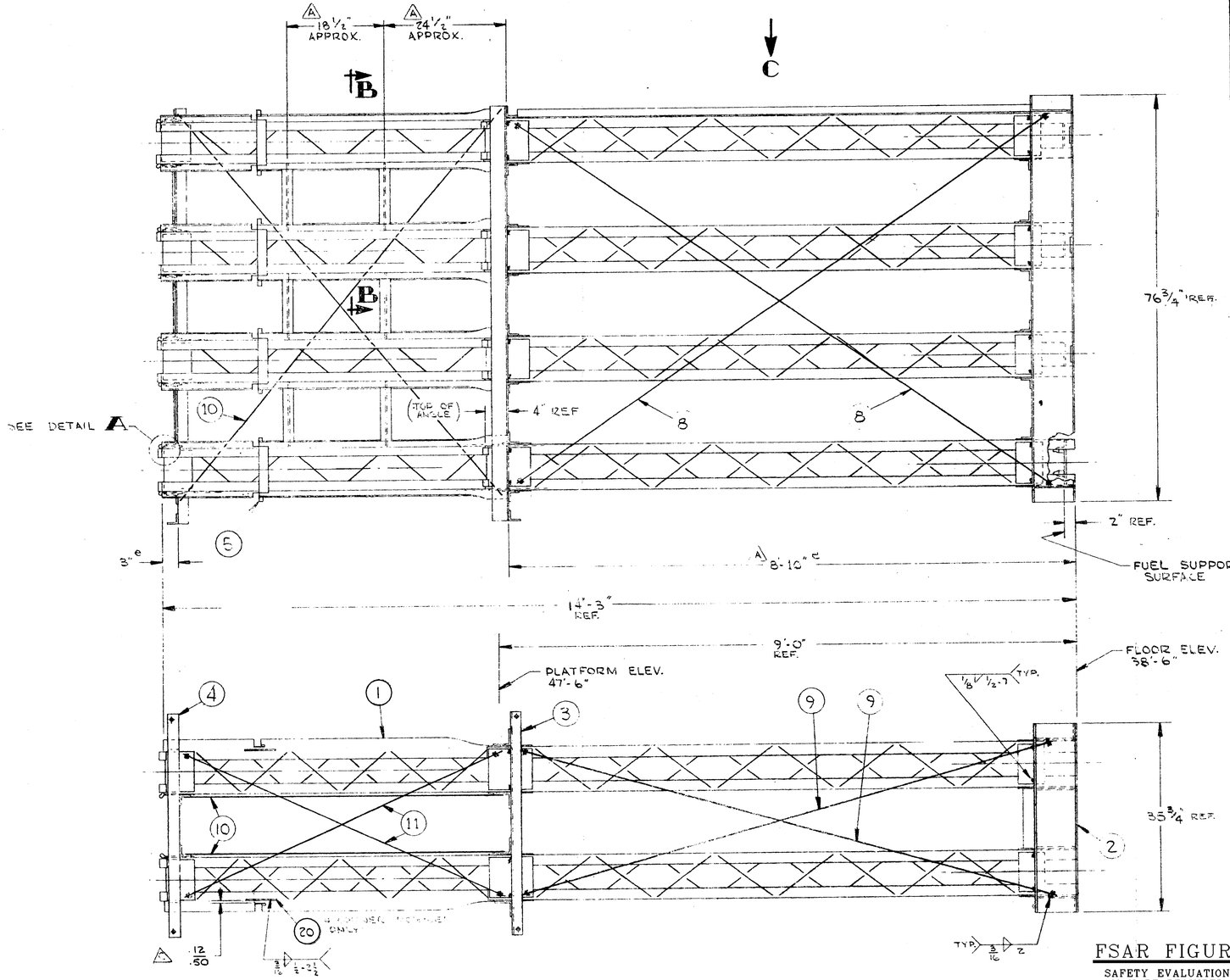
ORIG. DATE 1-24-73	NAME CLOSURE BAR NEW FUEL STORAGE RACKS
DR [illegible]	
CHK DLN	
ENG [illegible]	
APP [illegible]	SCALE 1/2 WT

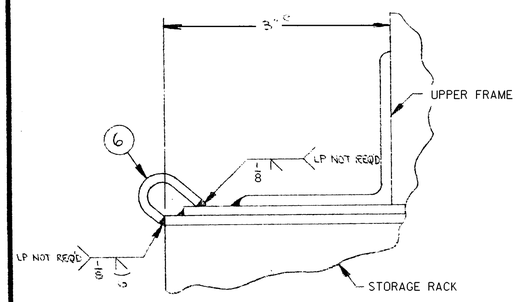
NEXT ASS'Y	STORAGE RACK ASS'Y
PROJECT NO	72414
SHEET 1 OF 1	
DWG. NO.	25203-13013 SH. 10

25203-13013 SH 0000001

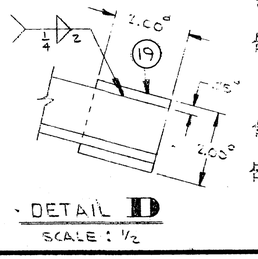
REVISIONS				
ECO NO	REV.	DESCRIPTION	DATE	APPROVAL
	A	R-11" WAS 9'-1" TO TOP OF ANGLE, 24½" WAS 18⅞", 18½" WAS 18⅜"	6-29-73	
	B	ADDED ITEMS #16 THRU #20; QTY. ITEM #9 WAS 4; ADDED VIEW "C" & DET. "D" .50 MAX. ON ITEM #10 WAS .50d	9-6-73	
	C	REVISED WELD ON #16 TO #18 (VIEW C); [illegible] WAS .50d	10-8-73	



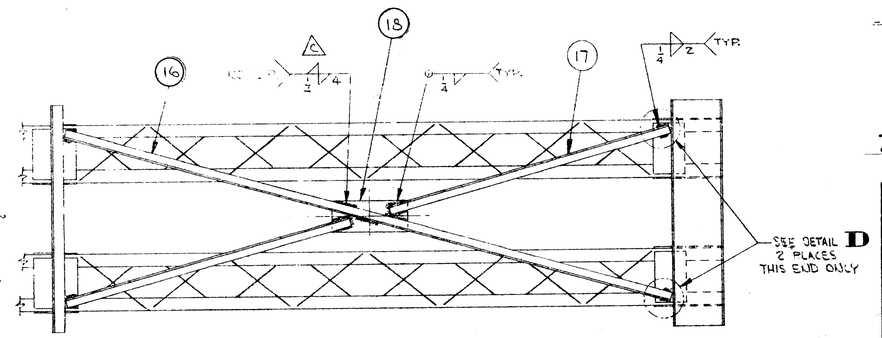




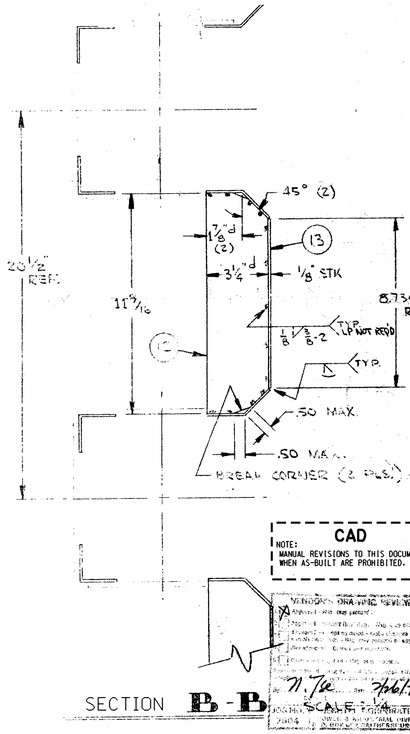
DETAIL A
SCALE: FULL
TYP: 24 PLACES



DETAIL D
SCALE: ½



VIEW C



SECTION B-B

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

NOTES:
1. ALL WELDS PER PaR QCP-1
2. MECHANICAL CLEANING OF ALL WELDS PER PaR QCP-17.
3. LIQUID PENETRANT INSPECT ALL WELDS PER PaR QCP-16 EXCEPT AS NOTED.
4. SOLVENT CLEAN PER PaR QCP-17, CLASS A

NON Q.A.

7604-C84B-15-4 JAN 21 1974
A-15043-D

MF	P.A.*	NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
	PO# 0203 1927	1	09/25/01	INCORPORATE DCN DM2-00-0025-00	RJH	GHH	TP	CAM

THE MILLSTONE POINT COMPANY
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BECHTEL-C-84B

PROGRAMMED AND REMOTE SYSTEMS CORP.
899 WEST HIGHWAY 96 ST. PAUL, MINNESOTA 55112 TELEPHONE 484-7361 AREA CODE 612
Amplifying the reach and strength of man

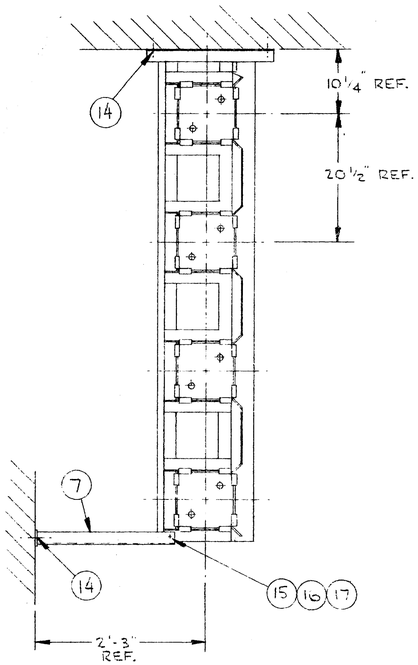
MATERIAL	UNLESS OTHERWISE SPECIFIED TOLERANCES IN INCHES ARE: a = ± .005 d = ± .060 b = ± .010 e = ± .120 c = ± .030	ORIG. DATE 9-18-73
		DR
HEAT TREATMENT	FINISH	CHK
		ENG
		APP

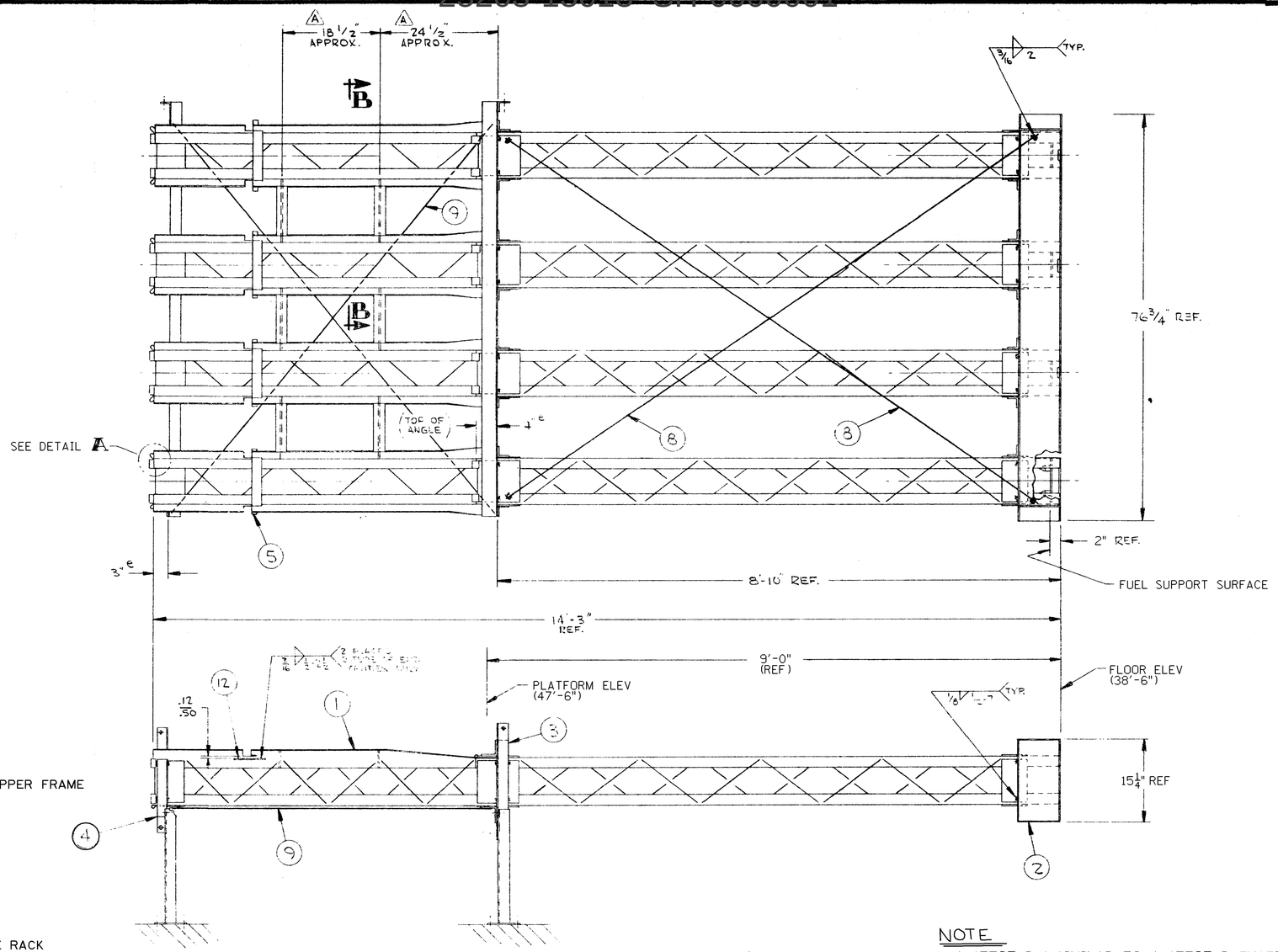
NAME: NEW FUEL STORAGE RACK ASSY (8 RACK MODULE)
SCALE 1"=1'-0" WT 1540

NEXT ASS'Y. I-15528-C
PROJECT NO. 72414
SHEET 1 OF
DWG. NO. 25203-13013 SH. 14

25203-13013 SH-0000001

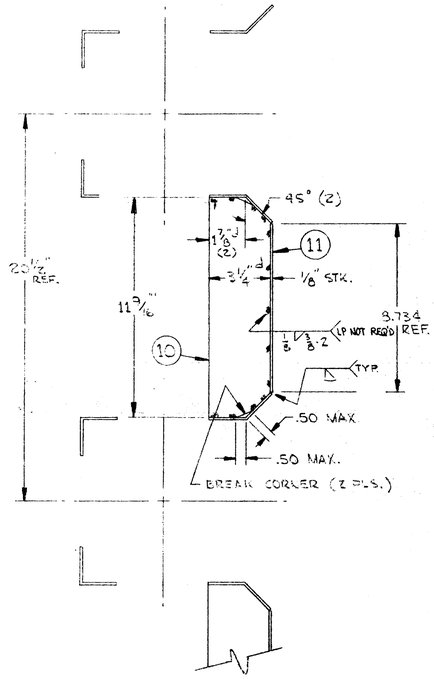
REVISIONS				
ECO NO	REV.	DESCRIPTION	DATE	APPROVAL
.	A	24½" WAS 19 5/16" TO TOP OF ANGLE, 18½" WAS 18 5/16"	6-29-73	
	B	ADDED ITEM # 17; .50 MAX. ON ITEM #10 WAS .50↕; 13/50 WAS .50↕	9-7-73	





DETAIL A
TYP. 24 PLACES

SCALE: FULL



SECTION B-B SCALE: ¼

NOTES:

1. ALL WELDS PER PaR QCP-1
2. MECHANICAL CLEANING OF ALL WELDS PER PaR QCP-17
3. LIQUID PENETRANT INSPECT ALL WELDS PER PaR QCP-16 EXCEPT AS NOTED.
4. SOLVENT CLEAN PER PaR QCP-17, CLASS A

NOTE

A-15527-D-1 SIMILAR TO A-15527-D EXCEPT OPPOSITE HANG POSITIONING OF ITEMS #3 &#4

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

JAN 21 1974

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

7604-C84B-20-3

A-15527-D

P.O.#	P.A.#	NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
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THE MILLSTONE POINT COMPANY
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
BECHTEL-C-84B

PROGRAMMED AND REMOTE SYSTEMS CORP.
899 WEST HIGHWAY 96 ST. PAUL, MINNESOTA 55112 TELEPHONE 484-7261 AREA CODE 612
Amplifying the reach and strength of man

MATERIAL	UNLESS OTHERWISE SPECIFIED TOLERANCES IN INCHES ARE: a = ± .005 d = ± .060 b = ± .010 e = ± .120 c = ± .030	ORIG. DATE 1-22-73	NAME: NEW FUEL STORAGE RACK ASS'Y (4 RACK MODULE)
HEAT TREATMENT	FINISH	DR [illegible]	
		CHK DLN	
		ENG [illegible]	
		APP [illegible]	
		SCALE 1"=1'-0" WT 750	

NEXT ASS'Y. 15528
PROJECT NO. 72414
SHEET 1 OF 1
DWG. NO. 25203-13013 SH. 19

25203-17001 SH-0000000
Contains Redacted Information

25203-17002 SH-0000000
Contains Redacted Information

25203-17003 SH-0000000
Contains Redacted Information

25203-17004 SH-0000000
Contains Redacted Information

25203-17005 SH-0000000
Contains Redacted Information

25203-17006 SH-0000000
Contains Redacted Information

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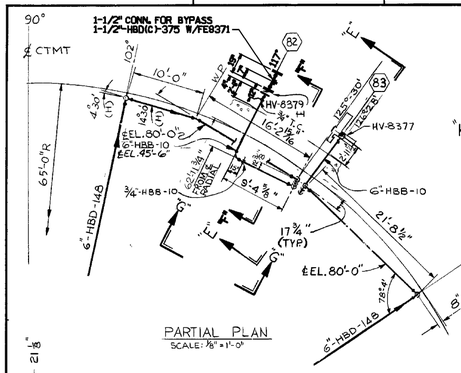
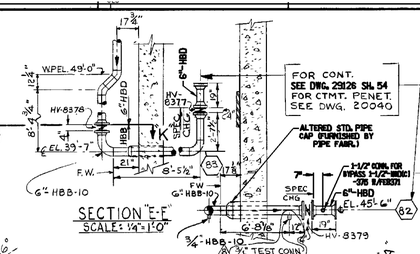
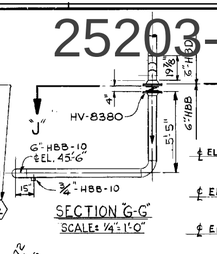
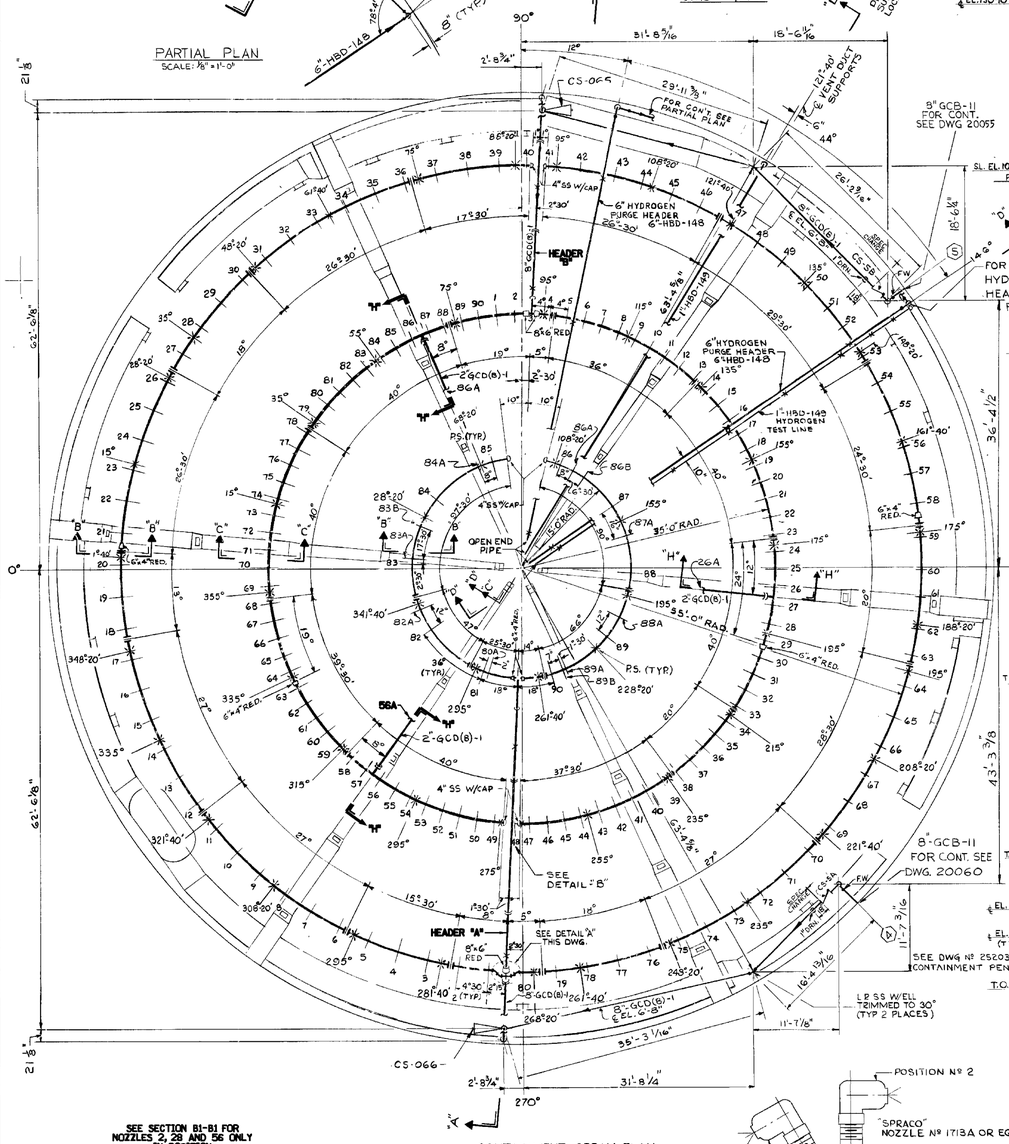
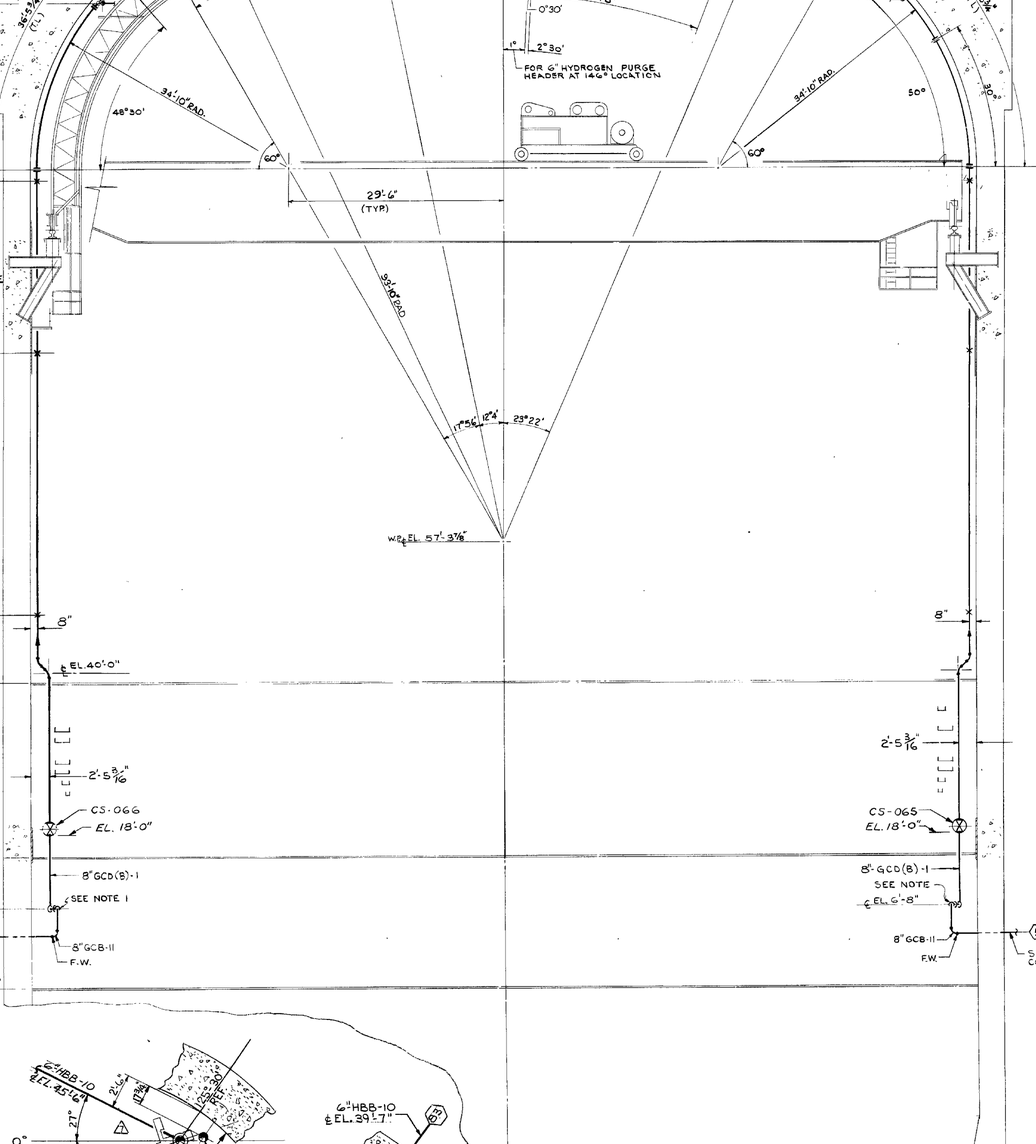
25203-20104 SH-0000000

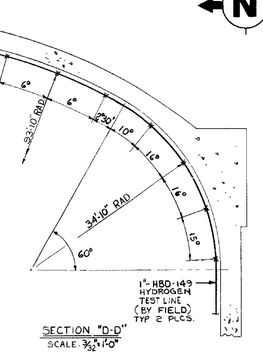


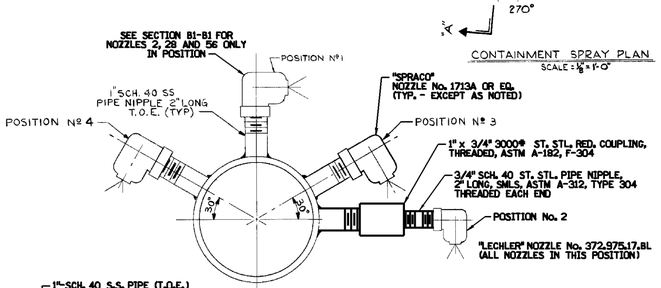


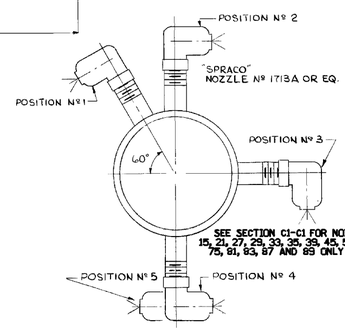


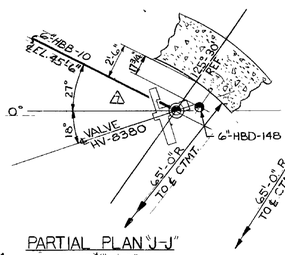


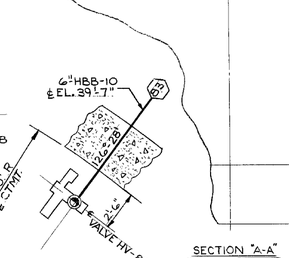


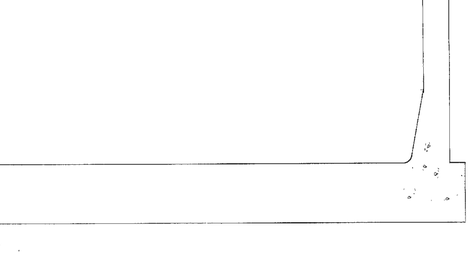


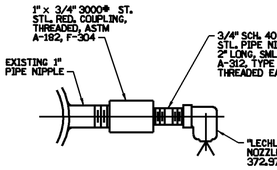


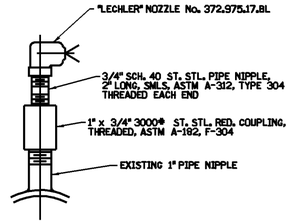


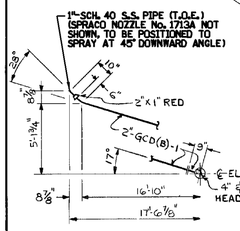












THE BELOW LISTED SPRAY NOZZLES ARE "SPRACO NO. 1713A" OR "LECHLER NO. 372.975.17.BL". WHERE LECHLER NOZZLES ARE USED, THE NOZZLE NUMBERS ARE PRECEDED BY AN ASTERISK (*).

NOZZLES IN POSITION Nº1			NOZZLES IN POSITION Nº2			NOZZLES IN POSITION Nº3	NOZZLES IN POSITION Nº4
*2	30	58	*1	*29	*57	83	83A	80A
4	32	60	*3	*31	*59	84	86B	82A
6	34	62	*5	*33	*61	85	89A	83B
8	36	65	*7	*35	*63	86		84A
10	38	67	*9	*37	*64	87		86A
12	40	70	*11	*39	*66	88		87A
14	42	72	*13	*41	*69	89		89A
16	44	74	*15	*43	*71	90		89B
18	46	76	*17	*45	*73			
20	48	78	*19	*47	*75			
22	50	80	*21	*49	*77			
24	52	83	*23	*51	*79			
26	54		*25	*53	81			
*28	*56		*27	*55	82			

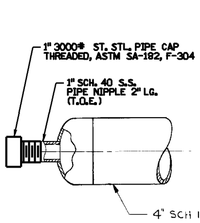
SECTION "B-B"
SCALED 3"=1'-0"
HEADER "A"

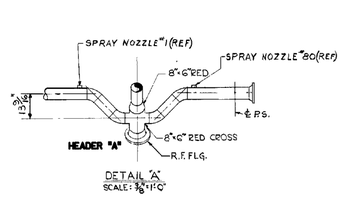
THE BELOW LISTED SPRAY NOZZLES ARE "SPRACO NO. 1713A" OR "LECHLER NO. 372.975.17.BL". WHERE LECHLER NOZZLES ARE USED, THE NOZZLE NUMBERS ARE PRECEDED BY AN ASTERISK (*).

NOZZLES IN POSITION Nº 1			NOZZLES IN POSITION Nº 2			NOZZLES IN POSITION Nº 3		NOZZLES IN POSITION Nº 4		NOZZLES IN POSITION Nº 5	
2	32	62	1			*3	47	*89	73	80	
4	34	64	7			*5	45				
6	36	66	13			9	*51				
8	38	68	19			*11	53				
10	40	70	25			*15	*57				
12	42	72	31			17	*60				
14	44	74	37			*21	63				
16	46	76	43			23	*65				
18		78	49			*27	*69				
20	50	A	55			*29	71				
22	52	82	61			*33	*75				
24	54	84	67			*35	77				
26	56	86	A			*39	*81				
28	58	88	79			41	*83				
30	59	90	85			*45	*87				

SECTION "C-C"
SCALED 3"=1'-0"
HEADER "B"

NOTE: THE POSITION OF NOZZLES 26A, 56A AND 86A, ASSOCIATED WITH HEADER "B", IS INDICATED ON SECTION "H-H"





NOTES
1. 90° OFFSETS ARE NOT ILLUSTRATED IN SECTION "A-A" FOR CLARITY.
2. ALL PIPE SUPPORT LOCATIONS IN CONTAINMENT SPRAY PLAN ARE REFERENCED FROM ℄.
3. FOR HANGER SUPPORT DETAILS SEE DWG. 25203-11172 DETAIL 2.
—X— DENOTES HANGER OR SUPPORT LOCATION.

GENERAL NOTES

1. FOR PIPING CLASS SUMMARY SHEETS SEE 7604-MS-1.
2. FOR PIPING CLASS SHEET SEE 7604-MS-3
3. FOR VALVE INFORMATION SEE MASTER VALVE LIST 7604-MS-7
4. FOR EQUIPMENT LIST SEE 7604-MS-50
5. FOR PIPING & INSTRUMENTATION DIAGRAM SEE DWG Nº 25203-26015 & 25203-26028 SHT 2
6. ALL ELBOWS ARE TO BE LONG RADIUS (1½ D) UNLESS OTHERWISE SPECIFIED.
7. ALL PIPING 2" & UNDER TO BE FIELD RUN FROM THE P&ID'S AND THOSE SHOWN ARE DIAGRAMMITIC ONLY, EXCEPT WHERE SPECIFICALLY LOCATED. USE BENDS WHERE POSSIBLE TO ELIMINATE FITTINGS.

Q.A.

FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

BECHTEL CORPORATION
GAITHERSBURG, MARYLAND
JOB 7604
THE MILLSTONE POINT COMPANY
SUBSIDIARY OF
NORTHEAST UTILITIES

	ECN 0000 1827	3	11-5-01	INCORPORATE DCN DM2-00-0025-00	AGC	GHH	TP	CAM
	N/A	2	5-25-99	INC. DCN DM2-00-0360-97 & DM2-01-0175-99	MCR	SJM	REB	REB
	N/A	1	7/31/81	AS-BUILT PER DCR-M-2.5-G10-81	RE	LM	LAC	
	N69-023	0	11/21/76	ISSUED AS-BUILT	DK	RT CK	DMB APP	
MF	P.A. NO.	NO.	DATE	REVISIONS	BY	CHK	APP	APP

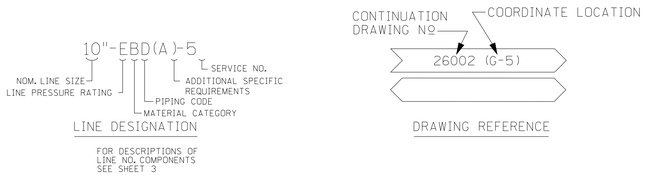
TITLE: MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2
AREA 5 PIPING
CONTAINMENT SPRAY & H_2 PURGE

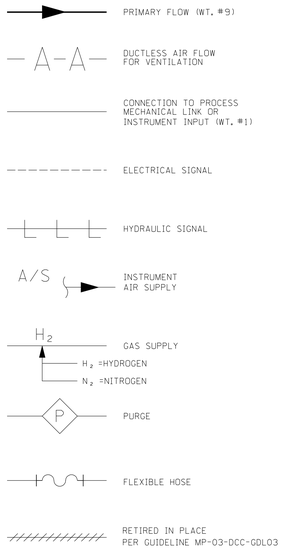
BY N.C.	CHKD AGC	APP'D AGC JOWL	APP GB
DATE 6-2-70	DATE 6-22-70	DATE 4/23/71	DATE 4/29/71
SCALE AS NOTED	MICROFILM DATE 1/20/76	DWG NO. 25203-20104	
P.A.NO. N69-023			

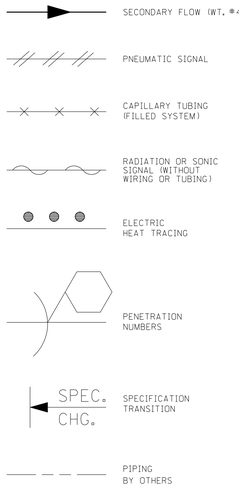
SERV CK MECH CK ELEC CK | CONC CK STRUC CK HYDRL CK | SERV. CK. MECH. CK. ELEC. CK. | CONC CK STRUC CK HYDRL CK | SERV. CK. MECH. CK. ELEC. CK. | CONC CK STRUC CK HYDRL CK | SERV CK MECH CK ELEC CK | CONC CK STRUC CK HYDRL CK | SERV CK MECH CK ELEC CK | CONC CK STRUC CK HYDRL CK

VARIABLES

LINES

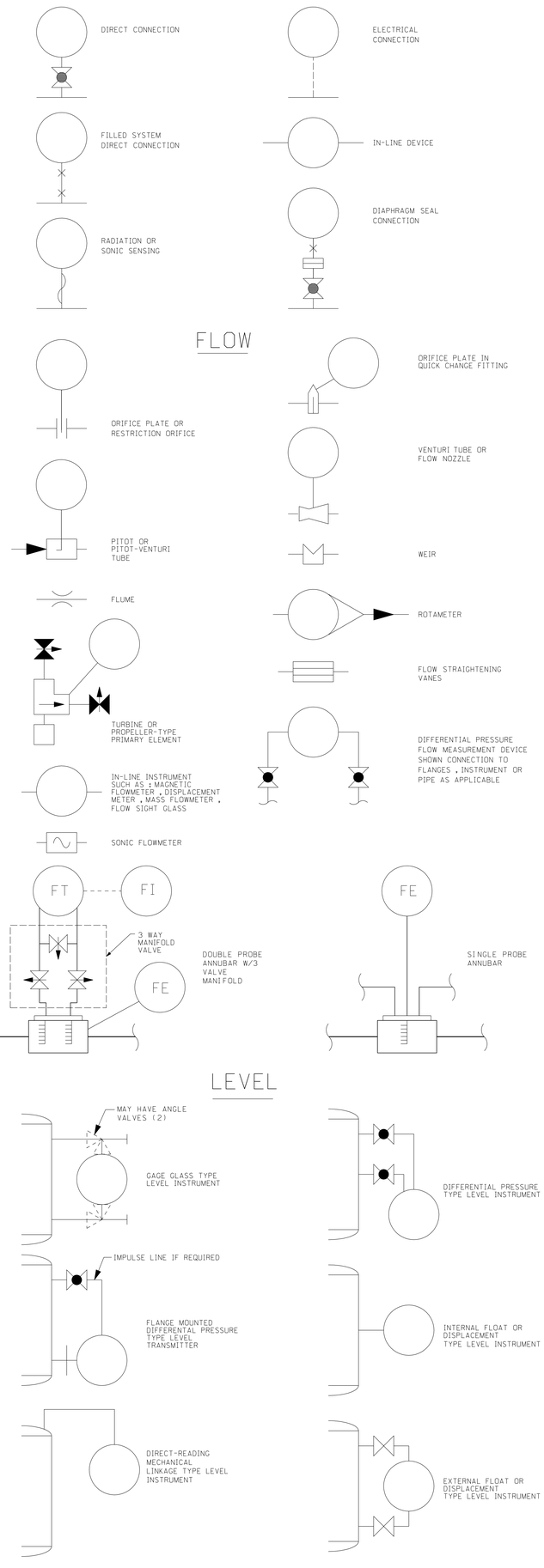






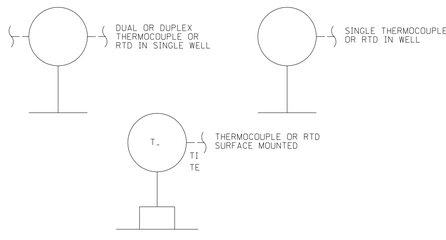
VALVES

TYPICAL CONNECTION - ANY VARIABLE



LEVEL

TEMPERATURE



ABBREVIATIONS (NOTE 9)

ABSOLUTE	ABS.	HYDROGEN	H_2
ALTERNATING CURRENT	A.C.	INCORPORATED	INC.
ACCUMULATOR	ACCUM.	INJECTION	INJ.
ADDITION	ADD.	INSTRUMENT(TATION)	INST.
AUX. EXH. ACTUATION SIGNAL	AEAS.	INSULATING FLANGE	I.F.
AIR COMPRESSOR	AIR.COMP.	LENGTH	LG.
AIR CONDITIONING)	A/C	LOCAL SUMP	LS.
ALTERNATE	ALT.	LOCATION	LOC.
APPROXIMATE	APPROX.	LOCKED CLOSED	L.C.
ARRANGEMENT	ARRGT.	LOCKED OPEN	L.O.
ASSEMBLY	ASSY.	LOW PRESSURE	L.P.
ATMOSPHERE	ATMOS.	LUBRICATION OIL	LUBE OIL
AUTOMATIC	AUTO.	MACHINE	MACH.
AUXILIARY	AUX.	MAIN	MN.
AVERAGE	AVG.	MANHOLE	M.H.
BEYOND DESIGN BASIS	B.D.B.	MANUFACTURER	MFR.
BLIND	BLD.	MAXIMUM	MAX.
BEARING	BRG.	MAIN CONTROL CENTER	M.C.C.
BREAKER	BKR.	MECHANICAL	MECH.
BUILDING	BLDG.	MINIMUM	MIN.
CONTROL ELEMENT DRIVE MECHANISM	CEDM	MISCELLANEOUS	MISC.
CONTROL ELEMENT DRIVE ASSY.	CEDA.	MOISTURE	MOIST.
CHANGE	CHG.	NOMINAL	NOM.
CONTAINMENT ISOLATION ACTUATION SIGNAL	CIAS.	NORMALLY CLOSED	N.C.
		NORMALLY OPEN	N.O.
CHEMICAL	CHEM.	NUMBER	NO.
CHLORINATION	CHLOR.	OPERATING	OP.
CIRCULATE(ING)	CIRC.	OUTSIDE AIR	O.A.
COLUMN	COL.	OUTSIDE DIAMETER	O.D.
COMPRESSOR	COMPR.	PANEL	PNL.
CONCRETE	CONC.	PENETRATION	PENT.
CONDENSATE	CONDS.	PRIMARY DRAIN TANK	PDT.
CONDENSER	COND.	PIPING & INSTRUMENT DIAGRAM	P & ID
CONNECTION	CONN.	PLATFORM	PLAT.
CONTAINMENT	CTMT.	PRIMARY WATER	PMW.
CONTAMINATED	CONTAM.	POWER	PWR.
CONTINUATION	CONT.	PRESSURE(IZED)	PRESS.
CONTROL VALVE	C.V.	PRIMARY	PRI.
COORDINATE	COORD.	PUMP	PP.
CHEMICAL & VOLUME CONTROL SYS.	CVCS.	PURIFICATION	PURIF.
DAMPER	DMPR.	QUENCH TANK	QT.
DEAERATOR	DEAER.	RADIATION	RAD.
DEBORATING	DEBOR.	REACTOR COOLANT SYSTEM	RCS.
DIRECT CURRENT	D.C.	REACTOR	REAC.
DEMINERALIZER(IED)	DEMIN.	RECEIVER	RCVR.
DETAIL	DET.	REDUCER	RED.
DIAMETER	DIA.	REINFORCE	REINF.
DIESEL GENERATOR	D.G.	REFERENCE	REF.
DISCHARGE	DISCH.	REHEAT	RHT.
DISTRIBUTION	DISTR.	RELIEF VALVE	R.V.
DOWN	DN.	REQUIRED	REQD.
DRAIN	DR./D.	RETURN	RET.
DRAWING	DWG.	REVISION (REVISE)	REV.
EACH	EA.	REACTOR BUILDING CLOSED COOLING WATER	RBCCW.
ENCLOSURE BUILDING FILTRATION ACTUATION SIGNAL	EBFAS.	RECIRCULATION	RECIRC.
		RETURN AIR	R.A.
ELECTRO HYDRAULIC CONTROL	EHC.	RADWASTE SYSTEM	RWS.
EJECTOR	EJECT.	SAFETY VALVE	S.V.
ELEVATION	EL.	SAFETY INJECTION ACTUATION SYSTEM	SIAS.
EMERGENCY	EMER.	SEALED CLOSED	S.C.
ENGINE	ENG.	SEALED OPEN	S.O.
EQUIPMENT	EQUIP.	SEPARATOR	SEP.
EVAPORATOR	EVAP.	SERVICE	SERV.
EXCHANGER	EXCH.	SUMP RECIRCULATOR ACTUATION SIGNAL	SRAS.
ENCLOSURE	ENCL.	SHEET	SH.
ELECTRIC WATER COOLER	EWC.	SHIELD(ING)	SHLD.
ETHANOLAMINE	ETA	STEAM GENERATOR FEED PUMP	SGFP.
EXCITER	EXC.	SOCKET WELD	S.W.
EXHAUST	EXH.	SPECIFICATION	SPEC.
EXPANSION JOINT	EXP. JT.	STAINLESS STEEL	SST.
EXTRACTION	EXTR.	STANDARD	STD.
FAIL CLOSED	F.C.	STATION	STA.
FAIL LOCKED/FAIL AS-IS	F.L.	STEAM	STM.
FAIL OPEN	F.O.	STEEL	STL.
FILTER	FILT.	STRAINER	STRN.
FLANGE	FLG.	STRUCTURE	STRUCT.
FLOOR	FL.	SUCTION	SUCT.
FOOT , FEET	FT.	SYSTEM	SYS.
FOUNDATION	FDN.	SAMPLING	SAMP.
FUTURE	FUT.	TANK	TK.
GALLON	GAL.	TEMPERATURE	TEMP.
GENERATOR	GEN.	TEST CONNECTION	TC
GROUND	GRD.	THERMAL MARGIN/LOW PRESSURE TRIP	TM/LPT.
HEADER	HDR.	THREADED	THRD.
HEAT	HT.	TRANSFER , TRANSFORMER	TRANS.
HEATER	HTR.	TURBINE	TURB.
HEATING & VENTILATING	H & V	TURBINE GENERATOR	TURB. GEN.
HEATING VENTILATION AND A/C	HVAC.	TYPICAL	TYP.
HIGH LEVEL	HI.LEV.	TYPICAL	TYP.
HIGH PRESSURE	H.P.	TURBINE BUILDING CLOSED COOLING WATER	TBCCW.
HIGH TEMPERATURE	HI. TEMP.	VOLUME CONTROL TANK	VCT.
HORIZONTAL	HOR.	VENTILATION	VENT.
HOT WATER	H.W.	WATER	WTR.
HOSE STATION	HS.	VENT	V.
HYDRAULIC	HYD.		

NOTES

1. DELETED
2. FOR PIPING & MATERIAL SPEC. SHEET SEE MS-1 AND MS-3.
3. FOR PROJECT MASTER VALVE IDENTIFICATION SEE MS-7.
4. FOR EQUIPMENT NUMBER SEE EQUIPMENT LIST MS-50.
5. FOR INSTRUMENT INDEX SEE MS-60.
6. DELETED
7. FOR ALL DRAIN LINE GLOBE VALVES SHOWN, GATE VALVES MAY BE USED REGARDLESS OF VALVE SYMBOL ON THE P & ID.
8. FOR LINE VENT AND DRAIN DETAILS SEE 7604-MS-6, P&ID DWG'S ONLY REFLECT PIPE CAPS THAT WERE INSTALLED AT THE TIME OF SYSTEM WALKDOWN, DUE TO SYSTEM MAINTENANCE AND OPERATION THESE CAPS MAY NO LONGER BE INSTALLED. OPERATIONS DEPARTMENT MAY REPLACE MISSING PIPE CAPS USING THE GUIDANCE PROVIDED PER ENGINEERING RECORD CORRESPONDENCE ER-97-011 & ER-97-011 REV.1. DESIGN ENGINEERING WILL UPDATE P & ID DWG'S TO REFLECT PIPE CAPS INSTALLED PER ER-97-011 & ER-97-011 REV.1 UPON NOTIFICATION BY OPERATIONS.
9. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
10. P & ID DWG'S REFLECT LOCKED VALVES IN ACCORDANCE WITH OPERATIONS DEPARTMENT INSTRUCTIONS 2-OPS-1.32 - LOCKED VALVES. P & ID DWG'S REFLECT VALVES IN THE POSITION REQUIRED FOR 100% SYSTEM OPERATION, DETERMINED BY VALVE LINE-UPS, SYSTEM OPERATING PROCEDURES, TECH. SPEC. AND DESIGN BASIS. P & ID DWG'S ARE NOT TO BE USED TO MAINTAIN OPERATIONAL CONFIGURATION CONTROL OF VALVES, SYSTEM OPERATING PROCEDURES SHALL BE USED WHEN DETERMINING PROPER VALVE POSITIONS. P & ID DWG'S WILL BE UPDATED TO REFLECT CHANGES IN VALVE POSITIONS AS REQUIRED BY CHANGES TO SYSTEM OPERATING PROCEDURES, VALVE LINE-UPS, TECH. SPECS, DESIGN BASIS AND ODI 2-OPS-1.32.
11. IT SHALL BE NOTED THAT NOT ALL VALVES DOWNSTREAM OF ROOT STOP VALVES HAVE VALVE ID NO'S, VALVES DOWNSTREAM OF ROOT STOP VALVES THAT HAVE BEEN ASSIGNED VALVE ID NO'S HAVE BEEN DONE SO AT THE REQUEST OF THE OPERATIONS DEPT. PER PORC COMMITMENT 295-030. DESIGN ENGINEERING WILL IDENTIFY VALVES DOWNSTREAM OF ROOT STOP VALVES ON A CASE BY CASE BASIS WHERE THERE IS A NEED/JUSTIFICATION OR AS REQUESTED BY OPERATIONS.
12. NOT ALL TAKE-OFF LINES ON P&ID'S ARE SHOWN IN THE SEQUENCE OF FLOW, AS THEY APPEAR IN THE FIELD. TAKE-OFFS ARE SHOWN DIAGRAMMATICALLY, WHEN SHOWING THEM IN SEQUENCE OF FLOW WILL RESULT IN A CONFUSING REPRESENTATION. TAKE-OFF LINES ARE NOT SHOWN DIAGRAMMATICALLY WHEN THERE IS EQUIPMENT BETWEEN THE TAKE-OFF LINE AND THE MAIN LINE, WHICH COULD LEAD TO IMPROPER ISOLATION OF EQUIPMENT. CHANGING P&ID'S TO SHOW TAKE-OFF LINES IN SEQUENCE OF FLOW WILL BE DONE AT THE DISCRETION OF THE DESIGN SUPERVISOR, ON A CASE BY CASE BASIS WITH REGARD TO THE IMPACT OF THE ASSOCIATED PIPING REPRESENTED ON THE P&ID.

REFERENCE DRAWINGS

25203-26001	SH.1	LEGEND
25203-26001	SH.2	LEGEND
25203-26001	SH.3	LEGEND
25203-26002	SH.1	MAIN STEAM FROM GENERATORS
25203-26002	SH.2	MAIN STEAM GENERATOR BLOWDOWN SYSTEM
25203-26002	SH.3	MAIN STEAM TURBINE & REHEATER NO. 1A
25203-26002	SH.4	MAIN STEAM TURBINE & REHEATER NO. 1B
25203-26002	SH.5	STEAM GENERATOR WET LAYUP RECIRCULATION SYSTEM
25203-26003	SH.1	L.P. TURBINE 1A & MOISTURE SEP. 1A REHEATER DRAINS & VENTS
25203-26003	SH.2	L.P. TURBINE 1B & MOISTURE SEP. 1B REHEATER DRAINS & VENTS
25203-26004	SH.1	HEATER DRAINS & VENTS SYSTEM
25203-26004	SH.2	HEATER DRAINS & VENTS SYSTEM
25203-26004	SH.3	HEATER DRAINS & VENTS SYSTEM
25203-26005	SH.1	CONDENSATE SYSTEM
25203-26005	SH.2	FEED SYSTEM
25203-26005	SH.3	CONDENSATE STORAGE & AUX. FEED
25203-26005	SH.4	CONDENSATE SYSTEM
25203-26006	——	TURBINE GLAND SEAL & DRAIN SYSTEM
25203-26007	SH.1	TBCCW SYSTEM
25203-26007	SH.2	TBCCW SYSTEM
25203-26007	SH.3	TBCCW SYSTEM
25203-26008	SH.1	CIRCULATING WATER
25203-26008	SH.2	SERVICE WATER
25203-26008	SH.3	SERVICE WATER TO VITAL AC SWITCHGEAR COOLING COIL & AC CHILLER
25203-26008	SH.4	SCREENWASH AND SODIUM HYPOCHLORITE
25203-26009	SH.1	INSTRUMENT AIR SYSTEM COMPRESSORS F-3E & F-3F
25203-26009	SH.2	INSTRUMENT AIR SYSTEM
25203-26009	SH.3	INSTRUMENT AIR SYSTEM
25203-26009	SH.4	INSTRUMENT AIR SYSTEM
25203-26009	SH.5	INSTRUMENT AIR SYSTEM
25203-26009	SH.6	INSTRUMENT AIR SYSTEM
25203-26009	SH.7	INSTRUMENT AIR SYSTEM
25203-26009	SH.8	INSTRUMENT & STATION AIR SYSTEM
25203-26009	SH.9	INSTRUMENT AIR COMPRESSOR F-3D
25203-26009	SH.10	STATION AIR - INTAKE STRUCTURE & CONTAINMENT
25203-26009	SH.11	STATION AIR - TURBINE BLDG.
25203-26010	SH.1	FUEL OIL SYSTEM
25203-26011	SH.1	FIRE PROTECTION
25203-26011	SH.2	DOMESTIC WATER
25203-26011	SH.3	HOSE STATION DETAILS & LEGEND
25203-26011	SH.4	HALON AND EXCITER CO_2 FIRE PROTECTION
25203-26011	SH.5	FIRE PROTECTION SYSTEM PIPING DETAILS
25203-26011	SH.6	DOMESTIC WATER SYSTEM
25203-26011	SH.7	FIRE PROTECTION AUTO SPRINKLER VALVE DETS.
25203-26012	——	COND. AIR REMOVAL WATER BOX PRIMING & TURBINE BLDG. SUMPS
25203-26013	——	TURBINE LUBE OIL SYSTEM
25203-26014	SH.1	REACTOR COOLANT SYSTEM
25203-26014	SH.2	REACTOR COOLANT SYSTEM
25203-26014	SH.3	REACTOR COOLANT PUMP
25203-26015	SH.1	L.P. SAFETY INJECTION SYSTEM
25203-26015	SH.2	HIGH PRESSURE SAFETY INJECTION PUMPS
25203-26015	SH.3	SAFETY INJECTION TANKS
25203-26017	SH.1	CHARGING SYSTEM
25203-26017	SH.2	DEBORATING & PURIFICATION SYSTEM
25203-26017	SH.3	BORIC ACID SYSTEM
25203-26018	SH.2	EMERGENCY DIESEL GENERATOR H-7A LUBE OIL, AIR & JACKET WATER COOLING
25203-26018	SH.3	EMERGENCY DIESEL GENERATOR H-7B LUBE OIL, AIR & JACKET WATER COOLING
25203-26018	SH.4	DIESEL GENERATOR H-7A & H-7B STARTING AIR
25203-26018	SH.5	DIESEL GENERATOR STARTING AIR
25203-26020	SH.1	CLEAN LIQUID RADWASTE SYSTEM
25203-26020	SH.2	CLEAN LIQUID RADWASTE SYSTEM
25203-26020	SH.3	CLEAN LIQUID RADWASTE SYSTEM
25203-26020	SH.4	CLEAN LIQUID RADWASTE SYSTEM
25203-26020	SH.5	CLEAN LIQUID RADWASTE SYSTEM
25203-26021	SH.1	AERATED LIQUID RADWASTE SYSTEM
25203-26021	SH.2	GASEOUS RADWASTE SYSTEM
25203-26021	SH.3	SPENT RESIN RADWASTE SYSTEM
25203-26022	SH.1	R.B.C.C.W. SYSTEM R.B.C.C.W. PUMPS & HEAT EXCHANGERS
25203-26022	SH.2	R.B.C.C.W. SYSTEM SPENT FUEL POOL AND SHUT-DOWN HEAT EXCHANGERS
25203-26022	SH.3	R.B.C.C.W. SYSTEM CNTMT. SPRAY PUMP AND S.I. PUMP SEAL COOLERS
25203-26022	SH.4	R.B.C.C.W. SYSTEM RCP THERMAL BARRIERS AND LO COOLERS
25203-26022	SH.5	R.B.C.C.W. SYSTEM CNTMT. AIR RECIRC. AND COOLANT UNIT
25203-26022	SH.6	R.B.C.C.W. SYSTEM REACTOR VESSEL SUPPORT CONCRETE COOLING COILS
25203-26023	SH.1	SPENT FUEL POOL COOLING & CLEANUP SYSTEM
25203-26023	SH.2	SPENT FUEL POOL COOLING & CLEANUP SYSTEM
25203-26024	SH.1	CONTAINMENT BUILDING DRAINS
25203-26024	SH.2	AUXILIARY BUILDING DRAINS
25203-26024	SH.3	AUXILIARY BUILDING DRAINS
25203-26024	SH.4	AUXILIARY YARD SUMP DRAIN
25203-26025	SH.1	SAMPLING SYSTEM
25203-26025	SH.2	SAMPLING SYSTEM
25203-26025	SH.3	SAMPLING SYSTEM
25203-26025	SH.4	SAMPLING SYSTEM
25203-26026	SH.1	AUX. STEAM & CONDENSATE
25203-26026	SH.2	AUX. STEAM & CONDENSATE
25203-26026	SH.3	AUX. STEAM & CONDENSATE
25203-26026	SH.4	AUX. STEAM HEATER INDEX
25203-26026	SH.5	AUX. STEAM & CONDENSATE
25203-26027	SH.1	HVAC SYSTEM TURB. BLDG. INTAKE STR. WHSE. & DIESEL GEN. ROOMS
25203-26027	SH.2	TURB. BLDG. INTAKE STR. WHSE. & D.G. RMS. CHILLED WATER SYSTEM
25203-26027	SH.3	CONTROL ROOM AIR CONDITIONING SYSTEM
25203-26027	SH.4	HEALTH PHYSICS FACILITY HEATING & VENTILATION SYSTEM
25203-26028	SH.1	CONTAINMENT AND ENCLOSURE BUILDING VENTILATION
25203-26028	SH.2	CONTAINMENT AND ENCLOSURE BUILDING VENTILATION
25203-26028	SH.3	CONTAINMENT AND ENCLOSURE BUILDING VENTILATION
25203-26028	SH.4	ENGINEERED SAFETY FEATURE ROOM AIR RECIRCULATION UNIT
25203-26028	SH.5	CONTAINMENT AND ENCLOSURE BUILDING VENTILATION
25203-26028	SH.6	STACK HOUSE VENTILATION WIDE RANGE GAS MONITORING
25203-26029	SH.1	DIAGRAM AUX.BLDG. VENTILATION SYSTEM
25203-26029	SH.2	DIAGRAM AUX.BLDG. VENTILATION SYSTEM
25203-26029	SH.3	DIAGRAM AUX.BLDG. VENTILATION SYSTEM
25203-26029	SH.4	BLDG 110 & 105 HVAC SH. 1
25203-26029	SH.5	BLDG 110 & 105 HVAC SH. 2
25203-26029	SH.6	BLDG 110 & 105 HVAC SH. 3
25203-26029	SH.7	CHEMISTRY LABORATORY HVAC
25203-26030	SH.1	WATER TREATMENT SYSTEM
25203-26031	SH.1	MISC. CHEMICAL FEED SYSTEM
25203-26031	SH.2	MISC. CHEMICAL FEED SYSTEM
25203-26031	SH.3	MISC. CHEMICAL FEED SYSTEM
25203-26033	——	N_2 SYSTEM
25203-26035	——	H_2 & CO_2 SYSTEMS
25203-26074	SH.1	POST ACCIDENT SAMPLING SYSTEM (P.A.S.S.)
25203-26074	SH.2	POST ACCIDENT SAMPLING SYSTEM (MODULES)
25203-26075	——	MISC. SYSTEM CTMT. TENDON GREASING SYSTEM
25203-26076	——	IFSI HELIUM, NITROGEN & ARGON GAS SYSTEM AND VACUUM DRYING SYSTEM
25203-26201	SH.1	ELECTRO HYDRAULIC TURBINE CONTROL SYSTEM
25203-26201	SH.2	ELECTRO HYDRAULIC TURBINE CONTROL SYSTEM
25203-26202	——	SHAFT HYDROGEN SEAL OIL SYSTEM
25203-26203	SH.1	TURBINE GENERATOR HYDROGEN SYSTEM & CO_2 PURGE
25203-26203	SH.2	MAIN TRANSFORMER & GENERATOR ISOPHASE BUS DUCT COOL. SYS.
25203-26204	——	STATOR WINDING COOLING SYSTEM
25203-26206	——	TURBINE BEARING VIBRATION MONITORING

FSAR FIGURE

OPERATIONS CRITICAL

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING AND INSTRUMENTATION DIAGRAM LEGEND

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: c:\fms32\tmpplot\C260011.DGN	
REVISED PER DC MP2-16-01149 & DUR 17-800369	DRAWING NO: 25203-26001	REV 34
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

FIRST LETTER		SECOND & SUCCEEDING LETTERS																			
MEASURED VARIABLE	SYMBOLS FOR MEASURED VARIABLES	DISPLAY DEVICES							CONTROLLING DEVICES							SENSING DEVICES			LOCAL OBSERVATION GLASS	TEST CONNECTION	RELAY OR CONVERTER (BLIND) (See Note 3)	SAMPLE NOZZLE
		INDICATING	RECORDING	INTEGRATING INDICATOR (See Note 5)	SCAN (See Note 7)	ALARM (See Note 12) LOW	ALARM (See Note 12) HIGH	ALARM (See Note 12) HIGH LOW	INDICATING	RECORDING	BLIND	CONTROL VALVE	SELF ACTUATED VALVE	FINAL CONTROL ELEMENT (See Note 11)	SWITCH (See Note 6)	PRIMARY ELEMENT	BLIND TRANSMITTERS	INDICATING TRANSMITTER				
Typical Symbol	( )	( ) I	( ) R	( ) QI	( ) J ( )	( ) AL	( ) AH	( ) AHL	( ) IC	( ) RC	( ) C	( ) V	( ) CV	( ) Z	( ) S	( ) E	( ) T	( ) IT	( ) G	( ) P	( ) Y	( ) N
Analysis (See Note 1)	A	AI	AR		AJ ( )	AAL	AAH	AAHL	AIC	ARC	AC	AV	ACV	AZ	AS	AE	AT	AIT		AP	AY	AN
Burner Flame	B	BI	BR		BJ ( )	BAL					BC	BV			BS	BE	BT		BG	BP	BY	
Conductivity	C	CI	CR		CJ ( )	CAL	CAH	CAHL	CIC	CRC		CV		CZ	CS	CE	CT	CIT		CP	CY	CN
Density	D	DI	DR		DJ ( )	DAL	DAH	DAHL	DIC	DRC		DV		DZ	DS	DE	DT	DIT		DP	DY	
Voltage (EMF)	E	EI	ER		EJ ( )	EAL	EAH	EAHL	EIC	ERC	EC			EZ	ES	EE	ET	EIT			EY	
Flow (See Note 10)	F	FI	FR	FQI	FJ ( )	FAL	FAH	FAHL	FIC	FRC	FC	FV	FCV	FZ	FS	FE	FT	FIT	FG	FP	FY	
Flow Ratio	FF	FFI	FFR		FFJ ( )				FFIC	FFRC	FFC	FFV		FFZ								
Gaging (Dimensional)	G	GI	GR		GJ ( )	GAL	GAH	GAHL	GIC	GRC	GC	GV		GZ	GS	GE	GT	GIT				
Hand	H								HIC		HC	HV	HCV	HZ	HS						HY	
Current	I	II	IR	IQI	IJ ( )	IAL	IAH	IAHL	IIC	IRC	IC			IZ	IS	IE	IT	IIT			IY	
Power	J	JI	JR	JQI	JJ ( )	JAL	JAH	JAHL	JIC	JRC	JC			JZ	JS	JE	JT	JIT			JY	
Time	K	KI	KR	KQI	KJ ( )	KAL	KAH	KAHL	KIC	KRC	KC			KZ	KS		KT	KIT			KY	
Level	L	LI	LR		LJ ( )	LAL	LAH	LAHL	LIC	LRC	LC	LV	LCV	LZ	LS	LE	LT	LIT	LG	LP	LY	
Moisture	M	MI	MR		MJ ( )	MAL	MAH	MAHL	MIC	MRC	MC	MV		MZ	MS	ME	MT	MIT		MP	MY	
Vibration	N	NI	NR		NJ ( )	NAL	NAH	NAHL							NS	NE	NT	NIT			NY	
Torque	O	OI	OR		OJ ( )	OAL	OAH	OAHL	OIC	ORC	OC	OV		OZ	OS	OE	OT				OY	
Pressure	P	PI	PR		PJ ( )	PAL	PAH	PAHL	PIC	PRC	PC	PV	PCV	PZ	PS	PE	PT	PIT		PP	PY	
Pressure Differential	PD	PDI	PDR			PDAL	PDAH	PDAHL	PDIC	PDRC	PDC	PDV	PDCV	PDZ	PDS		PDT	PDIT			PDY	
Quality or Event	Q	QI	QR	QQI	QJ ( )	QAL	QAH	QAHL	QIC	QRC	QC	QV		QZ	QS		QT	QIT			QY	
Radiation	R	RI	RR	RQI	RJ ( )	RAL	RAH	RAHL	RIC	RRC	RC	RV		RZ	RS	RE	RT	RIT		RP	RY	RN
Speed or Frequency	S	SI	SR	SQI	SJ ( )	SAL	SAH	SAHL	SIC	SRC	SC			SZ	SS		ST	SIT			SY	
Temperature	T	TI	TR		TJ ( )	TAL	TAH	TAHL	TIC	TRC	TC	TV	TCV	TZ	TS	TE (Note 8)	TT	TIT		TW (Note 8)	TY	
Temperature Differential	TD	TDI	TDR			TDAL	TDAH	TDAHL	TDIC	TDRC	TDC	TDV	TDCV	TDZ	TDS							
Multi-Variable	U	UI	UR		UJ ( )	UAL	UAH	UAHL	UIC	URC	UC	UV		UZ	US						UY	UN
Viscosity	V	VI	VR		VJ ( )	VAL	VAH	VAHL	VIC	VRC	VC	VV		VZ	VS	VE	VT	VIT			VY	
Weight	W	WI	WR	WQI	WJ ( )	WAL	WAH	WAHL	WIC	WRC	WC	WV		WZ	WS	WE	WT	WIT			WY	
Unclassified (See Note 4)	X	XI	XR		XJ ( )	XAL	XAH	XAHL	XIC	XRC	XC	XV		XZ	XS	XE	XT	XIT			XY	
Users Choice (See Note 2)	Y																				YY	
Position	Z	ZI	ZR		ZJ ( )	ZAL	ZAH	ZAHL	ZIC	ZRC	ZC			ZZ	ZS	ZE	ZT	ZIT			ZY	

NOTES

(1) "A"IS USED FOR ALL ANALYTICAL VARIABLES; FOR EXAMPLE: O_2 , H_2O , CO_2 ,pH , OCTANE INPROVEMENT, CHROMATOGRAPH ANALYZING ONE OR MORE STREAMS FOR ONE OR MORE COMPOUNDS , BOILING POINT, FEEEZING POINT , COMBUSTIBLES ETC. . THE CHEMICAL FORMULA RECOGNIZED SYMBOL (SUCH AS pH) OR A DESCRIPTION DENOTING THE FUNCTION OF THE ANALYZER MAY BE NOTED ON THE P&ID OUTSIDE THE INSTRUMENT SYMBOL . SAMPLE NOZZLE DETAILS ARE SHOWN ON DWG'S 25203-28402 SHT. SOI THRU SO4.

(2) A USER'S CHOICE LETTER IS INTENDED TO COVER A MEANING THAT WOULD BE USED REPETITIVELY IN A PARTICULAR PROJECT . WHEN USED , THE LETTER MAY HAVE ONE MEANING AS A FIRST LETTER AND ANOTHER MEANING AS A SUCCEEDING LETER .THE MEANINGS NEED BE DEFINED ONLY ONCE IN A LEGEND , OR OTHERWISE FOR THAT PROJECT . FOR EXAMPLE , THE LETTER "N" MAY BE DEFINED AS TURBIDITY AS A FIRST LETTER AND TELEVISION MONITOR AS A SECOND LETTER . "BN" WOULD BE A BURNER FLAME TELEVISION MONITOR .

(3) THE EQUATION OR DESCRIPTION DENOTING THE FUNCTION OF THE RELAY "Y" SHOULD BE SHOWN ON THE P&ID . FOR EXAMPLE: A B C K , LP SELECTOR , VOLUME BOOSTER .

(4) "X" IS USED TO REPRESENT ANY "SPECIAL" VARIABLES AND MAY BE DEFINED AS REQUIRED . FOR EXAMPLE: MASS FLOW RECORDERS WHICH RECEIVE A SIGNAL FROM A MULTIPLYING RELAY WHICH COMBINES THE PRODUCT OF DENSITY AND FLOW . THIS ITEM IS NOT TO BE CONFUSED WITH "U" MULTI-VARIABLE SYMBOL .

(5) WHEN "Q" IS USED AS A SECOND OR SUCCEEDING LETTER IT DENOTES AN INTEGRATING MODIFIER . FOR EXAMPLE: "FQI" IS AN INDICATING FLOW INTEGRATOR (OR TOTALIZER). NOTE THAT THE INTEGRATING FUNCTION SHALL BE SHOWN WITH SEPARATE IDENTIFICATION: FOR EXAMPLE: FQI/FRS OR FR/FQIS .

(6) IF THE SWITCH PERFORMS AN ON-OFF CONTROL FUNCTION , REPLACE "S" BY "C" . FOR EXAMPLE: "FS" BECOMES "FC" . STARTUP AND SHUTDOWN DEVICES ARE USUALLY BLIND , BUT MAY BE INDICATING OR RECORDING . IF SO ADD "I" OR "R" AFTER MEASURED VARIABLE . FOR EXAMPLE: "FIS" "TRS" .

(7) WHEN "J" IS USED AS A SECOND OR SUCCEEDING LETTER , IT DENOTES A SCANNING FUNCTION . FOR EXAMPLE , THE DESIGNATION "RJR" DENOTES A SCANNING RADIATION RECORDER .

(8) "TW" DENOTES AN EMPTY THERMOWELL . "TE" DENOTES A THERMOWELL WITH THERMOCOUPLE OR RESTISTANCE BULB AND HEAD SUITABLE FOR USE WITH A SECONDARY INSTRUMENT.

(9) PRESSURE RELIEF VALVES AND RUPTURE DISKS SHALL BE IDENTIFIED AS "PSV" AND "PSE" RESPECTIVELY .

(10) "FQ" AS AN INSTRUMENT IDENTIFICATION IS USED TO DESIGNATE A RESTRICTION ORIFICE (RO).

(11) FOR DEVICES OTHER THAN CONTROL VALVES , SUCH AS HYDRAULIC COUPLINGS , VARIABLE SPEED DRIVES , ETC. .

(12) HIGH-HIGH ALARMS WILL DESIGNATED "( )AHH" AND LOW-LOW ALARMS "( )ALL" . FOR EXAMPLE: LAHH DENOTES "HIGH-HIGH LEVEL ALARM".

(13) AO/AC IS USED ONLY IF VALVE FAILURE POSITION DOES NOT DEFINE ACTUATOR ACTION .

(14) "PP" DESIGNATES A PRESSURE POINT FOR CONNECTING TEMPORARY PRESSURE INDICATING INSTRUMENT.

GENERAL NOTES:

(A) INSTRUMENT IDENTIFICATIONS ARE BASED UPON ISA STANDARD "S5.1-1968" FOR FUTURE DETAILS REFER TO ISA STANDARD .

CONTROL VALVES

(USE WITH APPROPRIATE ACTUATOR SYMBOL)

FO — GLOBE , GATE OR OTHER IN-LINE TYPE VALVE AS IDENTIFIED BY APPROPRIATE SYMBOL
FO INDICATES FAIL OPEN
FC INDICATES FAIL CLOSED
FL INDICATES FAIL LOCKED (VALVE POSITION DOES NOT CHANGE)
FI INDICATES FAIL INDETERMINATE

X — UNCLASSIFIED (TYPE OF BODY WRITTEN IN , OR ADJACENT , TO SYMBOL)

FO — THREE-WAY FO=FAIL POSITION

FO — FOUR-WAY FO=FAIL POSITION

ACTUATORS

DIAPHRAGM TYPE PNEUMATIC ACTUATOR
AO AIR TO OPEN
AC AIR TO CLOSE
(SEE NOTE 13)

DIAPHRAGM , PRESSURE BALANCED

ELECTRIC — M — ROTARY MOTOR (SHOWN TYPICALLY WITH ELECTRIC SIGNAL)

CYLINDER , SINGLE ACTING; AO AIR TO OPEN AC AIR TO CLOSE (SEE NOTE 13)

S — A/S — CYLINDER , DOUBLE ACTING WITH ACTUATING PILOT VALVE

CYLINDER , DOUBLE ACTING ASSEMBLED WITHOUT PILOT VALVE

E/H — E/H =ELECTRO-HYDRAULIC

HAND ACTUATED (MOUNTED AT TOP , SIDE , OR BOTTOM OF ACTUATED DEVICE , AS APPLICIABLE)

S — SOLENOID

X — UNCLASSIFIED (TYPE OF ACTUATOR TO BE WRITTEN ADJACENT TO SYMBOL)

A/S — PNEUMATIC POSITIONER (SHOWN ON DIAPHRAGM TYPE ACTUATOR , ADDITIONAL PNEUMATIC LINE ADDED FOR DOUBLE ACTING ACTUATOR)

ELECTROPNEUMATIC POSITIONER (SHOWN ON DIAPHRAGM TYPE ACTUATOR , ADDITIONAL PNEUMATIC LINE ADDED FOR DOUBLE ACTING ACTUATOR)

A/S — INSTRUMENT AIR PRESSURE REGULATOR

SELF-ACTUATING DEVICES

PRESSURE

PCV — PRESSURE REDUCING REGULATOR , SELF-CONTAINED

PCV — PRESSURE REDUCING REGULATOR WITH EXTERNAL PRESSURE TAP

PDVC — DIFFERENTIAL PRESSURE REDUCING REGULATOR WITH INTERNAL AND EXTERNAL PRESSURE TAPS

PCV — BACK-PRESSURE REGULATOR SELF-CONTAINED

PCV — BACK-PRESSURE REGULATOR WITH EXTERNAL PRESSURE TAP

PSV — PRESSURE RELIEF OR SAFETY VALVE , ANGLE PATTERN SPRING OR WEIGHT LOADED , OR INTEGRAL PILOT

PSV — PRESSURE RELIEF OR SAFETY VALVE STRAIGHT THRU PATTERN SPRING OR WT. LOADED OR WITH INTEGRAL PILOT

PS — S — PSV — PRESSURE RELIEF OR SAFETY VALVE STRAIGHT THRU PATTERN TRIPPED BY INTEGRAL SOLENOID PILOT

PS — S — PSV — PRESSURE RELIEF OR SAFETY VALVE , ANGLE PATTERN , TRIPPED BY INTEGRAL SOLENOID

PSE — RUPTURE DISK OR SAFETY HEAD FOR VACUUM RELIEF

PSE — RUPTURE DISK OR SAFETY HEAD FOR PRESSURE RELIEF

PULSATION DAMPENER

VACUUM BREAKERS

PSV — EXISTING
EXISTING SYMBOL FOR VACUUM RELIEF, ANGLE PATTERN, SPRING OR WEIGHT LOADED OR WITH INTEGRAL PILOT

NEW
AS OF 10-31-96
NEW SYMBOL FOR VACUUM RELIEF ANGLE PATTERN , SPRING OR WEIGHT LOADED OR WITH INTEGRAL PILOT

VB — CHECK VALVE USED FOR VACUUM RELIEF

TEMPERATURE

TCV — TEMPERATURE REGULATOR , FILLED SYSTEM TYPE

LEVEL

LCV — LEVEL CONTROL VALVE WITH MECHANICAL LINKAGE

MISCELLANEOUS

1:1 — ( ) V — 1:1 RELAY (VOLUME BOOSTER)

( ) Y — SQUARE ROOT EXTRACTOR

( ) Y — > — HIGH MEASURED VARIABLE SELECTOR IF HIGH SIGNAL SELECTOR ADD (SIGNAL)

INDICATING LIGHT

INTERLOCK

PANEL MOUNTED ALARM WINDOW (THE STATUS OF ALL MAIN CONTROL BOARD ALARMS WILL BE INPUT TO THE PLANT COMPUTER)

I/P — ( ) Y — ELECTRIC CURRENT TO PENUMATIC SIGNAL CONVERTER

D — ( ) Y — DERIVATIVE OR RATE

INSTRUMENTS

LOCAL INSTRUMENT INCLUDING TRANSMITTER FOR SINGLE MEASURED VARIABLE

LOCAL INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION

INSTRUMENT MOUNTED INTERNAL TO OR ON REAR FACE OF BOARD

INSTRUMENT MOUNTED INTERNAL TO OR ON REAR FACE OF BOARD FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION

FRONT PANEL MOUNTED INSTRUMENT FOR SINGLE MEASURED VARIABLE

FRONT PANEL MOUNTED INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION

DIGITAL DATA SYSTEM-COMPUTER

VARIABLE INTO PLANT COMPUTER (THIS SYMBOL IS NOT TO BE USED TO SHOW ALARM STATUS OF PANEL MOUNTED ALARMS WINDOWS)

(1/2" X 7/8")

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REDRAWN SAME NUMBER THRU REV. 0

OPERATIONS CRITICAL

NON Q.A.

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	B	3-28-03 INC. DCN DM2-02-0217-02	MP	ASK	DLW	DED
N/A	7	2-16-01 INC. DCN DM2-00-0025-00	NJB	VJG	REB	GMO

REVISIONS DURING CONSTRUCTION

Dominion® Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION UNIT - 2
PIPING AND INSTRUMENTATION DIAGRAM
LEGEND
WATERFORD, CT

BY	CHKD.	APP.	APP.	APP.
R.D.K.	JBC	RSP JBL	CDM	JJY REB
DATE 12-20-84	DATE 3-4-85	DATE 3-7-85	DATE 3-7-85	DATE 3-12-85

SCALE: NONE
DWG. NO.: 84-024
25203-26001 SH.2 OF 3

PRESSURE SAFETY VA.	FLOW ELEMENT	FLOW SWITCH	COMPUTER POINT
2-MS-247	FE-4225	FS-4225	F4225
2-MS-249	FE-4226	FS-4226	F4226
2-MS-251	FE-4227	FS-4227	F4227
2-MS-253	FE-4228	FS-4228	F4228
2-MS-254	FE-4229	FS-4229	F4229
2-MS-252	FE-4230	FS-4230	F4230
2-MS-250	FE-4231	FS-4231	F4231
2-MS-248	FE-4232	FS-4232	F4232
2-MS-239	FE-4233	FS-4233	F4233
2-MS-240	FE-4234	FS-4234	F4234
2-MS-241	FE-4235	FS-4235	F4235
2-MS-242	FE-4236	FS-4236	F4236
2-MS-243	FE-4237	FS-4237	F4237
2-MS-244	FE-4238	FS-4238	F4238
2-MS-245	FE-4239	FS-4239	F4239
2-MS-246	FE-4240	FS-4240	F4240

DETAIL "A"
TYPICAL 16 PLACES

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SHTS. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STD. Y1.1.
2. DISCONNECT SWITCH IS TO BE IN OFF POSITION WHEN PLANT IS IN MODE 1.
3. DISCONNECT SWITCH IS TO BE IN OFF POSITION WHEN PLANT IS IN MODE 1, 2 & 3.
4. NORMALLY OPEN DRAIN VALVE TO REMOVE CONDENSATE.
5. DELETED.
6. SENTINEL RELIEF VALVE HAS BEEN REMOVED FROM THE TERRY TURBINE CASING.
7. NORMALLY OPEN NOZZLE CHECK VALVE INSTALLED AT 2-MS-48.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 1

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy®

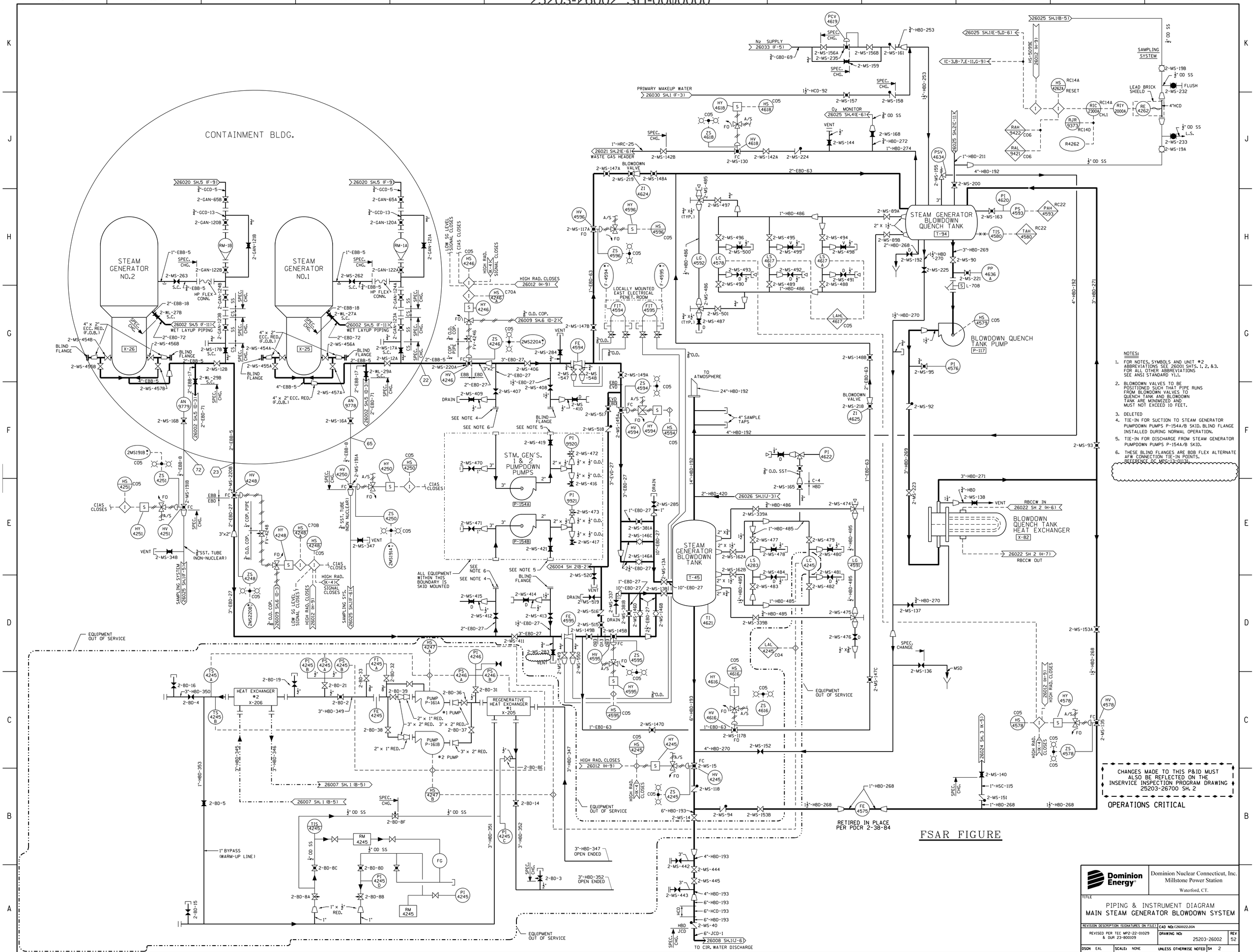
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
MAIN STEAM FROM GENERATORS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C26002I.DGN	
REVISED PER DC MP2-24-01023 AND DUR 24-800165	DRAWING NO: 25203-26002	REV 93
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

Authenticated By: MBULOGIN\scott84
2-NOV-2024 00:06
C26002I.DGN



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SHTS. 1, 2, &3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. BLOWDOWN VALVES TO BE POSITIONED SUCH THAT PIPE RUNS FROM BLOWDOWN VALVES TO QUENCH TANK AND BLOWDOWN TANK ARE MINIMIZED AND MUST NOT EXCEED 10 FEET.
3. DELETED
4. TIE-IN FOR SUCTION TO STEAM GENERATOR PUMPDOWN PUMPS P-154A/B SKID. BLIND FLANGE INSTALLED DURING NORMAL OPERATION.
5. TIE-IN FOR DISCHARGE FROM STEAM GENERATOR PUMPDOWN PUMPS P-154A/B SKID.
6. THESE BLIND FLANGES ARE BDB FLEX ALTERNATE AFW CONNECTION TIE-IN POINTS. REFERENCE DC MPG-13-01131.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 2

OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENT DIAGRAM
MAIN STEAM GENERATOR BLOWDOWN SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260022.DGN	
REVISED PER TEC MP2-22-01029 & DUR 23-800109	DRAWING NO: 25203-26002	REV 52
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

MBULOGIN\er1k004
2-MAY-2023 09:38
C260022.DGN

STEAM GENERATOR FEED PUMP TURBINE H-5A

STEAM GENERATOR FEED PUMP TURBINE H-5B

MOISTURE SEPARATOR/ REHEATER NO. 1A

#3 STEAM REHEAT STOP & INTERCEPT VALVE

#4 STEAM REHEAT STOP & INTERCEPT VALVE

H.P. TURBINE

L.P. TURBINE

L.P. TURBINE

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

OPERATIONS CRITICAL

SUPERSEDES 25203-26002 (THRU) REV. 33.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
MAIN STEAM TURBINE & REHEATER NO. 1A

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260023.DGN	
REVISED PER DC MP2-22-01114 & DUR 23-800107	DRAWING NO: 25203-26002	REV 43
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

MSULOGIN\er in004
1-MAY-2023 09:58
C260023.DGN

L.P. TURBINE H-2

#2 STEAM REHEAT STOP & INTERCEPT VALVE

#1 STEAM REHEAT STOP & INTERCEPT VALVE

H.P. TURBINE H-2

MOISTURE SEPARATOR/ REHEATER NO. 1B 28500 SH. 1157C

2nd. STAGE REHEATER 28500 SH. 1157B

1st. STAGE REHEATER 28500 SH. 1157A

MOISTURE SEPARATOR X-31B

LOCAL CONTROL 29218 SH.2 29218 SH.13

REACTOR PROTECTION 29193 SH.1

RRS STM. DUMP TREF 28200 SH.1271A

RRS STM. DUMP BYPASS CONTROL SYS. 28200 SH.1271J

EHC SYSTEM 28500 SH 1320 THROTTLE STEAM PRESS.

LIV 5069 26004 SH. 3 (H-5)

LV-4145/4149 26003 SH. 2 (F-2XF-4) 26003 SH. 1 (F-3XF-4) LV-4134/4130

LV-4141 26003 SH. 2 (F-5) 26003 SH. 1 (F-6) LV-4126

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

OPERATIONS CRITICAL

SUPERSEDES 25203-26002 THRU REV. 33

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
MAIN STEAM TURBINE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260024.DGN	
REVISED PER DC MP2-17-00112 & DUR 18-800228	DRAWING NO: 25203-26002	REV 32
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 4

MBULOGIN\dav1714
16-OCT-2018 16:16
C260024.DGN

HEATER NO. 4A X-4A
HEATER NO. 3A X-3A
HEATER NO. 2A X-2A
HEATER NO. 4B X-4B
HEATER NO. 3B X-3B
HEATER NO. 2B X-2B
HEATER NO. 5A X-5A
HEATER NO. 6A X-6A
HEATER NO. 5B X-5B
HEATER NO. 6B X-6B
CONDENSER "A" X-8A
CONDENSER "B" X-8B
DRAINS COOLER X-7A
DRAINS COOLER X-7B
HEAT EXCH. STM. JET AIR EJECTOR X-9A
HEAT EXCH. STM. JET AIR EJECTOR X-9B
STEAM PACKING EXHAUSTER X-10
CNDS. PUMPS

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FOR STEAM BYPASS SPRAY PIPING INSIDE CONDENSERS X-8A & X-8B SEE DWGS. 29001 SHTS. 85 & 86.
3. CONDENSATE PUMP RTDS
   P-2A
   TE-5167 UPPER GUIDE BEEARING RTD
   TE-5168 LOWER GUIDE BEARING RTD
   TE-5353 MOTOR WINDING TEMP. RTD
   P-2B
   TE-5178 UPPER GUIDE BEEARING RTD
   TE-5180 LOWER GUIDE BEARING RTD
   TE-5352 MOTOR WINDING TEMP. RTD
   P-2C
   TE-5187 UPPER GUIDE BEEARING RTD
   TE-5188 LOWER GUIDE BEARING RTD
   TE-5351 MOTOR WINDING TEMP. RTD
4. WORK THIS DRAWING WITH 25203-26005 SH. 4
5. LOCATED BEHIND SECONDARY SAMPLE STATION/SINK H-53, SEE DWG. 25203-28408 SH.6327.
6. ALTERNATE CONDENSATE FLOW PATH INSTALLED VIA 2-CN-407 TO CONTROL & MAINTAIN WATER TROUGH LEVEL AT DESIRED LEVEL PER TEC MP2-24-01080.

REFERENCE DRAWINGS:

1. PIPING STEAM SEAL & DRAINS 29221 SH.2
2. TWO-215,000 SQ. FT. SURF. COND. BY WESTINGHOUSE ..... 29001 SERIES

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENT DIAGRAM
CONDENSATE SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C26005I.DGN	
REVISED PER TEC MP2-24-01080 & DUR 24-800174	DRAWING NO: 25203-26005	REV 40
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

MSULOGIN\er ik804
19-NOV-2024 12:09
C26005I.DGN

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. LOCATED BEHIND SECONDARY SAMPLE STATION/SINK H-53, SEE DWG. 25203-28408 SH.6327.
3. EXTERNALLY MOUNTED LEADING EDGE FLOW METER (LEFM) TRANSDUCERS INSTALLED PER EC MP2-23-01099.

STEAM GENERATOR FEEDWATER PUMPS

SGFP SEAL INJ. BOOSTER PUMP

ATMOSPHERIC COLLECTION DRAIN TANK

HEATER NO. 1A X-1A

HEATER NO. 1B X-1B

(3)ELEMENT FEEDWATER CONTROLS

STEAM GENERATOR NO.1

STEAM GENERATOR NO.2

REACTOR PROTECTION SYSTEM AUTO AUXILIARY FEEDWATER SYSTEM

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 3

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
FEED SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260052.DGN	
REVISED PER TEC MP2-24-01015 AND DUR 24-800155	DRAWING NO: 25203-26005	REV 86
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

Authenticated By: MBULOGIN\jomm694
19-OCT-2024 07:52
C260052.DGN

CONDENSATE SURGE TANK

CONDENSATE SURGE TANK PUMP

CONDS. SURGE TANK HEAT EXCHANGER X-168

CONDENSATE TRANSFER PUMP

CONDENSATE STORAGE TANK CIRC. PUMP

CONDS. STORAGE TANK HEAT EXCHANGER X-46

CONDENSATE STORAGE TANK

STEAM GENERATOR AUXILIARY FEED PUMPS

COMPRESSED AIR BOTTLES

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ~~LC-5280A & B RETIRED IN PLACE PER PDCR 2-027-93~~
3. PORTABLE FLOW COMPUTER CONNECTED TO THE TRANSDUCER DURING PERIODIC TESTING.
4. FOR SIPHON BREAKER ON 3"-HBD-188, INSIDE CST (T40) SEE 25203-13006 SH. 44.
5. 2-CN-241 (LV-5417) AND LY-5280C HAVE BEEN ABANDONED IN PLACE PER DCR M2-98073.
6. MANUAL AFW VALVE TAMPER SWITCH CONTROL PANEL C05 ANNUNCIATOR D-15.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 4

SUPERSEDES 25203-26005 THRU REV.36

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

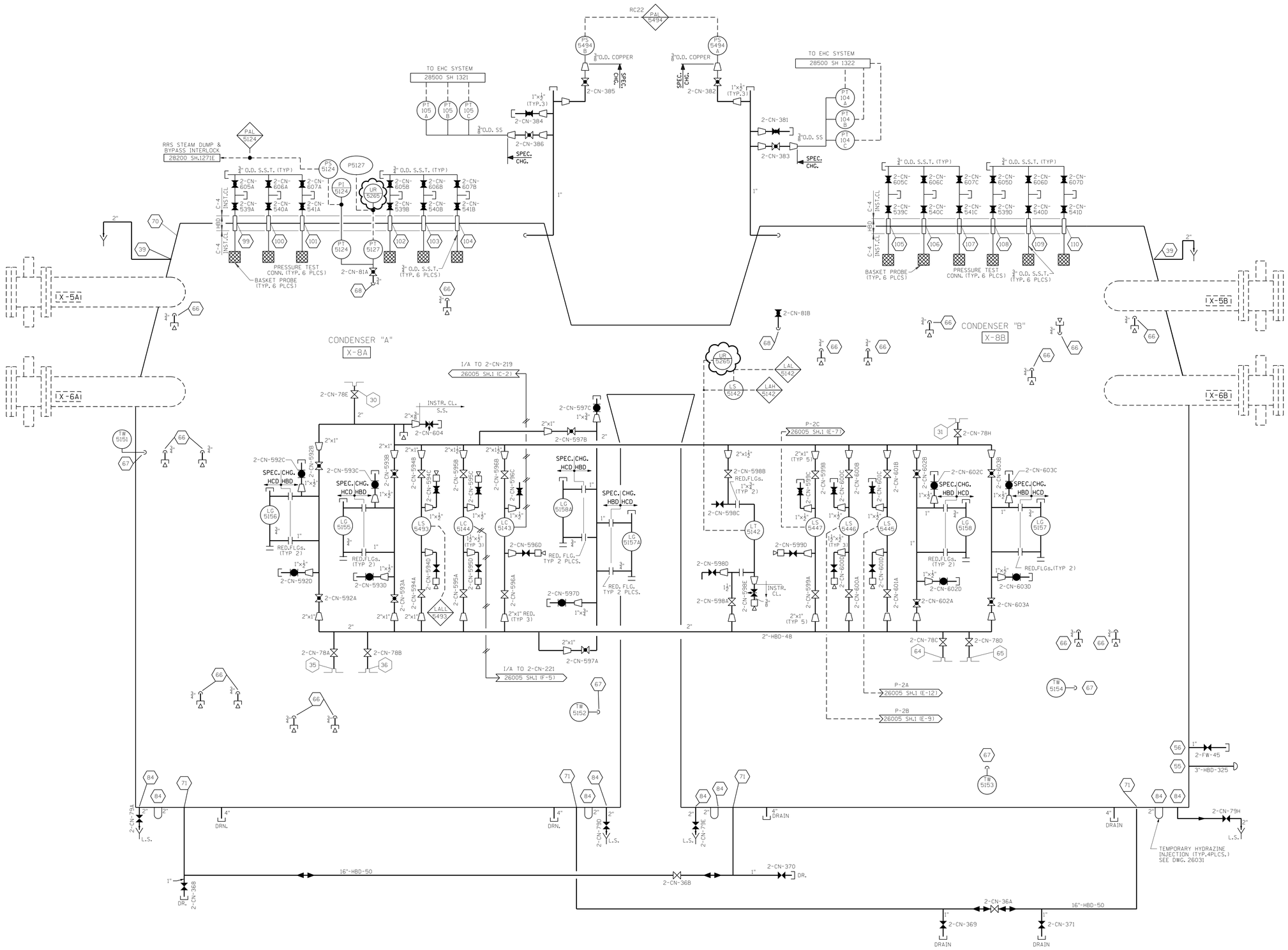
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
CONDENSATE STORAGE & AUX. FEED

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C26005.DGN	
REVISED PER DC MP2-12-01174 & DUR 14-800162	DRAWING NO: 25203-26005	REV 60
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 3



NOTE:
WORK THIS DRAWING WITH
25203-26005 SH.1

OPERATIONS CRITICAL

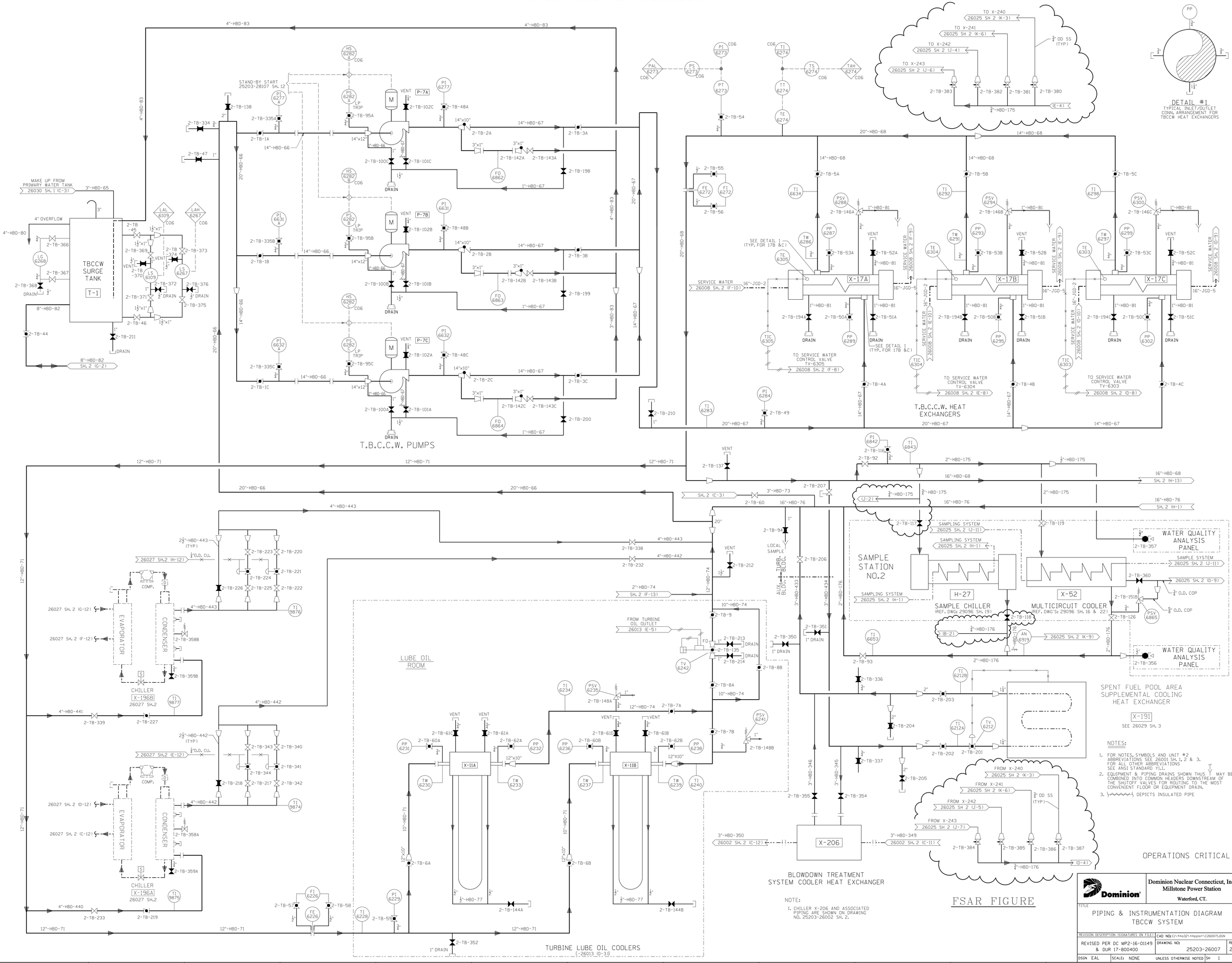
Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENT DIAGRAM
CONDENSATE SYSTEM

REVISION DESCRIPTION SIGNATURES ON FILE	CAD NO: C260054.DGN	
REVISED PER DC WP2-17-00171 AND DUR 18-800210	DRAWING NO: 25203-26005	REV 7
DSGN DLW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 4

MBLOGIN\dlw+40
27-OCT-2018 20:43
C260054.DGN



Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
TBCCW SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C26007L.DGN	
REVISED PER DC MP2-16-01149 & DUR 17-800400	DRAWING NO: 25203-26007	REV 25
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

HEATER DRAIN PUMPS

ELECTRO-HYDRAULIC OIL COOLERS

STEAM GENERATOR F.W. PUMP LUBE OIL COOLERS (4)2 SPARES

STATION AIR COMPRESSOR AFTERCOOLER

STATION AIR COMPRESSOR

CHEMICAL ADDITION TANK T-35

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FOR ALL OTHER NOTES SEE DRAWING NUMBER 25203-26007 SH. 1

OPERATIONS CRITICAL

THIS DRAWING REDRAWN FROM DRAWING NUMBER 25203-26007 REVISION 15

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE POWER STATION UNIT 2
PIPING AND INSTRUMENTATION DIAGRAM
TBCCW SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER MSDUR 14-000004 AND DUR 14-800004

CAD NO: ...\PROJECT\20000\P260072.DGN

DRAWING NO: 25203-26007 REV 19

DSGN DWJ | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 2

STATOR LIQUID COOLERS

STATOR LIQ COOLING 26204 (C-9)

X-14A

X-14B

SAMPLE COOLERS

X-180 X-179 X-178

RETIRED IN PLACE PER DCR M3-99042

FEEDWATER HEATERS 1 & 2 SAMPLE COOLERS

STEAM PACKING EXHAUSTER SAMPLE COOLER

PANEL C-98B

X-185 X-186 X-184

CONDENSATE PUMP MOTOR COOLING

P-2A P-2B P-2C

TO WAREHOUSE No. 5 A/C UNIT 3CCS-004-364-4 25212-26934 SH.2 (B-10)

FROM WAREHOUSE #5 A/C UNIT 3CCS-004-365-4 25212-26934 SH.2 (B-9)

X-13D X-13C X-13B X-13A

GENERATOR H_2 COOLERS

FROM GEN. H2 COOLER (TAP NO. GTG-3) GE#114D8334 (7604-M945-32-3)

FROM EXCITER AIR COOLER TAP 7604-M955-505 7604-M955-502

X-15A

EXCITER AIR COOLER

CONTINUES VENT LINE

OPEN TROUGH

F-119B F-119A

X-16B X-16A

BUS DUCT

26203 SH.2 (E-4) 26203 SH.2 (E-2) 26203 SH.2 (G-2)

32005 SH. 37 39008 SH. 29

32005 SH. 36 39008 SH. 29

MAIN TRANSFORMER & GENERATOR ISOPHASE BUS DUCT COOLING SYSTEM

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FOR ALL OTHER NOTES SEE DRAWING NUMBER 25203-26007 SH.1

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
TBCCW SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260073.DGN	
REVISED PER MSDUR 24-801036 & DUR 24-800098	DRAWING NO: 25203-26007	REV 11
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

Authenticated By: MBULOGIN\scot84
25-JUN-2024 12:47
C260073.DGN

RELIEF VENT DISCHARGE TRANSITION BOX
2-CW-42
2-CW-40
3"-JW-7
2-CW-43
2-CW-41
16"-JW-7
MP2FLANGECW-01
JGD JW
ABOVE GROUND
BELOW GROUND
CUT BELOW GRADE FILLED W/CONC.
36"-JGD-40
16" "YOLOY" STEEL
SERVICE WTR. FROM SWGR RM. COOLING COILS
SH. 3 (E-3)
4"-HUD-32
SERVICE WTR. FROM CHILLERS
SH. 3 (G-8)
3"-HUD-29
2-SW-284
2-SW-192
2-SW-283
2-SW-197
SPEC. CHG.
2-SW-6
SERVICE WTR. FROM TBCCW HT. EXCH.
SH. 2 (C-6)
24"-JGD-5
3"-JGD-29
SERVICE WTR. FROM "A" D/G HT. EXCH.
SH. 2 (F-5)
8"-JGD-7
D.G. BLDG. "A" YARD
SERVICE WTR. FROM "B" D/G HT. EXCH.
SH. 2 (J-5)
D.G. BLDG. "B" YARD
10"-JGD-7
SERVICE WTR. FROM "A" RBCCW HT. EXCH.
SH. 2 (F-6)
24"-JGD-6
SERVICE WTR. FROM "C" RBCCW HT. EXCH.
DISCHG. FROM AERATED LIQUID RADWASTE SYS.
26021 SH.1 (J-4)
2½"-HSD-1
DISCHG. FROM RBCCW ROOM SUMP
26024 SH.2 (E-8)
2"-HSD(R)-13
ATMOSPHERIC COLLECTION DRAIN TANK (T-41)DRAIN
26005 SH. 2 (D-10)
3"-HBD-326
COND. DUMP
26005 SH. 1 (B-6)
10"-HBD-53
2-CW-9
ABOVE GROUND
BELOW GROUND
SUCTION LINE IN PROTECTING WELL
1"-HCD-93
GRAB SAMPLE
10"-JGD-7
12"-KE-4
8"-JGD-7
BLOWDOWN DISCHARGE
26002 SH.2 (A-6)
6"-JCD-1
DISCHG. FROM CLEAN LIQUID RADWASTE SYS.
26020 SH.2 (C-1)
3"-HSD-2
4"-JGD-33
24"-JGD-5 (COATED & WRAPPED)
10"-JGD-53 (COATED & WRAPPED)
DISCHARGE CONDUIT TO QUARRY FROM UNIT #2
SEAL WELL
CIRCULATING WATER PUMP
¾" CPVC
CIRCULATING WATER FLOW MEASURING DEVICE
DETAIL "A"
CIRC WATER FLOW MEASURING (A-4, 6, 8, 10)

NOTE
ALL PENETRATION NO.S ⬡ ARE TYPICAL FOR 4 DISCH. NOZZLES
RADIALLY SPACED
84"-JGD-17
2CW11A*
2CW11B*
2CW11C*
2CW11D*
2-CW-7A
2-CW-7B
2-CW-7C
2-CW-7D
2-CL-26A
2-CL-26B
2-CL-26C
2-CL-26D
2-HCD-89
2-CW-11A
2-CW-11B
2-CW-11C
2-CW-11D
2-CW-4A
2-CW-4B
2-CW-4C
2-CW-4D
2-CW-37A
2-CW-38A
2-CW-39A
2-CW-5A
2-CW-5E
2-CW-5F
2-CW-5H
1"-JGD-12
3"-JDD-63
66"-JGD-10
2-CW-12A
2-CW-12B
2-CW-12C
N.C.
2-CW-37B
2-CW-38B
2-CW-39B
2-CW-5C
2-CW-5G
2-CW-5B
2-CW-5D
26012 (F-8) TO WATER BOX PRIMING TANK TK-61
26012 (F-7) TO WATER BOX PRIMING TANK TK-61
26012 (F-6) TO WATER BOX PRIMING TANK TK-61
26012 (C-8) TO WATER BOX PRIMING TANK TK-61
26012 (C-7) TO WATER BOX PRIMING TANK TK-61
26012 (C-6) TO WTR. BOX PRIMING TANK TK-61
CONDENSER NO. 1A
CONDENSER NO. 1B
X-8A
X-8B
78" I.D. REINF. CONC.
VALVE PIT WALL
RETIRED IN PLACE
TO RECIRC. LINE DISTRIBUTION TROUGH SEE NOTE 5
2CW13
2-CW-13
L.C.
HS 6620
ZS 6620
HV 6620
XJ 5
T6574
T6585
T6600
T6617
TE 6574
TE 6585
TE 6600
TE 6617
TI 6573
TI 6578
TI 6601
TI 6609
HV 6575
HV 6586
HV 6608
HV 6618
HV 6619
ZS 6575
ZS 6586
ZS 6608
ZS 6618
ZS 6619
HS 6575
HS 6586
HS 6608
HS 6618
HS 6619
PI 6574
PI 6587
PI 6599
PI 6610
XJ 2A
XJ 2B
XJ 2C
XJ 2D
XJ 4A
XJ 4B
AN 6911
AN 6912
AN 6913
AN 6914
LG 6635
LG 6636
LG 6637
LG 6638
2-CW-36A
2-CW-36B
2-CW-36C
2-CW-36D
TW 6572
TW 6571
TW 6570
TW 6569
TW 6568
TW 6567
TW 6584
TW 6583
TW 6582
TW 6581
TW 6580
TW 6579
TW 6607
TW 6606
TW 6605
TW 6604
TW 6603
TW 6602
TW 6616
TW 6615
TW 6614
TW 6613
TW 6612
TW 6611
2-CW-11E
2-CW-11F
2-CW-11G
2-CW-11H
2-CW-8A
2-CW-8B
2-CW-8C
2-CW-8D
2-CW-30
2-CW-3A
2-CW-3B
2-CW-3C
2-CW-3D
XJ 1A
XJ 1B
XJ 1C
XJ 1D
XJ 3A
XJ 3B
HV 6538
HV 6542
HV 6547
HV 6551
HV 6561
HV 6593
ZS 6538
ZS 6542
ZS 6547
ZS 6551
ZS 6561
ZS 6593
HS 6538
HS 6542
HS 6547
HS 6551
HS 6561
HS 6593
PI 6560
PI 6566
PI 6588
PI 6592
PI 6594
T6557
T6564
T6596
T6589
T6562
T6563
T6597
T6590
T6598
T6591
TE 6557
TE 6564
TE 6596
TE 6589
TE 6562
TE 6563
TE 6597
TE 6590
TE 6598
TE 6591
TI 6556
TI 6565
TI 6595
TI 6588
TW 6559
TW 6558
TW 6562
TW 6563
TW 6597
TW 6590
TW 6598
TW 6591
91
92
93
94
2-CW-11H
2-CW-11G
2-CW-11F
2-CW-11E
2CW11H*
2CW11G*
2CW11F*
2CW11E*
PAL 6523
PS 6523
CO6
VENT
2-SW-202
2-SW-27
FROM 2-WW-101
26011 SH. 2 (A-12)
2"-HCD-60
TRANS. BOX
REFER TO LOGIC DIAGRAM 25203-28108 SH. 11 FOR PUMP AND VALVE INTERLOCKS
ABOVE GROUND
BELOW GROUND
2"-HCD-61
2-SW-19
FROM 2-1A-61
26009 SH. 3 (J-11)
PIC 6522
PV 6522
2-SW-26
2-SW-138A
2-SW-138B
FD 2-SW-98
2-SW-203
2-SW-137
2-SW-17
PAL 6514
PC 6514
TO PV-6514
26011 SH. 2 (B-11)
2-SW-67
2-SW-195
2-SW-132B
2-SW-134B
2-SW-133B
2-SW-131B
2-SW-130B
PDI 6472
L-64B
RO 6670
FROM HDR. "A" 24"-JGD-1
26008 SH. 2 (C-11)
2-HLD-60
JGD
HLD
HCD
2-SW-213
2-SW-299
2-SW-132A
2-SW-134A
PDI 6471
L-64A
2-SW-131A
2-SW-133A
2-SW-130A
RO 6669
FROM HDR. "B" 24"-JGD-1
26008 SH. 2 (C-12)
2-SW-298
2-HCD-61
2-HCD-60
1"-HCD-61
2-SW-139A
2-SW-139B
2-SW-139C
2-SW-139D
2-SW-139E
2-SW-139F
2-SW-139G
2-SW-139H
FI 6529
FI 6531
FI 6533
FI 6535
2-SW-140A
2-SW-140B
2-SW-140C
2-SW-140D
VFD-P6A
VFD-P6B
VFD-P6C
VFD-P6D
HS 6658A
HS 6659A
HS 6660A
HS 6661A
HS 6658B
HS 6659B
HS 6660B
HS 6661B
HS 6658C
HS 6659C
HS 6660C
HS 6661C
HS 6658
HS 6659
HS 6660
HS 6661
SIC 6A
SIC 6B
SIC 6C
SIC 6D
A106
A201
A107
A202
C-179
C-180
2-CW-28A
2-CW-28B
2-CW-28C
2-CW-28D
2-HCD-382
2-JDD-52
IT-P6A
IT-P6B
IT-P6C
IT-P6D
IT-P6A
IT-P6B
IT-P6C
IT-P6D
TE 6505
TE 6504
TE 6544
TE 6662
TE 6550
TE 6665
TE 6553
TE 6668
TE 6539
TE 6543
TE 6334
TE 6123
T6505
T6504
T6544
T6662
T6550
T6665
T6553
T6668
T6539
T6543
T6334
T6123
P-6A
P-6B
P-6C
P-6D
2-CW-19A
2-CW-20A
2-CW-21A
2-CW-22A
2-CW-19B
2-CW-20B
2-CW-21B
2-CW-22B
1"-HCD-70
P6526
P6545
P6555
P6554
PT 6526
PT 6545
PT 6555
PT 6554
PI 6808
PI 6809
PI 6810
PI 6811
2-CW-1A
2-CW-1B
2-CW-1C
2-CW-1D
2-CW-23
2-CW-24
2-CW-25
2-CW-26
2-CW-27
2-CW-28
2-CW-29
2-CW-30
LG 6019
LG 6020
LG 6021
LG 6022
SEE DETAIL "A" (J-2)
SPEC. CHG.
2"-JDD-54
2"-JDD-25
STATION AIR 26009 SH.10 (E-2)
L-347 AIR EDUCTOR
4"-JBD-72
2-CW-31
PI 6076
CIRCULATING WATER PUMPS
FROM LS 6901A-D 26012 (K-11)
FROM LS6500E 26008 SH.4 (J-1)
FROM LS6502E 26008 SH.4 (H-1)
FROM LS6512E 26008 SH.4 (G-1)
FROM LS6510E 26008 SH.4 (E-1)
2 QUARRY
1 QUARRY
UNIT 2 DISCHARGE
UNIT 2 INTAKE
'C' BAY
'B' BAY
TE 6694
AE 6694
AIT 6694
A6694
T6695
TU1QW
AU1QW
PHIT 4-3
PHE 4-3
TEMP
pH
TE 6693
AE 6693
AIT 6693
A6693
T6691
TE 6692
AE 6692
AIT 6692
T6690
A6692
TE 6696
AE 6696
AIT 6696
A6696
T6696
WATER QUALITY MONITORING EQUIP.

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. JGD PIPING CAN CONSIST OF EITHER LINED CARBON STEEL, STAINLESS STEEL OR HIGH PERFORMANCE STAINLESS STEEL (UNS N08367) MATERIAL. FOR ACTUAL INSTALLED MATERIAL, REFER TO APPLICABLE PIPE SPOOL DRAWINGS.
3. FLANGED ENDS MATING WITH BUTTERFLY VALVES MAY REQUIRE COUNTERBORING AFTER ASSEMBLY OF PIPE SPOOL. FOR COUNTERBORING INFORMATION, REFER TO CLASS JGD OF SPECIFICATION SP-ME-668.
4. DELETED.
5. FOUR (4) 2'-0" x 3'-0" SLUICE GATE OPENINGS AT THE INTAKE STRUCTURE WERE SEALED WITH CONCRETE. SEE DRAWING 25205-11100 SH. 1.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 5

FSAR FIGURE

OPERATIONS CRITICAL

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENT DIAGRAM
CIRCULATING WATER

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260081.DGN	
REVISED PER DC MP2-17-01030 & DUR 18-800126	DRAWING NO: 25203-26008	REV 106
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

RBCCW HEAT EXCHANGERS
(3)ONE SPARE

TBCCW HEAT EXCHANGERS
(3)ONE SPARE

SERVICE WATER COOLING PUMPS (3)ONE SPARE

JACKET WATER COOLER

LUBE OIL COOLER

AIR COOLER

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. JGD PIPING CAN CONSIST OF EITHER LINED CARBON STEEL, STAINLESS STEEL OR HIGH PERFORMANCE STAINLESS STEEL (UNS N08367) MATERIAL. FOR ACTUAL INSTALLED MATERIAL, REFER TO APPLICABLE PIPE SPOOL DRAWINGS.
3. FLANGED ENDS MATING WITH BUTTERFLY VALVES MAY REQUIRE COUNTERBORING AFTER ASSEMBLY OF PIPE SPOOL. FOR COUNTERBORING INFORMATION, REFER TO CLASS JGD OF SPECIFICATION SP-ME-668
4. THIS BLIND FLANGE IS A BDB FLEX SERVICE WATER CONNECTION TIE-IN POINT. REFERENCE DC MPG-13-01131.

SERVICE WATER COOLING TO DIESEL GENERATOR HEAT EXCHANGER

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 6

OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
SERVICE WATER

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260082.DGN	
REVISED PER DC MP2-20-01176 & DUR 21-800163	DRAWING NO: 25203-26008	REV 125
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 2

MBULOGIN\jame694
13-SEP-2021 12:08
C260082.DGN

CHILLED WATER SUPPLY 26027 SH. 2 (C-8) 1½"-JBD-59
1½"-JBD-55 CHILLED WATER RETURN 26027 SH. 2 (C-9)
EVAPORATOR X-169C
CHILLER X-169A
PI 4668A
PDI 4668A
PI 4669A
2-SW-278B
1½"-HUD-101
2-SW-306
PSV 4668
2-SW-119
2½"-HUD-101
1½"-HUD-104
2-SW-278A
(NOTE 4)
2-SW-101
FROM X-169A COMPRESSOR 26027 SH. 2 (B-10)
CAP TUBE
2-SW-111
2-SW-125
FCV 8943
2½"-HUD-101
2-SW-122
TI 4668
2-SW-232A
FI 4668
FE 4668
2-SW-232B
2-SW-105
2½"-HUD-29
2½"-HUD-104
BLIND FLANGE
2-SW-160
SH. 1 (K-11)
3"-HUD-29
CHILLED WATER SUPPLY 26027 SH. 2 (C-4) 1½"-JBD-64
1½"-JBD-54 CHILLED WATER RETURN 26027 SH. 2 (C-4)
EVAPORATOR X-169D
CHILLER X-169B
PI 4669B
PDI 4668B
PI 4668B
PSV 4670
2-SW-307
2-SW-279B
1½"-HUD-102
2-SW-279A
(NOTE 5)
1½"-HUD-105
2-SW-121
2½"-HUD-102
2-SW-103
FROM X-169B COMPRESSOR 26027 SH. 2 (B-5)
2-SW-113
2-SW-127
FCV 8945
2-SW-124
TI 4670
2-SW-234A
FI 4670
FE 4670
2-SW-234B
2-SW-107
2½"-HUD-102
BLIND FLANGE
2"
2½"-HUD-105
2-SW-286
6"-HUD-24
4"-HUD-24
2-SW-195
SERVICE WATER HEADER B
SH. 2 (E-11)
2"
BLIND FLANGE
TI 4671
2½"-HUD-104
6"-HUD-28
4"-HUD-30
4"-HUD-24
2-SW-285
2-SW-174
2-SW-175 NC
2-SW-176
TURBINE BUILDING FL. EL. 14'-6"
T.B. 31'-6"
2-SW-194
6"-HUD-28
SERVICE WATER HEADER A
SH. 2 (D-11)
3"-HUD-130
2½"-HUD-113
1"-HCD-134
PSV 6942
2-SW-191
SPEC CHG
C515
TI 6933
TT 6933
PI 6939
TE 6933
FI 6924
PI 6924
PDI 6925C
PI 6925C
VALVE CLOSES ON MOISTURE SIGNAL
PY 6926
C80
HS 6926
MAH 6925
PI 6936
2-SW-183C
T.B. 54'-6" SWITCH GEAR RM 56'-6"
2½"-HUD-113
2"-HUD-113
XJ 32
2-SW-116
VENT
SWITCH GEAR RM 56'-6"
T.B. 54'-6"
XJ 33
2-SW-184C
SEE NOTE 3
2-SW-188A
2-SW-188B
2-SW-282A
SEE NOTE 3
2-SW-282B
4"-HUD-32
SH. 1 (K-11)
TO DISCHARGE CONDUIT
2½"-HUD-114
2-SW-181C
FE 6924
T.B. 31'-6"
T.B. 14'-6"
T.B. 54'-6"
T.B. 31'-6"
2-SW-177C
2-SW-178C FO
2-SW-179C
SPEC. CHANGE
HUD
HCD
2-SW-180C NC
TE 6930
TI 6930
TT 6930
C515
DRIP PAN
X-183
2"
1"-JCD-10
1"
2-SW-117
DRAIN
MS 6926
2"-HUD-114
1"-HCD-134
PSV 6941
2-SW-190
SPEC. CHG.
2"-JCD-10
C515
TI 6932
TT 6932
TE 6932
PI 6938
SEE NOTE 3
2-SW-184B
FI 6923
2-SW-281A
2-SW-187B
2-SW-187A
2-SW-181B
FE 6923
PI 6923
PDI 6925B
PI 6925B
2-SW-281B
VALVE CLOSES ON MOISTURE SIGNAL
PY 6927
C80
HS 6927
MAH 6926
PI 6935
2-SW-183B
T.B. 31'-6" CABLE VAULT 45'-0"
2½"-HUD-111
2"-HUD-111
VENT
2-SW-114
HUD
VENDOR
HUD
VENDOR
2"-HUD-112
2-SW-177B
2-SW-178B FO
2-SW-179B
SPEC. CHANGE
HUD
HCD
2-SW-180B
TE 6929
TI 6929
TT 6929
C515
XJ 30
XJ 31
X-182
DRIP PAN
2-SW-115
DRAIN
1"-JCD-11
MS 6927
CABLE VAULT 45'-0" T.B. 31'-6"
2"-JCD-11
3"-HUD-32
VALVE CLOSES ON MOISTURE SIGNAL
PY 6925
C80
HS 6925
MAH 6926
PI 6934
2-SW-183A
SPEC. CHG.
2-SW-129
VENT
HLD
HUD
XJ 26
XJ 27
XJ 26A
XJ 27A
X-181A
2"-HLD-1
2"-HLD-2
1"-HCD-131
PSV 6940
2-SW-189
SPEC. CHG.
TI 6931
TT 6931
C514
PI 6937
FI 6922
2-SW-280A
TE 6931
SEE NOTE 3
2-SW-184A
2-SW-186A
2-SW-186B
2-SW-181A
FE 6922
PI 6922
PDI 6925A
PI 6925A
2-SW-280B
2-SW-177A
2-SW-178A FO
2-SW-179A
HUD
HLD
2-SW-180A NC
TE 6928
TI 6928
TT 6928
C514
T.B. 31'-6" SUB STA. 36'-6"
2-SW-128
XJ 28
XJ 29
X-181B
DRIP PAN
1"-HBD-421
MS 6925
SUB STA. 36'-6" T.B. 31'-6"
3"-HLD-2
3"-HUD-131
3"-HUD-130
VITAL AC SWITCHGEAR ROOM COOLING COILS

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FOR CHILLED WATER PIPING SEE 25203-26027 SH. 2.
3. TUBING AND FITTINGS IN THIS AREA ARE MIXED COPPER NICKEL, MONEL AND STAINLESS STEEL.
4. TUBE PLUG MAY BE REMOVED TO SUPPORT X169A SERVICE WATER DRAINING ACTIVITIES. TAGGING ID IS 2-SW-278A-TEE-PLUG.
5. TUBE PLUG MAY BE REMOVED TO SUPPORT X169B SERVICE WATER DRAINING ACTIVITIES. TAGGING ID IS 2-SW-279A-TEE-PLUG.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 7

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

SUPERSEDES IN PART 25203-26027 SH. 2 THRU REV. 8

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENT DIAGRAM
SERVICE WATER TO VITAL AC SWITCHGEAR COOLING COIL AND AC CHILLERS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260083.DGN	
REVISED PER MSDUR 23-802008 & DUR 23-800291	DRAWING NO: 25203-26008	REV 36
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

Authenticated By: MBULOGIN ver ik004
7-NOV-2023 11:41
C260083.DGN

SCREEN WASH PUMPS (2)
ONE STANDBY

CIRCULATING WATER
SODIUM HYPOCHLORITE SYSTEM

STOP LOG GUIDE
TRAVELING SCREEN

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. SPILL CONTAINMENT DIKE SIZED FOR CONTENTS OF ONE TANK ONLY. ONE TANK MUST BE KEPT ISOLATED.
3. 2-HP-3 IS THROTTLED AS NEEDED TO CONTROL SODIUM HYPOCHLORITE TO CIRCULATING WATER PUMPS.

OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
SCREEN WASH AND SODIUM HYPOCHLORITE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260084.DGN	
REVISED PER EC MP2-20-01132 AND DUR 20-800224	DRAWING NO: 25203-26008	REV 44
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 4

MBULOGIN\dav1714
21-SEP-2020 11:48
C260084.DGN

INSTRUMENT AIR COMPRESSOR F3F

INSTRUMENT AIR COMPRESSOR F3E

INSTRUMENT AIR DRYER SKID T-52F

INSTRUMENT AIR DRYER SKID T-52E

INSTRUMENT AND STATION AIR P & ID LEGEND			
SYSTEM	DRAWING NO.	BUILDING	BUILDING ELEVATION
INSTRUMENT AIR	25203-26009 SH. 1	TURBINE	14'-6"
	25203-26009 SH. 2	TURBINE	14'-6"
	25203-26009 SH. 3	TURBINE INTAKE STRUCTURE	31'-6" 14'-0"
	25203-26009 SH. 4	TURBINE	31'-6"/54'-6"
	25203-26009 SH. 5	AUXILIARY	14'-6"/38'-6"
	25203-26009 SH. 6	AUXILIARY	(-)5'-0"/(-)45'-6"
	25203-26009 SH. 7	AUXILIARY	(-)25'-6"
	25203-26009 SH. 8	CONTAINMENT	14'-6"/38'-6"/(-)3'-6"/(-)22'-6"
STATION AIR	25203-26009 SH. 8	TURBINE AUXILIARY	1'-6"/14'-6" ALL ELEVATIONS
INSTRUMENT AIR	25203-26009 SH. 9	TURBINE	14'-6"
STATION AIR	25203-26009 SH. 10	INTAKE STRUCTURE CONTAINMENT	14'-6" ALL ELEVATIONS
STATION AIR	25203-26009 SH. 11	TURBINE	4'-10"/14'-6"/31'-6"/54'-6"/YARD

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES OPERATIONS CRITICAL

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
INSTRUMENT AIR SYSTEM COMPRESSORS F-3E & 3F

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: DGNSPEC FOR FILE VERIFICATION	
REVISED PER MSDUR 11-800145 AND DUR 11-800561	DRAWING NO: 25203-26009	REV 34
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

INSTRUMENT AIR COMPRESSOR F3D
(INGERSOLL RAND-ROTARY SCREW)

SH.1 (E-13)

BREATHER

CO6 UA 9903 COMPRESSOR AND DRYER COMMON TROUBLE

OIL BY-PASS LIFT CHECK 2-IA-972

OIL SUMP

2-IA-964

OIL PUMP BY-PASS 2-IA-970

CONTROLLER

DRIVE MOTOR

OIL PUMP

PT 9899 F

PT 9899 E

L-790 OIL FILTER

OIL TEMP. CONTROL VALVE 2-IA-971

TE 9899 F

OIL COOLER

FAN MOTOR

2-IA-962

BLOWDOWN SILENCER 2-IA-976

AFTER-COOLER

TE 9899 D

PT 9899 C

PT 9899 B

TE 9899 C

2-IA-961

H.P. AIR END

INTERSTAGE MOISTURE SEPERATOR

2-IA-963

2-IA-978

L-791A

TE 9899 B

L.P. AIR END

L.P. DISCHARGE SILENCER

INTER-COOLER

BLOWER COOLING AIR FILTER

AIR

L-832D

INLET AIR FILTER L-789

PT 9899 A

TE 9899 A

AIR

TE 9899 F

PT 98990

AFTERCOOLER MOISTURE TRAP

2-IA-975

2-IA-974

3"-HCD-380

HCD SPEC. CHG.

HBD

4"-HBD-468

8"

2-IA-977

SILENCER BLOWDOWN

2-IA-967

L-791B

2-IA-968

2"-HCD-380

2-IA-973

4"x3" RED.

2-IA-766

3"-HBD-468

2-IA-767

TURB. BLDG. WALL

PORTABLE COMPRESSOR TIE-IN

2-IA-768

2-IA-969

½"-HCD-380

2-IA-966

FLOOR DRAIN

HS 1150

2-IA-965

2-IA-511

2-IA-512

2" C.S.

2" C.S.

INSTRUMENT AIR RECEIVER T-51A

PT 9900

1½" TEST

1" ½" 2-IA-513

2-IA-514

8" 3" IF HBD JDD (MOD.)

¼" O.D. S.S.

2-IA-518

2-IA-1128 RO 7972 SPEC. CHG.

¾" S.S. FLEX HOSE S.S. SPEC. CHG.

1" ½" 2-IA-515

½"-HBD-470

¼"-HBD-470

2-IA-516

L-360C

HV 9900

2-IA-517

3"-JDD-67

FLOOR DRAIN

COMPRESSOR INTERNALS LEGEND

OIL

DRIVE

AIR

PDI 9901C ⅜" O.D. S.S. VENT 2-IA-877

2-IA-536A ¼" O.D. 2-IA-535A ¼" O.D. 2" 2-IA-533A

AFTER FILTER L-359A

SPEC. CHG.

2-IA-534A

PDI 9901D ⅜" O.D. S.S. VENT 2-IA-878

2-IA-536B ¼" O.D. 2-IA-535B ¼" O.D.

AFTER FILTER L-359B

SPEC. CHG.

2-IA-534B

SPEC. CHG. JDD (MOD.) C.S. 2"-JDD-68

2-IA-531

2-IA-532

½" REPRESSURIZATION VALVE

2-IA-527C

2-IA-527B ⅜" O.D. COPPER

2-IA-527A ⅜" O.D. COPPER

DRYERS CONTROLS CPT52

INLET SWITCHING AND PURGE EXHAUST VALVE(SEE NOTE 3)

¼" O.D. S.S.

2-IA-525

SPEC. CHG.

2-IA-526C 2-IA-526A 2-IA-526B 2-IA-526D

PI 9901A

HUMIDITY SENSOR PROBE

MS 9901 D

INSTRUMENT AIR DRYER T-52D

RIGHT CHAMBER

PSH 9901 D PSL 9901 D PI 9901D

PSH 9901 C PSL 9901 C PI 9901C

PURGE EXHAUST MUFFLER

HUMIDITY SENSOR PROBE

MS 9901 C

INSTRUMENT AIR DRYER T-52C

LEFT CHAMBER

2-IA-528C 2-IA-528A

FD 9901 A

PI 9901B ¼" C.S.

2-IA-529

2-IA-528D 2-IA-528B

¼" O.D.

OUTLET VALVE ASSEMBLY SEE NOTE 2

PI 9901E

MI 9901E

SILICA GEL MOISTURE INDICATOR

2-IA-530 ¼" O.D. S.S. ¼" S.S.

SPEC. CHG.

PILOT AIR FILTER L-456

PDI 9901A ⅜" O.D. S.S. VENT 2-IA-879

2-IA-522A ¼" O.D. 2-IA-521A

PRE FILTER L-358A

2-IA-520A SPEC. CHG.

¼" O.D. C.S. JDD (MOD.) 3"-JDD-67

2-IA-519A

3"X2" RED. FLG

PDI 9901B ⅜" O.D. S.S. VENT 2-IA-880

2-IA-522B ¼" O.D. 2-IA-521B ¼" O.D.

PRE FILTER L-358B

2-IA-520B SPEC. CHG.

2-IA-519B

2-IA-523A

L-360A

2-IA-523B

L-360B

2-IA-524A

2-IA-524B

HV 9901 A

HV 9901 B

¼" O.D. S.S.

SPEC. CHG.

FLOOR DRAIN

2"-JDD-68

FR/FQI 9978

FE 9978 FIT 9978

2-IA-772 ½"

3"x2" RED

3"-JDD-68

2-IA-537

2-IA-539

2-IA-538

P1 99024 ¼" TEST

2-IA-540

2-IA-541

FUTURE CONNECTION FOR DEW POINT MONITOR

SH.1 (B-4)

½"-JDD-69A

2-IA-543 2-IA-544

INSTRUMENT AIR DRYERS
(INLET, AFTER FILTERS AND DRYERS BY PALL PNEUMATIC PRODUCTS CORP.) (SEE REF 1)

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. VALVES 2-IA-528A THRU 2-IA-528D AND FLOW ORIFICE FO-9901A ARE INTERNAL TO THE OUTLET VALVE ASSEMBLY.
3. VALVES 2-IA-526A THRU 2-IA-526D ARE INTERNAL TO THE INLET SWITCHING AND PURGE EXHAUST VALVE.

REFERENCE DRAWINGS:

25203-29489 SH.1 - INSTRUMENT AIR DRYERS T-52C AND D FLOW SCHEMATIC

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

OPERATIONS CRITICAL

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
20	9-21-10	INCORP. DCN DM2-00-0145-09	EAL	RFL	DLW	MP
19	5-21-09	INCORP. DCN DM2-06-0049-06	JSJ	RFL	WBZ	DED

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE POWER STATION-UNIT 2
PIPING & INSTRUMENT DIAGRAM
INSTRUMENT AIR COMPRESSOR F3D
WATERFORD, CT

BY S.E. BESADE	CHKD S.REINHARDT	APP. SRF	RCB	APP. BJD
DATE 6/1/88	DATE 6/8/88	DATE 8-1-88	8-1-88	DATE 8-1-88
SCALE NONE	DWG. NO. 84-024	25203-26009 SH.9		

SPEED INPUT FROM MAG. PICKUP -26018 SH. 4 (F-3) | WOODWARD 2301A GOVERNOR | WOODWARD DIGITAL REF. UNIT (DRU) | D/G CONTROLS SEE -32041 SH.3

SPEED INPUT FROM MAG. PICKUP -26018 SH. 4 (C-3) | WOODWARD 2301A GOVERNOR | WOODWARD DIGITAL REF. UNIT (DRU) | D/G CONTROLS SEE -32041 SH.15

CLEAN OIL STORAGE TANKS

CLEAN OIL TRANSFER PUMPS

DIESEL GENERATOR H-7A

DIRTY FUEL TANK T-46A

DIESEL OIL DUPLEX FILTER L-79

ENGINE DRIVEN F.O. PUMP P-178A

HAND PRIMING PUMP P-179A

DIESEL OIL SUPPLY TANK T-48A

DIESEL GENERATOR H-7B

DIRTY FUEL TANK T-46B

DIESEL OIL DUPLEX FILTER L-81

ENGINE DRIVEN F.O. PUMP P-178B

HAND PRIMING PUMP P-179B

DIESEL OIL SUPPLY TANK T-48B

DIESEL FUEL OIL STORAGE TANK (25,000 GALLONS) T-148

DIESEL FUEL OIL TRANSFER PUMP P-174A

DIESEL FUEL OIL TRANSFER PUMP P-174B

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT 2 ABBREVIATIONS, SEE 25203-26001, SHEETS, 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. PIPE SIZES ARE IDENTIFIED ON THE P&ID AS FOLLOWS:
   1/2 " = PIPE; 1/2 " O.D. = TUBE
3. CONTAINMENT PIPE IS USED FOR ENCAPSULATING UNDERGROUND LINES 1-1/2-HBC-449 & 450. MATERIAL OF THIS CONTAINMENT PIPE IS IN ACCORDANCE WITH SPECIFICATION SP-ME-688, CLASS HBD.
4. STRAINER BASKET HAS BEEN REMOVED.

OPERATIONS CRITICAL

Q.A.

SUPERSEDES DWG. 25203-26010 THRU REV.4

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
26	2-16-11	INCORP. DCN DM2-00-0046-09	EAL	DWJ	DLW	DLW
25	12-29-09	INC. DCN DM2-00-0311-09	JMS	RFL	WBZ	REB
24	6/17/08	INC. DCN DM2-00-0213-08	JMS	RFL	DLW	MP
23	9-8-04	INC. DCN DM2-00-0207-04	SMC	MP	DLW	DED

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE POWER STATION UNIT-2
PIPING & INSTRUMENTATION DIAGRAM
FUEL OIL SYSTEM
WATERFORD, CT

BY	MRB	CHK'D.	SBR	APP.	WJQ	JGR	APP.	SBR
DATE	1-10-87	DATE	3-30-87	DATE	4/8/87	4/7/87	DATE	4-9-87
SCALE	NONE	DWG. NO.	25203-26010 SH. 1				REV	26

84-024

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL HOSE STATION CONNECTIONS ARE 2½".
3. A HOSE HOUSE SHALL BE PROVIDED WITH EACH YARD HYDRANT.
4. VERTICAL CHASE IS BETWEEN SPREADING AREA (AUX. BLDG.) AND CABLE VAULT (TURBINE BLDG.)
5. DIESEL GENERATOR ROOM PRE-ACTION SYSTEM IS ARRANGED FOR FIRE SUPRESSION AND COOLING OF CEILING STRUCTURAL STEEL TO MEET APPENDIX "R" CRITERIA.
6. VALVE POSITION LIMIT SWITCHES SHALL BE NON-FUNCTIONAL PER PA 87-043
7. ALARMS IN NAP FIRE PANEL.
8. ALL DELUGE AND SPRINKLER VALVES HAVE A LOCAL HEADER AND DRAIN PRESSURE GUAGE, NOT SHOWN.
9. TWO FIRE HYDRANTS ARE USED AS EMERGENCY X-TIE FROM CITY WATER TO MAIN FIRE LOOP VIA HYDRANT HOSE. FOR HYDRANT LOCATION SEE REF. 25205-20003
10. VENT VALVE 2-FIRE-405 IS LEFT OPEN TO PROVIDE LEAK DETECTION PAST VALVE 2-FIRE-34 TO INSURE THAT THE CST IS NOT CONTAMINATED BY FIRE WATER.
11. ALARMS IN UNIT #3 CONTROL ROOM VIA 3CES-PNLFPS.

REFERENCE DRAWINGS:

1. SPRINKLER SYST. DWGS

25203-29242	SH 8	TRANSFORMER
25203-29242	SH 11	AUX. BLDG.
25203-29242	SH 12	T.B. & DIESEL OIL TANK RMS
25203-29242	SH 13	H2 SEAL OIL UNIT
25203-29242	SH 17	T.B. BELOW 54'-6"
25203-29242	SH 18	T.B. BELOW 31'-6"
25203-29242	SH 20	LUBE OIL ROOM
25203-29242	SH 21	FEED PUMP
25203-29242	SH 22	TRANSFORMERS
25203-29242	SH 24	TRANSFORMERS
25203-29242	SH 25	T.B. NE STORAGE AREA
25203-29242	SH 26	TURBINE BEARINGS
25203-29242	SH 28	CTMT. BLDG STANDPIPE
25203-29242	SH 29	CTMT. BLDG STANDPIPE
25203-29242	SH 30	RR. TRUCK ACCESS
25203-29242	SH 31	AUX. FD. PUMP CABLE TRAYS
25203-29242	SH 34	AUX. BLDG. CABLE VAULT
25203-29242	SH 35	AUX. BLDG. CABLE VAULT
25203-29242	SH 36	AUX. BLDG. CABLE TRAYS (-)45'-6"
25203-29242	SH 37	AUX. BLDG. CABLE TRAYS 14'-6"
25203-29242	SH 38A	AUX. BLDG. CABLE VAULT
25203-29242	SH 38B	AUX. BLDG. CABLE VAULT
25203-29242	SH 41	TRANSFORMER SPRAY SYSTEM LAYOUT
25203-29242	SH 42	TRANSFORMER SPRAY SYSTEM DETAILS
25203-29242	SH 43	TRANSFORMER HYDRAULIC REFERENCE DIAGRAM
25203-29242	SH 44	SITE TRANSFORMER LAYOUT
25203-29548		AUX. BLDG. CHG. PUMP WATER CURTAIN
25203-29549		AUX. BLDG. MCC-B-61 WATER CURTAIN
25203-29550	SH 1	DIESEL GEN. ROOM
25203-29550	SH 2	DIESEL GEN. ROOM
25203-29553		T.B. EAST/WEST CABLE VAULT
25203-29554		T.B. VERTICAL CABLE SHAFT
25203-29556		DIESEL GEN. CABLE VAULT
25203-29565		2-FIRE-80, 81, & 145 TRIM DETAILS
25203-29566	SH 1&2	SUPERVISORY AIR (SA) TO D.G. ROOM SPRINKLER VALVES

2. FIRE ZONE DRAWINGS

25203-29378	SH 1	AUX. BLDG. EL. (-)45'-6"
25203-29378	SH 2	AUX. BLDG. EL. (-)25'-6"
25203-29378	SH 3	AUX. BLDG. EL. (-)15'-6"
25203-29378	SH 4	AUX. BLDG. EL. 14'-6"
25203-29378	SH 5	AUX. BLDG. EL. 36'-6"
25203-29378	SH 6	CABLE VAULT & INTAKE STRUCTURE
25203-29378	SH 7	TURB. BLDG. EL. 14'-6"
25203-29378	SH 8	TURB. BLDG. EL. 31'-6"
25203-29378	SH 9	TURB. BLDG. EL. 54'-6"

3. 25203-26011 SH 5 — TYPICAL DETAILS OF DELUGE VALVE ASSEMBLIES AND ALARM CHECK VALVE ASSEMBLIES
4. 25203-26011 SH 3 — HOSE STATION LEGEND
5. 25203-26011 SH 7 — FIRE PROTECTION AUTO SPRINKLER VALVE DETS.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 8

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
FIRE PROTECTION

REVISION DESCRIPTION SIGNATURES ON FILE	CAD NO: C26011.DGN	
REVISED PER DC MP2-19-01111 & DUR 23-800104	DRAWING NO: 25203-26011	REV 59
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

Authenticated By: MBULOGIN ver. ik004
1-MAY-2023 10:35
C26011.DGN

EXISTING 12" CITY WATER LINE

25202-26018 SH.7 (J-10) CITY WATER SUPPLY FROM METER PIT — 12" — 2-FIRE-93 LO — 25202-26018 SH.6 (G-10) TO FIRE WATER STORAGE TANKS M7-6A & 6B & DOMESTIC WATER TANK M7-10

25202-26030 SH.1 (K-12) TO MP#1 & 2 WATER TREATMENT SYSTEMS — 2"-JH-31 — 3-WTS-004-358-4 — FROM UNIT 2 AND UNIT 3 CONTRACT WATER TREATMENT FACILITY EM-1206 (H-6) 25212-26920 SH. 7

3-WTS-004-359-4 TO OR FROM UNIT 3 WATER TREATMENT SYSTEM EM-1206 (G-10) 25212-26920 SH. 7

2-WW-65 LO — 8"-KE-1 — 2" CONT. NOT VERIFIED

UNIT 2 TURBINE BLDG. ACCESS AREA — WAREHOUSE & UNIT 2 POLISHING FACILITY — YARD

2-WW-251 — 8"-KE-1 — 2-WW-252

UNDERGROUND / ABOVEGROUND — SPEC. CHG. — 8"-JW — 1"JH-62

2-WW-373 — 2-WW-371 BACKFLOW PREVENTER — 2-WW-372 — 8" CITY WATER HEADER

2-WW-439 — 2-WW-446 BACKFLOW PREVENTER — 2-WW-447 — 2-WW-448 — EMERG. SHOWER EYEWASH

3" SUPPLY VALVES (TYP. 11) — VENDOR

HYDRA-STOP FLANGED STYLE FITTING (INTERNALS REMOVED)

3" DISCHARGE VALVES (NC TYP. 15) — VENDOR — 8" CST HEADER

8"-JH-51 — UNDER GROUND — TO CONDENSTATE STORAGE TANK 26005 SH.3 (J-5) — 2-CN-612 DRN. — FLOW METER

CONTRACT WATER TREATMENT FACILITY BUILDING (BLDG. #215)

8"-KE-1 UNDERGROUND — ABOVE GROUND / UNDERGROUND — SPEC. CHG. — 2-WW-242A — BLIND FLANGE — NOTE 6

VENDOR — 2-PMW-433 — 4"-JH-52 — UNDERGROUND PIPING

TO #5 WAREHOUSE 25212-26947 SH. 2 (M-7) — 3-DWS-V993 — 2" — SPEC. CHG. — 2-WW-19 — 3"-JDD-10

6"-JBD-50 — 2-WW-97 — 3"-JDD-11 — 4"-JBD-52 — 2-WW-90 — 2-WW-268

TO 3" HEADER 26011 SH.6 (B-6)

ELECTRIC HYDROLASER — PI 7965 B — 2-WW-311 — PI 7965 A — 2" (TYP.) — 2-WW-225 — 2-WW-226 — 2-WW-227 — 2-WW-228 — 2-WW-250B — 2-WW-250A — 2-WW-375 BACKFLOW PREVENTER — 3"-JDD-65 — 2-WW-222 — 2-WW-177

MAINT. DECON RM. — RECIRC. SINK — TO CLEAN WATER SPRAY VALVE — WASH SINK — SAMPLE SINK -45'-0"

TURBINE BUILDING

AUX. BLDG. — HEALTH PHYSICS MEZZANINE

2-WW-23 — 2-WW-55 — 2-WW-24 — 2-WW-54 — 2-WW-33 — 2-WW-53 — 2-WW-26 — 2-WW-58 — 2-WW-285 — 2-WW-29 — 2-WW-51 — 2-WW-31 — 2-WW-52

1"-JDD-15 — AUX 14'-6" RR. ACCESS — EMERG. SHOWER — 2-WW-324 — 2-WW-284 — 1"-JDD-10 — 2-WW-256

2-WW-25 — 2-WW-57 — 2-WW-286 — VENT — 2-WW-184 — 2-WW-176 — 2-WW-145 — 2-WW-146 — 2-WW-147 — WW-19 — WW-21 — 1"-JDD-10

VENT — 2-WW-175 — 2-WW-48 — 1"-JDD-15 — 2"-JDD-10

2-WW-347 — 2-WW-308 — 2-WW-341 — 2-WW-43 — 2-WW-50 — 2-WW-148 — 2-WW-173 DRN. — 1"-JDD-13 — WW-20 (DISCONNECTED) — 2-WW-41 — VENT 2-WW-172 — WW-32 — WW-30 — WW-34 (DISCONNECTED)

2-WW-44 — 1"-JDD-14 — WW-28 — 2-WW-45 — VENT 2-WW-174 — 1"-JDD-13 — WW-22 — 2-WW-49 — 2-WW-42 — 2-WW-171

AUXILIARY BUILDING

2-WW-170 — 2-WW-169 — 2-WW-281 — 2-WW-282 — 2-WW-280 — 2-WW-283 — WW-37 — 2-WW-316 — AUX 14'-6" BORIC ACID TK — 2-WW-317 — EMERG. SHOWER EYEWASH

WW-17 — 1"-JDD-13 — 2-WW-37 — 2-WW-309 — 2-WW-320 — 2-WW-47 — 2-WW-319 — WW-18 — 2-WW-321 — 2-WW-322 — RECIRC. SINK — TO CLEAN WATER SPRAY VALVE — WASH SINK — 2-WW-323 — MAINT. DECON RM.

2-WW-168 — 2-WW-276 — 2-WW-312 — 2-WW-38 — 2-WW-278 — 2-WW-279 — WW-38 — WW-16 — 2-WW-318 — 2-WW-277 — SAMPLE SINK -45'-0" — 2-WW-39 — WW-15

2-WW-143 — EWC 2 — 2-WW-351 — 2-WW-350 — PW-2 — 2-WW-78 — 2-WW-77 — 2-WW-36 — 2-WW-35 — 2-WW-76 — EWC 1 — 2-WW-34 — P/W-1 — 2-WW-348

2-WW-157 — 2-WW-155 — 2-WW-156 — 2-WW-349

1"-JDD — 2-WW-325 — SUPPLY TO OUTAGE TRAILER — 2-WW-260 — 2-WW-263 — PW-6 — 2-WW-287 — 2-WW-328 — DRAIN — L-444 — 2-WW-288 — PSV 7843 — PW-4 — HW-6 — HW-7 — 2-WW-289 — 2-WW-327 — X-81 — DRAINS TO FL 14'-6" — ELECTRIC HOT WATER HEATER — 2-WW-262 — 2-WW-261 — 2-WW-259 — TO COMPUTER ROOM HUMIDIFIER — WW-27 — 2-WW-460 — SOLIDIFICATION BUILDING — WW-35 — 2-WW-258 — 2-WW-329 — 2-WW-326 — 2-WW-292 — 2-WW-291 — WW-36 — 2-WW-290 — EMERG. SHOWER EYEWASH

RETIRED IN PLACE PER DCR M2-00001

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT 2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL WASHWATER CONNECTIONS ARE PROVIDED WITH 1" RUNOUTS TO QUICK DISCONNECTS.
3. POTABLE WATER EQUIPMENT IS LOCATED IN MISCELLANEOUS STORAGE AREA.
4. DELETED.
5. TECHNICAL REQUIREMENTS FOR PIPE MATERIAL& SIZES SHALL BE IN ACCORDANCE WITH SPECIFICATION SP-ME-819.
6. A BACK-FLOW PREVENTER (BFP) IS REQUIRED TO BE INSTALLED PRIOR TO USE OF THIS CONNECTION TO MEET CT. PUBLIC HEATH CODE SECTION-3-R38A.

LV 8606 — A/S — 2" — 2-WW-253 — 2-WW-92 — 2-WW-91 — 2-WW-98 — 1" — 2-WW-254 — DRN. — OPEN ENDED — 4"-JBD-48 — 2-WW-93 — STOPS ON LOW-LEVEL — 2-WW-57A — VENT — DOMESTIC WATER STORAGE TANK T-85 — 4" — OVERFLOW — 3"-JDD-33

(A-11) — 2-WW-369 — 2-WW-354 — L-722 — BACKFLOW PREVENTOR 2-WW-355 — 3/4"-JDD-71 — TO 2-HP-43 26008 SH.4 (D-11)

LC 8606 — LC 8642 — LC 8643 — LC 8606 — A/S — JBD JDD — 2-WW-434A — 2-WW-434C — 2-WW-434D — 2-WW-434B — 2-WW-433A — 2-WW-433C — 2-WW-433D — 2-WW-433B — 2-WW-94A — 2-WW-432A — 2-WW-432C — 2-WW-432D — 2-WW-94B — 2-WW-432B — 2-WW-94C — 2-WW-57B

DOMESTIC WATER BOOSTER PUMPS — 26008 SH.1 (C-12) — TY 8666 — TS 8666 — HS 8666 — PY 6514 — FO — S — A/S — PI 6910 — 2-WW-2 — 2-WW-89 FO — 2"X1½" RED — 2-WW-313 — 2-WW-314 — BLIND FLG. — 2"-JDD-6

2-WW-181 — FC — 2-WW-95 — 1"-JDD-34 — 3"-JDD-34 — 3"X2" RED — 2-WW-67B — 2-WW-158 — 2-WW-154 — 2-WW-568 — 2-WW-56A — PI 7294 — 2-WW-100A — 2" — 3" — 2-WW-67A — 4" O.D. — L-522A — 2-WW-14A — 1½" X 1" — 2-WW-166 — P-77A — NP77A — HS 7292 — NP77A — HS 7293 — NP77B — PI 7295 — 3/4" O.D. — 2-WW-100B — 2" — 3"-JDD-34 — L-522B — 2-WW-14B — 1" — 2-WW-165 — P-77B — DRN. — MECH. VACUUM PUMPS 26012 (K-4)

FD 8606 B — FD 8606 A — BLIND FLANGE — 1"-JDD-34 — 3"-JDD-34 — 3"-JDD-33 — 3"-JDD-7 — 2-WW-104 — 2-WW-105 — 2-WW-107 — 2-WW-163 — 2-WW-103 — 2-WW-108 — 2-WW-182 — 2-WW-428 BACKFLOW PREVENTER — 2-WW-427 — 2-WW-429 — 2-WW-165 — 2-WW-1 — 2-WW-14B — 2-WW-164 — PCV 8614 — 2-WW-200 — 1½" — JDD JBD — 2-WW-201 — ABOVE GND BELOW GND — 1" — 2-WW-241 — 1"-JDD-8 — 2-WW-239 — 2-WW-334 — T.B. 31'-6" ACCESS CONTROL CENTER — EMERG. SHOWER/ EYEWASH — 26012 (K-4) — TO WATER BOX PRIMING PUMP — 3"-JDD-7 — 3"X1" RED. — 1"X½" RED. — VENT — 2-WW-353

2-WW-424 — 2-WW-425 BACKFLOW PREVENTER — 2-WW-426 — 2-WW-128 — 2-WW-200 — 1½" — JDD JBD — 2-WW-201 — T.B. 14'-6" T.B. 31'-6"

2-WW-183 — 3/4"-JDD-11 — 2-WW-178 — 1½"X1" RED. — WW-11 — 2-WW-23 — 1½"-JDD-11 — 2-WW-167 1½"X1" RED. — 1" — WW-7 — 2-WW-24 — 2-WW-32 — WW-9 — 2-WW-382 DRAIN — 1½"X1" RED. — 2-WW-21 — WW-10 — T.B. 14'-6" CHEMICAL ADDITION AREA — EMERG. SHOWER EYEWASH — 2-WW-161 — 1½" — 1½"-JDD-11 — 2-WW-22 — 2-WW-162 — 2-WW-33 — WW-13 — 2-WW-215 — 1½" — 1½"-JDD-11

2-WW-79 — EWC 6 — 2-WW-80 — 2-WW-81 — WW-8 — WW-12 — 2-WW-27 — 2-WW-26 — WW-14 — WW-15

2-WW-271 — 2-WW-31 — 1"-JDD-11 — WW-5 — 2-WW-272 — 2-WW-30 — WW-1

2-WW-18 — WW-6 — 2-WW-20 — 1" — 2-WW-214 — 1½"-JDD-11 — 2-WW-338 — PCV 7805 — 2-WW-423 — 2-WW-337 — 2-WW-335 — 2-WW-421 — PI 7805 — 2-WW-336 — 2-WW-422 — 2-WW-151 — 2-WW-270 — 2-WW-269 — 3"-JDD-2 — 4"-JBD-52

1"-JDD-37 — 2-WW-129 — 1½"-JDD-37 — 3"-JDD-37 — PSV 7906 — 2-WW-125 — PI 7903 — 2-WW-211 — LS 7906 — CONTROL PANEL C82 — HEATER ELEMENT — DOMESTIC WATER HOT WATER TANK T-28 — NORTH SHOWER COLD WATER SUPPLY — EAST SHOWER COLD WATER SUPPLY — 2-WW-274 — 2-WW-275 — 2-WW-39 — 1½"-JDD-39 — TS 7909 — TS 7908 — 2-WW-124 — 2-WW-450 — 2-WW-117 — 2-WW-273 — HS 186 — KC 114 — EXP TK T-149 — P1 6952 — S.S. JDD — P1 6951 — 2-WW-455 — 2-WW-454 — 2-WW-453 — 2-WW-452 — P186 — HOT WATER CIRC. PUMP — PCV 7888 — 2-WW-116 — 2-WW-142 — 2-WW-154 — 2"-JDD-39 — 3/4"-JDD-40 — 1"X3/4" RED. — 2-WW-134 — 1"-JDD-37 — 1½"X1" RED. — 2-WW-133 — ¾"-JDD-37 — 2-WW-132 — ¾"X¼" RED. — 2-WW-180 — HW-5 — 26011 SH.6 (F-5) FROM 2-WW-123 DOMESTIC HOT WATER RECIRC.

TO 2-WW-122 DOMESTIC HOT WATER SYSTEM 26011 SH.6 (F-4) — 2-WW-153 — ¾"X½" RED. — 1½" — HW-1 — 2-WW-84 — HW-2 — 2-WW-85 — HW-3 — 2-WW-86

2-WW-25 — 2-WW-28 — 2-WW-159 — 2-WW-29 — WW-4 — 1"-JDD-11 — 2-WW-160 — DRN. — 1"-JDD-6 — ABOVE GR. — BELOW GR. — 2-WW-158 — SUPPLY TO GUARDHOUSE BLDG. 201 — 1½"GALV. PIPE UNDERGROUND

AUX. STEAM SURGE TK. SAMPLE COOLER X-211 — 26026 SH.1 (C-12) AUX .STM.& COND.

TO CIRC. PUMPS 26008 SH. 1 (E,F-13) — 2"-HCD-60 — 2-WW-101 — 1"-JDD-73 — 2-WW-330 — 2-WW-294 — 2-WW-307 — 2-WW-356 — 2-WW-296 — 2" — JDD JBD — 2-WW-304 — 2-WW-305 — 2-WW-303 — 2-WW-302 — 2-WW-298 — 2-WW-297 — 3" X 2" RED. — INTAKE STR BELOW GND — 2-WW-306 OUTSIDE BLDG. — INSIDE BLDG. — 2-WW-301

INTAKE STRUCTURE — EMERG. SHOWER/ EYEWASH — 2-WW-265 — 2-WW-295 — 2-WW-221 — 3/4"-JDD-6 — 2-WW-220 — 3/4"-JDD-71 (D-13)

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy® — Dominion Nuclear Connecticut, Inc. Millstone Power Station — Waterford, CT.

TITLE: MILLSTONE POWER STATION-UNIT 2
PIPING & INSTRUMENT DIAGRAM
DOMESTIC WATER

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260112.DGN	
REVISED PER DC M2-23-01133 & DUR 24-800132	DRAWING NO: 25203-26011	REV 64
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

Authenticated By: MBULOGINver1x004
4-DEC-2024 11:13
C260112.DGN

PIPING & INSTRUMENTATION DIAGRAM
COND. AIR REMOVAL WATER BOX PRIMING & TURBINE BLDGS. SUMP

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C26012.DGN
REVISED PER EC MP2-21-01172 AND DUR 24-800021 | DRAWING NO: 25203-26012 | REV 76
DSGN DWJ | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 0

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

THIS DRAWING IS REDRAWN FROM 25203-26012 THRU REV.28

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. 18 GAUGE / 8 MESHES S.S. SCREEN / STRAINER INSTALLED
3. TYGON TUBING ONLY INSTALLED WHEN REQUIRED BY PROCEDURE.

MBULOGIN\er ik004
18-JAN-2024 12:24
C26012.DGN

REACTOR COOLANT PUMP 1B (SH. 3)

REACTOR COOLANT PUMP 2A (SH. 3)

REACTOR COOLANT PUMP 1A (SH. 3)

REACTOR COOLANT PUMP 2B (SH. 3)

STEAM GENERATOR 1
REFER TO DRAWING 26002 & 26005 FOR SECONDARY SIDE INSTRUMENTATION

STEAM GENERATOR 2
REFER TO DRAWING 26002 & 26005 FOR SECONDARY SIDE INSTRUMENTATION

REACTOR VESSEL

LOOP 1B

LOOP 2A

LOOP 1A

LOOP 2B

PRESSURIZER SPRAY LINE
SH. 2 (H-3)

PRESSURIZER SURGE LINE
SH. 2 (F-6)

PRESSURIZER PRESSURE CONTROL
25203-29199 SH.5

INADEQUATE CORE COOLING MONITOR

TM/LPT SETPOINT &ΔT POWER REFERENCE CALCULATORS

T. AVG. CALCULATOR

RRS T. AVG. CALCULATOR
25203-28200 SH.1271B

RRS T. AVG. CALCULATOR
25203-28200 SH.1271C

INDICATION & ALARM
25203-29199 SH.8

SAFETY INJECTION & SHUTDOWN COOLING
26015 SH. 3 (C-12)
2-SI-237

SAFETY INJECTION & SHUTDOWN COOLING
2-SI-247

SAFETY INJECTION & SHUTDOWN COOLING
26015 SH. 3 (E-12)
2-SI-227

SHUTDOWN COOLING
26015 SH. 3 (G-5)

LOOSE PARTS MONITOR

SPEAKER RC-24

FLOW INDICATION & LOW FLOW TRIP
25203-28500 SH. 72, 76, 77, 78

CHARGING LINE
26017 SH. 1 (J-13)

LETDOWN LINE
26017 SH. 2 (J-13)

PRIMARY DRAIN HEADER
26024 SH. 1 (D-10)

SYSTEM VENT
26028 SH. 3 (E-4)

TO CONTAINMENT BUILDING DRAINS
26024 SH. 1 (D-5)

PRESSURIZER VENT
26014 SH. 2 (K-8)

N² SUPPLY
26020 SH. 5 (F-9)

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 9

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. THIS SECTION OF ¾" -HRC-6 WILL BE REMOVED AS AN INTERFERENCE WHEN CACS/CONTAINMENT PURGE SYSTEM CROSS CONNECT DUCTWORK IS INSTALLED DURING MODES 5 OR 6.
3. THE PRESSURIZER SURGE LINE MATERIAL IS 12" SCH. 160, STAINLESS STEEL, TYPE 316, SEAMLESS ASTM-A-351, GR CF8M
4. OPEN-ENDED GOOSENECK VENT INSTALLED FOR FME AND TESTING.

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION - UNIT 2
PIPING AND INSTRUMENTATION DIAGRAM
REACTOR COOLANT SYSTEM
WATERFORD, CT

REVISION DESCRIPTION (SIGNATURES ON FILE)
REVISED PER DC MP2-19-01181 & DUR 21-800209

CAD NO: C260141.DGN
DRAWING NO: 25203-26014
REV: 43

DSGN EAL | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 1

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27-OCT-2021 14:37
C260141.DGN

SAMPLING 26025 SH. 1 (K-13) X-64

2RC45 #=

CIAS CLOSES

HS 7690 C72

HY 7690 A/S

ZS 7690

HV 7690

21

PRIMARY DRAINS & QUENCH TANKS COOLING LINE SAMPLE

26020 SH.5 (E-9)

3/8" O.D.

2-S-739

1" X 3/8"

2-S-738 L-737 2-S-737

1" X 3/8" DR. 2-S-740

3/8" O.D. SST

2-RC-98 SPEC CHG

3/4"X3/4" 2-RC-45 FC

3/4"-CCB-10

1/2"X3/4" DELAY COIL

3/4"-CCB-10 2-RC-65

3/8" O.D. SST 2-RC-434 LC

2-RC-435 LC

26074 SH. 1 (H-9) P.A.S.S.

3/8" O.D. SST

2-RC-431

2-RC-432

3/8" SS O.D.

HS 1060 CIAS CLOSES

HY 1060 A/S

2RC003 #=

ZS 1060 FO

HV 1060

3/8" SS O.D.

3/4"-CCB-10 2-RC-003 FC

SPEC CHG

3/8"-CCB-10

SH. 1 (G-10) PRESSURIZER SURGE LINE

SH. 1 (B-10) REACTOR COOLANT

3/4"-CCB-10

2-RC-066

CCB

INSTR. CL. C-2

AN 9782

3/4" S.S. TUBE

MANUAL VENT

SH.1 (J-10) TO EBFS

INSTR. CL. C-4

CCB

INSTR. CL. C-2

L.C. 2-RC-062

1/2" T.C.

3/4"-ECD-3

ECD

2-RC-51

2-RC-50

BCA

3/4"-BCA-3

1"X3/4"

1"X1/2"

2-RC-52

3/4"-BCA-3

3/8" S.S. TUBING

INSTR. CL. C-2 BCA

ZAH 200

ZS 200

TAH 107

TI 107 C03

TT 107

TE 107

10"-ECC-1

6"-ECC-1

1"-BCA-3

ZAH 201

ZS 201

TE 108

TT 108

TI 108 C03

TAH 108

2-RC-200 SPEC CHG 2 1/2" 4"-BCA-4

2-RC-201 SPEC CHG 2 1/2" 4"-BCA-4

6"-ECC-1

4"-BCA-2

4"-CCA-11

2" TEE (TYP)

2"X1" RED (TYP)

1"-BCA

2-RC-204

2"X1/2" RED (TYP)

1"-BCA

2-RC-205

1/2" O.D. S.S.

PT 102 D

PI 102 B-1 C-10

P102B1 PCA5

PT 102 B-1

PR 102 B-1 C-02

SH. 1 (K-6) PRESSURIZER SPRAY LINE

2"-CCA-12

26017 SH. 1 (F-13) AUXILIARY SPRAY

HS 1403 HS 1402 HS 1404 HS 1405 CO3F

4"-BCA-2

4"X2 1/2" 2-RC-403

M

ZS 403

2RC403 #=

HS 1402B

HS 1404B

2RC405 #=

ZS 405

4"X2 1/2" 2-RC-405

2 1/2"-BCA-2

25203-29199 SH.4 PRESSURIZER PRESSURE SAFETY CHANNEL

BCA 2-RC-402 FC ECC

HS 1402 A C70A

ZS 402A

ZAH 402

HS 1404 A C70B

ZS 404A

2-RC-404 BCA FC ECC

TW SPARE

TAH 114

TE 114 TT 114 TI 114 C03

ZAH 404

C03

TAH 106 TI 106 TT 106 TE 106 C03

TT-115 26014 SH. 1 (G-9)

TT-125 26014 SH. 1 (H-4)

ATWS AMSAC

25203-29199 SH.4 SAFETY CHANNELS WITH INDICATION

PT 102 A

1/2" O.D. S.S.

1"-BCA 2-RC-206

25203-29199 SH.5 PRESSURE CONTROL

PT 100 X

P103-1

PT 103-1

1/2" O.D. S.S.

2-S1-651, 624, 614 26015 SH.3 (F-4,F-10,H-10)

PI 103-1 B C-21

1"-BCA 2-RC-208

25203-28500 SH 69A LEVEL CONTROL

LT 110 X

PRESSURIZER T-37

1"-BCA 2-RC-207

BCA (TYP 4 PLACES) INSTR. CL. C-2

TE 109 TT 109 TI 109 C03

T109

TE 101 TT 101 TI 101 C03

T101

PT 102 B

25203-29199 SH.4 SAFETY CHANNELS WITH INDICATION

ATWS AMSAC

PT 100 Y

25203-29199 SH.5 PRESSURE CONTROL

1/2" O.D. S.S.

PT 103

2-S1-652, 634, 644 26015 SH.3 (G-5,D-10,B-10)

PI 103 PI 103B C-21 P103

1"-BCA 2-RC-209

LT 110 Y

25203-28500 SH 69A LEVEL CONTROL

HS 304C I 304C CO3F

LT 103 PCA-5 L103

HS 404A I 404A CO3F

LI 103-1

IE 304C IE 404A

LI 103 C-03

25203-28500 SH 50B/50C BACK UP HEATER CIRCUIT BREAKERS

25203-28500 SH 50D PROPORTIONAL HEATER CONTROLS

12"-BCA-1 NOTE 3 (SH. 1)

SH. 1 (H-8) PRESSURIZER SURGE LINE

4"-ECC-1

1-323-01

HS 7311 CIAS CLOSES A/S HY 7311 FO 2PMW43 #= ZS 7311

VENT 2-PMW-318 HV 7311 FC 2-PMW-43 SPEC CHG LC 2-PMW-165 T.C.

VENT 2-PMW-2

26020 SH. 5 (F-6) NITROGEN SUPPLY

3/4"-HCD-31

2-PMW-1

2"-HCD-37 SPEC CHG HCD HCC 2-RC-243 2"-HCC-48

26030 SH. 1 (E-3) PRIMARY MAKE-UP WATER

2-PMW-319 2"-HCD-37 2-PMW-3 3/4" 2"-HCB-4 SPEC CHG

1 1/2"-HCD-37

26023 SH. 1 (H-11) REACTOR VESSEL DECON. SYSTEM

1 1/2"-HCD-82

WASTE GAS HEADER 26021 SH. 2 (C-3)

2"-HRC-15 SPEC CHG 2-RC-242

HRD HSD HRD SPEC. CHG. HRC 3/4"-HRD-8 WT-13 2-GR-153 3/4"

HS 400 HY 400 A/S FO ZS 400 SPEC CHG 1 1/2"-HCC-47

CONTAINMENT SUMP 2-PMW-213

2-PMW-420 2-PMW-421 1"-HCD-37 2-GR-100 DRAIN

2-PMW-315 2-PMW-314 2-PMW-313 2-PMW-422

3/4"-HRC-15 2-RC-400 FC

T-105 SH. 3 (H-1) TO 2-RC-408

SPEC CHG HCC HCD 2-RC-430

2-RC-005 VENT 3/4"-HCC-47 1 1/2"

2-PMW-302 2-PMW-303 2-PMW-304 2-PMW-306 2-PMW-301

1 1/2"-HCD-82

2-PMW-307 2-PMW-308 2-PMW-309 2-PMW-316 2-PMW-312 2-PMW-310

1 1/2"-HCD-82

2"-HCC-57 26015 SH. 3 (J-11) HPSI TEST LINE RELIEF

3"-HSC-22 26020 SH. 5 (D-11) PDT & QT COOLER

3"X2" RED. 2-RC-466 HCD ECD(C) ECD(C) 2"-ECD(C)-9 ECC

3" SPEC CHG 2-RC-41

2" SPEC CHG 2-RC-42

3" 10"-ECC-1

2"-HCC-48

18" 10"

P-28 26020 SH. 5 (C-9)

2"X1" RED (TYP)

2"X1/2" RED

1" 2-RC-240

1/2" O.D. S.S.

SPEC. CHG.

QUENCH TANK T-38

P116 PI 116 PT 116 3/8" O.D. S.S.

PAH 116 PA 116

LT 116 1/2" O.D. S.S. SPEC. CHG. 2-RC-241

LAHL 116 LA 116 L116 LI 116

HS 401 HY 401 2RC401 #= A/S FO ZS 401

3" 2" 2-RC-004 2-RC-401 FC

2"-HSC-58 3"-HSC-59

26024 SH. 1 (E-10) PRIMARY DRAIN HEADER

26020 SH. 5 (A-6) PDT & QT COOLER

TAH 116 TA 116 T116 TT 116 TI 116 C03

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 10

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion Energy®

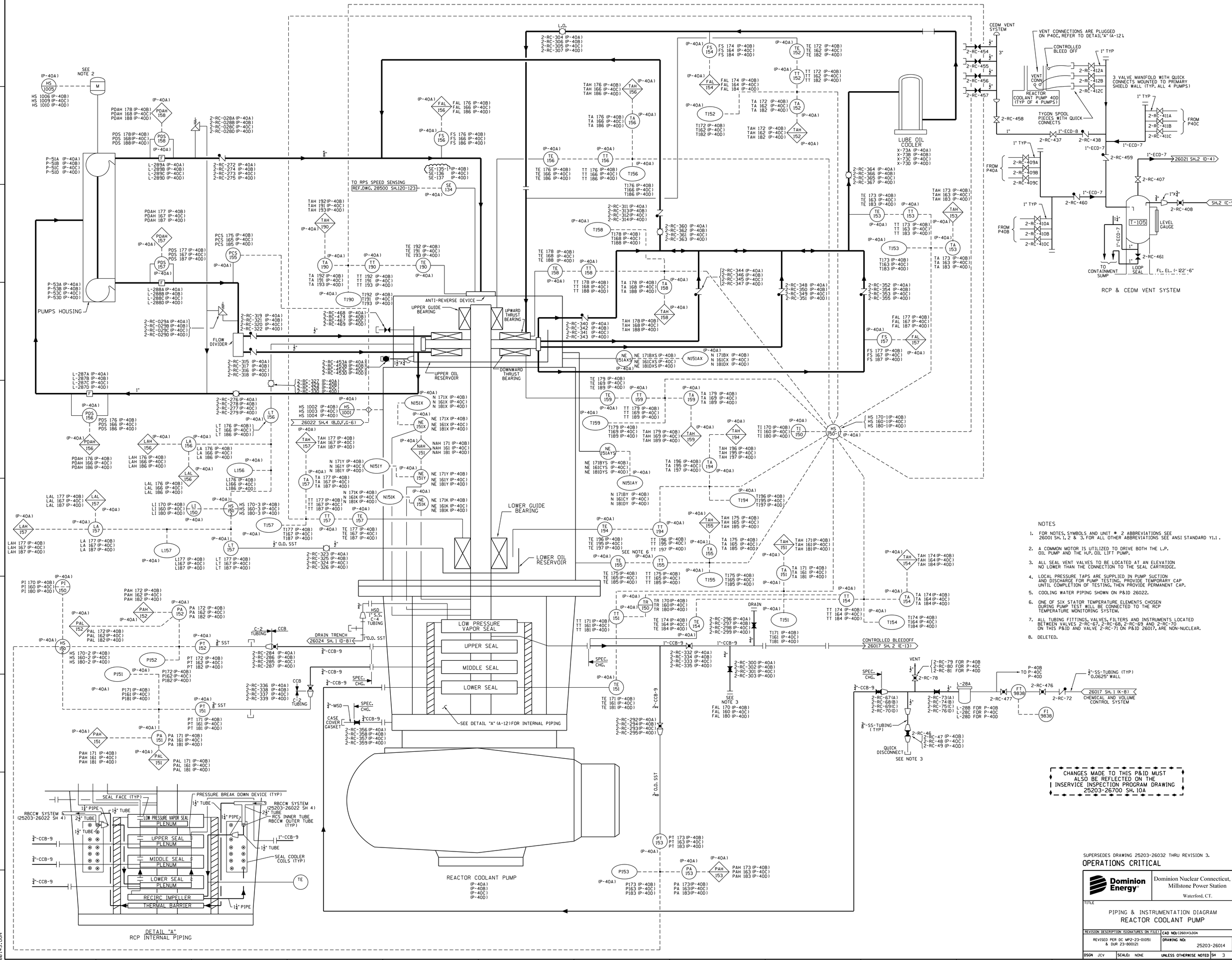
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
REACTOR COOLANT SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260142.DGN	
REVISED PER TEC MP2-24-01017 & DUR 24-800159	DRAWING NO: 25203-26014	REV 52
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 2

Authenticated By: MBULOGIN\scot84
23-OCT-2024 22:39
C260142.DGN



NOTES

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. A COMMON MOTOR IS UTILIZED TO DRIVE BOTH THE L.P. OIL PUMP AND THE H.P. OIL LIFT PUMP.
3. ALL SEAL VENT VALVES TO BE LOCATED AT AN ELEVATION NO LOWER THAN THE CONNECTION TO THE SEAL CARTRIDGE.
4. LOCAL PRESSURE TAPS ARE SUPPLIED IN PUMP SUCTION AND DISCHARGE FOR PUMP TESTING. PROVIDE TEMPORARY CAP UNTIL COMPLETION OF TESTING, THEN PROVIDE PERMANENT CAP.
5. COOLING WATER PIPING SHOWN ON P&ID 26022.
6. ONE OF SIX STATOR TEMPERATURE ELEMENTS CHOSEN DURING PUMP TEST WILL BE CONNECTED TO THE RCP TEMPERATURE MONITORING SYSTEM.
7. ALL TUBING FITTINGS, VALVES, FILTERS AND INSTRUMENTS LOCATED BETWEEN VALVES 2-RC-67, 2-RC-68, 2-RC-69 AND 2-RC-70 ON THIS P&ID AND VALVE 2-RC-71 ON P&ID 26017, ARE NON-NUCLEAR.
8. DELETED.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 10A

SUPERSEDES DRAWING 25203-26032 THRU REVISION 3.

OPERATIONS CRITICAL

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
REACTOR COOLANT PUMP

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260143.DGN	
REVISED PER DC MP2-23-01051 & DUR 23-800121	DRAWING NO: 25203-26014	REV 46
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

Authenticated By: MBULOGIN_ame694
8-MAY-2023 16:52
C260143.DGN

25203-26015 SH-0000000

NOTES:

1. FOR NOTES SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1
2. PORTABLE FLOW COMPUTER CONNECTED TO THE TRANSDUCER DURING PERIODIC TESTING.
3. PORTIONS OF LINE HCD-3(S.I. TEST HEADER) WAS UPGRADED TO SEISMIC CLASS 1 PER PDCR 2-222-76. THE UPGRADED PORTIONS OF THE LINE TO SEISMIC CLASS 1 ARE NOW REFLECTED AS HCD-3(C)1
4. DELETED.
5. VALVES 2-CS-3A, & 3B ARE BDB FLEX ALTERNATE RCS MODES 5 & 6/ CONTAINMENT SPRAY CONNECTION TIE-IN POINTS. REFERENCE DC MP2-13-01131.

FSAR FIGURE

OPERATIONS CRITICAL

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 11

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
L.P. SAFETY INJECTION SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260151.DGN	
REVISED PER MSDUR 17-000128 & DUR 17-800268	DRAWING NO: 25203-26015	REV 50
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

HIGH PRESSURE SAFETY INJECTION PUMPS

HIGH PRESSURE SAFETY INJECTION PUMPS

REFUELING WATER STORAGE TANK T-41

RWST CIRC. PUMP

RWST HEAT EXCHANGER

SAFEGUARDS ACTUATION SYSTEM

CONTAINMENT SUMP

SUMP SCREEN STRAINER L-795 SEE DWG 25203-11173

ENCAPSULATION SEE DWG 25203-20080

M2TSP-BASKETS TRISODIUM PHOSPHATE BASKETS (TYP 5)

TWO (2)@ 105.76 CUBIC FT

THREE (3)@ 31.25 CUBIC FT

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. DELETED PER DCN#DM2-00-0740-00
3. THIS GAUGE INSTALLED ONLY PER AOP2579A. NORMALLY STORED IN THE FIRE SHUTDOWN STORAGE CABINET.
4. PORTIONS OF LINE HCD-3(S.I. TEST HEADER) WAS UPGRADED TO SEISMIC CLASS 1 PER PDCR 2-222-76. THE UPGRADED PORTIONS OF THE LINE TO SEISMIC CLASS 1 ARE NOW REFLECTED AS HCD-3(C).
5. CHECK VALVE INTERNALS HAVE BEEN REMOVED FOR VALVE 2-CS-26
6. LINES 6"-CCA-6, 3"-CCA-7, 2"-CCA-7 & 2"-CCA-9 UPSTREAM OF VALVES 2-SI-706A/B/C/D ARE ASME III CLASS 2 PER DCN DM2-00-855-98.
7. ¼" DIAMETER HOLE IN 18" SUPPLY LINE AT VENT CONNECTION
8. PP IDENTIFIES SI HEADER BEING MONITORED.
9. 4" NPT X 5" STORZ HOSE CONNECTION SPOOL TO BE INSTALLED AS REQUIRED AND BE STORED AS FOLLOWS: (1) LOCATED ON HANGER NO. 505464 AND (1) LOCATED IN THE BDB STORAGE LOCATION.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 12

OPERATIONS CRITICAL

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
HIGH PRESSURE SAFETY INJ. PUMPS

REVISION DESCRIPTION SIGNATURES ON FILE	CAD NO: C260152.DGN	
REVISED PER EC MRP-21-01117 & DUR 21-800232	DRAWING NO: 25203-26015	REV 51
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

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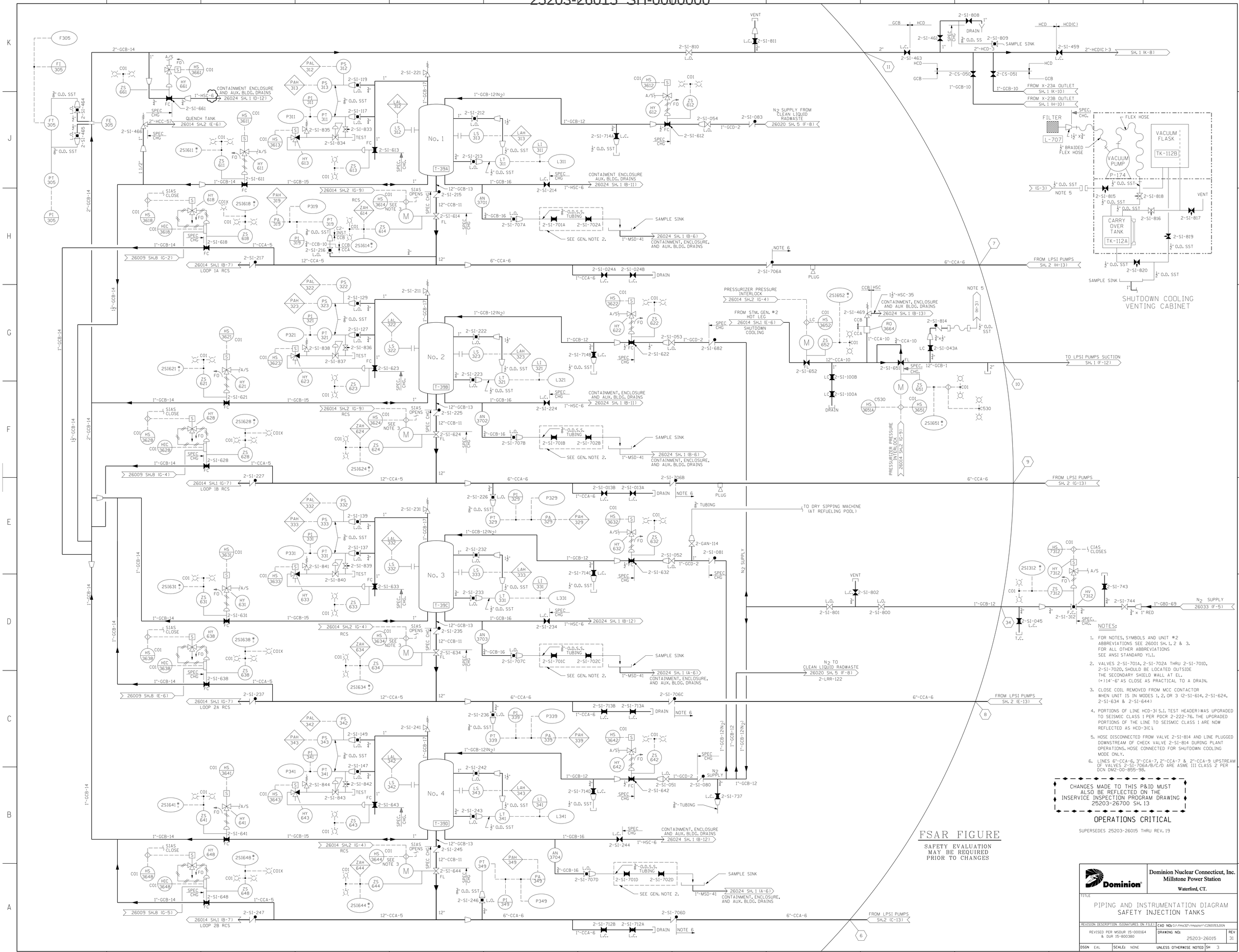
04-NOV-2021 09:32
C260152.DGN


No. 1
No. 2
No. 3
No. 4
T-39A
T-39B
T-39C
T-39D
SHUTDOWN COOLING VENTING CABINET
VACUUM FLASK TK-112B
VACUUM PUMP P-174
CARRY OVER TANK TK-112A
FILTER L-707
QUENCH TANK 26014 SH.2 (E-6)
CONTAINMENT ENCLOSURE AND AUX. BLDG. DRAINS 26024 SH. 1 (D-12)
N2 SUPPLY FROM CLEAN LIQUID RADWASTE 26020 SH. 5 (F-8)
FROM X-23A OUTLET SH. 1 (K-10)
FROM X-23B OUTLET SH. 1 (H-10)
TO LPSI PUMPS SUCTION SH. 1 (F-12)
FROM LPSI PUMPS SH. 2 (H-13)
FROM LPSI PUMPS SH. 2 (G-13)
FROM LPSI PUMPS SH. 2 (E-13)
FROM LPSI PUMPS SH. 2 (C-13)
TO DRY SIPPING MACHINE (AT REFUELING POOL)
LOOP 1A RCS 26014 SH.1 (B-7)
LOOP 1B RCS 26014 SH.1 (G-7)
LOOP 2A RCS 26014 SH.1 (G-7)
LOOP 2B RCS 26014 SH.1 (B-7)
PRESSURIZER PRESSURE INTERLOCK 26014 SH.2 (G-4)
SHUTDOWN COOLING 26014 SH.1 (E-6)
CONTAINMENT, ENCLOSURE AND AUX. BLDG. DRAINS 26024 SH. 1 (B-11)
CONTAINMENT, ENCLOSURE, AND AUX. BLDG. DRAINS 26024 SH. 1 (B-6)
SEE GEN. NOTE 2.
SAMPLE SINK
DRAIN
PLUG
NOTE 6
NOTE 5
SEE NOTE 3
SIAS CLOSE
SIAS OPENS
RCS
N2 SUPPLY
N2 TO CLEAN LIQUID RADWASTE 26020 SH. 5 (F-8)
VENT
SPEC CHG
SPEC. CHG.
TEST
INST.
CCB
CCA
NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. VALVES 2-SI-701A, 2-SI-702A THRU 2-SI-701D, 2-SI-702D, SHOULD BE LOCATED OUTSIDE THE SECONDARY SHIELD WALL AT EL. (+)14'-6" AS CLOSE AS PRACTICAL TO A DRAIN.
3. CLOSE COIL REMOVED FROM MCC CONTACTOR WHEN UNIT IS IN MODES 1, 2, OR 3 (2-SI-614, 2-SI-624, 2-SI-634 & 2-SI-644)
4. PORTIONS OF LINE HCD-31 (S.I. TEST HEADER) WAS UPGRADED TO SEISMIC CLASS 1 PER PDCR 2-222-76. THE UPGRADED PORTIONS OF THE LINE TO SEISMIC CLASS 1 ARE NOW REFLECTED AS HCD-31C\1.
5. HOSE DISCONNECTED FROM VALVE 2-SI-814 AND LINE PLUGGED DOWNSTREAM OF CHECK VALVE 2-SI-814 DURING PLANT OPERATIONS. HOSE CONNECTED FOR SHUTDOWN COOLING MODE ONLY.
6. LINES 6"-CCA-6, 3"-CCA-7, 2"-CCA-7 & 2"-CCA-9 UPSTREAM OF VALVES 2-SI-706A/B/C/D ARE ASME III CLASS 2 PER DCN DM2-00-855-98.
CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 13
OPERATIONS CRITICAL
SUPERSEDES 25203-26015 THRU REV. 19
FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.
TITLE
PIPING AND INSTRUMENTATION DIAGRAM
SAFETY INJECTION TANKS
REVISION DESCRIPTION (SIGNATURES ON FILE)
REVISED PER MSDUR 15-000164 & DUR 15-800380
CAD NO: C:\fms32\tmpplot\C260153.DGN
DRAWING NO: 25203-26015
REV 31
DSGN EAL
SCALE: NONE
UNLESS OTHERWISE NOTED
SH 3

REGENERATIVE HEAT EXCH.

VOLUME CONTROL TANK T-5

CHARGING PUMP #3

CHARGING PUMP #2

CHARGING PUMP #1

CHARGING PUMP SEAL LUBRICATION SYSTEM

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. DELETED
3. VALVE 2-CH-435 NO LONGER FUNCTIONS AS A CHECK VALVE. PART NO.4 (PISTON DISC) AND PART NO. 7 (SPRING) WERE REMOVED.
4. PORTABLE FLOW COMPUTER CONNECTED TO THE TRANSDUCER DURING PERIODIC TESTING.
5. A STAINLESS STEEL FUNNEL IS LOCATED AT THE OUTLET OF 1" DRAIN LINE (1"-HSC-129) FROM THE "B" CHARGING PUMP SEAL LUBRICATION SYSTEM PIPE OUTLET FEEDING INTO THE FLOOR DRAIN
6. RAD WASTE LINES DESIGNATED AS HSC-35, HSC-113, AND HSC-114 AS SHOWN ON THIS DRAWING ARE NOT PART OF THE ISI PROGRAM. RAD WASTE LINES DESIGNATED AS HSC-35, HSC-113, AND HSC-114 WHICH ARE ASME SECTION III CODE CLASS 3 LINES AS SHOWN ON THIS DRAWING ARE DOWNGRADED TO ANSI B31.1 PER REG GUIDE 1.143 REV. 1 "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT WATER NUCLEAR POWER PLANTS". PIPE CLASSES FOR THE LINE NUMBERS WILL REMAIN THE SAME FOR HISTORICAL PURPOSES. ALL NEW PIPING SHALL BE INSTALLED TO THE APPLICABLE ANSI B31.1 PIPE CLASS, WITH A (R) FORTH LETTER CODE FOR RAD WASTE QA. PER DCN DM2-00-0216-03.
7. 3-WAY VALVES 2-CH-993, 2-CH-994, & 2-CH-995 ARE ALIGNED AND LOCKED OPEN TO CLEAN WASTE.
8. WHERE NOTED, INSTRUMENT AVAILABLE FOR MANUAL MONITORING OF SYSTEM.
9. PIPING INSTALLED WITHIN THE CHARGING PUMP SEAL LUBRICATION SYSTEM DESIGNATED AS PIPE CLASS HSC, WHICH IS AN ASME SECTION III, CODE CLASS 3 LINE, HAS BEEN DOWNGRADED TO NON-QA IN ACCORDANCE WITH MEPL DETERMINATION MP2-CD-1403. THIS LINE NUMBER PIPE CLASS WILL REMAIN THE SAME FOR HISTORICAL PURPOSES, EXCEPT THE MATERIAL CAN NOW BE MAINTAINED UNDER THE AUSPICES OF THE ANSI B31.1 CODE. ALL NEW MATERIAL THAT WILL BE UTILIZED FOR THE SEAL LUBRICATION SYSTEM PIPING MODICATIONS CAN NOW BE IN ACCORDANCE WITH THE REQUIREMENTS OF SPECIFICATION SP-ME-668, PIPE CLASS HSD.
10. THIS BLIND FLANGE IS A BDB FLEX ALTERNATE RCS MODES 1-4 CONNECTION TIE-IN POINT. REFERENCE DC MPG-13-01131.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 14

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
CHARGING SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260171.DGN	
REVISED PER DC MP2-17-00119 AND DUR 18-800219	DRAWING NO: 25203-26017	REV 67
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

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12-OCT-2018 16:21
C260171.DGN

FSAR FIGURE

OPERATIONS CRITICAL

Dominion®	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE NUCLEAR POWER STATION-UNIT 2 PIPING & INSTRUMENTATION DIAGRAM DEBORATING & PURIFICATION SYSTEM
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO:s...\PROJECT\20000\P260172.DGN
REVISED PER MSDUR 16-000164 AND DUR 16-800380	DRAWING NO: 25203-26017 REV 38
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED SH 2

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 15

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STD. Y1.1.
2. RAD WASTE LINES DESIGNATED AS HSC-35, HSC-75, HSC-113 AND HSC-117 AS SHOWN ON THIS DRAWING ARE NOT PART OF THE ISI PROGRAM. RAD WASTE LINES DESIGNATED AS HSC-35, -75, -113, AND -117 WHICH ARE ASME SECTION III CODE CLASS 3 LINES AS SHOWN ON THIS DRAWING ARE DOWNGRADED TO ANSI B31.1 PER REG GUIDE 1.143 REV. 1 "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT WATER NUCLEAR POWER PLANTS". PIPE CLASSES FOR THE LINE NUMBERS WILL REMAIN THE SAME FOR HISTORICAL PURPOSES. ALL NEW PIPING SHALL BE INSTALLED TO THE APPLICABLE ANSI B31.1 PIPE CLASS, WITH A (R) FORTH LETTER CODE FOR RAD WASTE QA. PER DCN DM2-00-0216-03.
3. EQUIPMENT ELECTRICALLY RETIRED.

VOLUME CONTROL TANK
SH. 1
T-5

VENT
2-CH-089A
3"-HCC-33
2-CH-188
SPEC. CHG.
2-CH-512 NC
FC
FO
A/S
2CH512 #
ZS 512
C04
FY 512
HS 2512 A
HS 2512
MAKEUP CONTROL SYSTEM DIAGRAM
25203-29203 SH.7
SEE NOTE 5
SIAS CLOSES

PMW
26030 SH. 1 (E-3)
1"-HCD-24
2-CH-368
PMW
26022 SH. 1 (K-13)
FROM 2-RB-272
½"-HCD-138
2-CH-915
L.C.
1" x ½" RED.

I/A
26009 SH.7 (K-12)
2-IA-665
BACK UP I/A
26009 SH.6 (E-6)
T-131 ACCUM.
2-IA-1143
½"-HCD-36
PMW
26030 SH. 1 (B-3)
SH. 1 (G-3)
TO CHARGING PUMPS
1"-MSD-9
26024 (MSD)
TO CNTMT ENCL. BLDG AND AUX BLDG. DRAIN SYS
2-CH-968
26030 SH. 1 (D-3)
PMW
3"-HCB-19
Z-192
HS 2192
FY 192
ZS 192
C02
FO
S
SPEC. CHG.
2-CH-080
3"-HCD-36
HS 2504
2CH504 #
ZS 504
2-CH-191
2-CH-504 FL
M
LALL 227
TO/FROM LC-227
SH. 1 (J-2)
TO HS-2501
SH. 1 (H-5)
1"-HCB-17
3"-HCB-25
2-CH-705
DRAIN
FAHL 210X
C04
FC
2-CH-192
1"
PWS T
26015 SH.2 (A-5)
2-CH-972
¾" O.D. SS
F210X
2-CH-992
SAMPLE SINK
FC
FQIS 210X
2-CH-970
CL C-4 TUB. HCD
2-CH-971
FT 210 XB
FT 210 XA
2-CH-181
⅜" SS TUBING (TYP.)
HCD (TYP.)
2-CH-182
2-CH-090
3"-HCD-36
2-CH-195
2-CH-183
FE 210
FC 2-CH-210X
HCD
HCB
SPEC. CHG.
4"-HCD-126
ATM
2-CH-376
FY 210 X
A/S
FO
2-CH-193
2CH196
ZS 196
2-CH-185
½"-HCC-44
L.S.
2-CH-670
VENT
INOTE 6]
2-CH-196
3"-HCB-21
2-CH-184
2-CH-673
T-132 ACCUM.
2-IA-666
2-IA-1144
DRAIN
HS 2196 A
HS 2196
SIAS CLOSES
FY 196
2-CH-186
TO LC 227
PMW
26030 SH. 1 (D-3)
IA
26005 SH.7 (H-13)
½"-HCD-36
2-CH-210X
FC
2-CH-210Y
A/S
FO
FY 210 Y
FQIS 210Y
C04
HS 210
FR 210
2-CH-172
SPEC. CHG.
2-CH-773
½" O.D. SS
2-CH-772
⅜" O.D. SS
2-CH-780
2-CH-774
TO VCT LEVEL TRANSMITTER HEADER
SH. 1 (H-2)
2-CH-781
⅜" RUBBER HOSE
LC 2-CH-180
2-CH-179
1½"-HCB-12
TO CHARGING PUMPS
SH. 1 (G-4)
4"-HCB-12
1"-HCB-15
1"-HCB-18
2-CH-174
2-CH-175
F210Y
FE 210 X-A
FT 210 Y
FE 210 X-B
FC 210Y
FAHL 210Y
C04
2-CH-514
FL
M
HS 2514
ZS 514
2CH514 #
C02
SIAS OPENS
NOTE 3
FP
FE 9870 A
FE 9870 B
2CH510 #
HY 510
ZS 510
HS 2510
SIAS CLOSES
2-CH-159
2"-HCB-16
2-CH-510
1"-HCB-16
2-CH-728
1"-HCB-16
DRAIN
2"-HCB-18
2-CH-164
PDI 209
LS S
2-CH-165
2-CH-162
2-CH-166
2-CH-986
1"X3/4" RED
BORIC ACID STRAINER
3"-HCB-18
VENT
2-CH-088
2-CH-155
2-CH-153
3"-HCB-15
2-CH-161
2-CH-748
¾"
1"
2-CH-037
2-CH-708
DRAIN
HCB
HSC
3"-HCB-15
2-CH-036
3"-HSC-103
2-CH-757
PMW
26030 SH. 1 (D-3)
2-CH-034
½"-HCD-36
SPEC. CHG.
½"-MSD-9
TO CHARGING PUMPS
SH. 1 (F-3)
2-CH-091
VENT
2-CH-727
2-CH-177
SPEC. CHG.
2-CH-176
½"-HCC-44
LS
3"-HCB-18
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS
26024 SH.3 (F-7)
2-CH-514
2-CH-159
2"-HCB-16
2-CH-510
1"-HCB-16
2-CH-728
1"-HCB-16
DRAIN
2-CH-158
2-CH-691
2-CH-783
BORIC ACID PUMP
P19A
2-CH-145
3"-HCB-13
2-CH-146
¾"-MSD-15
FRAME ADAPTER DRAIN
HCB CL C-4 TUB.
P1 206
PT 206
P206
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS.
26024 SH.3 (F-7)
½" TUBING (MSD-15)
2-CH-749
HCB MSD
¾"-MSD-15 SEAL DRAIN
BORIC ACID PUMP CONTROLS
SIAS STARTS
32009 SH. 4 & 5
2CH511 #
HY 511
ZS 511
HS 2511
HS 226
SIAS CLOSES
2-CH-156
2-CH-511
1"-HCB-16
2-CH-729
2-CH-151
DRAIN
2-CH-692
2-CH-784
BORIC ACID PUMP
P19B
2-CH-143
2-CH-154
2-CH-152
3"-HCB-15
2-CH-147
¾"-MSD-15
FRAME ADAPTER DRAIN
HCB CL C-4 TUB.
PI 208
PT 208
P208
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS.
26024 SH.3 (F-7)
½" TUBING (MSD-15)
2-CH-388
HCB MSD
½"-MSD-15 SEAL DRAIN
26024 SH.3 (F-7)
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS.

PMW
26030 SH. 1 (E-3)
FROM 2-PMW-333
1½"-HCD-25
¾"-HCD-25
2-CH-976
2-CH-977
¾"
2-CH-792
HS 2034
M
2-CH-119
BORIC ACID BATCHING TANK
2"-MSD-11
TIC 213
T7
HS 213
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS.
26024 SH.3 (D-5)
2-CH-120
½"-HCD-8
L.S.
2-CH-123
1"-HCD-7
2-CH-121-
2"X1"
1"-MSD-11
SPEC. CHG.
1"
2-CH-122
2"-HCD-8
L2 S
BORIC ACID BATCHING STRAINER
LALL 208
LALL 206
C126
LS 206
½"-HCD-36
CO2
LI 206
LA 206
LAHL 206
PCA 5
L206
C10
LT 206
NOTE 4
LI 208B
LI 206 A
2-CH-990
2"-HCB-16
SPEC. CHG.
C-4 TUBING
2-CH-126
LS 207
SPARE
BORIC ACID TANK
TBA
NOTE 2
HS 207
NP142
HS 206
NP141
2"-HCD-8
4"-HCC-41
LS 208
½" O.D. SST
1"-HCD-125
SPEC. CHG.
2-CH-731A
4"-HCC-41
2-CH-124
4"
2-CH-732A
1"
PCA 5
L208
LA 208
LAHL 208
LI 208
LT 208
LI 208 A
2-CH-135
2-CH-731B
2-CH-732B
6" MIN. TYP.
2"-HCB-16
C-4 TUBING
2-CH-136
LS 209
SPARE
BORIC ACID TANK
T-8B
HS 209
NP144
HS 208
NP143
2-CH-139
2-CH-979
½"-HCC-44
L.S.
NOTE 2
2-CH-127
1"-HCC-41
2"-HCB-16
4"-HCC-41
4"-MSD-15
TO CNTMT ENCL. BLDG. AND AUX BLDG. DRAIN SYS.
26024 SH.3 (G-7)
3"-HCB-13
LO 2-CH-130
LO 2-CH-140
2CH508 #
ZS 508
HS 2508
M
FL 2-CH-508
3"-HCB-14
SIAS OPENS
2CH509 #
HS 2509
ZS 509
SIAS OPENS
FL 2-CH-509
2-CH-693
2-CH-694
½"-HCD-36
3"
2-CH-128
SPEC. CHG.
2-CH-978
¾"-HCC-44
L.S.
PCV 249
2-CH-131
CL C-4 TUB. HCD
¾" X ½" RED
½" P X ¾" T
¾" O.D. SST
2-CH-783
2-CH-691
2-CH-621
1"-MSD-15
SPEC. CHG.
DRAIN
2-CH-730
2-CH-142
2-CH-189
2-CH-190
½"-HCC-44
L.S.
SH. 1 (G-3)
TO CHARGING PUMPS

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. BAST HEATERS HAVE BEEN DE-ENERGIZED PER PDCR 2-15-88
3. PORTABLE FLOW COMPUTER CONNECTED TO THE TRANSDUCER DURING PERIODIC TESTING.
4. THIS GAUGE INSTALLED ONLY PER AOP2579A. NORMALLY STORED IN THE FIRE SHUTDOWN STORAGE CABINET.
5. LOSS OF CONTROL POWER TO VALVE 2-CH-512 WILL RESULT IN THE CLOSURE OF VALVES 2-CH-210X & 2-CH-210Y.
6. VALVE 2-CH-196 FAILS "CLOSED" ON LOSS OF CONTROL POWER OR FAILS "AS-IS" ON LOSS OF CONTROL AIR (I.E, NON-SAFETY RELATED INSTRUMENT AIR AND SAFETY RELATED ACCUMULATOR AIR.)

FSAR FIGURE

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 16

SUPERSEDES 25203-26017 THRU REV. 20

OPERATIONS CRITICAL

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
BORIC ACID SYSTEM
WATERFORD, CT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\mpp1o1\C260173.DGN	
REVISED PER DC MP2-15-01121 AND DUR 17-800167	DRAWING NO: 25203-26017	REV 46
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

ENGINE DRIVEN AIR COOLER COOLANT PUMP

P-127A

AIR COOLING HEAT EXCHANGER

X-83A

SERVICE WATER SUPPLY 26008 SH. 2 (D-4)

SERVICE WATER RETURN 26008 SH. 2 (D-5)

AIR COOLING SYSTEM

AIR INTERCOOLERS

JACKET WATER EXPANSION TANK

T-80A

JACKET WATER HEAT EXCHANGER

X-45A

SERVICE WATER SUPPLY 26008 SH. 2 (E-4)

SERVICE WATER RETURN 26008 SH. 2 (E-5)

JACKET COOLING SYSTEM

STAND-BY JACKET COOLANT HEATER

X-54A

ELECTRIC MOTOR DRIVEN JACKET COOLANT PUMP

P-88A

LOW SPEED RELAY FROM SS-8780A SH.4 (G-2)

DIESEL ENGINE

JACKET HEADER

H-7A

NORTH HEADER

SOUTH HEADER

TURBOCHARGER

TURBOCHARGER

DIESEL ENGINE DRIVEN JACKET COOLANT PUMP

P-49A

LUBE OIL HEAT EXCHANGER

X-53A

SERVICE WATER SUPPLY 26008 SH. 2 (D-5)

SERVICE WATER RETURN 26008 SH. 2 (D-4)

VENT TO CRANKCASE (D-7)

L-704 LUBE OIL FILTER

L-701 LUBE OIL STRAINER

DIESEL ENGINE DRIVEN L.O. PUMP

P-48A

STANDBY L.O. CIRC. PUMP

P-89A

ELECTRIC MOTOR DRIVEN PRE-LUBE PUMP

P-120A

LUBE OIL HEATER

X-55A

GENERATOR

DRAIN TO FLOOR

EXHAUST TO ATMOSPHERE

FROM ENGINE CONTROL

SAMPLE

NOTE:

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. ALL HOSES SHOWN ARE RUBBER/FABRIC UNLESS OTHERWISE NOTED.
3. THE FOLLOWING INSTRUMENTS INTEGRAL TO THE DIESEL GENERATOR ARE NOT REPRESENTED ON THIS P&ID AND ARE REFERENCED ON THE FOLLOWING DRAWINGS:

SI-8997, ST-8997 AND TI-8855 ON	25203-32041 SH. 031
SI-8998, ST-8998 AND TI-8856 ON	25203-32041 SH. 032
SY-8780A ON	25203-28500 SH. 11C
SY-8780B ON	25203-28500 SH. 12C
ZT-8780A ON	25203-28500 SH. 11B
ZT-8780B ON	25203-28500 SH. 12B
SI-8997 AND TI-8855 ON	25203-39038 SH. 26A
SI-8998 AND TI-8856 ON	25203-39038 SH. 26B
TE-9370A AND C ON	25203-28500 SH. 1070A AND C
	25203-32041 SH. 7
TE-9371A AND C ON	25203-28500 SH. 1071A AND C
	25203-32041 SH. 7
TIS-9370A, C AND TIS-9371A, C ON	25203-32041 SH. 7
	25203-39038 SH. 6A

4. SEE PDCR 2-46-90 FOR VALVE/REDUCER INSTALLATION

REFERENCE DRAWINGS:
25203-29035 SH 12: GENERAL ARRANGEMENT DIESEL GEN RM. (H-7A & H-7B)
25203-29035 SH 6: AIR COOLANT SYSTEM SCHEMATIC

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

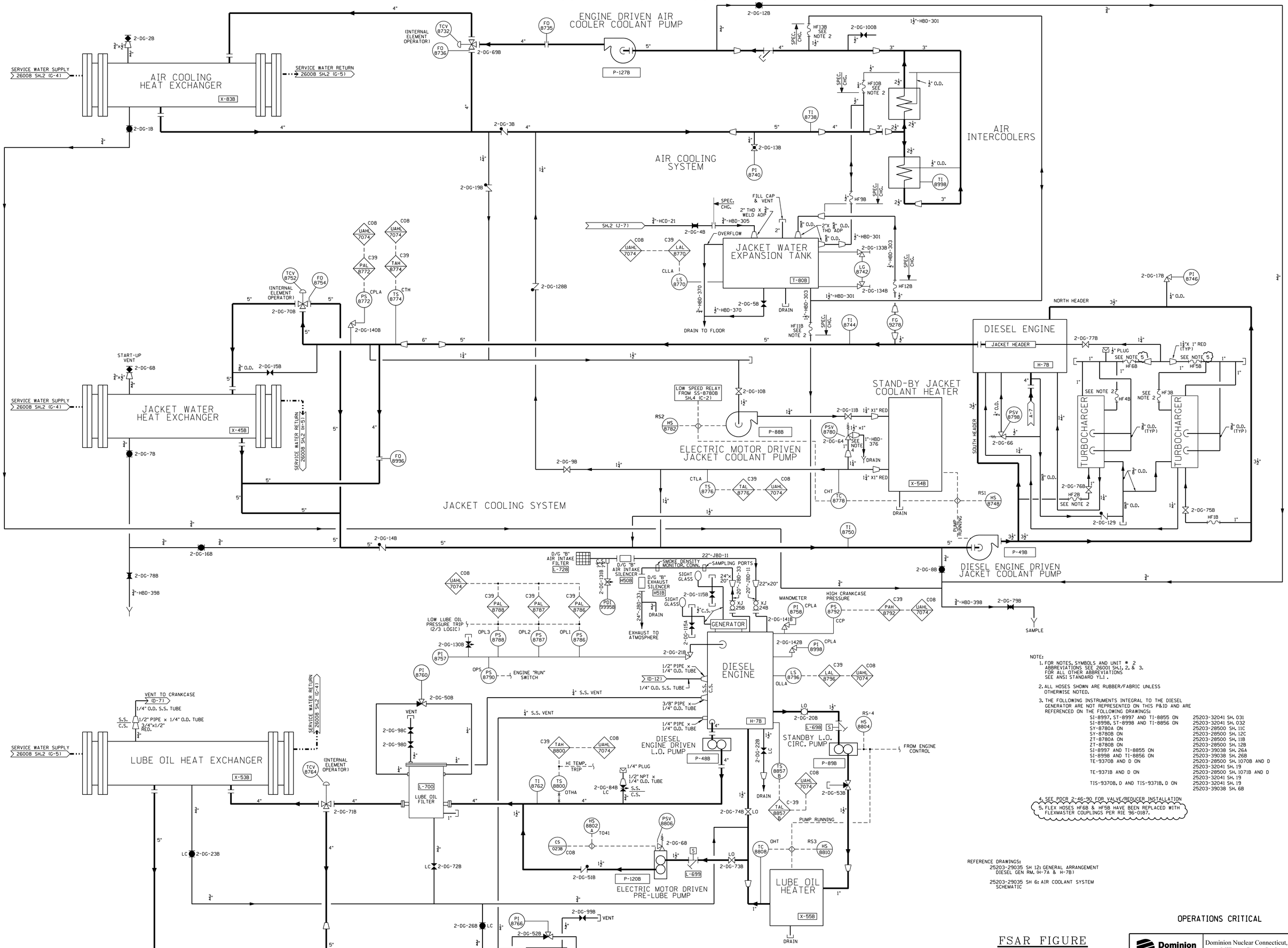
THIS DWG SUPERSEDES -26018 THRU REV 13.

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
EMERGENCY DIESEL GENERATOR H-7A
LUBE OIL,AIR,AND JACKET WATER COOLING
WATERFORD, CT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: c:\fms32\tmpplot\C260182.DGN	
REVISED PER MSDUR 18-000050 & DUR 18-800114	DRAWING NO: 25203-26018	REV 26
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2



NOTE:

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL HOSES SHOWN ARE RUBBER/FABRIC UNLESS OTHERWISE NOTED.
3. THE FOLLOWING INSTRUMENTS INTEGRAL TO THE DIESEL GENERATOR ARE NOT REPRESENTED ON THIS P&ID AND ARE REFERENCED ON THE FOLLOWING DRAWINGS:

SI-8997, ST-8997 AND TI-8855 ON	25203-32041 SH. 031
SI-8998, ST-8998 AND TI-8856 ON	25203-32041 SH. 032
SY-8780A ON	25203-28500 SH. 11C
SY-8780B ON	25203-28500 SH. 12C
ZT-8780A ON	25203-28500 SH. 11B
ZT-8780B ON	25203-28500 SH. 12B
SI-8997 AND TI-8855 ON	25203-39038 SH. 26A
SI-8998 AND TI-8856 ON	25203-39038 SH. 26B
TE-9370B AND D ON	25203-28500 SH. 1070B AND D
	25203-32041 SH. 19
TE-9371B AND D ON	25203-28500 SH. 1071B AND D
	25203-32041 SH. 19
TIS-9370B, D AND TIS-9371B, D ON	25203-32041 SH. 19
	25203-39038 SH. 68

4. SEE PDCR 2-46-90 FOR VALVE/REDUCER INSTALLATION
5. FLEX HOSES HF6B & HF5B HAVE BEEN REPLACED WITH FLEXMASTER COUPLINGS PER RIE 96-0187.

REFERENCE DRAWINGS:
25203-29035 SH 12: GENERAL ARRANGEMENT DIESEL GEN RM. (H-7A & H-7B)
25203-29035 SH 6: AIR COOLANT SYSTEM SCHEMATIC

OPERATIONS CRITICAL

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

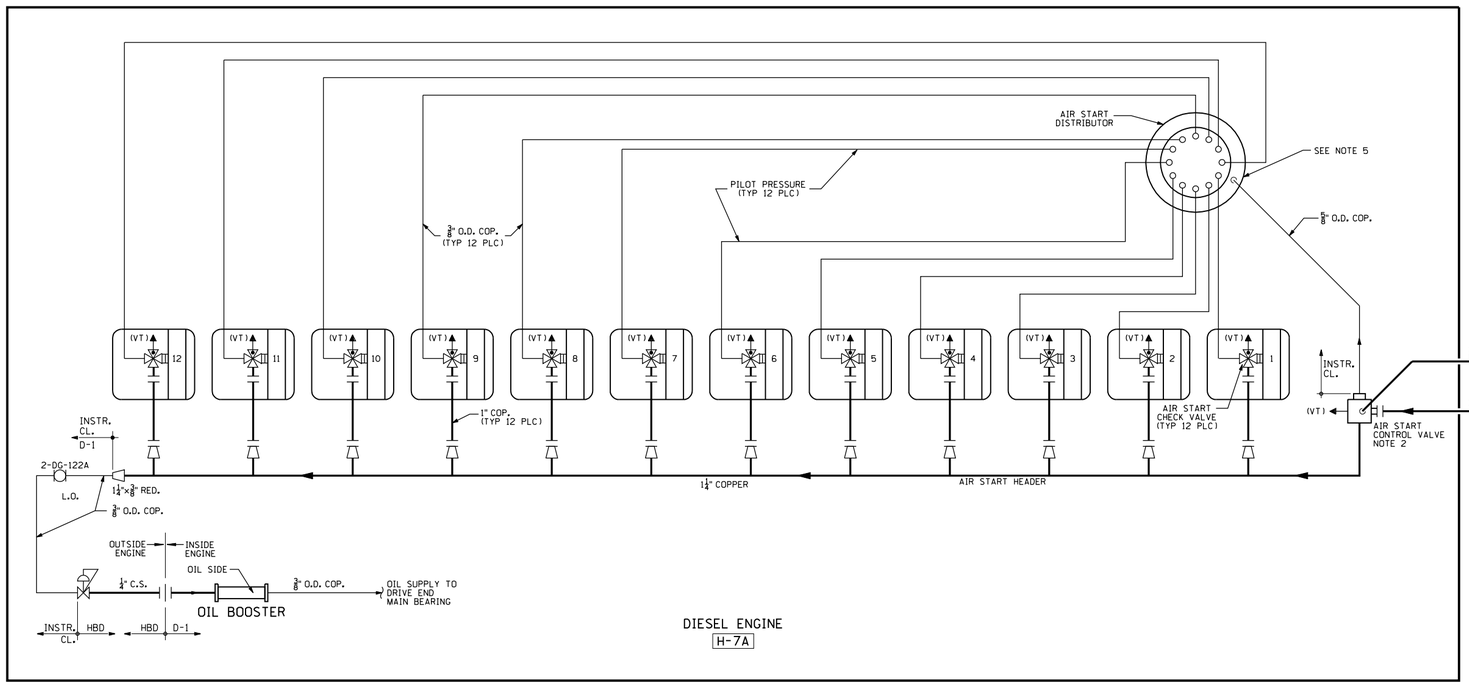
Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

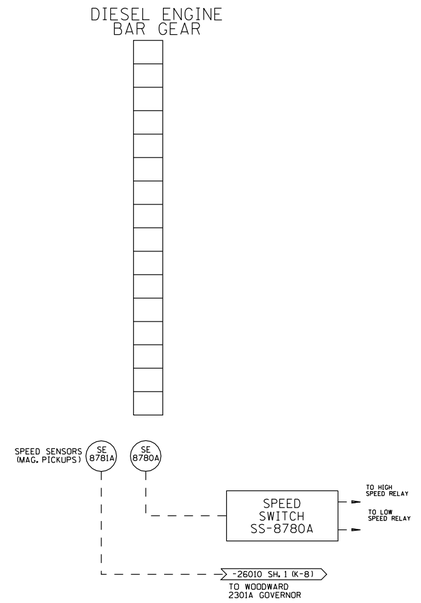
TITLE
PIPING & INSTRUMENTATION DIAGRAM
EMERGENCY DIESEL GENERATOR H-7B
LUBE OIL, AIR, AND JACKET WATER COOLING

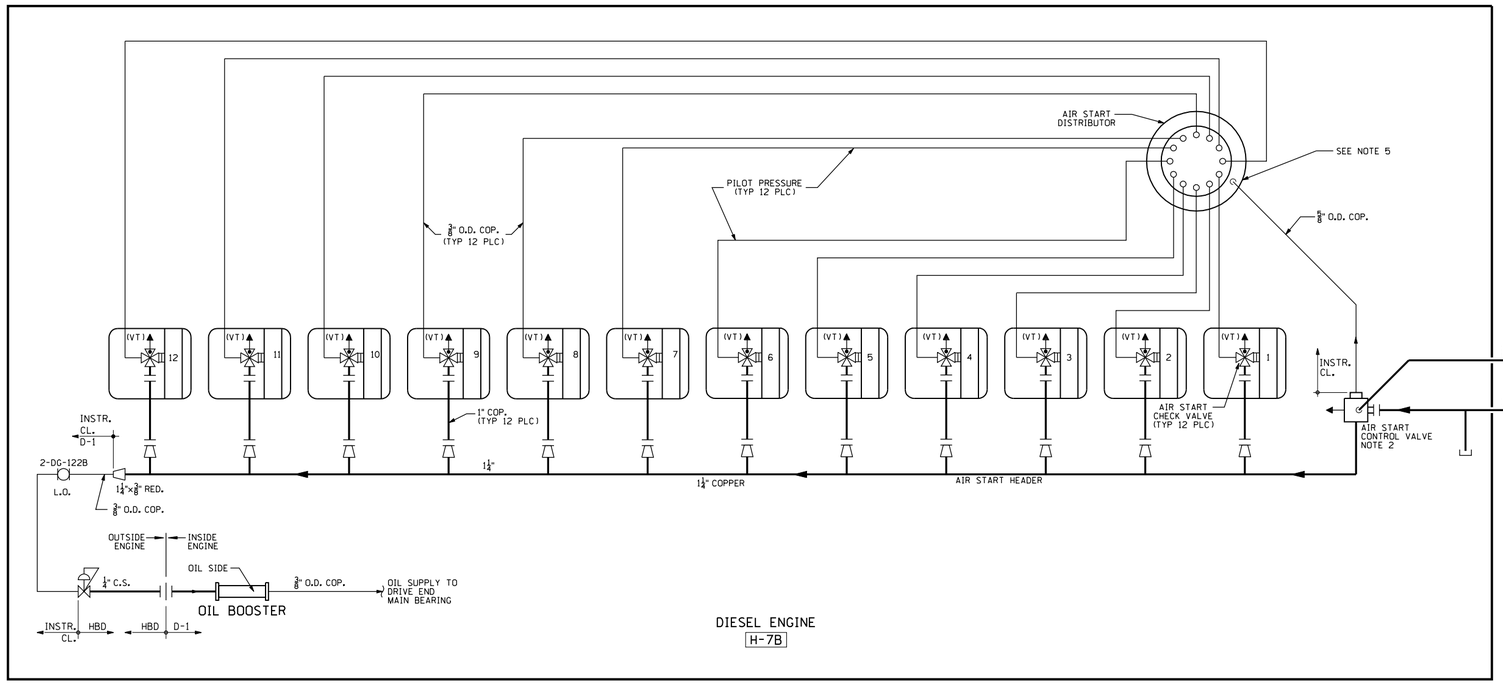
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260183.DGN	
REVISED PER MSDUR 25-801002 & DUR 25-800002	DRAWING NO: 25203-26018	REV 30
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 3

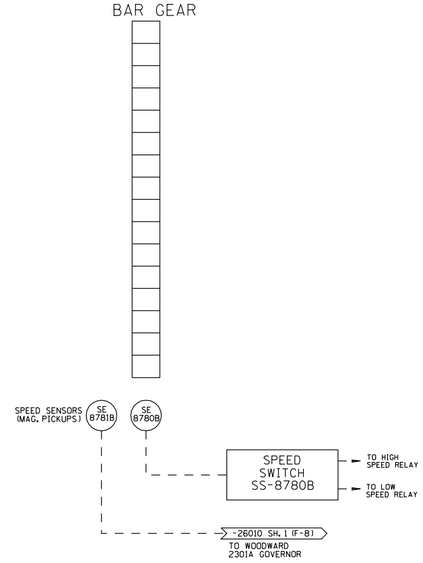
MBULOGIN\er in004
14-JAN-2025 10:19
C260183.DGN

25203-26018 SH-0000000









NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1,2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. THESE VALVES HAVE THE INTERNALS REMOVED.
3. * SUPPLY WITH EQUIPMENT
4. FOR APPLICABLE SYSTEM NOTES SEE DRAWING 25203-26108 SH. 2.
5. ARRANGEMENT OF AIR START TUBING IS DIAGRAMMATIC AND DOES NOT REFLECT THE FIRING ORDER. FOR FIRING ORDER, SEE VENDOR MANUAL 25203-138-0024.

OPERATIONS CRITICAL

THIS DRAWING PARTIALLY SUPERSEDES DWG NO. 25203-26018 SH. 1A REV 1 & DWG NO. 25203-26018 SH. 1B REV 1.

Q.A.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	7	2-10-01	INC. DCN DM2-00-0025-00	NJB	JSJ	REB	GMO
N/A	6	5-26-98	INC. DCN DM2-S-0401-96. ADDED TITLE TO DIESEL "B" (B-9)	GZ	KSS	REB	REB
N/A	5	9-18-96	INC. DCN DM2-S-394-96. ADDED NOTE 5.	MGD	KSS	DAD	DAD
84-024	4	10-17-95	INC. DCN DM2-S-0208-94 PER DCR M2-S-0136-94. (C-H-6) DEL'D VALVES 2-DG-123A & B	LLJ	SAH KSS DME	WAP	WAP
91-056	3	11-21-94	AS BUILT PER: DCR-M2-S-0355-94 (Incorp. DCN DM2-S-0848-94)	KSS	SAH KSS	WAP	WAP
N/A	2	3-17-94	AS BUILT PER DCR M2-S-077-94 (EDITORIAL)	MGD	DAD	WAP	WAP
N/A	1	3-31-93	AS BUILT PER: DSR-M2-S-469-93 (Incorp. DCN DM2-S-159-93)	JJ	MGD DAD	TGQ	WAP

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
DIESEL GENERATOR H-7A & H-7B
STARTING AIR
WATERFORD, CT

BY	CHKD.	APP.	APP.
KSS	SAH	TGQ	WAP
DATE 3-15-93	DATE 3-31-93	DATE 3-31-93	DATE 3-31-93

SCALE: NONE
DWG. NO.: 91-056
25203-26018 SH.4 OF 5
REV 7

C:\Vms50\Vmsplot\Cs0018d.DGN
08/06/93

SH.4 (H -4)

DIESEL START/RUN LOGIC

HY 8313 — S — VENT — 2-DG-27A — 2-DG-88A LO — 2-DG-86A LO — 2-DG-91A FO — 2-DG-29B

HY 8171 AS-1 — 2-DG-94A — 2-DG-96A FO — L-231 — 2-DG-87A LO — 2-DG-92A FO — 2-DG-29A — S L-693 — 2-DG-90A — 1"-HBD-307

HY 8170 — SPEC. CHG. — 2-DG-95A FO — L-230 — 2-DG-93A — DIESEL START — HY 8170 AS-2

L-694 S — 2-DG-89A — 1"-HBD-307

HBD D-1 — PS 8314 A APL1 — 2-DG-143 — PI 8314 A CPLA — P8314 — PAL 8314 C38 — UAHL 7463 A C08 — UAHL 7073 C08

PS 8314 B APL2 — 2-DG-144 — PI 8314 B CPLA — LO 2-DG-30B — HF20B — 1½"-HBD-307 — 2"-HBD-307 — 2"-HBD-309

2"-HBD-307 — 2"-HBD-309 — 1"-HBD-308 — 2-DG-49A (DOUBLE CHECK VALVE)

LO 2-DG-31A — 2-DG-32A SEE NOTE 2 — LO 2-DG-31B — 2-DG-32B SEE NOTE 2

PI 8833 — 2-DG-33A — STARTING AIR TANK T-49A — PSV 8950 — 2-DG-54 — 2-DG-35A — 1½"-HCD-306 — LO 2-DG-36A

PSV 8951 — 2-DG-55 — STARTING AIR TANK T-49B — PI 8834 — 2-DG-33B — 2-DG-35B — LO 2-DG-36B

½"-HBD-314 — 2-DG-34A — H.C. — ½"-HBD-314 — 2-DG-34B — H.C.

½"-HBD-308 — 1½"-HCD-306

1½"-HCD-306 — PI 8833 A — 2-DG-110A — DRYER L-356A — PI 8833 B — 2-DG-113A F — 2-DG-104B — PRE-FILTER L-355A — 2-DG-103A — 2-DG-38B LC — SPEC. CHG.

AFTER FILTER L-355B — 2-DG-110B — MI 8833 B — 2-DG-104A DRAIN — 2-DG-112A DRAIN — 2-DG-102A — 1½"-HBD-313

¾"-HCD-306 — 2-DG-101A

PSV 8819 — 2-DG-59 — PI 7985 A — 2-DG-107A — 2-DG-106A — INLET FILTER L-705A — F-10A — HS 7985 A C38 — PS 7985 A — 2E — 2-DG-37A — 2-DG-108A DRAIN — A.C. MOTOR DRIVEN AIR COMPRESSOR

¾"-HCD-306 — ¼" O.D. COP. — ⅜" O.D. COP. — 2-DG-97A — SPEC. CHG. — 2-DG-97C — SPEC. CHG.

PS 7985 C — HS 7985 C C38 — INLET FILTER L-705C — F-10C — 2-DG-127A — D.C. MOTOR DRIVEN AIR COMPRESSOR

2-DG-106C — 2-DG-37C — DRAIN — 2-DG-108B — PI 7985 C — 2-DG-107C — PSV 8821 — 2-DG-60

2-DG-42 — PSV 8832 — SPEC. CHG. — 2-DG-58 — 26009 SH.5 (B-8) INSTRUMENT AIR SYSTEM — 2"-JDD-23 — 2"-HBD-309 — 2-DG-40 — 2-DG-41

2-DG-46 — 2-DG-48 — 2-DG-45 — 2-DG-44 — 2-DG-47 — 2-DG-43 — 2½"-HBD-309

26018 SH.4 (C-4)

DIESEL START/RUN LOGIC

HY 8817 — S — VENT — 2-DG-27B — 2-DG-88B LO — 2-DG-86B LO — 2-DG-91B FO — 2-DG-29D

HY 8173 AS-1 — DIESEL START — 2-DG-94B — 2-DG-96B FO — L-233 — 2-DG-87B LO — HY 8173 — 2-DG-92B FO — 2-DG-29C — LL-695 S — 1"-HBD-311 — 2-DG-90B

HY 8172 — SPEC. CHG. — 2-DG-95B FO — L-232 — 2-DG-93B — HY 8172 AS-2

L-696 S — 1"-HBD-311 — 2-DG-89B

HBD D-1 — PS 8818 A APL1 — 2-DG-145 — PI 8818 A CPLA — P8818 — PAL 8818 C39 — UAHL 7463 B C08 — UAHL 7074 C08

PS 8818 B APL2 — 2-DG-146 — PI 8818 B CPLA — LO 2-DG-30D — HF20D — 2"-HBD-309 — 2"-HBD-309 — 1"-HBD-312 — 2-DG-49B DOUBLE CHECK VALVE

LO 2-DG-31C — 2-DG-32C SEE NOTE #2 — LO 2-DG-31D — 2-DG-32D SEE NOTE #2

PI 8729 — 2-DG-33C — STARTING AIR TANK T-49C — PSV 8952 — 2-DG-56 — 2-DG-35C — 1½"-HCD-310 — LO 2-DG-36C

PSV 8953 — 2-DG-57 — STARTING AIR TANK T-49D — PI 8730 — 2-DG-33D — 2-DG-35D — LO 2-DG-36D

½"-HBD-314 — 2-DG-34C — H.C. — ½"-HBD-314 — 2-DG-34D — H.C.

½"-HBD-312 — 1½"-HCD-310

1½"-HCD-310 — PI 8729 A — 2-DG-111A — DRYER L-356B — PI 8729 B — 2-DG-113B F — 2-DG-104D — PRE-FILTER L-355C — 2-DG-103B — 2-DG-38A LC — SPEC. CHG.

AFTER FILTER L-355D — 2-DG-111B — MI 8729 B — 2-DG-104C DRAIN — 2-DG-112B DRAIN — 2-DG-102B — 1½"-HBD-313

¾"-HCD-310 — 2-DG-101B

PSV 8820 — 2-DG-61 — PI 7985 B — 2-DG-107B — 2-DG-106B — INLET FILTER L-705B — F-10B — HS 7985 B C39 — PS 7985 B — 2F — 2-DG-37B — 2-DG-109A DRAIN — A.C. MOTOR DRIVEN AIR COMPRESSOR

¾"-HCD-310 — ¼" O.D. COP. — ⅜" O.D. COP. — 2-DG-97B — SPEC. CHG. — 2-DG-97D — SPEC. CHG.

PS 7985 D — HS 7985 D C39 — INLET FILTER L-705D — F-10D — D.C. MOTOR DRIVEN AIR COMPRESSOR

2-DG-106D — 2-DG-37D — DRAIN — 2-DG-109B — PI 7985 D — 2-DG-107D — PSV 8822 — 2-DG-62

D.G. ROOM A / D.G. ROOM B

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1 & 2 FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. THESE CHECK VALVES HAVE THE INTERNALS REMOVED.
3. * SUPPLY WITH EQUIPMENT
4. FOR APPLICABLE SYSTEM NOTES SEE DRAWING NO. 25203-26018 SH. 2.

OPERATIONS CRITICAL
THIS DRAWING PARTIALLY SUPERSEDES DWGS 25203-26018 SH. 1A, REV. 1 & SH 1B, REV. 1

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION							

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	15	6-15-10	INC. DCN DM2-00-0012-10	DWJ	RFL	DLW	MP
N/A	14	1-28-10	INC. DCN DM2-00-0008-10	EAL	REB	DLW	RFL

Dominion — Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
DIESEL GENERATORS
STARTING AIR
WATERFORD, CT.

BY	CHKD.	APP.	APP.
KSS	SAH	TGQ	WAP
DATE 2-22-93	DATE 3-31-93	DATE 3-31-93	DATE 3-31-93

SCALE: NONE — DWG. NO.: 91-056 — 25203-26018 SH.5 — REV. 15

PRIMARY DEMIN. T-22A

PRIMARY DEMIN. T-22B

PRIMARY DEMIN. (FUTURE) T-22C

SECONDARY DEMIN. T-23A

SECONDARY DEMIN. T-23B

PRE-DEGASIFIER FILTER L-12

SPENT FUEL POOL COOLING AND CLEAN UP 26023 SH.2 (C-2) 2-RW-356

SAMPLE SYSTEM 26025 SH.1 (K-6) TO 2-SYS-61

COOLANT WASTE RCVR TANK PUMP P-29 SH. 2 (B-8)

GASEOUS RADWASTE SYSTEM 26021 SH.2 (D-7)

TO WASTE GAS SURGE TANK 26021 SH.2 (F-2)

FROM RESIN ADDITION TANK T-19 26017 SH.2 (E-3)

N_2 SYSTEM 26033 (F-13)

CONTAINMENT, ENCLOSURE AND AUXILIARY BUILDING DRAINS 26024 SH.3 (K-2)

SAMPLE SYSTEM 26025 SH.1 (K-2)

FROM DEGASIFIER EFFLUENT COOLER SH. 4 (F-3)

PWW & N_2 SYSTEM 26021 SH.1 (F-7)

SPENT RESIN TANK 26021 SH.3 (G-12)

SPENT RESIN TANK 26021 SH.3 (H-12)

SPENT RESIN TANK 26021 SH.3 (J-12)

TO COOLANT WASTE RECEIVER TANKS T-14A & T-14B SH. 2 (G-9)

TO COOLANT WASTE RECEIVER TANKS T-14A & T-14B SH. 2 (K-6)

TO T-15A & 15B SH. 2 (K-7)

FROM P-30 DISCHARGE 2 SH.2 (K-3)

FROM P-29 DISCH SH. 2 (B-8)

AERATED LIQUID RWS 26021 SH. 1 (J-4) FROM 2-LRA-62

TO LV-9184 & LV-9185 SH. 2 (G-10)

SH. 5 (G-10)

SH. 4 (H-12)

SH. 2 (B-7)

NOTES:

1. FOR NOTES, SYSTEMS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3 FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. RAD WASTE LINES SHOWN ON THIS DRAWING ARE NOT PART OF THE ISI PROGRAM. ALL RAD WASTE ASME SECTION III CODE CLASS 3 LINES ON THIS DRAWING ARE DOWNGRADED TO ANSI B31.1, PER REG GUIDE 1.143 REV 1, "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES AND COMPONENTS INSTALLED IN LIGHT WATER NUCLEAR POWER PLANTS". PIPE CLASSES FOR LINE NOS. WILL REMAIN THE SAME FOR HISTORICAL PURPOSES. ALL NEW PIPING SHALL BE INSTALLED TO THE APPLICABLE ANSI B31.1 PIPE CLASS, WITH A (R)FOURTH LETTER CODE FOR RAD WASTE QA.

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
CLEAN LIQUID RADWASTE SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C26020I.DGN	
REVISED PER DC MP2-21-01038 & DUR 23-800009	DRAWING NO: 25203-26020	REV 27
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

MBILDGIN Ver 14.004
13-JAN-2023 12:13
C26020I.DGN

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. (DELETED)
3. RAD WASTE LINES SHOWN ON THIS DRAWING ARE NOT PART OF THE ISI PROGRAM. ALL RAD WASTE ASME SECTION III CODE CLASS 3 LINES ON THIS DRAWING ARE DOWNGRADED TO ANSI B31.1, PER REG GUIDE 1.143 REV 1, "DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES AND COMPONENTS INSTALLED IN LIGHT WATER NUCLEAR POWER PLANTS". PIPE CLASSES FOR LINE NOS. WILL REMAIN THE SAME FOR HISTORICAL PURPOSES. ALL NEW PIPING SHALL BE INSTALLED TO THE APPLICABLE ANSI B31.1 PIPE CLASS, WITH A (R) FOURTH LETTER CODE FOR RAD WASTE QA.

COOLANT WASTE MONITOR TANK T-15A

COOLANT WASTE MONITOR TANK T-15B

COOLANT WASTE RECEIVER TANK T-14A

COOLANT WASTE RECEIVER TANK T-14B

COOLANT WASTE RECEIVER TANKS PUMP P-29

COOLANT WASTE MONITOR TANKS PUMP P-30

COOLANT WASTE FINAL FILTER L-15

CONTROLATRON (FLOW MEASURING LOCATION)

CLEAN LIQUID RADWASTE SAMPLE PUMP (L18-13)

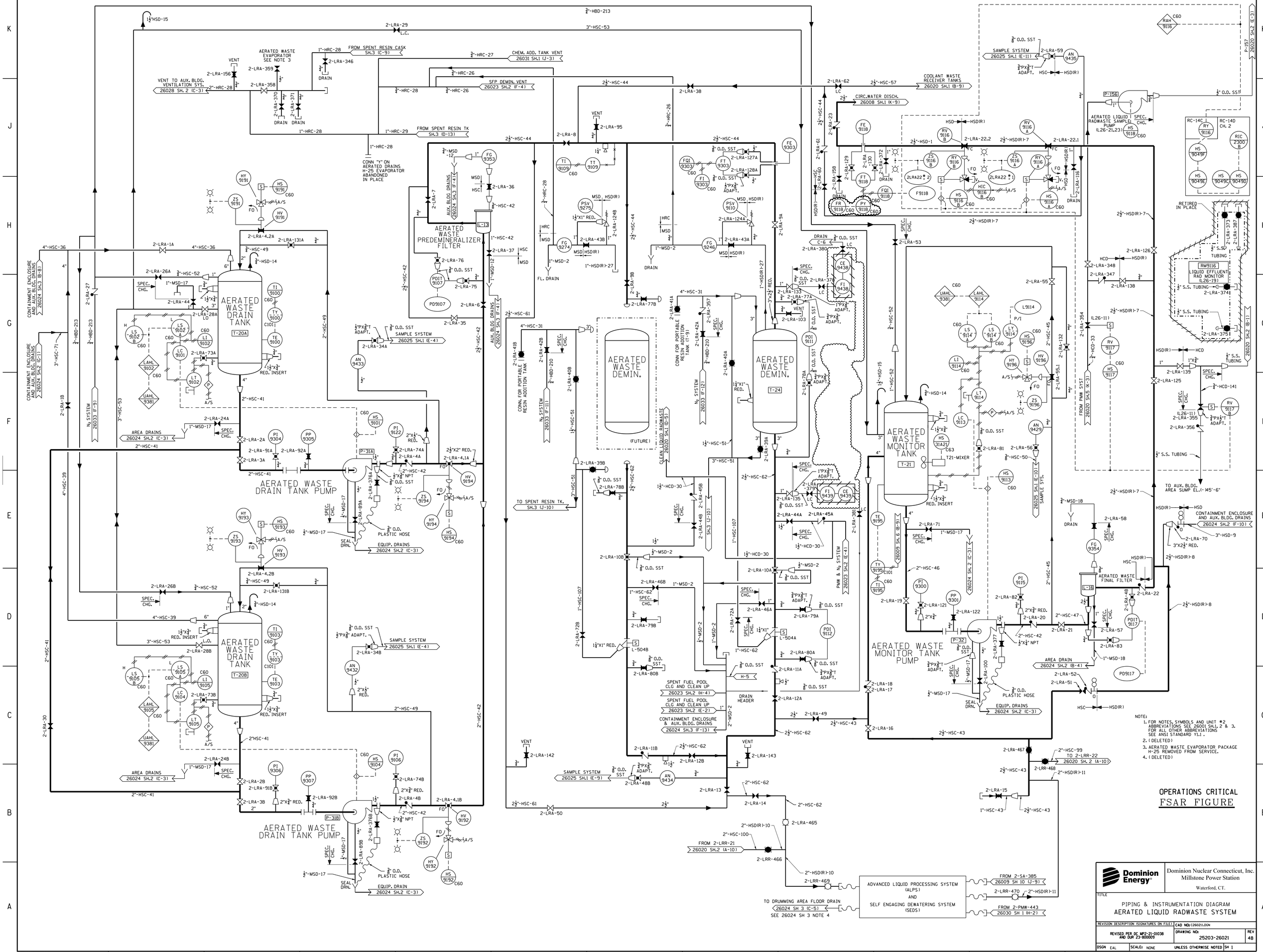
LIQUID EFFLUENT RAD MONITOR (L26-19)

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy	Dominion Nuclear Connecticut, Inc. Millstone Power Station, Waterford, CT.
TITLE	PIPING AND INSTRUMENTATION DIAGRAM CLEAN LIQUID RADWASTE SYSTEM
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260202.DGN
REVISED PER DC MP2-19-01091 & DUR 23-800023	DRAWING NO: 25203-26020 REV 37
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED SH 2

MBULOGIN\scott84
4-FEB-2023 14:48
C260202.DGN


AERATED WASTE DRAIN TANK
T-20A
AERATED WASTE DRAIN TANK PUMP
AERATED WASTE DRAIN TANK
T-20B
AERATED WASTE DRAIN TANK PUMP
AERATED WASTE PREDEMINERALIZER FILTER
AERATED WASTE DEMIN.
(FUTURE)
AERATED WASTE DEMIN.
T-24
AERATED WASTE MONITOR TANK
T-21
AERATED WASTE MONITOR TANK PUMP
AERATED WASTE FINAL FILTER
ADVANCED LIQUID PROCESSING SYSTEM (ALPS) AND SELF ENGAGING DEWATERING SYSTEM (SEDS)
OPERATIONS CRITICAL
FSAR FIGURE
NOTE:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. (DELETED)
3. AERATED WASTE EVAPORATOR PACKAGE H-25 REMOVED FROM SERVICE.
4. (DELETED)
Dominion Energy
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.
TITLE
PIPING & INSTRUMENTATION DIAGRAM
AERATED LIQUID RADWASTE SYSTEM
REVISION DESCRIPTION (SIGNATURES ON FILE)
REVISED PER DC MP2-21-01038 AND DUR 23-800009
CAD NO: C260211.DGN
DRAWING NO: 25203-26021
REV 48
DSGN EAL
SCALE: NONE
UNLESS OTHERWISE NOTED
SH 1

WASTE GAS DECAY TANK T-19A

WASTE GAS DECAY TANK T-19B

WASTE GAS DECAY TANK T-19C

WASTE GAS DECAY TANK T-19D

WASTE GAS DECAY TANK T-19E

WASTE GAS DECAY TANK T-19F

WASTE GAS FILTER L-17

MOISTURE SEPARATOR

MOISTURE SEPARATOR

WASTE GAS COMPRESSOR F-1B

WASTE GAS COMPRESSOR F-1A

WASTE GAS SURGE TANK T-18

ACCOOLER

ACCOOLER

FROM COMPRESSOR CONTROLLER

SAMPLING SYSTEM 26025 SH.4 (E-12)

SAMPLING SYSTEM 26025 SH.4 (E-11)

SAMPLING SYSTEM 26025 SH.4 (E-10)

SAMPLING SYSTEM 26025 SH.4 (E-8)

SAMPLING SYSTEM 26025 SH.4 (A-9)

TO SAMPLE SYSTEM

CVCS ION EXCH. VENT 26017 SH.2 (E-1)

FROM VOLUME CONTROL TANK 26017 SH.1 (K-5)

FROM VOLUME CONTROL TANK GAS SUPPLY RELIEF 26017 SH.1 (J-5)

FROM COOLANT WASTE RECEIVER AND MONITOR TANKS RELIEF VENT 26020 SH.2 (F-10)

FROM STM. GEN. BLWDN. QUENCH TANK 26002 SH.2 (J-7)

FROM DEGASIFIER 26020 SH.4 (F-8)

PREVIOUSLY FROM BORIC ACID EVAPORATOR

FROM EQUIPMENT DRAIN SUMP TANK 26020 SH.5 (H-9)

SAMPLE SYSTEM 26025 SH.4 (E-13)

SAMPLE SYSTEM 26025 SH.4 (E-9)

TO COOLANT WASTE RECEIVER AND MONITOR TANKS 26020 SH.2 (C-9)

TO COMPRESSOR F-1A

TO COMPRESSOR F-1B

TO SAMPLE SYSTEM 26025 SH.4 (E-12)

RWS DEMIN. VENT 26020 SH.1 (J-1)

AUX BLDG. DRAIN SYSTEM 26024 SH.3 (F-8)

CLEAN LIQUID RADWASTE 26020 SH.1 (J-1)

CONTAINMENT ENCLOSURE AND AUX. BLDG. DRAINS 26024 SH.3 (G-7)

TO CONTAINMENT DRAINS

FROM QUENCH TANK 26014 SH.2 (D-9)

FROM RCP & CEDM VENT SYS. TK. 105

TO/FROM RBCCW 26022 SH.2 (D-11) / 26022 SH.2 (C-10)

TO/FROM RBCCW 26022 SH.2 (B-12) / 26022 SH.2 (C-11)

FROM PRIMARY DRAIN TANK 26020 SH.5 (G-4)

TO UNIT 1 STACK 26028 SH. 5 (E-6)

N2 SYSTEM 26033 (E-9)

N2 SUPPLY 26020 SH. 2 (J-1)

TEMPORARY NITROGEN SPLY

VENDOR SUPPLIED

BULKHEAD CONNECTOR

INJECTION PP W/SIGHT GLASS

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .

REDRAWN FROM 25203-26021 SH.2 THRU REV.7

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 34

OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
GASEOUS RADWASTE SYSTEM

REVISION DESCRIPTION SIGNATURES ON FILE	CAD NO: C26021 2.DGN	
REVISED PER MSDUR 19-000012 & DUR 19-800028	DRAWING NO: 25203-26021	REV 33
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

MBUILDING_one694
06-FEB-2019 14:34
C26021 2.DGN

25203-26021 SH-0000000

HDR. FLUSH CONN.

FC 9276A

2-SRW-70

1½"

SPENT FUEL POOL DEMINERALIZER
26023 SH.2 (E-4)
3"-HSC-88

3"-HSC-51 → SH.1 (E-8)
3"-HSC-51 → SH.1 (E-6)
AERATED WASTE DEMIN. TANKS

FUTURE SECONDARY DEMINERALIZER
26020 SH.1 (F-1)
3"-HSC-27

SECONDARY DEMINERALIZER
26020 SH.1 (F-3)
3"-HSC-27

FUTURE PRIMARY DEMINERALIZER
26020 SH.1 (F-6)
3"-HSC-27

3"-HSC-40

PRIMARY DEMINERALIZER
26020 SH.1 (F-10)
3"-HSC-27

PRIMARY DEMINERALIZER
26020 SH.1 (F-7)
3"-HSC-27

PURIFICATION & DEBORATING ION EXCHANGERS
26017 SH.2 (C-5)
3"-HSC-60

2-SRW-1

1"-HSC-124

FG 9276

3"-HSC-40

2-SRW-3 LC

PAH 9120 C60

UAHL 9381

PS 9120 C60

PT 9120

PI 9120 C60

⅜" O.D. SST

SPEC. CHG.

2-SRW-21

¾"-HSC-134

2-SRW-4

2-SRW-30 ¾" O.D.

P A/S

LT 9427

LS 9427

LAL 9427 C60

PI 9121

½"-MSD-12

FROM P-29 & P-30 CASING DRAINS
26020 SH.2 (B-6)

L-152 F

½"-MSD-49

FG 9135A

⅜" O.D.

FG 9135

MSD

DRAIN

1"HRD-9

HRD

HRD

2-SRW-86 A

¾"

VENT
SH.1 (J-9)

HRC

1"-HRC-29

2-SRW-52

2-SRW-18

SPEC. CHG.

1"

2-SRW-29

PSV 9121

SPEC. CHG.

¾"-HSC-87

1"-HSC-87

2-SRW-8

2-SRW-2

SPENT RESIN TANK T-25

2-SRW-14

1½"-HSC-86

2-SRW-12

2-SRW-13

2-SRW-5

3"-HSC-54

2-SRW-15

1½"-HSC-86

2-SRW-79

2-SRW-17

¾"-HBD-214

N₂ SYSTEM
26033 (F-9)

SPEC. CHG.

2-SRW-16

1½"-HCD-39

WATER TREATMENT SYSTEM
26030 SH.1 (F-3)

2"-HSC-56

2-SRW-9 LC

2"-HSC-56

3"-HSC-55

LC 2-SRW-6

3"-HSC-55

3"-HSC-55

S A/S FC SPEC. CHG. 1"X¾" TYP 2-SRW-51

¾"-HCD-39

2-SRW-53

S A/S FC ½"PX½" HOSE ADAPT. 2-SRW-51C

½"-HCD-144

½"

S A/S FC ½"PX½" HOSE ADAPT. 2-SRW-51B

½"

1"X¾" RED. 2-SRW-76

¾"-HCD-39

2-SRW-7

SPEC. CHG.

A/S S

⅜" O.D. SST

2-SRW-51A ½"PX⅜" ADAPT.

2-SRW-48

¾"X¾"

2-SRW-82

¾"X¾"

2-SRW-83

1½"X¾"

PI 9815

SPENT RESIN FILL HEAD

LT 9814

LIC 9814

LAHL 9814

DRIP PAN

CONTAINMENT ENCLOSURE AND AUX. BLDG. DRAINS
26024 SH.3 (G-6)

2"-HSC-56

2-SRW-11

2-SRW-10

L-505 S

1"-HSC-124

2-SRW-31

1½"-HSC-86

2-SRW-69

2-SRW-68

TO VENTILATION SYSTEM SH.1 (K-10)

1"-HRD-6

1"X¾" RED.

2-LRA-345

SPEC. CHG.

S A/S FO SPEC. CHG. 1"X¾" TYP 2-SRW-46

PS 9799

PI 9799

⅜" O.D. SST

¾"PX⅜"T 2-SRW-78

2-SRW-77

¾"-HSD-4

2"-HSC-56

P-108

PUMP SKID DRIP PAN

⅜" O.D. SST. (WSD)

DRAIN

⅜" TUBING

S A/S FO SPEC. CHG. 1"X¾" 1" 2-SRW-47 1"X¾"

2-SRW-81

PI 9814

¾"-HCD-39

SPENT RESIN SHIPPING CASK DEWATERING PUMP

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3 FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .

REDRAWN FROM 25203-26021 SH.1 THRU REV.15
25203-26021 SH.2 THRU REV.7

OPERATIONS CRITICAL

Q.A.

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
12	7-27-07	INCORP. DCN DM2-00-0146-07	EAL	REB	DLW	RFL
11	12-17-02	INCORP. DCN DM2-00-0226-01	JBB	TF	DED	RJY
10	10-11-01	INCORP. DCN DM2-00-0591-00	JSJ	TF	DLW	RJY
N/A	9	11-17-99	INC. DCN DM2-00-1547-98	SJM	MECH. ENGR.	REB	REB
84-024	8	9-20-95	INC. DCN DM2-S-1013-95 PER DCR M2-S-0348-95	LLJ	DRAFTING SUPV.	REB	MANAGER

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE NUCLEAR POWER STATION UNIT-2
PIPING & INSTRUMENTATION DIAGRAM
SPENT RESIN RADWASTE SYSTEM
WATERFORD, CT

BY ASK	CHKD S.REINHARDT	APP. WRK	BJD	APP. SBR	
DATE 1-5-87	DATE 6-4-87	DATE 6/16/87	6/16/87	DATE 6/16/87	
SCALE NONE	DWG. NO. 25203-26021 SH.3			REV 12	

TO 2-CH-915 26017 SH. 3 (K-8) — ½"-HCD-138

TO CHARGING PUMP/ SEAL LUBRICATION SYSTEM 26017 SH. 1 (B-12)

¾" O.D. TUBING — CWRT & CWMT REF LEG FILL 26020 SH. 2 (B-4)

SPEC. CHG. — ½" S.S. TUBING — TO RM-9049 26020 SH. 2 (E-4) — 2-RB-399

1"-HBD(B)-239 — 2-RB-213 — 4"-HBD(B)-239

FLUSH — 2-RB-271 — ½" — ½"-HBD(B)-116 — ½" OD — 2-RB-268 — HBD(B) C-4 — RM-6038 — C-4 HBD(B) — ½" OD — ½"-HBD(B)-116 — 2-RB-269

RAH 9422 C06 — HS 6038 — RC14D — 2-RB-270 — ½"-MSD — HBD(B) — MSD — RE 6038 — FIS 6038 — RIY 2000 B — RIC 2300 B — RC14D CH-1 — R6038 — RJR 9373 — RC14D CH-5

PAH 6894 — G-8,E-8 — T6269 — T6270 — TE 6270 — TE 6269 — SIAS STARTS — PS 6008 — P6008 — PI 6008 — PT 6008

2½"-HBD(B)-239 — 2-HCD(B)-239 — IF — FO 6749 — HCD(B) HBD(B) — 2-RB-107A — 2-RB-182A — 2-RB-2A — 2-RB-3A — 4"-HBD(B)-239

2-RB-240A — PS 6119 A — PI 6743 — 2-RB-111A — HS 6119 A — M — VENT — 2-RB-74A — 12" — P-11A — 14" — PI 6744 — 2-RB-111B — 2-RB-75A — 2-RB-75D — ½"-MSD-14 — EQUIP DRAIN SYSTEM 26024 SH.3 (E-7) — SPEC. CHG.

2-RB-212A — 20"-HBD(B)-116 — 2-RB-56A — AN 9764 — SAMPLE SYSTEM 26025 SH.1 (E-8) — ⅜" S.S. TUBING

2-RB-214A — 1" — FI 6312 — ½"-HBD(B)-116 — 2-RB-42 — 2-RB-43 — VENT — 2-RB-203A — SERVICE WATER 26008 SH.2 (K-10)

(H,F-4) — TAH 6895 — TS 6031 — TT 6031 — TI 6031 — T6031 — PI 6028 — 2-RB-113A — TE 6031 — TIC 6308 — TO TV-6308 26008 SH.2 (K-8) — AN 9767 — 2-RB-57A — SAMPLE SYSTEM 26025 SH.1 (E-9) — 20"-HBD(B)-117

PSV 6642 — 1" — OPEN ENDED — 2-RB-303A — ¾"-HBD(B)-116 — VENT CAP — 2-RB-76A — HBD(B) HCD(B) — X-18A — 2-RB-77B — 2-RB-77A — PI 6022 — 2-RB-112A — TE 6120 — T6120 — SERVICE WATER 26008 SH.2 (J-9)

2-RB-4A — 1"-HBD(B)-116 — 2-RB-204A — HS 6011 — HY 6011 — TT 6032 — TI 6032 — T6032 — TS 6032 — J-5 — AN 9768 — 2-RB-57B — SAMPLE SYSTEM 26025 SH.1 (E-8)

NC — 2-RB-251A — HV 6011 — ZS 6011 — FL — 20"-HBD(B)-117 — TIC 6307 — TO TV-6307 26008 SH.2 (J-8) — PI 6029 — TE 6032 — PSV 6643 — 1" — OPEN ENDED — 2-RB-303B — ¾"-HBD(B)-116 — VENT CAP — 2-RB-76B — HBD(B) HCD(B) — 2-RB-113B — X-18B — 2-RB-77C — 2-RB-77D — PI 6023 — 2-RB-112B — TE 6121 — T6121 — SERVICE WATER 26008 SH.2 (H-9)

2-RB-4B — 1"-HBD(B)-116 — 2-RB-204B — HS 6012 — HY 6012 — TT 6033 — TI 6033 — T6033 — TS 6033 — J-5 — AN 9769 — 2-RB-57C — SAMPLE SYSTEM 26025 SH.1 (E-7)

NC — 2-RB-251B — HV 6012 — ZS 6012 — FL — 20"-HBD(B)-117 — TIC 6306 — TO TV-6306 26008 SH.2 (H-8) — PI 6030 — TE 6033 — PSV 6644 — 1" — OPEN ENDED — 2-RB-303C — ¾"-HBD(B)-116 — VENT CAP — 2-RB-76C — HBD(B) HCD(B) — 2-RB-113C — X-18C — 2-RB-77E — 2-RB-77F — PI 6024 — 2-RB-112C — TE 6027 — T6027 — SERVICE WATER 26008 SH.2 (G-9)

2-RB-4C — 1"-HBD(B)-116 — 2-RB-204C — 2-RB-203C — VENT — 3" — 20"-HBD(B)-116 — SERVICE WATER 26008 SH.2 (G-10) — FI 6314 — 1" — 2-RB-39 — 2-RB-38 — 2-RB-214C — 1" — 20"-HBD(B)-116 — 2-RB-3C — 2-RB-2C — 12" — P-11C — 14" — 2-RB-111F — PI 6748 — 2-RB-75C — 2-RB-75F — ½"-MSD-14 — EQUIP DRAIN SYSTEM 26024 SH.3 (D-7) — SPEC. CHG.

2-RB-56C — AN 9766 — SAMPLE SYSTEM 26025 SH.1 (E-7) — ⅜" S.S. TUBING

2-RB-240C — PS 6119 C — PI 6747 — 2-RB-111E — HS 6119 C — T6310 — T6317 — TE 6317 — TE 6310 — J-7 — PS 6010 — P6010 — PI 6010 — PT 6010 — 2-RB-74C — 2-RB-212C — 2-RB-107C — 2-RB-182C — FO 6751 — HCD(B) HBD(B) — IF — 2½"-HBD(B)-239

2-RB-214B — 1" — FI 6313 — 1" — 2-RB-41 — 2-RB-40 — 2½"-HBD(B)-239 — 2-HCD(B)-239 — IF — FO 6750 — HCD(B) HBD(B) — 2-RB-107B — 2-RB-182B — 2-RB-2B — 2-RB-3B — 12" — P-11B — 14" — 2-RB-111D — PI 6746 — 2-RB-75B — 2-RB-75E — ½"-MSD-14 — EQUIP DRAIN SYSTEM 26024 SH.3 (D-7) — SPEC. CHG.

2-RB-56B — AN 9765 — SAMPLE SYSTEM 26025 SH.1 (E-8) — ⅜" S.S. TUBING

2-RB-240B — PS 6119 B — PI 6745 — 2-RB-111C — HS 6119 B — T6271 — T6309 — TE 6309 — TE 6271 — J-7 — PS 6009 — P6009 — PI 6009 — PT 6009 — 2-RB-74B — 2-RB-212B — HEADER "A" OR "B" 20"-HBD(B)-116 — 2-RB-2B

HEADER "A" 20"-HBD(B)-115 — HEADER "B" 20"-HBD(B)-115 — VENT — 2-RB-201 — ¾" — 2-RB-44 — HY 6002 — HS 6002 — A/S — ZS 6002 — 2-RB-45 — HEADER "A" 20"-HBD(B)-115 — FL — 2-RB-211A — 2-RB-202A — HV 6002 — 1"-HBD(B)-115

1"-HBD(B)-115 — 2-RB-202B — HS 6003 — A/S — ZS 6003 — NC — 2-RB-211B — HV 6003 — HY 6003

1"-HBD(B)-115 — 2-RB-202C — HV 6004 — HY 6004 — HS 6004 — A/S — ZS 6004 — 2-RB-46 — 2-RB-47 — HEADER "A" OR "B" 20"-HBD(B)-115 — FL — 2-RB-211C

1"-HBD(B)-115 — 2-RB-202D — HV 6005 — HS 6005 — A/S — ZS 6005 — NC — 2-RB-211D — HY 6005

1"-HBD(B)-115 — 2-RB-202E — HV 6006 — HS 6006 — A/S — ZS 6006 — 2-RB-48 — 2-RB-49 — HEADER "B" 20"-HBD(B)-115 — FL — 2-RB-211E

1"-HBD(B)-115 — 2-RB-202F — HV 6007 — HS 6007 — A/S — ZS 6007 — NC — 2-RB-211F — HY 6007

HEADER "A" 20"-HBD(B)-115 — HEADER "B" 20"-HBD(B)-115 — 4"-HBD(B)-239 — 2-RB-274 — VENT — 1" — 2-RB-109B — 2-RB-109A — 4-HBD(B)-241 — 3" — 3" — VENT — 2-RB-376 — R.B.C.C.W. SURGE TANK T-3 — 1½" — 2-RB-55A — 2-RB-55B — 1½" — 1½" — 2-RB-377 — 1/2" PIPE X 1/4" TUBE RED. — DRAIN — LC 6000 — LS 6730 — LI 6730 — LT 6730 — 2-RB-110B — WEIR — 2-RB-110A — LT 6001 — LI 6001 — LS 6001 — L6001 — 4-HBD(B)-241 — 2"-HBD(B)-241 — 2-RB-548 — 2-RB-544 — LAHL 6892

2-RB-1B — 2-RB-1A — FQ 6039 — FQ 6040 — 6"-HBD(B)-115 — 8"-HBD(B)-115 — CHEMICAL ADDITION TANK 26031 SH.1 (H-1) — ¾"-HCD-80 — SPEC. CHG. — ¾"-HCD(B)-74 — 2-RB-50A — 2-RB-50B — SPEC. CHG. — T6040 — TE 6040 — 20"-HBD(B)-115 — PLUG — T6039 — TE 6039 — ¾"-HCD(B)-74

VENT — 2-RB-200A — HEADER "A" RETURN SH. 4 (C-13) — 16"-HBD(B)-128 — VENT — 2-RB-200B — HEADER "A" RETURN SH. 2 (A-11) — 20"-HBD(B)-132 — HEADER "B" RETURN SH. 5 (B-9) — 16"-HBD(B)-128 — 20"-HBD(B)-115 — HEADER "B" RETURN SH. 2 (B-7) — 20"-HBD(B)-132

ZAH 6000 — 2RB215 — LV 6000 — A/S — FO — S — ZS 6000 — HS 6000 — TEST CONN. — 1" — 2-RB-421 — 2-RB-417 — 2-RB-418 — 3"-HBD(B)-240 — 1"-HCD-138 — 2-RB-398 — DRAIN — 2-RB-272 — 3"-HCD-35 — 26030 SH. 1 (B-3) — 2-RB-66 — 2-RB-215 — SPEC. CHG. — 2-RB-67 — 2-RB-273 — 1"-HBD(B)-240 — SPEC. CHG. — 3"-HCD-35 — 2-RB-53

R.B.C.C.W. PUMPS (3)

PAL 6037 — PS 6037 — 2-RB-129B — HEADER "A" TO CAR FAN UNITS & RV SUPPORT COOL COILS — 16"-HBD(B)-117 — SH. 5 (K-8) — HEADER "A" HPSI/LPSI — SH. 3 (D-13) — 20"-HBD(B)-117

R.B.C.C.W. HEAT EXCHANGERS (3)

TO HEADER "A" SH. 2 (G-12) — 12"-HBD(B)-133 — HEADER "A" 20"-HBD(B)-117 — HEADER B — HEADER A

PAL 6036 — PS 6036 — 2-RB-129A — HEADER B TO X-20 SH. 2 (K-11) — 12"-HBD(B)-133 — 2-RB-209 — 1"-HBD(B)-133 — DRAIN — HEADER B SH. 5 (B-11) — 16"-HBD(B)-117 — HEADER B 18"-HBD(B)-131 — SH. 2 (H-5) — 20"-HBD(B)-117 — HEADER A

FAH 6034 — FS 6034 — F6034 — FI 6034 — 2-RB-114A — FT 6034 — FE 6034 — 2-RB-114B — HEADER B — HEADER A

VENT — 2-RB-205A — 2-RB-115A — FE 6035 — FT 6035 — 2-RB-115B — FI 6035 — F6035 — FAH 6035 — FS 6035 — VENT — 2-RB-205B

TO F-1A & F-1B SH. 2 (D-12) — 1"-HBD(B)-114

PIPING LINE SYMBOLS
HEADER A
HEADER B
HEADER A OR B (SWING FACILITY)

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1,2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL DRAINS FROM THE R.B.C.C.W SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID SEE DRAWING 26024, (SHTS 1-3) A FUNNEL (Y)DENOTES THE OPEN (MSD)DRAIN SYSTEM.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 17

FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL
REDRAWN INPART FROM 25203-26022 SH. 1 REV 15.

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM
R.B.C.C.W. PMPS & HEAT EXCHANGERS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260221.DGN	
REVISED PER MSDUR 23-802015 & DUR 23-800309	DRAWING NO: 25203-26022	REV 47
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

Authenticated By: MBULOGIN
20-NOV-2023 13:01
C260221.DGN
ver ik004

SPENT FUEL POOL HEAT EXCHANGER

AERATED WASTE EVAPORATOR PACKAGE
RETIRED IN PLACE PER PDCR 2-103-79

QUENCH TANK HEAT EXCHANGER

SHUTDOWN HEAT EXCHANGER

SPENT FUEL POOL HEAT EXCHANGER

BORIC ACID EVAPORATOR PACKAGE
RETIRED IN PLACE PER PDCR 2-95-054

AFTERCOOLER

AFTERCOOLER

WASTE GAS COMPRESSORS

ENGINEERED SAFETY FEATURES ROOM AIR COOLING COIL

ENGINEERED SAFETY FEATURES ROOM AIR COOLING COIL

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL DRAINS FROM THE R.B.C.C.W SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID SEE DRAWING 26024, (SHTS 1-3) A FUNNEL (Y) DENOTES THE OPEN (MSD) DRAIN SYSTEM.

PIPING LINE SYMBOLS
- HEADER A
- HEADER B
- HEADER A OR B (SWING FACILITY)

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 18

OPERATIONS CRITICAL

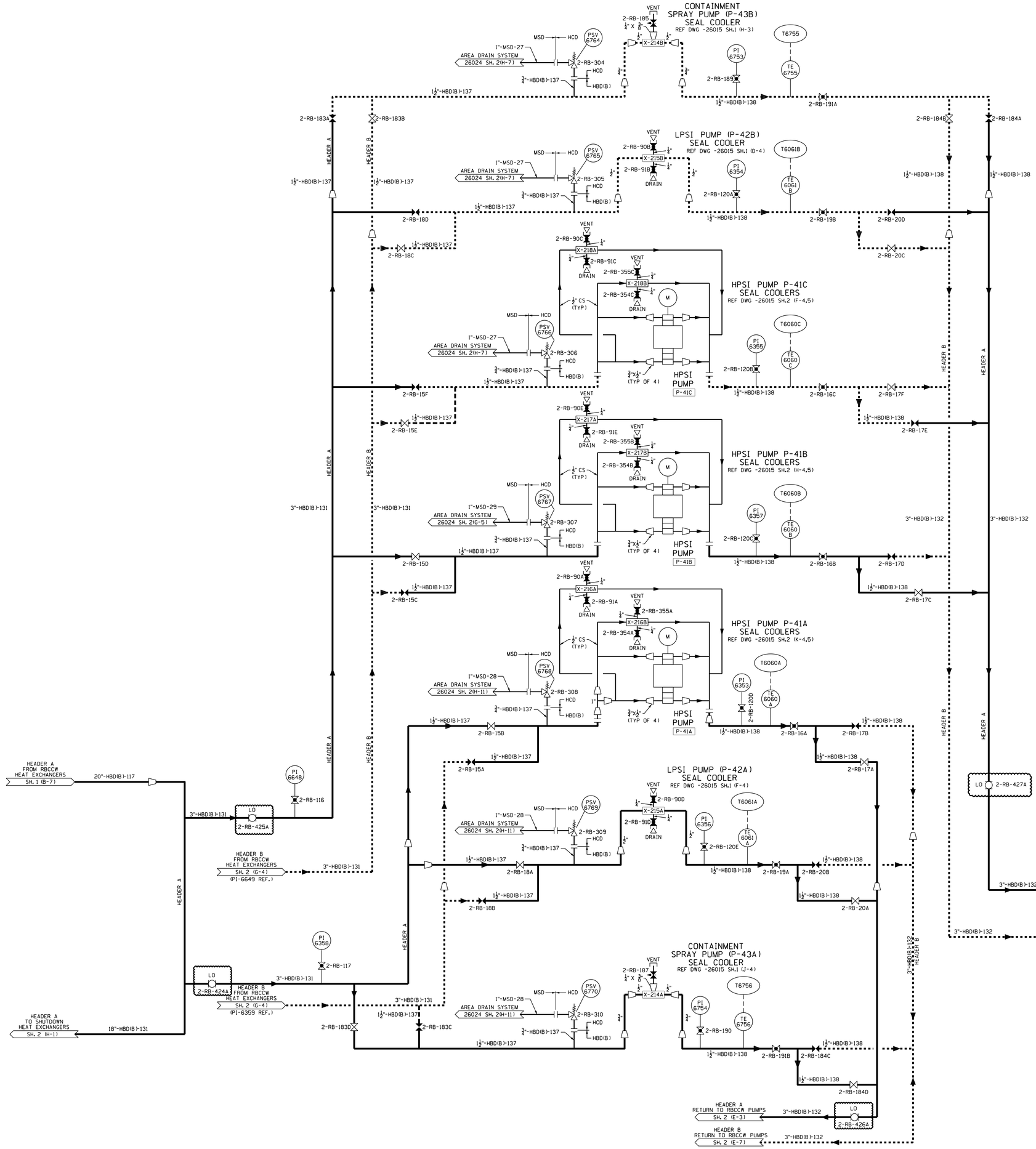
FSAR FIGURE

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM SPENT FUEL POOL AND SHUT-DOWN HEAT EXCHANGERS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplo1\C260222.DGN	
REVISED PER MSDUR 17-000255 & DUR 17-800563	DRAWING NO: 25203-26022	REV 28
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL DRAINS FROM THE R.B.C.C.W SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID SEE DRAWING 26024, (SHTS 1-3) A FUNNEL (Y) DENOTES THE OPEN (MSD) DRAIN SYSTEM.

PIPING LINE SYMBOLS

- HEADER A
- HEADER B
- HEADER A OR B (SWING FACILITY)

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 19

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM CNTMT
SPRAY PMP AND SI. PMP SEAL COOLERS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: PA260223.DGN	
REVISED PER DC MP2-18-00105 & DUR 18-800201	DRAWING NO: 25203-26022	REV 15
DSGN MKN	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

MBLOGIN\dav1714
08-OCT-2018 16:16
PA260223.DGN

PRIMARY DRAIN TANK AND QUENCH TANK COOLER

CEDM COOLERS (3)

RCP LUBE OIL COOLERS (4)

REACTOR COOLANT PUMPS AND THERMAL BARRIERS (4)

HEADER A

HEADER B

PIPING LINE SYMBOLS

HEADER A
HEADER B
HEADER A OR B (SWING FACILITY)

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL DRAINS FROM THE R.B.C.C.W SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID SEE DRAWING 26024, (SHTS 1-3) A FUNNEL (Y) DENOTES THE OPEN (MSD) DRAIN SYSTEM.
3. ORIFICE "D" IS PERMANENTLY INSTALLED BY RCP DWG. #25203-20014, COOLING WATER PIPING PHYSICAL ARRANGMT. (BYRON JACKSON)

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

REDRAWN INPART FROM 25203-26022 SHEET 2 REVISION 8.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 20

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.
N/A	24	5-30-08 INC. DCN DM2-00-0160-08	DWJ	REB	DLW
N/A	23	5-18-05 INC. DCN DM2-00-0127-05	SMC	ASK	WBZ

DED MP

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM
RCP. THERMAL BARRIERS & LO COOLERS
WATERFORD, CT

	BY	CHK'D.	APP.	APP.	APP.	SBR
	M.R. BIBISI	SBR	SRF	GEC	RCB	BJD
DATE	4-27-87	10-28-87	8-1-88	8-1-88	8-1-88	8-1-88

SCALE NONE
DWG. NO. 84-024
25203-26022 SH. 4
REV 24

SAMPLE COOLERS (2)
(SAMPLE LINES SHOWN ON DWG #26025 SH.1)

SAMPLE COOLER X-192

CTMT. AIR RECIRC. AND COOLANT UNIT X-35A

CTMT. AIR RECIRC. AND COOLANT UNIT X-35C

CTMT. AIR RECIRC. AND COOLANT UNIT X-35B

CTMT. AIR RECIRC. AND COOLANT UNIT X-35D

MULTICIRCUIT COOLER X-64

MULTICIRCUIT COOLER X-65

DEGASIFIER EFFLUENT COOLER X-51

LETDOWN HEAT EXCHANGER X-22

DEGASIFIER VENT CONDENSER (26020 SH.4, F-10)

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y14.
2. ALL DRAINS FROM THE R.B.C.C.W. SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID SEE DRAWING 26024, SH. 1-3 A FUNNEL (Y) DENOTES THE OPEN MSD/DRAIN SYSTEM.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 21

PIPING LINE SYMBOLS
HEADER A
HEADER B
HEADER A OR B (SWING FACILITY)

REDRAWN INPART FROM 25203-26022 SH. 2 THRU REV. 8

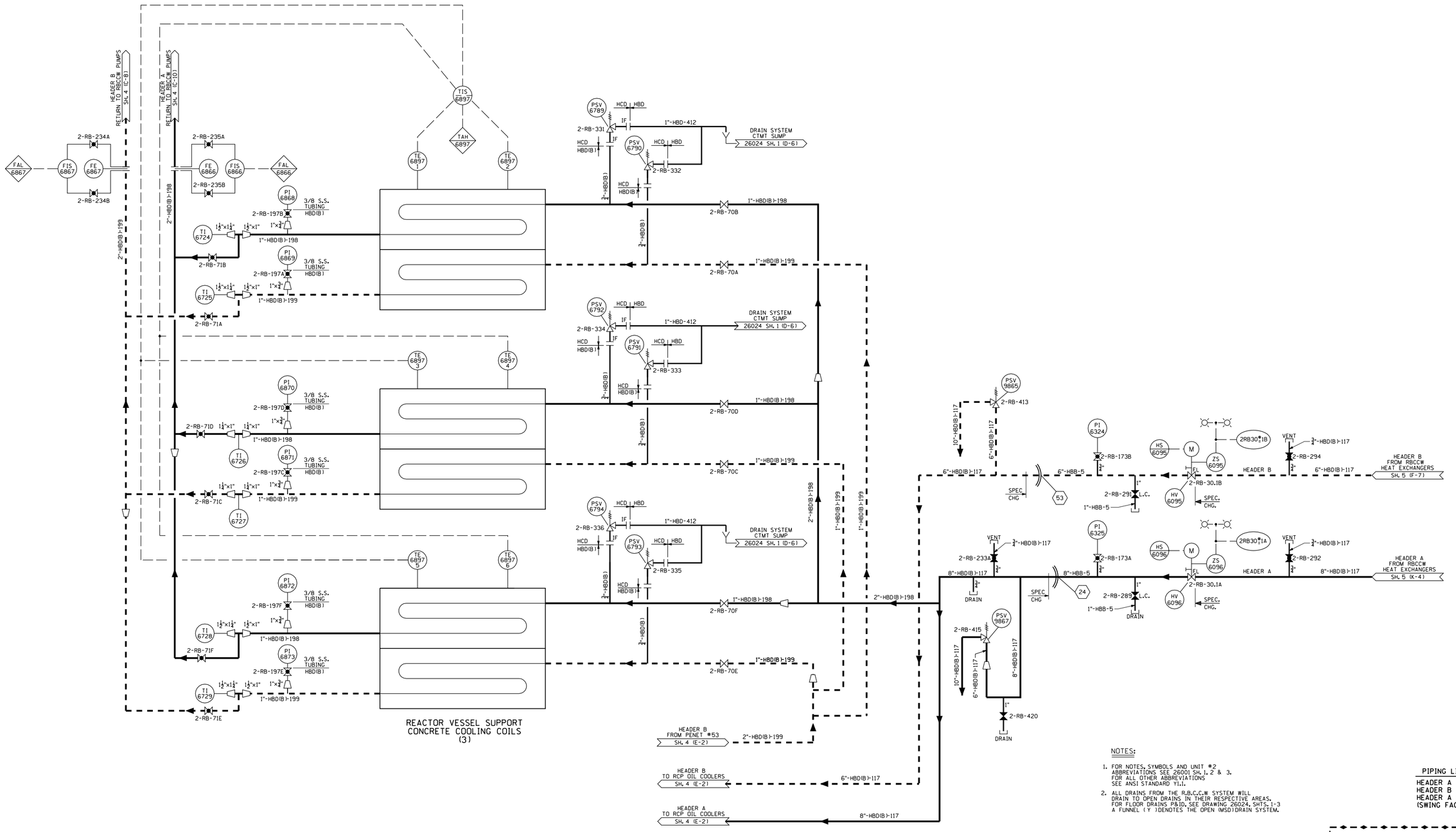
OPERATIONS CRITICAL

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING AND INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM
CNTMT AIR RECIRC. AND COOLANT UNIT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260225.DGN	
REVISED PER MSDUR 14-000096 & DUR 14-800288	DRAWING NO: 25203-26022	REV 27
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 5



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL DRAINS FROM THE R.B.C.C.W. SYSTEM WILL DRAIN TO OPEN DRAINS IN THEIR RESPECTIVE AREAS. FOR FLOOR DRAINS P&ID, SEE DRAWING 26024, SHTS. 1-3 A FUNNEL (Y)DENOTES THE OPEN (MSD)DRAIN SYSTEM.

PIPING LINE SYMBOLS

- HEADER A
- HEADER B
- HEADER A OR B (SWING FACILITY)

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 22

OPERATIONS CRITICAL

Q.A.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REDRAWN INPART FROM 25203-26022 SHEET 2 REVISION 8.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	5-18-05	INC. DCN DM2-00-0127-05	SMC	ASK	WBZ	MP
N/A	5-10-02	INC. DCN DM2-00-0547-01	BLM	TP	SBB	CAM
N/A	3-7-02	INC. DCN DM2-00-0156-01	JCV	TF	DLW	RJY
N/A	3-3-02	INC. DCN DM2-00-0157-01	JCV	TF	NMN	PFL
N/A	5-29-00	INC: DCN'S DM2-00-0345-97, DM2-00-0025-00 & DM2-00-0355-00	ASK	FM	REB	REB
N/A	8-18-98	INC. DCN DM2-00-0298-97	MCR		REB	REB
N/A	3-12-98	INC. DCN DM2-00-0438-98	GZ		REB	REB

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING AND INSTRUMENTATION DIAGRAM
R.B.C.C.W. SYSTEM REACTOR VESSEL SUPPORT CONCRETE COOLING COILS
WATERFORD, CT

BY	CHKD.	APP.	APP.	APP.
M. R. BIBISI	SBR	SRF	GEC	RCB / SBR / RJD
DATE 7-24-87	DATE 10-28-87	DATE 8-1-88	DATE 8-1-88	DATE 8-1-88 / 8-1-88

SCALE NONE
DWG. NO. 84-024
25203-26022 SH. 6 OF 6
REV 15

25203-26023 SH-0000000

REACTOR VESSEL HEAD DECONTAMINATION SYSTEM

REACTOR VESSEL HEAD IN HEAD STAND

RETIRED IN PLACE

(5) 3/4" REVOLVING SPRAY NOZZLES

SPEC. CHG.

2" 2" 2" 2" 2"

3"-HCC-61

2"-MSD-36 CONTAINMENT BLDG. DRAINS 26024 SH.1 (G-5)

CHEMICAL ADDITION & VENT

2-RW-185

HS 8878

AGITATOR

1"-HCC-62

4"-HCC-58

2-RW-214

MIXING TANK T-95

PMW 26014 SH. 2 (D-10)

1 1/2"-HCD-37

2-RW-197

2-RW-198

1 1/2"-HCC-63

SPEC. CHG.

TIC 8710

HS 8710

2-RW-192

A/S

P

2"-HCC-60

2-RW-199

1"-MSD-36

CONTAINMENT BLDG. DRAINS 26024 SH.1 (F-5)

2-RW-177A

SPEC. CHG.

L-509

S

2-RW-284

3"-HCC-59

1"-HCC-59

LIS 8711

PI 8706

2-RW-204

3"-HCC-60

2-RW-179A

2-RW-178A

2 1/2"

1 1/2"

VENT

2-RW-275

3/4"

2-RW-260

1"

DRAIN

1/2"-MSD-36

2-RW-200A

1/2"-MSD-36 CONTAINMENT BLDG. DRAINS 26024 SH.1 (G-5)

SPEC. CHG.

REMOVABLE FLANGED SPOOL PIECE

3"-HCC-60

SPEC. CHG.

3"-HCC-65

3"-HSC-65

CONTAINMENT BLDG. DRAINS 26024 SH.1 (K-9)

1"

2-RW-183

2-RW-261

2-RW-262

DRAIN

DRAIN

2-RW-180A

2-RW-180B

2-RW-207

SPEC. CHG.

2-RW-205A

2-RW-205B

2 1/2"

FILTERS

3/4"-MSD-36

FG 8712

FG 8713

PDI 8707

L-58A

L-58B

1/2"

2-RW-215

L.S.

3"-HCC-61

1"

2-RW-208A

2-RW-181A

2-RW-181B

2-RW-206

CONTAINMENT BLDG. DRAINS 26024 SH.1 (G-5)

1"-MSD-36

3"-HCC-61

1"-MSD-36

2-RW-208B

FE 8709

FI 8709

2-RW-211

2-RW-210

REFUELING POOL SKIMMER FILTERS

CONTAINMENT BLDG. DRAINS 26024 SH.1 (J-5)

3/4"-MSD-35

FG 7956

2-RW-162B

MSD HCC

3/4"-HCC-55

2-RW-162A

2-RW-74B

2-RW-74A

2-RW-73B

2-RW-73A

L-22B

L-22A

2 1/2"-HCC-55

2-RW-75

2 1/2"-HCC-55

2-RW-290

VENT

2-RW-274

2-RW-99

3/8" O.D. SST

PDI 7326

2-RW-98

1"-HCC-55

2-RW-160B

2-RW-160A

HCC MSD

1"-MSD-35

2 1/2"-HCC-55

4"-HCC-8

PLUG

4"-HCC-2

SPEC. CHG.

4"-HCB-11

68

2-RW-64

LC

2-RW-63

4"-HCC-8

SH. 2 (B-4)

2-RW-154

2-RW-159

2-RW-232

2-RW-21

4"-HCC-2

SH. 2 (C-13)

67

2-RW-158

4"-HCB-10

2-RW-34

CONTAINMENT SUMP 26024 SH.1 (J-8)

3"-HCC-16

2-RW-272

DRAIN

VENT EL 38'-6" MIN

2-RW-249

1"-HCC-65

2-RW-394

2-RW-393

2-RW-362

2-RW-390

LS 9750

LS 9751

2-RW-389

2-RW-391

2-RW-392

2-RW-395

2-RW-363

2-RW-396

24"-HCC-54 DETRAINING SECTION

2 1/2"-HCC-54

PI 7313

2-RW-97

3/4"

2 1/2"-HCC-55

P-96

SPEC. CHG.

1 1/2"

1/2"-MSD-35

2-RW-170

3/4"-MSD-35 SEAL DRAIN

2"

4"

HS 7320

REFUELING POOL SKIMMER PUMP

3/8" O.D. SST

PDI 7330

3/8" O.D. SST

2-RW-96

2-RW-95

L-507

S

3/4"

2"-MSD-35

CONTAINMENT BLDG. DRAINS 26024 SH.1 (J-5)

2 1/2"-HCC-54

1"-HCC-54

2-RW-281

2-RW-233

DRAIN

2 1/2"-HCC-54

SPEC. CHG.

2-RW-72

2 1/2"-HCC-54

2-RW-155

(6" BELOW WATER LEVEL)

1"-MSD-33

2-RW-157

2 1/2"

CONTAINMENT BLDG. DRAINS 26024 SH.1 (D-5)

1"-MSD-33

2-RW-156

4"-HCC-8

1"

2-RW-220

DRAIN

SKIMMER L-100A

SKIMMER L-100B

REFUELING POOL

FUEL TRANSFER CANAL

1"

1 1/2"

2-RW-364

2-RW-365

T.C.

1/2"-MSD-44

REACTOR VESSEL

H-1

3/4"-HCB-27

2-RW-292

LC

42

T.C.

2-RW-31

1/2"-HCB-27

T.C.

2-RW-291

FUEL TRANSFER TUBE ISOLATION VALVE

36"

2-RW-280

4"-HCC-2

2-RW-22

LC

1"

2-RW-259

DRAIN

FUEL TRANSFER TUBE

MONITOR TRENCH DRAINS

TO CONTAINMENT SUMP

FLEX HOSE SEE NOTE 2

L-362

FLEX HOSE SEE NOTE 2

1 10 2 9 7 8

6 5 4 3 11

SPEC. CHG.

MSD-44 HSC-67

2-RW-238 2-RW-235 2-RW-244 2-RW-243 2-RW-241 2-RW-245

2-RW-237 2-RW-242 2-RW-240

2-RW-236 2-RW-239

2-RW-252

MSD-47 HSC-67

CONTAINMENT SUMP 26024 SH.1 (J-5)

4"-MSD-22

4"-HCC-2

2-RW-253 2-RW-254 2-RW-255

3"

2-RW-354

4"X3"

3"-HCC-15

2-RW-23

3-HCD-128

CONTAINMENT SUMP 26024 SH.1 (G-11)

VENT

2-RW-258

LEAK CHASE TEST MANIFOLD

1/2"-HSC-67

LAH 7912

2-RW-373

LS 7911

2-RW-374

2-RW-371

2-RW-372

2-RW-246

2-RW-247

2-RW-24A

4"-MSD-21

CONTAINMENT BLDG. DRAINS 26024 SH.1 (K-5)

LO

2-RW-124

LC

2-RW-123

4"-HCC-2

2-RW-24B

4"-MSD-21

CONTAINMENT BLDG. DRAINS 26024 SH.1 (K-5)

LO

SPEC. CHG.

1/2"-MSD-33

1/2"-MSD-1

2"-MSD-1

CONTAINMENT BLDG. DRAINS 26024 SH.1 (D-4)

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 23A

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1 & 2. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FLEX HOSES SHALL BE REMOVED AND FITTINGS CAPPED DURING THE OPERATING CYCLE.

THIS DRAWING SUPERCEDES 25203-26023 THRU REV 17

OPERATIONS CRITICAL

Q.A.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	18	9-23-04 INC. DCN DM2-00-0131-04	DLW	MP	JCV	DED
N/A	17	2-27-02 INC. DCN DM2-00-0094-02	JCV	LEJ	KEC	RJY
N/A	16	2-16-01 INC. DCN DM2-00-0025-00	NJB	VJG	REB	GMO
N/A	15	1-6-99 INC. DCN DM2-00-1017-98	GZ	XXX	REB	REB
N/A	14	2-5-98 INC. DCN DM2-S-0583-96	GZ	XXX	REB	REB
N/A	13	12-11-97 INC. DCN DM2-00-1377-97, (Pg. 6)	GZ	XXX	DAD	DAD
N/A	12	10-28-97 INC. DCN DM2-00-0843-97	MGD	XXX	REB FOR WAP	REB FOR WAP

Dominion® Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION - UNIT NO. 2
PIPING & INSTRUMENTATION DIAGRAM
SPENT FUEL POOL COOLING & CLEANUP SYS
WATERFORD, CT

BY	CHKD	APP.	APP.
M PUPILLO	S.REINHARDT	WCF	JGR	SBR
DATE 6-1-87	DATE 6-1-87	DATE 6/9/87	DATE 6/9/87	DATE 6/9/87

SCALE NONE | DWG. NO. 25203-26023 SH.1 OF 2 | REV 18

84-024

SPENT FUEL POOL

ZONE 7 | ZONE 6 | ZONE 2 | ZONE 1 | ZONE 3 | ZONE 4 | ZONE 9 | ZONE 8 | ZONE 5 | ZONE 10

LEAK TEST CHASE MANIFOLD

SPENT FUEL POOL SKIMMER PUMP

SPENT FUEL POOL SKIMMER FILTERS

SPENT FUEL POOL FILTERS

REFUELING WATER PURIFICATION PUMPS

SPENT FUEL POOL COOLING PUMPS

SPENT FUEL POOL HEAT EXCHANGERS

SPENT FUEL POOL DEMINERALIZER.

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FLEX HOSES SHALL BE REMOVED AND FITTINGS CAPPED DURING THE OPERATING CYCLE.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

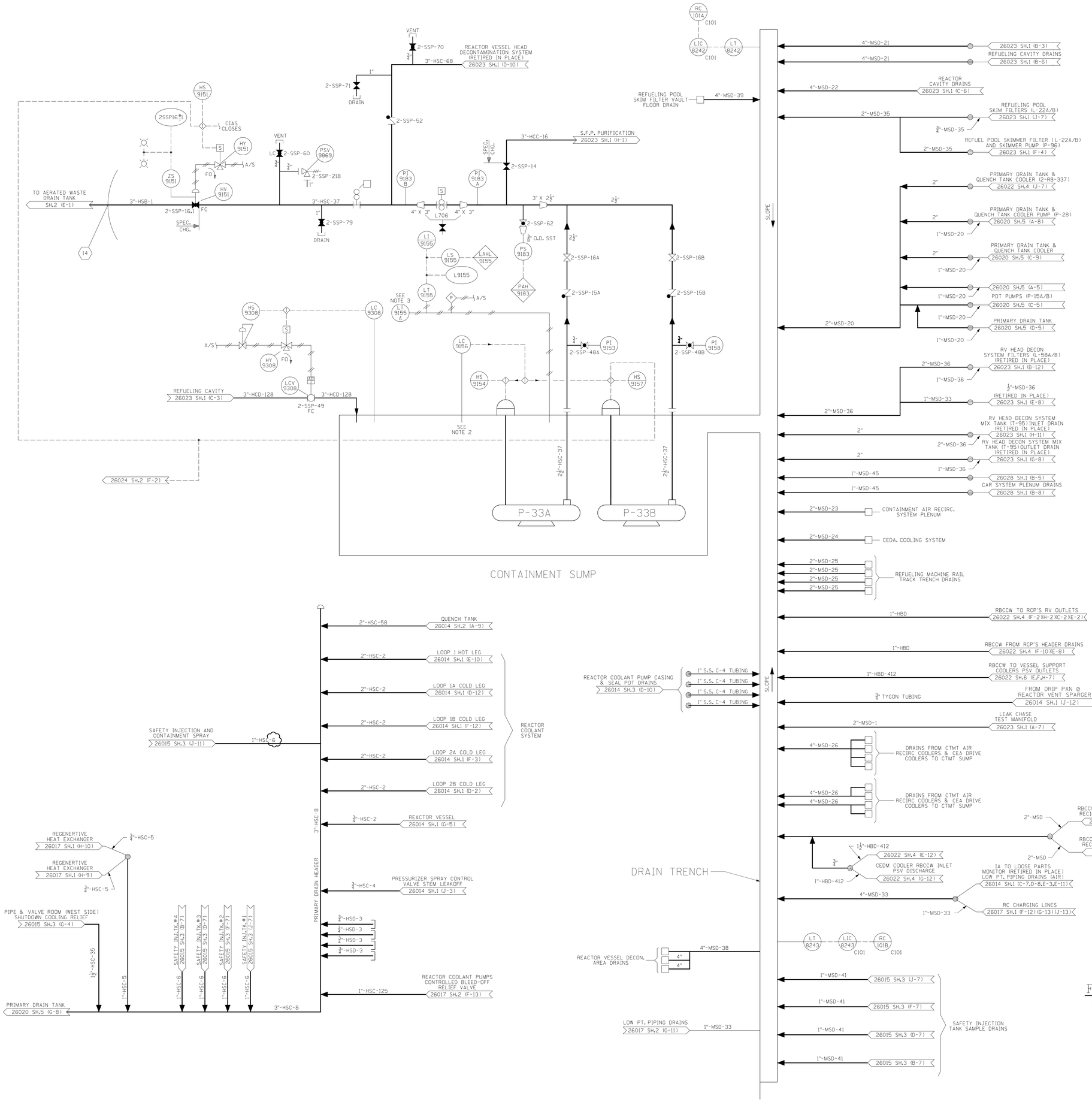
CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 23

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
SPENT FUEL POOL COOLING & CLEANUP SYS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: c:\fms32\tmpplot\C260232.DGN	
REVISED PER DC MP2-13-01011 AND DUR 15-800332	DRAWING NO: 25203-26023	REV 33
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2



Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
CONTAINMENT BUILDING DRAINS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260241.DGN	
REVISED PER MSDUR 15-000164 & DUR 15-800380	DRAWING NO: 25203-26024	REV 21
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

SAFETY INJECTION ROOM "A" SUMP (-)45'-6"

SAFETY INJECTION ROOM "B" SUMP (-)45'-6"

SAFETY INJECTION ROOM "C" SUMP (-)45'-6"

H.P. FACILITY DECON ROOM COLLECTION TANK (TK-139) EL.14'-6"

SERVICE WATER TUNNEL

RBCCW ROOM SUMP ELEV. (-)25'-6"

RAILROAD ACCESS AREA SUMP PUMP 14'-6"

AUXILIARY BUILDING AREA SUMP (-)45'-6"

T48A & B TANK ROOM EMERGENCY DRAINAGE PIPING

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. FOR DRAIN SYMBOLS SEE 26024 SH. 1

OPERATIONS CRITICAL

FSAR FIGURE

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 30

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
AUXILIARY BUILDING DRAINS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260242.DGN	
REVISED PER EC MP2-24-01112 & DUR 25-800081	DRAWING NO: 25203-26024	REV 22
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

MBLOGIN=gr 14884
8-MAY-2025 08:45
C260242.DGN

AREA DRAIN EL.(-)5'-0" 4"
LOCAL SAMPLE DRAIN 2"
SHELL SIDE SPENT FUEL POOL HEAT EXCH. & RELIEFS X-20B 26022 SH.2(H-9,J-11) 2½"
SHELL SIDE SPENT FUEL POOL HEAT EXCH. & RELIEFS X-20A 26022 SH.2(E-10,F-12) 2½"
CORRIDOR EL.(-)15'-0" 4"
FROM 2-CH-999 VCT GAS SAMPLE TUBING DRAIN 26025 SH.4 (E-6) 2½"
SPENT FUEL POOL PUMP 26023 SH.2 (J-4) 2½"
SPENT FUEL POOL PUMP 26023 SH.2 (H-4) 2½"
CHANNEL SIDE SPENT FUEL POOL HEAT EXCHANGERS & RELIEFS 26023 SH.2 (J-1) 2½"
CORRIDOR EL.(-)15'-0" 4"

REACTOR COOLANT PUMP CONTROLLED BLEED OFF 26017 SH.2 (A-8) 2½"
RELIEF VALVE DISCHARGE DRAIN, 2-CH-345 26017 SH.2 (K-7) 2½"
LETDOWN PRE & POST FILTERS, L-10 & L-20 VALVE CH-520 LEAK-OFF 26017 SH.2 (G-5) 2½"
CAPPED 2½"
BORONOMETER 26017 SH.2 (G-4) 2½"
CH-110P & Q LEAK-OFFS 26017 SH.2 (K-10) 2½"
RELIEF VALVE DISCHARGE DRAIN 2-CH-986 26017 SH.1 (H-7) 2½"
2½"-HSC-35

CVCS ION EXCH. DRN & VENT DRN 26017 SH.2 (D-1)
CH-500 LEAK-OFF REC & MON TANK VENTS 26017 SH.2 (A-5)
CHARGING PUMP C VCT & VCT INST. DRN 26017 SH.1 (F-4) 2½"
CHARGING PUMP B 26017 SH.1 (C-5) 2½"
CHARGING PUMP A 26017 SH.1 (A-5) 2½"
SPARE (CAPPED) ION EXCH. ROOM 2½"
VENT 2-SSP-37A ¾"

2-SSP-68 LO 1"-HSC-76 EQUIP. DRAIN SUMP TANK VENT 26020 SH.5 (J-8)
24"-HSC-35
COLLECTION SECTION
2-SSP-63 ⅜" O.D. SST
REMOTE SENSORS
LT 9796
L9796
2-SSP-64 ⅜" O.D. SST
2-SSP-65
LS 9796
LC 2-SSP-69
HY 9796
LV 9796
S
2-SSP-66
FO
A/S
QR 9796
ZS 9796
2SSP66*
2-SSP-67
2½"-HSC-115
2½"-HSC-114
4"-HSC-35
EQUIP. DRAIN SUMP TANK 26020 SH.5 (K-2)
4"-HSC-113

VENT 2-SSP-36A ¾"
2½" SPARES (CAPPED) EAST PIPE & VALVE ROOM
2½" FROM 2-MS-140 26002 SH.2 (C-4)

2½" SPARE (CAPPED) (DEGASIFIER PUMP RM.)
2½" DEGASIFIER PUMPS P-16A & P-16B 26020 SH.4 (A-6)(C-6)
SI-409, 417, 439 LEAK-OFFS 26015 SH.1 (B-8) 26015 SH.2 (J-9)(C-10)
DRAIN FROM EFFLUENT COOLER 26020 SH.4 (C-4)
2½" DEGASIFIER DRAINS & DEGASIFIER PREHTR 26020 SH.4 (D-8)
SAMPLE RM 26025 SH.1 (F-13)
DEGASIFIER FILTER L-12 26020 SH.1 (A-12)(B-10)
2½"-HSC-113

PRIMARY & SECONDARY DEMIN. DRAINS & VENT 26020 SH.1 (E-7,-10)(F-5)(H-2)
PRIMARY & SECONDARY DEMIN. DRAINS & VENT 26020 SH.1 (E-1)
INSTR. VALVE LRR-24A & LRR-24B LRR-11A, 11B & 11C LEAK-OFF 26020 SH.1 (E-2,-4,-7,-9,-11) 2½"
2½"-HSC-118
SPARE (CAPPED) (COOLANT WASTE TANK RM.) 2½"
COOLANT WASTE TANK PUMPS P-29 & P-30 26020 SH.2 (B-7) 2½"
COOLANT WASTE FINAL FILTER L-15 26020 SH.2 (B-5)(D-5)

AERATED WASTE EVAPORATOR RETIRED IN PLACE 2½"
SPARE 2½"
SPARE 2½"
SPARE 2½"
AERATED WASTE EVAPORATOR ROOM 4"
FROM 2-CH-998 VCT GAS SAMPLE TUBING DRAIN 26025 SH.4 (E-6) 2"
CORRIDOR EL.(-)5'-0" 4"
DRAIN HEADER 26021 SH.1 (C-6) 2½"
SPENT FUEL POOL FILTERS 26023 SH.2 (B-6,C-5) 2½"
VALVE 2-RW-188 STEM LEAKOFF 26023 SH.2 (D-4) 1"-MSD-2
REFUELING WATER PURIFICATION PUMPS 26023 SH.2 (C-4) 1"-MSD-2
VALVE 2-RW-40A STEM LEAKOFF 26023 SH.2 (A-8)
VALVE 2-RW-40B STEM LEAKOFF 26023 SH.2 (A-6)
SPENT FUEL POOL FILTER COMPARTMENT 4"
REFUELING WATER PURIFICATION PUMP 26023 SH.2 (C-10) 2½" 4"-MSD-2
REFUELING WATER PURIFICATION PUMP 26023 SH.2 (B-10) 2½"-MSD-1
SPENT FUEL POOL LEAK CHASE 26023 SH.2 (F-10) 2"-MSD-1
26023 SH.2 (F-4) 1"-MSD-2
SPENT FUEL POOL DEMINERALIZER COMPARTMENT 4"-MSD-1
AERATED WASTE DEMINERALIZER COMPARTMENT 4"-MSD-1
SPENT FUEL POOL SKIMMER PUMP 26023 SH.2 (E-12) 2½"-MSD-3
SPENT FUEL POOL SKIMMER FILTERS 26023 SH.2 (E-7,G-8) 2½"-MSD-3
4"-MSD-1

RETIRED PER PDCR 2-95-054
RECOVERED BORIC ACID STORAGE TK 2½"-MSD-5
RECOVERED BORIC ACID STORAGE TK 2½"-MSD-5
BORIC ACID EVAPORATOR 26020 SH.3 (A-8) 2-BAE-2
BORIC ACID EVAPORATOR 2½"-MSD-5
BORIC ACID EVAPORATOR ROOM 4"
LOCAL SAMPLE DRAIN 2"
CORRIDOR EL.(-)5'-0" 4"
LOCAL SAMPLE DRAIN 2"
CORRIDOR EL.(-)5'-0" 4"
CORRIDOR EL.(-)5'-0" 4"
LOCAL SAMPLE DRAIN 2"
4"-MSD-5
4"-MSD-6
4"-MSD-6
4"-MSD-6

RETIRED PER PDCR 2-95-054
BORIC ACID EVAPORATOR
BORIC ACID EVAPORATOR ROOM 4"
PRIMARY WTR TRANS PP'S & HEAD TK 26030 SH.1 (E-10,E-5) 2½"
FUTURE PRIMARY DEMINERALIZER COMPARTMENT 4"
PRIMARY DEMINERALIZER COMPARTMENT 4"
PRIMARY & FUTURE SECONDARY DEMIN COMPARTMENT 4"
SECONDARY DEMINERALIZER COMPARTMENT 4"
4"-MSD-3

RELIEF FROM SHUTDOWN HEAT EXCHANGER TO H.P.S.I. SUCTION 26015 SH.1 (G-10) 2½"
H.P. SAFETY P-41A INJECTION PUMP 26015 SH.2 (H-4,-5) 2½"
L.P. SAFETY P-42A INJECTION PUMP 26015 SH.1 (E-5) 2½"
LOW POINT PIPE DRAIN SPARE (CAPPED) 2½"
SHUTDOWN HEAT EXCHANGER X-23A 26015 SH.1 (J-7) 2½"
CONTAINMENT SPRAY PUMP P-43A 26015 SH.1 (H-1) 2½"
2½"-HSC-112

SPARE (CAPPED) 2½"
SHUTDOWN HEAT EXCHANGER X-23B 26015 SH.1 (G-8) 2½"
H.P. SAFETY INJECTION PUMP P-41C 26015 SH.2 (E-4,-5) 2½"
SPARE (CAPPED) 2½"
L.P. SAFETY P-42B INJECTION PUMP 26015 SH.1 (B-5) 2½"
CONTAINMENT SPRAY PUMP P-43B 26015 SH.1 (G-1) 2½"
2½"-HSC-120

2½"-HSC-112
SPARE (CAPPED) 2½"
H.P. SAFETY INJECTION PUMP P-41B 26015 SH.2 (F-4,-5) 2½"
2½"-HSC-121
2½"-HSC-122
SPARE (CAPPED) 2½"
GASEOUS WASTE EQUIPMENT DRAIN 26021 SH.2 (D-7) 2½"
4"-HSC-112
EQUIPMENT DRAIN SUMP TANK 26020 SH.5 (H-4)
4"-HSC-56 SPENT RESIN SHIPPING CASK DEWATERING PUMP & SPENT RESIN TANK OVERFLOW 26021 SH.3 (C-13)

ROOF RAD WASTE BLDG. 4"X3"
3" HVAC DRAIN
CARBON STL JH 3"
HOOD SINK 1½"
LAB SINK 1½" 1½"
LAB SINK 2"
FLOOR DRAIN JH
LAB SINK 1½"
3"-JH-60
HVAC DRAIN 3" 1½"
HVAC DRAIN 1½"
EYE WASH STATION 1½"
LAB SINK 1½"
LAB SINK 2" 1½"
LAB SINK 2" 1½"
CHEM. LAB EL. 14'-6"
JH JGD
3"-JGD-35
SERVICE WATER PIPE TUNNEL
JGD JH
3"-JH-61

BORIC ACID TANK AREA
BORIC ACID TANK OVERFLOW & DRAIN 26017 SH.3 (D-1) 4"
BORIC ACID PP. DISCH. TO VOLUME CONTROL TK. RELIEF LINE DISCH. 26017 SH.3 (G-9) 2"
BORIC ACID TRANSFER PUMPS 26017 SH.3 (A-5,A-7,C-5) 2½"
4"
2" LOOP FILL
2" VENT
LOOP SEAL EL. 14'-6" AUX. BLDG FL.
ELECTRICAL PENETRATION ROOM (WEST SIDE)
BORONOMETER ROOM
SAMPLE ROOM 26025 SH.1 (A-12) 2½"
SAMPLE ROOM 26022 SH.5 (J-12) 4"
4"-MSD-11

CORRIDOR EL.(-)5'-0" 4"
CORRIDOR EL.(-)5'-0" 4"
4"-MSD-15
RBCCW PUMP 26022 SH.1 (C-8) 2½"
SPARE 2½"
SPARE 2½"
SPARE 2½"
2½"-MSD-14
RBCCW PUMP 26022 SH.1 (E-8) 2½"
RBCCW PUMP 26022 SH.1 (C-8) 2½"
LOCAL SAMPLE DRAIN 2"
LOCAL SAMPLE DRAIN 2"

AREA DRAIN 4"
WASTE GAS SURGE TANK ROOM 4"
LOCAL SAMPLE DRAIN 2"
WASTE GAS COMPRESSOR ROOM 4"
LOCAL SAMPLE DRAIN 2"
WASTE GAS FILTER COMPARTMENT 4"
DEHUMIDIFIER 4"
WASTE GAS DECAY TANK ROOM 4"
LOCAL SAMPLE DRAIN 2"
PIPEWAY EL.(-)25'-6" 4"
LOCAL SAMPLE DRAIN 2"
WASTE GAS COMPRESSOR ROOM 4"
PIPEWAY EL.(-)25'-6" 4"
PIPEWAY EL.(-)25'-6" 4"
4"-MSD-4

CORRIDOR EL.14'-6" 4"
CHEMICAL STORAGE 4"
CHEMICAL ADDITION TANK & PUMPS AREA 4"
BORIC ACID STRAINER DRAIN & BORIC ACID BATCH TANK OVERFLOW 26017 SH.3 (J-3) 2½"
BORIC ACID BATCH TANK AREA 4"
CORRIDOR EL.14'-6" 4"
RCP SEAL ROOM 2½"-MSD-50
DRUMMING AREA (PLUGGED) 4"
SEE NOTE 4
1"-JBD-84 4"-MSD-9
4"-MSD-13
4"-MSD-12
4"-MSD-11

ELECTRICAL PENETRATION ROOM (EAST SIDE) 14'-6"
2" LOOP FILL
2" VENT
LOOP SEAL (AUX. BLDG. EL. 14'-6")
CORRIDOR EL.14'-6" 4"
SKIMMER PUMP ROOM 4"
4"-MSD-3
4"-MSD-3

PIPE & VALVE ROOM (EAST SIDE)
2" LOOP FILL
2" VENT
LOOP SEAL (-)5'-6"
4" PIPEWAY EL.(-)25'-6" PLUGGED
4" PIPEWAY EL.(-)25'-6"
4"-MSD-7
2½"-MSD-7
LOCAL SAMPLE DRAIN 2"
EVAPORATOR STEAM CONDENSATE PUMP 2"

1½" 2"-MSD-7
LAHL 9748
2-SSP-7
SPEC. CHG.
2-SSP-6
PIPE & VALVE ROOM (EAST SIDE)
FROM 2-SI-808 & 2-SI-809
4"
¾" O.D. SST
2-SSP-5
PI 9750
1½"-HCD-136
LC 9748
HS 9749
LE 9748
2"
4"x3" RED.
3"
P-126
JCD JBD
2-SSP-185
⅜"
2"-JBD-23
OFF-GAS LINE DRAIN 26028 SH.5 (B-7)
OFF-GAS PIPEWAY SUMP PUMP
JDD JCD

MAIN STEAM PIPE TUNNEL EL. 36'-6" (TURB. BLDG.) PLUGGED 4"
WEST PENET. RM. EL.38'-6" 4"
2" FILL
2" VENT
LOOP SEAL CABLE VAULT 25'-6"
CONTROL RM. HVAC EQUIP. EL.36'-6" 4"
AUX. BLDG.VENT. EQUIP. EL.38'-6" 4"
LAYDOWN AREA EL.38'-6" 4"
SPENT FUEL POOL AREA EL.38'-6" 4"
SOUTH TRACK EL.38'-6" 2"
SPENT FUEL POOL AREA EL.38'-6" 4"
4"-MSD-8

4"-MSD-8
1" 2-SSP-192
L-510B S
2-SSP-23B
2-SSP-22B
2-SSP-78 1"
CHG. SPEC.
1"-MSD-42
TO 2-LRA-1A AWDT(T-20A) 26021 SH.1 (G-13)
4"-HSC-36
L-510A S
2-SSP-23A
2-SSP-22A
2-SSP-77 1"
CHG. SPEC.
1"-MSD-42
2-SSP-61B ¾"
2-SSP-61A ¾"
¾" O.D. SST
PDIS 9268
¾" O.D. SST
PAH 9268
UAHL 9381

4"-HSC-108 CLEAN LIQUID RADWASTE SYSTEM 26020 SH.5 (J-1)
SPEC. CHG.
2"-HSC-127
2"-HBD-61 CONDENSATE RECOVERY PUMPS 26026 SH.2 (E-3)
2-SSP-53
3" SPEC. CHG.
2-SSP-12
4"-MSD-9

2" LOOP FILL
4"-MSD-9
2" VENT
PIPE & VALVE ROOM (WEST SIDE)
LOOP SEAL (-)5'-6" IN FLOOR
SPARE (SEALED) 2"
CORRIDOR EL.(-)25'-6" 4"
AERATED WASTE PRE-FILTER, PRE-FILTER COMPARTMENT 26021 SH.1 (H-9,G-9) 4"
LOCAL SAMPLE DRAIN 2"
EAST VENT ROOM EL.38'-6" (PLUGGED) 4"
EAST VENT EQUIP. ROOM EL.38'-6" 4"
2" VENT 2" FILL
LOOP SEAL 14'-6" OVERHEAD
EAST VENT EQUIP. ROOM EL.38'-6" 4"
FROM 2-RB-315 26022 SH.2 (J-7) 2"-MSD-8
2" VENT 2" FILL
NOZZLE DAM WORK AREA EL. 38'-6"
LOOP SEAL 14'-6" OVERHEAD
NORTH TRACK DRAIN 2" FILL
2" VENT
EAST MAIN STM. PENET. ROOM 4"
EAST MAIN STM. PENET. ROOM 2½"
EAST MAIN STM. PENET. ROOM 4"
LOOP SEAL 14'-6" OVERHEAD 4"-MSD-8
NEW FUEL STORAGE AREA 4"-MSD-8
CASK WASH PIT
EL.14'-6"
DRAIN FOR COIL X-191 2"
4"-MSD-13
4"-MSD-12

PRE DEGASIFIER FILTER COMPARTMENT 4"-MSD-9
CORRIDOR EL.(-)25'-6" 4"-MSD-9
SPENT RESIN TANK COMPARTMENT 4"-MSD-9
2-SSP-17
LOCAL SAMPLE DRAIN 2"
DEGASIFIER PUMP ROOM 4"-MSD-10
LOCAL SAMPLE DRAIN 2"
DEGASIFIER ROOM 4"
CHARGING PUMP ROOM 4"-MSD-10
LOCAL SAMPLE DRAIN 2"-MSD-10
CHARGING PUMP ROOMS 4"-MSD-10
4"-MSD-10
4"-MSD-9

2" LOCAL SAMPLE DRAIN
4" PURIFICATION ION EXCHANGER COMPARTMENT
4" VOLUME CONTROL TANK ROOM
2" LOCAL SAMPLE DRAIN
4" DEBORATING ION EXCHANGER COMPARTMENT
4"-MSD-9
4" LETDOWN HEAT EXCHANGER ROOM
4" LETDOWN STRAINER, LETDOWN & PURIFICATION FILTER COMPARTMENT

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. FOR DRAIN SYMBOLS SEE 26024 SH. 1
3. RAD WASTE LINES DESIGNATED AS HSC-35 AS SHOWN ON THIS DRAWING ARE NOT PART OF THE ISI PROGRAM. RAD WASTE LINES DESIGNATED AS HSC-35 WHICH ARE ASME SECTION III CODE CLASS 3 LINES AS SHOWN ON THIS DRAWING ARE DOWNGRADED TO ANSI B31.1 PER REG GUIDE 1.143 REV. 1 DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES, AND COMPONENTS INSTALLED IN LIGHT WATER NUCLEAR POWER PLANTS. PIPE CLASSES FOR THE LINE NUMBERS WILL REMAIN THE SAME FOR HISTORICAL PURPOSES. ALL NEW PIPING SHALL BE INSTALLED TO THE APPLICABLE ANSI B31.1 PIPE CLASS, WITH A (R) FORTH LETTER CODE FOR RAD WASTE QA. PER DCN DM2-00-0216-03.
4. ALPS DRAIN HEADER INSTALLED ON FLOOR DRAIN. SEE DC MP2-15-01045 FOR ADDITIONAL DETAILS.

FSAR FIGURE

OPERATIONS CRITICAL

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING & INSTRUMENTATION DIAGRAM
AUXILIARY BUILDING DRAINS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260243.DGN	
REVISED PER MSDUR 21-000058 & DUR 21-800193	DRAWING NO: 25203-26024	REV 39
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

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17-AUG-2021 18:25
C260243.DGN

SAMPLE COOLER

SAMPLE SINK

SAMPLE COOLER

MULTI CIRC SAMPLE COOLER

FSAR FIGURE

OPERATIONS CRITICAL

AUXILIARY BUILDING
SAMPLING ROOM
CENTRALIZED SAMPLING STATION NO.1

NOTES:

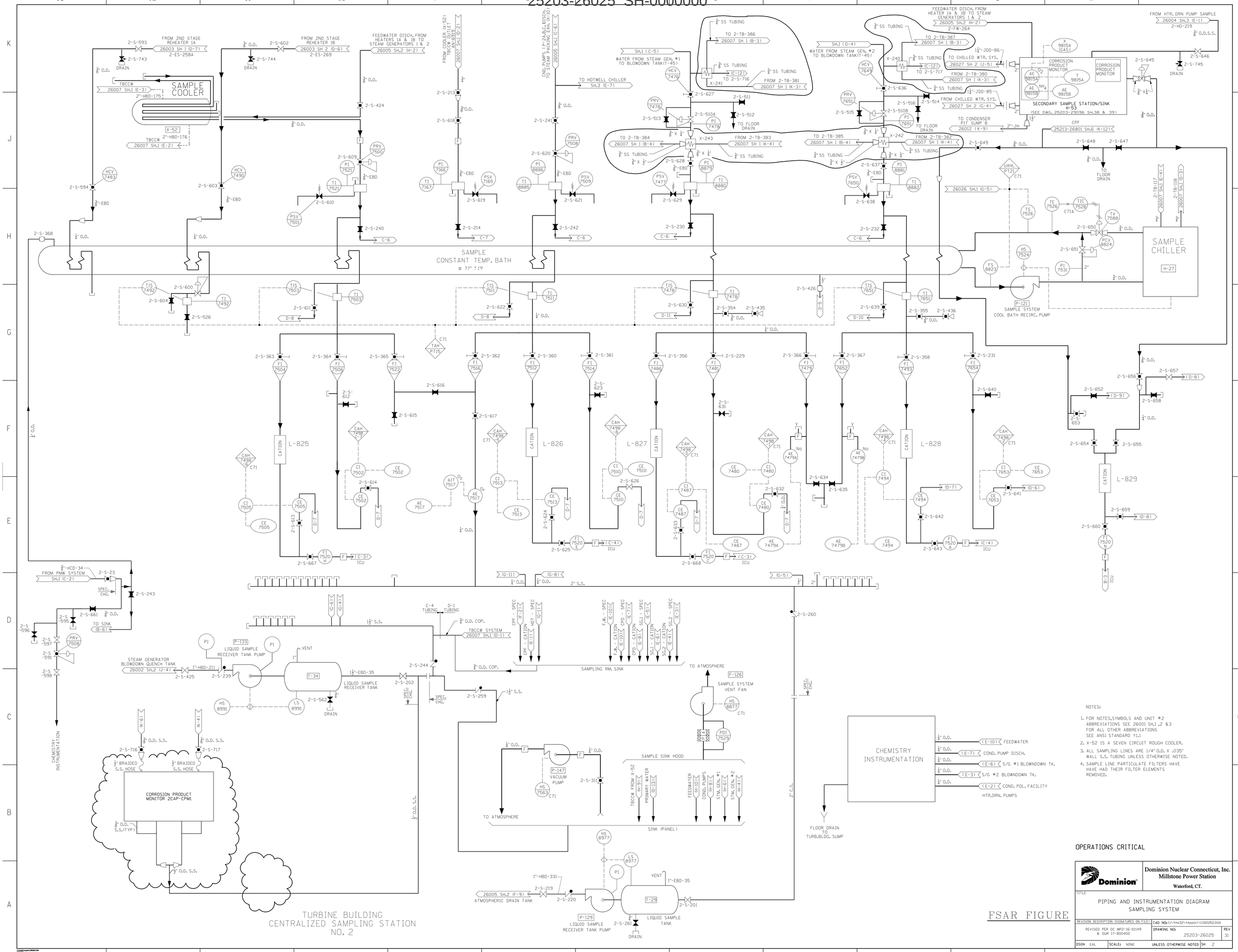
1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. FOR MAIN WATER SUPPLY TO THE SYSTEM SEE 25203-26007.
3. ALL SAMPLING LINES ARE 3/8" O.D. X .065" WALL S.S. TUBING UNLESS OTHERWISE NOTED.
4. SAMPLE LINE PARTICULATE FILTERS HAVE HAD THEIR FILTER ELEMENTS REMOVED.

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION - UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
SAMPLING SYSTEM
WATERFORD, CT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260251.DGN	
REVISED PER DC MP2-16-01149 AND DUR 17-800406	DRAWING NO: 25203-26025	REV 38
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1



NOTES:

1. FOR NOTES,SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1 ,2 &3 FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1
2. X-52 IS A SEVEN CIRCUIT ROUGH COOLER.
3. ALL SAMPLING LINES ARE 1/4" O.D. X .035" WALL S.S. TUBING UNLESS OTHERWISE NOTED.
4. SAMPLE LINE PARTICULATE FILTERS HAVE HAVE HAD THEIR FILTER ELEMENTS REMOVED.

OPERATIONS CRITICAL

FSAR FIGURE

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING AND INSTRUMENTATION DIAGRAM
SAMPLING SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260252.DGN	
REVISED PER DC MP2-16-01149 & DUR 17-800400	DRAWING NO: 25203-26025	REV 31
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

TO CONDENSER 'B' PENETRATION #76 26005 SH.1 (E-91)
FROM AN-5241 ("C" HOTWELL) 26005 SH.1 (F-71)
TO CONDENSER 'B' PENETRATION #76 26005 SH.1 (E-71)
FROM AN-5242 ("D" HOTWELL) 26005 SH.1 (E-61)
TO CONDENSER 'A' PENETRATION #76 26005 SH.1 (D-131)
FROM AN-5136 ("A" HOTWELL) 26005 SH.1 (E-111)
TO CONDENSER 'A' PENETRATION #76 26005 SH.1 (E-101)
FROM AN-5137 ("B" HOTWELL) 26005 SH.1 (E-101)

CONDENSER SAMPLE PUMPS (4)

HOTWELL CHILLER H-46

COOLFLOW CFT-75 REFRIGERATED RECIRCULATOR

TURBINE BUILDING SUMP

TURBINE BUILDING WEST SIDE
CORROSION MONITORING SAMPLE STATION

NOTES

1. FOR NOTES,SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3 FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1
2. FOR COOLING WATER SUPPLY TO SYSTEM SEE P&ID 25203-26007.
3. ALL SAMPLING LINES ARE $\frac{3}{8}$ O.D.X.065 S.S. TUBING UNLESS OTHERWISE NOTED.

OPERATIONS CRITICAL

REDRAWN INPART FROM 25203-26025 SH.2 THRU REV. 13.

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENT DIAGRAM
SAMPLING SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260253.DGN	
REVISED PER WSDUR 21-000062 AND DUR 21-800161	DRAWING NO: 25203-26025	REV 24
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 3

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17-AUG-2021 10:59
C260253.DGN

25203-26026 SH-0000000

TURBINE BLDG. EL. 31'-6"

AUX. BLDG. EL. 38'-6"

AUX. BLDG. EL. (-)25'-6"

TURBINE BLDG. EL. 14'-6"

RETIRED IN PLACE PER DCR M3-99042

DEAERATOR

REBOILER DEAERATOR FEEDWATER TANK T-63

CONDENSATE RECOVERY TANK T-76

REBOILER BLOWDOWN TANK T-64

AUXILIARY STEAM FEEDWATER SURGE TANK T-62

CHEMICAL ADDITION TANK T-44

AUXILIARY STEAM REBOILER X-50

X-192

P-182

P-52A

P-52B

REBOILER FEEDWATER PUMPS

P-84A

P-84B

CONDENSATE RECOVERY PUMPS

P-91A

P-91B

AUXILIARY STEAM DEAERATOR FEED PUMPS

X-211 LOCAL SAMPLE

FAN COIL UNIT X-173 F-151

X-178

X-179

X-180

SAMPLE COOLERS

NOTE STRAINERS L-436 THRU 441 DRAIN CAPPED FOR INSTALLATION OF CHICAGO FITTINGG AND HOSE WHEN DRAINING IS REQUIRED.

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

OPERATIONS CRITICAL

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
AUXILIARY STEAM AND CONDENSATE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260261.DGN	
REVISED PER MSDUR 21-000057 & DUR 21-800152	DRAWING NO: 25203-26026	REV 57
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

MSULOGIN\er ik004
06-AUG-2021 06:40
C260261.DGN

25203-26026 SH-0000000

AUXILIARY BUILDING

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. VALVE #2-AS-71 HAS HAD INTERNALS REMOVED.

TURBINE BUILDING

OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
AUXILIARY STEAM AND CONDENSATE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260262.DGN	
REVISED PER EC MP2-20-01129 & DUR 20-800270	DRAWING NO: 25203-26026	REV 26
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 2

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06-NOV-2020 08:03
C260262.DGN

FUEL HANDLING AREA HEATING COIL

X-41

VB

2-AS-900

2-AS-850

PI 9923

2-AS-851

2"-HBD-489

L-732 AIR VENT

2-AS-852

4"-HBD-490

2-AS-853

2"-HBD-491

2-AS-857

2-AS-858

L-733

2-AS-854

LC 8232

2-AS-855

CONDENSATE LIFT PUMP P-184

ST-226

2-BW-696

2"-HBD-492

DRN.

2-BW-698

26029 SH. 3 (H-9) FROM TE-8232

TCV 8232

2-AS-733

2-AS-732

2-AS-731

2-AS-734

4"-HBD-153

L-136

2-AS-780

2-AS-735

2-BW-670

2-AS-848

2-AS-856

2-AS-847

L-731

ST-225

2-BW-697

TRUCK ACCESS AREA

X-153

L-554

ST-43

2-BW-683

2-BW-646

2-AS-730

X-152

L-553

ST-42

2-AS-728

2-AS-741

OUTSIDE DIESEL BAY TANK RM.

"A" DIESEL ROOM

L-715

2-BW-641

2-AS-726

ST-192

2-AS-727

3"-HBD-144

2-HBD-144

SOUTH ENCLOSURE BLDG. (B-7)

ENCLOSURE BUILDING VENTILATION ROOM

L-560

ST-193

2-AS-736

2-BW-649

3" X 2"

2-BW-651

ST-51

L-562

2-AS-770

2-AS-769

2-BW-648

ST-49

2-AS-742

L-560

X-162

X-159

ABOVE TEMP. MAINT. WORK AREA

ABOVE NOZZLE DAM ROOM

2-BW-644

2-BW-645

ST-44

L-555

X-154

2-AS-729

2-AS-768

4"-HBD-153

FROM TE-8235 -26028 SH.1 (B-2)

VB

2-AS-872

PI 9924

TCV 8235

X-43 (TOP)

X-43 (BOTTOM)

2-AS-875 AIR VENT

2-AS-874 AIR VENT

2-AS-869

2-AS-864

2-AS-867

2-AS-865

2-AS-880

2-AS-881

2-AS-870

2-AS-873

2-AS-724

2-AS-723

2-AS-722

2-AS-764

2-AS-725

2-AS-721

2-BW-695

ST-227

2-BW-694

ST-228

L-734

L-735

L-419

6"-HBD-153

ST-221

2-BW-684

L-718

2-AS-832

2-AS-765

2-BW-624

ST-45

L-556

X-155

2-AS-720

1"-HBD-153

3"-HBD-144

AUXILIARY BUILDING

BY CL OTHERS HBD

RETIRED IN PLACE PER DC MPG-17-00116

26026 SH. 5 (D-13)

1½"-HBD-144

2-BW-716

2-BW-715

HP OFFICE

2-AS-899

2-AS-898

3"-HBD-153

X-140

X-139

"B" DIESEL GEN. ROOM

L-545

ST-34

L-544

ST-33

2-AS-712

2-BW-619

2-AS-713

2-BW-621

2-BW-620

1½"-HBD-153

L-716

ST-190

2-AS-740

2-BW-685

2-BW-623

ST-191

L-418

2-AS-718

2-AS-763

6"-HBD-153

YARD (TRENCH)

TANKAGE AREA

6"x4"

4"-HBD-153

2½"-HBD-153

2-AS-702

2-AS-24

2-AS-833

2-IA-949B

FROM TIC 3034 26015 SH. 2 (C-9)

PCV 3034

A/S 26009 SH. 6 (G-4)

2-IA-949A

TO RWST (T-41) -26015 SH.2 (C-5)

3"-HCD-5

2-AS-839

2-AS-76 FO

TV 3034

L-67

2-AS-25

L-68

½"-HBD-153

FROM TIC 7308 26030 SH. 1 (H-6)

2-AS-75 FO

TV 7308

TO PWST (T-42) -26030 SH.1 (J-1)

3"-HCD-65

2-AS-703

2½"

2-AS-719

6"

X-48 REFUEL WTR. STOR. TK. HEAT EXCH.

X-47 PRIMARY WATERSTOR. TK.HEAT EXCH.

2-AS-302

FROM RWST (T-41) -26015 SH.2 (B-5)

3"-HCD-64

FROM PWST (T-42) -26030 SH.1 (J-1)

RECEIVER

PI 9932

PI 9931

L-793

2-AS-300

L-794

2-AS-301

2-AS-701

L-420

2-AS-772

ST-194

2-BW-627

2-BW-628

ST 160A

ST 160B

2-AS-304

2-AS-305

2-AS-303

ST-246

ST-245

2-BW-92

2-BW-93

2-BW-94

2-BW-95

½"-HBD-166

ST 195

ST-247

2-BW-96

2-BW-141

2-BW-98

2-BW-97

ST-248

2-BW-140

2-BW-142

2-BW-100

2-BW-99

1½"-HBD-166

2-BW-435

YARD (TRENCH)

DRAIN

4"-HBD-144

SH.1 (J-3)

L-417

2-BW-626

ST-189

2-BW-611

2-AS-669

2-AS-668

FROM ST-224 SH.2 (J-9)

2-BW-686

SH.2 (K-7)

SH.2 (J-7)

10"-HBD-151

8"-HBD-153

10"x8"

2-AS-656

2-AS-655

2-BW-608

ST-188

L-416

2-AS-657

2-AS-762

L-432

2-AS-658

2-AS-659

X-56

2-BW-609

ST-59

L-570

2-AS-661

FROM TE-8233 26029 SH. 2 (H-10)

2-BW-610

ST-61

L-572

2-AS-667

2-AS-662

2-BW-653

2-BW-654

2" X 1½"

2-HBD-144

ENCLOSURE BLDG. VENTILATION ROOM (J-5)

2-BW-652

2-AS-117

2-AS-677

ST-39

L-550

X-146

1½"x1¼"

X-131

L-538

2-AS-838

2-AS-678

ST-27

X-135

L-542

2-AS-778

2-AS-679

ST-31

X-143

L-548

2-AS-777

2-AS-680

ST-37

X-141

L-546

2-AS-841

2-AS-682

ST-35

2-BW-671

2-AS-776

X-160

L-561

2-AS-773

ST-50

X-134

L-541

2-AS-774

ST-30

X-145

L-549

2-AS-775

ST-38

2-BW-643

2-BW-642

1"-HBD-144

2-AS-681

2-AS-683

L-422

ST-197

L-421

ST-196

2-BW-647

2-BW-570

2-BW-586

2-BW-692

2-BW-594

2-BW-655

2-BW-589

4"x1½"

TO CONDENSATE RECOVERY TANK T-76 4" SH.1 (J-5)

FUEL POOL AREA

3"

ENCLOSURE SOUTH BUILDING

FROM TE-8234 26029 SH. 3 (G-8)

2-BW-670

2-BW-671

2-AS-671

2-AS-672

2-AS-673

X-57

TCV 8234

L-431

2-AS-760

2-AS-670

ST-60

2-AS-674

L-571

2-AS-666

2-AS-664

2-AS-665

X-58

TCV 8237

L-717

ST-61

2-AS-667

L-572

FROM TE-8237 26029 SH. 3 (G-12)

DUCT BOOSTER COILS (3)

X-130

X-126

X-133

X-128

X-136

X-129

X-132

X-127

1½"-HBD-153

2-AS-699

2-AS-698

2-AS-697

2-AS-696

2-AS-694

2-AS-692

2-AS-693

2-AS-689

L-537

L-533

L-540

L-535

L-543

L-536

L-539

L-534

ST-26

ST-22

ST-29

ST-24

ST-32

ST-25

ST-28

ST-23

2-AS-700

2-AS-771

2-BW-637

2-BW-636

2-BW-635

2-BW-634

2-BW-632

2-BW-631

2-BW-629

L-426

ST-201

2-BW-638

2-AS-695

L-425

ST-200

2-AS-690

L-424

ST-199

2-AS-691

2-AS-687

L-423

2-AS-767

ST-198

2-AS-688

2-BW-615

2-BW-633

2-BW-639

2-AS-766

2-AS-829 (V)

2-AS-830

2"-HBD-153

2½"-HBD-153

3"-HBD-153

1"-HBD-144

2½"-HBD-144

4"-HBD-153

ENCLOSURE NORTH BUILDING

2-BW-351

2-BW-640

2-BW-668

2-BW-350

ST-202

2-AS-761

L-427

2-AS-675

2-AS-676

4"x2½"

4"x3"

6"

2-AS-200

2-AS-834

8"x6"

6"-HBD-153

REMOVABLE SPOOL

AUX. BLDG. E. PENNELL ROOM

2-AS-201

DRAIN

6"x4"

2-AS-117

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

REDRAWN IN PART FROM DRAWING 25203-26026 SH. 1 THRU REV. 32.

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
AUX. STEAM AND CONDENSATE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260263.DGN	
REVISED PER DC MPG-17-00116 & DUR 19-800331	DRAWING NO: 25203-26026	REV 28
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

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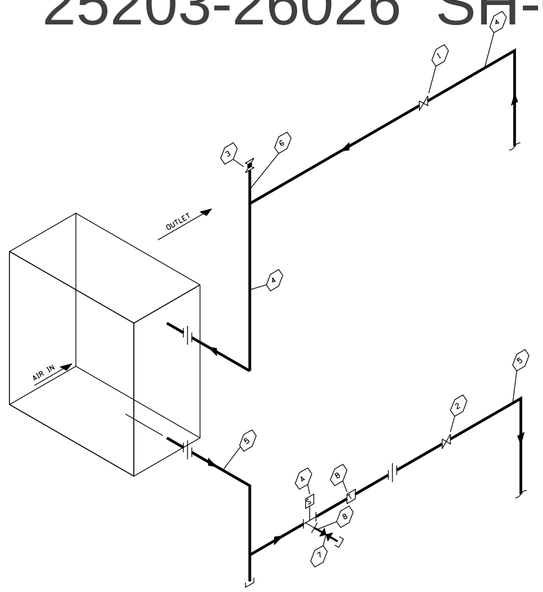
20-DEC-2019 16:56
C260263.DGN

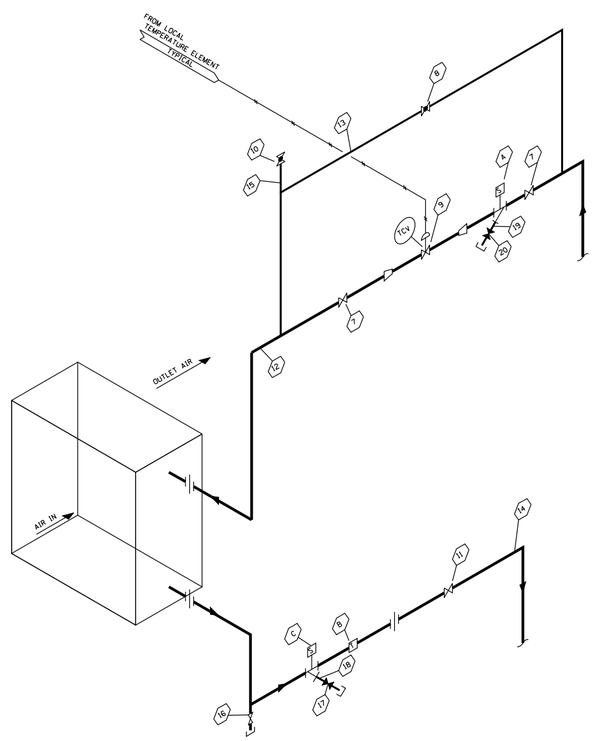
UNIT HEATER ARRANGEMENT INDEX

P&ID LOCATION	UNIT HEATER Nº	LOCATION COL.	LOCATION OUTLET ELEV.	VALVE SIZE & MARK Nº 1	2	3	7	STRAINER A	TRAP B	LINE SIZE & CLASS 4	5	6	8	CAPACITY BTU/HR
SH.2 (C-8,9)	X-101	A-24	29'-0"	¾"-19	¾"-19	¾"-130	½"-19	L-82	ST-2	¾"-HBD	¾"-HBD	¾"-HBD	½"-HBD	51,000
SH.2 (C-6)	X-102	A-21	↓	↓	↓	↓	——	L-83	ST-3	↓	↓	↓	——	↓
SH.2 (C-2)	X-103	A-16	↓	↓	↓	↓	——	L-84	ST-4	↓	↓	↓	——	↓
——	X-104	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.2 (C-4)	X-105	A-18	29'-0"	¾"-19	¾"-19	¾"-130	——	L-85	ST-5	¾"-HBD	¾"-HBD	¾"-HBD	——	51,000
SH.2 (C-12)	X-106	D-26	↓	↓	↓	↓	½"-19	L-86	ST-6	↓	↓	↓	½"-HBD	↓
SH.2 (D-8)	X-107	A-24	46'-0"	↓	↓	↓	½"-19	L-87	ST-7	↓	↓	↓	½"-HBD	↓
SH.2 (D-6)	X-108	A-21	↓	↓	↓	↓	½"-19	L-88	ST-8	↓	↓	↓	½"-HBD	↓
SH.2 (D-2)	X-109	A-16	↓	↓	↓	↓	——	L-89	ST-9	↓	↓	↓	——	↓
SH.2 (D-4)	X-110	A-18	↓	↓	↓	↓	——	L-90	ST-10	↓	↓	↓	——	↓
SH.2 (D-12)	X-111	D-26	↓	↓	↓	↓	——	L-91	ST-11	↓	↓	↓	——	↓
SH.2 (D-10)	X-112	B-26	↓	↓	↓	↓	½"-19	L-92	ST-12	↓	↓	↓	½"-HBD	↓
SH.2 (D-10)	X-113	B-26	69'-0"	1½"-19	↓	↓	——	L-93	ST-13	1¼"-HBD	↓	↓	——	108,000
SH.2 (D-6)	X-114	A-21	↓	↓	↓	↓	——	L-94	ST-14	↓	↓	↓	——	↓
SH.2 (D-2)	X-115	A-16	↓	↓	↓	↓	——	L-95	ST-15	↓	↓	↓	——	↓
SH.2 (H-2)	X-116	E.5-17	↓	¾"-19	↓	↓	——	L-96	ST-16	¾"-HBD	↓	↓	——	24,600
SH.2 (H-1)	X-117	E.5-23	66'-0"	↓	↓	↓	——	L-97	ST-17	↓	↓	↓	——	51,000
SH.2 (D-12)	X-118	D-26	↓	1½"-19	↓	↓	——	L-98	ST-18	1¼"-HBD	↓	↓	——	108,000
——	X-119	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.2 (F,G-12)	X-120	F.8-25.9	29'-0"	¾"-19	¾"-19	¾"-130	——	L-99	ST-19	¾"-HBD	¾"-HBD	¾"-HBD	——	50,500
SH.2 (G-12)	X-121	F.8-25.9	69'-0"	↓	↓	↓	——	L-100	ST-20	↓	↓	↓	——	↓
SH.2 (C-10)	X-122	B-26	29'-0"	↓	↓	↓	½"-19	L-532	ST-21	↓	↓	↓	½"-HBD	51,000
——	X-123	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-124	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-125	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.3 (G-12)	X-126	E.5-24	29'-0"	1½"-19	¾"-19	——	——	L-533	ST-22	1¼"-HBD	¾"-HBD	——	——	108,000
SH.3 (G-6)	X-127	M.4-23	↓	↓	↓	——	——	L-534	ST-23	↓	↓	——	——	↓
SH.3 (G-10)	X-128	H.2-24.9	↓	↓	↓	——	——	L-535	ST-24	↓	↓	——	——	↓
SH.3 (G-8)	X-129	L.5-24.9	↓	↓	↓	——	——	L-536	ST-25	↓	↓	¾"-HBD	——	↓
SH.3 (G-13)	X-130	E.5-23	↓	↓	1"-19	——	——	L-537	ST-26	↓	1"-HBD	——	——	211,000
SH.3 (D-7,8)	X-131	M.4-20	72'-0"	↓	↓	——	——	L-538	ST-27	↓	↓	——	——	↓
SH.3 (G-7)	X-132	M.4-24	↓	2½"-29	↓	¾"-130	——	L-539	ST-28	2½"-HBD	↓	¾"-HBD	——	398,000
SH.3 (G-11)	X-133	F.2-24.9	29'-0"	1½"-19	↓	——	——	L-540	ST-29	1½"-HBD	↓	¾"-HBD	——	211,000
SH.3 (D-12)	X-134	E.5-18.9	72'-0"	↓	↓	——	——	L-541	ST-30	↓	↓	——	——	↓
SH.3 (D-8)	X-135	L.5-19	↓	↓	¾"-19	——	——	L-542	ST-31	↓	¾"-HBD	——	——	108,000
SH.3 (G-9)	X-136	K.6-24.9	↓	2½"-29	1"-19	——	——	L-543	ST-32	2½"-HBD	1"-HBD	——	——	398,000
SH.3 (H-2,3)	X-139	R-18.9	45'-0"	1½"-19	1"-19	——	——	L-544	ST-33	1½"-HBD	1"-HBD	——	——	208,000
SH.3 (H-3)	X-140	N.3-18.9	↓	↓	¾"-19	——	——	L-545	ST-34	↓	¾"-HBD	——	——	98,000
SH.3 (D-10)	X-141	F.8-18.9	72'-0"	↓	1"-19	——	——	L-546	ST-35	↓	1"-HBD	——	——	211,000
SH.2 (H-1)	X-142	E.5-20	69'-0"	¾"-19	¾"-19	¾"-130	——	L-547	ST-36	¾"-HBD	¾"-HBD	¾"-HBD	——	24,600
SH.3 (D-9)	X-143	K.6-19	72'-0"	1½"-19	1"-19	——	——	L-548	ST-37	1½"-HBD	1"-HBD	¾"-HBD	——	211,000
——	X-144	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.3 (D-13)	X-145	E.5-20	72'-0"	1½"-19	¾"-19	——	——	L-549	ST-38	1¼"-HBD	¾"-HBD	——	——	108,000
SH.3 (D-6,7)	X-146	M.4-22	↓	↓	¾"-130	——	——	L-550	ST-39	↓	¾"-HBD	——	——	↓
——	X-147	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-148	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-149	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-151	Q-16.6	29'-0"	¾"-19	¾"-19	¾"-130	——	L-552	ST-41	¾"-HBD	¾"-HBD	¾"-HBD	——	50,500
SH.3 (H-6,7)	X-152	R-22	53'-0"	↓	↓	——	——	L-553	ST-42	↓	↓	↓	——	24,600
SH.3 (K-8)	X-153	R-18.9	↓	1½"-19	↓	——	½"-19	L-554	ST-43	1¼"-HBD	↓	↓	½"-HBD	98,000
SH.3 (H-7,8)	X-154	S-20	↓	¾"-19	↓	——	——	L-555	ST-44	¾"-HBD	↓	↓	——	24,600
SH.3 (J-3)	X-155	N.3-20	↓	1"-19	↓	——	——	L-556	ST-45	1"-HBD	↓	↓	——	74,200
SH.2 (G-12)	X-156	F.8-25.9	46'-0"	¾"-19	↓	¾"-130	——	L-557	ST-46	¾"-HBD	↓	¾"-HBD	——	50,500
SH.2 (D-4)	X-157	A-18	69'-0"	1½"-19	↓	¾"-130	——	L-558	ST-47	1½"-HBD	↓	¾"-HBD	——	108,000
SH.2 (D-8,9)	X-158	A-24	↓	↓	↓	¾"-130	——	L-559	ST-48	↓	↓	¾"-HBD	——	↓
SH.3 (H-8,9)	X-159	S-18.9	53'-0"	1"-19	↓	——	——	L-560	ST-49	1"-HBD	↓	¾"-HBD	——	67,900
SH.3 (D-11)	X-160	F.2-18.9	72'-0"	1½"-19	↓	——	——	L-561	ST-50	1½"-HBD	↓	——	——	108,000
——	X-161	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.3 (H-10)	X-162	S-17.2	53'-0"	1"-19	¾"-19	——	½"-19	L-562	ST-51	1"-HBD	¾"-HBD	——	½"-HBD	67,900
——	X-163	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-164	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-165	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-166	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
——	X-171	——	——	——	——	——	——	DELETED	——	——	——	——	——	——
SH.2 (J-1)	X-172	E.5-23	80'-6"	¾"-19	¾"-19	¾"-19	——	L-633	ST-106	¾"-HBD	¾"-HBD	¾"-HBD	——	108,000
SH.2 (E-10)	X-212	YARD	——	¾"-19	¾"-19	¾"-19	——	L-411	ST-180	¾"-HBD	¾"-HBD	¾"-HBD	——	——



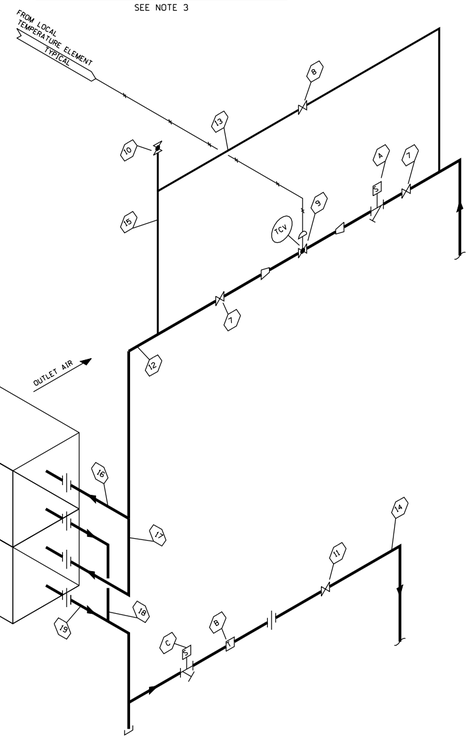
UNIT HEATER ARRANGEMENT

SEE NOTE 3



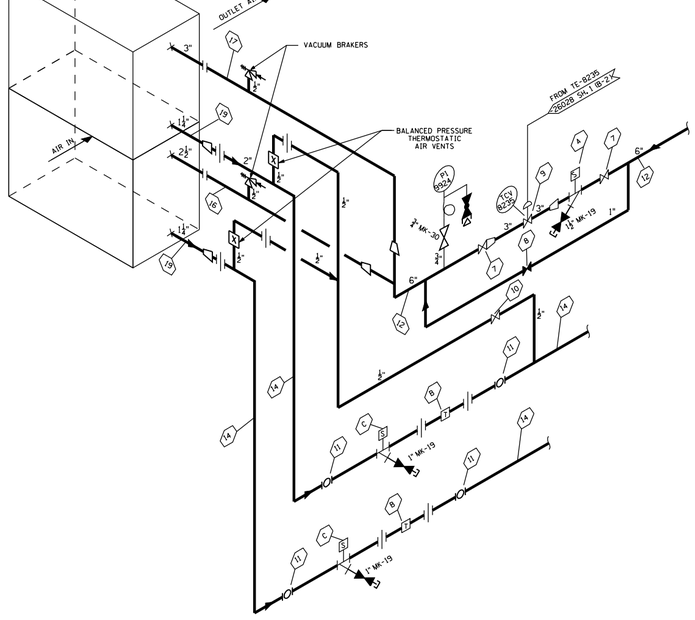
SINGLE HEATING COIL ARRANGEMENT

SEE NOTE 3



DOUBLE HEATING COIL ARRANGEMENT

SEE NOTE 3



DOUBLE HEATING COIL ARRANGEMENT FOR X-43

SEE NOTE 3

DOUBLE HEATING COIL ARRANGEMENT INDEX

P&ID LOCATION	COIL Nº	SYSTEM	VALVE SIZE & MARK Nº 7	8	9	10	11	STRAINER A	TRAP B	LINE SIZE & CLASS 12	13	14	15	16	17	18	19	STRAINER C	CAPACITY MBH
									ST-227									L-734	
SH.3 (K-3)	X-43	CONTAINMENT PURGE	6"-29	1"-19	TCV-8235	½"-19	2"-987	L-419	ST-228	6"-HBD	1"-HBD	2"-HBD	-	2½"-HBD	3"-HBD	1¼"-HBD	1¼"-HBD	L-735	242
SH.2 (J-4,5,6)	X-37 A&B	RADWASTE VENT SYSTEM	FOR VALVE, STRAINER, LINE ARRG'T SEE DWGS #24067 SHS. 186-188																2620 EA
SH.2 (E-3)	X-167	ACCESS CONTROL AREA A/C	2"-19	¾"-19	TCV-8207	——	1½"-19	L-573	ST-62	2"-HBD	¾"-HBD	1½"-HBD	¾"-HBD	2"-HBD	2"-HBD	1½"-HBD	¾"-HBD	L-714	209,000

SINGLE HEATING COIL ARRANGEMENT INDEX

P&ID LOCATION	COIL Nº	SYSTEM	VALVE SIZE & MARK Nº 7	8	9	10	11	16	17	20	STRAINER A	TRAP B	STRAINER C	LINE SIZE & CLASS 12	13	14	15	18	19	CAPACITY BTU/HR
SH.3 (K-12)	X-41	FUEL HANDLING	4"-29	1"-19	TCV-8232	——	NT. 4	NT. 4	NT. 4	1½"-19	L-136	NT. 4	NT. 4	3"-HBD	1"-HBD	NT. 4	——	NT. 4	1½"-HBD	1,640,000
——	X-42A	——	——	——	——	——	——	——	——	DELETED	——	——	——	——	——	——	——	——	——	——
——	X-42B	——	——	——	——	——	——	——	——	DELETED	——	——	——	——	——	——	——	——	——	——
SH.3 (B-2)	X-56	RADWASTE	¾"-19	¾"-130	TCV-8233	——	¾"-19	——	½"-19	½"-19	L-432	ST-59	L-570	¾"-HBD	¾"-HBD	¾"-HBD	——	½"-HBD	½"-HBD	8,930
SH.3 (C-5)	X-57	RADWASTE	1½"-19	1"-130	TCV-8234	——	¾"-19	——	½"-19	½"-19	L-431	ST-60	L-571	1"-HBD	1"-HBD	¾"-HBD	——	½"-HBD	½"-HBD	94,770
SH.3 (C-3)	X-58	RADWASTE	¾"-19	¾"-130	TCV-8237	——	¾"-19	——	½"-19	——	L-717	ST-61	L-572	¾"-HBD	¾"-HBD	¾"-HBD	——	½"-HBD	——	10,450

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. VENT VALVES MARK Nº: 10 AND 3 ARE OPTIONAL.
3. FOR SPECIFIC HEATING COIL PIPING ARRANGEMENT REFER TO HEATER AS INDICATED IN THE P&ID LOCATION COLUMN.
4. FOR X-41 CONDENSATE RETURN COMPONENT CONFIGURATION SEE P&ID 25203-26026 SH. 3 (K-12).

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

OPERATIONS CRITICAL

INFORMATION ON THIS DRAWING WAS FORMERLY SHOWN ON DWG. 25203-26026 SH. 2.

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	7	4-12-01 INCORP. DCN DM2-02-0056-00	NJB	JT	DLW	RJY
MMOD M2-96574	6	10-30-00 INCORP. DCN #'S DM2-00-0025-00 & DM2-00-0293-98	FRM	FRM	REB	REB
N/A	5	4-7-99 INC. DCN'S DM2-01-1027-98 & DM2-00-1277-97	MCR	DAD FOR WAP	REB	REB
84-024	4	9-28-95 INC. DCN DM2-S-0250-94 PER DCR M2-S-0160-94	LLJ	DAD FOR WAP	DAD FOR WAP	DAD FOR WAP
N/A	3	4-1-93 AS BUILT PER DCR-M2-S-182-93 (DCN-DM2-S-169-93)	AMS	SAH	WAP	WAP
86-242	2	6-23-92 AS BUILT PER DCR-M2-S-182-91	MJ	DAD FOR WAP	AGL	GAF
84-024	1	11-22-89 AS BUILT PER DSR-M2-S-1419-89	KNE	DAD FOR WAP	DAD FOR WAP	DAD FOR WAP

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION UNIT 2
PIPING & INSTRUMENT DIAGRAM
AUX. STEAM HEATER INDEX
WATERFORD, CT

BY	CHKD.	APP.	APP.
RJC	SBR MEL	CJA RCB	BJD SBR
DATE 5-15-85	DATE 11-22-89	DATE 11-22-89	DATE 11-22-89

SCALE NONE | DWG. NO. 25203-26026 SH. 4 OF 5 | REV 7

84-024

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING & INSTRUMENTATION DIAGRAM
HVAC SYSTEM TURB. BLDG., INTAKE STR, WHSE & DIESEL GEN. RMS.

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C26027LDGN	
REVISED PER EC MP2-21-01142 AND DUR 23-800007	DRAWING NO: 25203-26027	REV 54
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

MBULOGIN\dav1714
18-JAN-2023 09:56
C26027L.DGN

NOTES:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. "O.A." DENOTES OUTSIDE AIR
3. "R.A." DENOTES RETURN AIR
4. "S.A." DENOTES SUPPLY AIR
5. "E.A." DENOTES EXHAUST AIR
6. DUCT SIZE GIVEN AS DIA. DENOTES EQUIV. ROUND DUCT SIZE.
7. DETAIL OF HEATER DRAIN PUMP MOTOR VENTILATION SYSTEM SEE DETAIL "A" THIS DWG., AND 25203-29453 SH.1 & 2 (CUMMINGS INSULATION VENDOR DWGS. SM-1,SM-2.)
8. F-19 LOCATED HERE SHOWN ON 26029 SH. 1.
9. FOR THE ISOLATED PHASE BUS COOLER SEE P&ID 25203-26203 SH. 2.
10. FANS F-157A, B, C & D ARE TO BE UTILIZED FOR COMPENSATORY ACTIONS IN THE EVENT OF THE LOSS OF VENTILATION TO UNIT 2 ELECTRICAL SWITCHGEAR AREAS PER DCR M2-98035.
11. DELETED
12. DELETED

CABLE VAULT RECIRC. UNIT
EL. 56'-6"

DC SWITCHGEAR ROOM A/C UNIT
AUX. BLDG. EL. 14'-6"

ACCESS CONTROL AREA A/C UNIT
EL. 54'-6"

VENTILATION EQUIP. ROOM

DC SWITCHGEAR ROOM A/C UNIT
EL. 14'-6"

CHILLED WATER SURGE TANK
EL. 54'-6"

HYDRAZINE FEEDER TANK
T-36

CONDENSER

EVAPORATOR

COMPRESSOR

NON VITAL CHILLER
X-196B

NON VITAL CHILLER
X-196A

P-149C

P-149B

P-149A

CHILLED WATER PUMP
P-122A

CHILLED WATER PUMP
P-122B

X-169C
EVAPORATOR

X-169D
EVAPORATOR

CHILLER
X-169A

CHILLER
X-169B

NON VITAL CHILLED WATER PUMPS

ABANDONED IN PLACE

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1 , 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. PIPING HAS BEEN CAPPED PER PDCR 2-18-73.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 24

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

REDRAWN INPART FROM 25203-26027 SH.2 THRU REV.9

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE NUCLEAR POWER STATION UNIT-2
PIPING & INSTRUMENTATION DIAGRAM
TURBINE BLDG., INTAKE STR., WHSE. & D.G. RMS.
CHILLED WATER SYSTEM
WATERFORD, CT

REVISION DESCRIPTION SIGNATURES ON FILE | CAD NO: C260272.DGN

REVISED PER DC MP2-20-01013 & DUR 21-800205 | DRAWING NO: 25203-26027 | REV 52

DSGN DWJ | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 2

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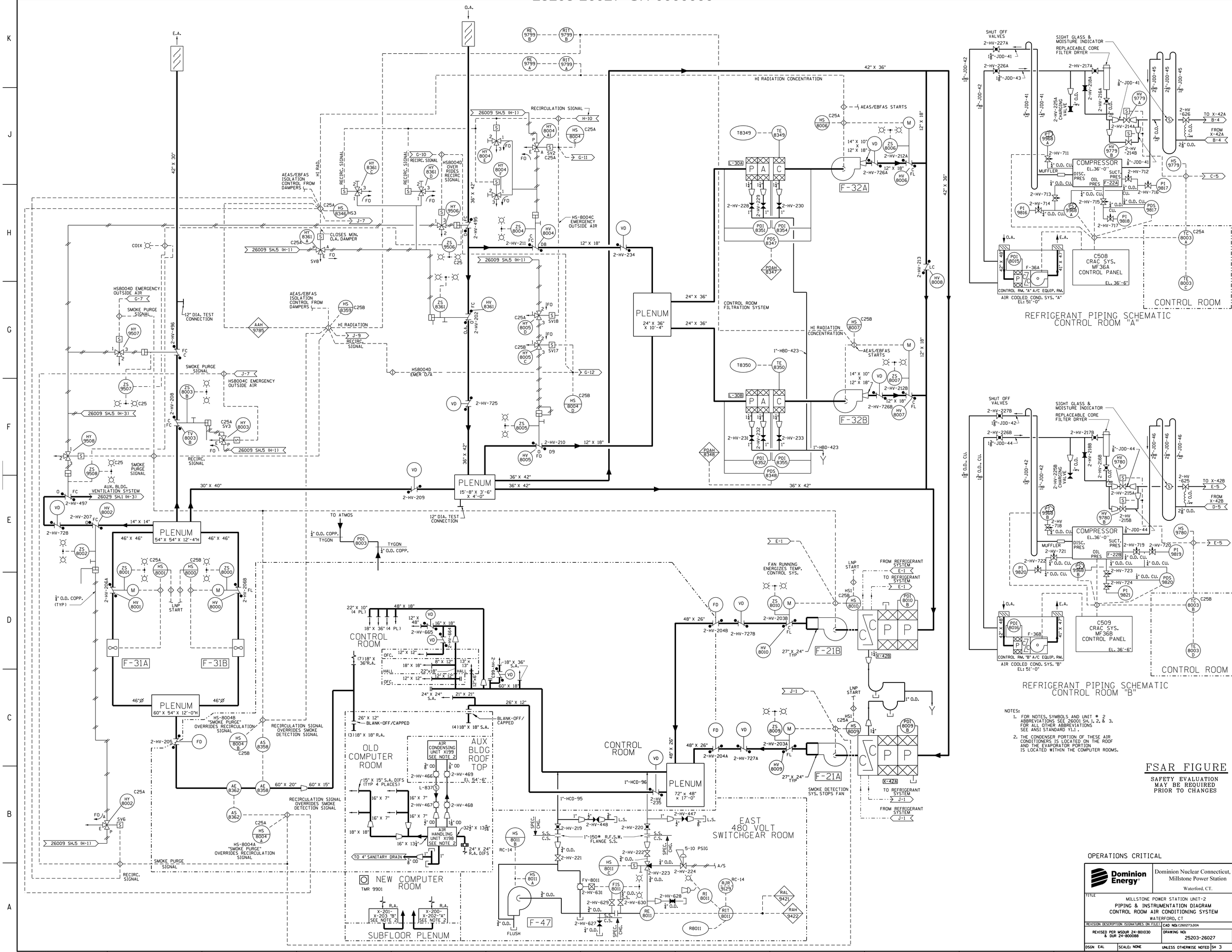
25-OCT-2021 08:28
C260272.DGN



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH. 1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. THE CONDENSER PORTION OF THESE AIR CONDITIONERS IS LOCATED ON THE ROOF AND THE EVAPORATOR PORTION IS LOCATED WITHIN THE COMPUTER ROOMS.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

Dominion Energy	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE POWER STATION UNIT-2 PIPING & INSTRUMENTATION DIAGRAM CONTROL ROOM AIR CONDITIONING SYSTEM WATERFORD, CT
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260273.DGN
REVISED PER MSDUR 24-801030 & DUR 24-800088	DRAWING NO: 25203-26027 REV 45
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED SH 3

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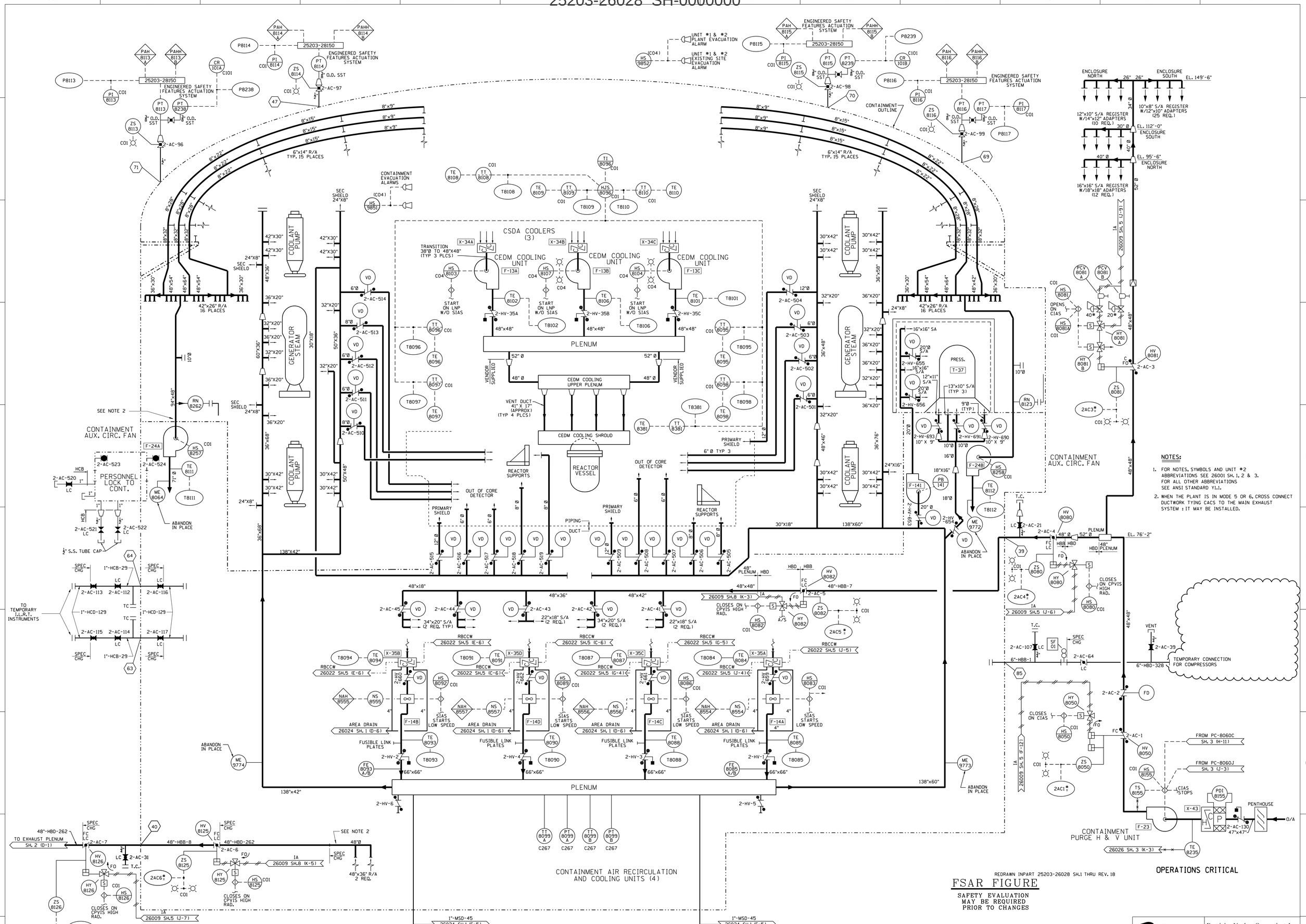
7-JUN-2024 11:00
C260273.DGN



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. WHEN THE PLANT IS IN MODE 5 OR 6, CROSS CONNECT DUCTWORK TYING CACS TO THE MAIN EXHAUST SYSTEM : IT MAY BE INSTALLED.

REDRAWN INPART 25203-26028 SH.1 THRU REV. 18

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CONTAINMENT VENTILATION SYSTEM

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 25

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING AND INSTRUMENTATION DIAGRAM
CONTAINMENT AND ENCLOSURE BUILDING VENTILATION

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C26028I.DGN	
REVISED PER TEC MP2-24-01035 & DUR 24-800164	DRAWING NO: 25203-26028	REV 49
DSGN EAL	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

Authenticated By: MBULOGIN\er-ik004
31-OCT-2024 15:41
C26028I.DGN

25203-26028 SH 0000000

CONTAINMENT AIR MONITORING SYSTEM

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. WHERE PIPE IS DESIGNATED ANSI B31.1 (1967) IT IS DESIGNED AND PROCURED TO THESE REQUIREMENTS HOWEVER IT IS MAINTAINED IN ACCORDANCE WITH 10CFR50 APPENDIX J AND ASME SECTION XI REQUIREMENTS APPLICABLE TO ASME CLASS 2 SYSTEMS.

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 25A

REDRAWN INPART 25203-26028 SH.1 THRU REV. 16

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
55	5-14-08	INCORP. DCN DM2-00-0162-08	EAL	RFL	DLW	DED
54	2-28-05	INCORP. DCN DM2-00-0237-03 DM2-01-0237-03	DLW	RFL	WBZ	MP
53	2-1-05	INCORP. DCN DM2-00-0024-05	DLW	ASK	WBZ	MP
52	11-23-04	INCORP. DCN DM2-00-0004-04, DM2-01-0004-04	DLW	MP	DED	DED
51	12-20-01	INCORP. DCN DM2-00-1834-98	JCV	TF	DLW	RJY
50	6-12-01	INCORP. DCN DM2-00-0243-01	JSJ	NJB	DLW	RJY

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
PIPING AND INSTRUMENTATION DIAGRAM
CONTAINMENT AND ENCLOSURE BUILDING VENTILATION
WATERFORD, CT

BY	CHK'DS	APP.	APP.
M. R. BIBISI	REINHART	SRF/BJD	SBR
DATE 1-12-87	DATE 8-31-87	DATE 8-14-87/9-15-87	DATE 9-15-87

SCALE NONE | DWG. NO. 25203-26028 SH.2 | REV. 55

84-024

ENCLOSURE BUILDING FILTRATION SYSTEM EQUIPMENT ROOM

ENCLOSURE BUILDING

PD8060E — PDT 8060 E

PD8060J — PDT 8060 J — 2-EB-26 — TURBINE BLDG. — 2-EB-27 — O. A.

FROM HY-8128 SH. 2 (E-3) — PC 8060 J — TO F-23 SH. 1 (C-1)

ENCLOSURE BUILDING

36"Ø

PD8060D — PDT 8060 D

HY-8128 SH. 2 (E-3)

TO FAN F-23 SH. 2 (E-3) — PC 8060 C

PD1 8060 — CO1

PDT 8060 H,I SH. 4 (H-9) — HJS 8060 — CO1

PD8060C

CR H&V ROOM

2-EB-23 — PDT 8060 C — 2-EB-24

ELECTRICAL PENETRATION ROOM

O. A.

PDT 8060 B — PD8060B

20"Ø

22"Ø

16"x16" R/A — VD — 2-EB-12

FD 2-HV-56 — FD 2-HV-57

PIPING PENETRATION ROOM

CONTAINMENT PENETRATION COOLING FANS (2)

16"x16" R/A — VD — 2-EB-11

HS 8055 — CO6 — 36"x36" — M — ZS 8055 — CO6 — HV 8055 — 2-HV-59 — 33"Ø — F-37A

HS 8054 — CO6 — 36"x36" — M — ZS 8054 — CO6 — HV 8054 — 2-HV-60 — 33"Ø — F-37B

PLENUM — PLENUM

16"Ø

ENGINEERED SAFETY FEATURES ROOM

PDT 8060 A — PD8060A

FROM 2-EB-1C (-)25' WEST PENE. RM. SH. 4 (H-5) — 14"Ø

CONTAINMENT PENETRATION COOLING SYSTEM

20"x18" R/A — VD — 2-EB-17 — VD — 20"x18" R/A — 2-EB-18 — 36"Ø — (E-3)

20"x18" R/A — VD — 2-EB-14

20"x18" R/A — VD — 2-EB-13

14"Ø — VD — 2-HV-65 — TE 8061 — T8061 — 20

10"Ø — VD — 2-HV-66 — 11¼" x 5½" OVAL x 10"Ø — TE 8058 — T8058 — 16

FD 2-HV-51 — 2-EB-92,2-EB-99 — G-3,F-3

HI RAD CLOSES — RC 101C — C101 — RIT 8240 — C101 — RE 8240

R8240 — RAH, 8020 — CO2 — RAH — CO1

R8241 — C101 — RIT 8241 — RE 8241

HI RAD CLOSES — 2-EB-91,2-EB-100 — G-4,F-4

FD 2-HV-50 — 16"Ø

8"Ø — VD — 2-HV-71 — 21

8"Ø — VD — 2-HV-70 — 2

8"Ø — VD — 2-HV-69 — 72

8"Ø — VD — 2-HV-68 — 65

9"Ø — VD — 2-HV-67 — 10

18"Ø — (D-3)

12"HBD(B)-146 — 12"HBD(B)-146 — 12"HBD(B)-147 — 12"HBD(B)-147

33"Ø — F-18A — 33"Ø — F-18B

POST INCIDENT RECIRCULATION SYSTEM

HS 8255 — CO1 — HS 8256 — CO1

POST INCIDENT RECIRCULATION FANS (2)

TE 9768 — T9768 — TE 9769 — T9769 — TE 9770 — T9770

H_2 RECOMB. HTR H-29A

TE 8721 — HJS 8721 — CO1 — TIS 8721 — CO1 — TO H-29A POWER CABINET — J1 8723 — CO1 — JC 8723 — CO1

H_2 RECOMB. HTR H-29B

TE 8722 — HJS 8722 — CO1 — TIS 8722 — CO1 — TO H-29B POWER CABINET — J1 8724 — CO1 — JC 8724 — CO1

TE 9765 — T9765 — TE 9766 — T9766 — TE 9767 — T9767

TE 9771 — T9771

CONTAINMENT

DOME SAMPLE PIPING ABANDONED IN PLACE

VD — 20"x18"R/A — 2-EB-15 — 2-EB-12 — T.C. — 3/4" — 1"-HBB-11 — LC — 2-AC-46 — 1"-HBB-149 — SPEC. CHG. — 2-EB-135 — VENT — ¼"x⅜" RED. — ⅜" S.S.-TUBING

HYDROGEN MONITORING SYSTEM (DOME SAMPLE NOT IN SERVICE)

VD — 20"x18"R/A — 2-EB-16 — 2-EB-122 — 2-AC-51 — 2-EB-136

HBD — HBB — 1"-HBD-149 — 1"-HBB-11 — 89 — 88

6"-HBD-148 — 2-IA-1142 — T-122 ACCUM. — 2-IA-689 — SPEC. CHG. — HY 8378 — HI RAD CLOSES — RIT 8241 E-8 — CIAS CLOSE — 2EB100 # — ZS 8378 — HV 8378 — HS 8378 — CO1 — 6"-HBB-10 — 83

2-EB-100 — FC — FL — SPEC. CHG.

2EB91 # — SPEC. CHG. — 2-IA-1141 — T-121 ACCUM. — 2-IA-688 — HY 8380 — HI RAD CLOSES — RIT 8241 E-8 — CIAS CLOSE — ZS 8380 — HV 8380 — HS 8380 — CO1 — 6"-HBB-10 — 82

2-EB-91 — FC — FL — SPEC. CHG.

2-EB-88 — 19 — LC

26014 SH.1 (J-12) — ¾" — ¾"-HRC-6 — 2-EB-81 — DRAIN

CIAS CLOSES — HY 8377 — HS 8377 — CO1 — 2-IA-725 — IA — 26009 SH.5 (K-4) — C527 — FT 8065 — FI 8065 — 3 WAY MANIFOLD VALVE — FE 8065 — F-8 RIT 8240 — HI RAD CLOSES — 2EB99 # — ZS 8377 — HV 8377 — ACCUM. T-134 — 2-IA-725 — 1½"-HBD(C)-375 — 6"-HBD(C)-375 — 2-EB-194 — 6"Ø — SPEC. CHG.

2-EB-99 — FC — SPEC. CHG.

HYDROGEN PURGE SYSTEM

L.C. — ¾" — 2-EB-121 — T.C. — CIAS CLOSES — HY 8379 — HS 8379 — CO1 — 2-IA-724 — IA — 26009 SH.5 (H-5) — C527 — FT 8371 — FI 8371 — 3 WAY MANIFOLD VALVE — FE 8371 — F-8 RIT 8240 — HI RAD CLOSES — 2EB92 # — ZS 8379 — HV 8379 — ACCUM. T-133 — 2-IA-724 — 1½"-HBD(C)-375 — 6"-HBD(C)-375 — 2-EB-193 — 6"Ø — SPEC. CHG.

2-EB-92 — FC — SPEC. CHG.

L.C. — ¾" — 2-EB-120 — T.C. — (G-9) — 14"Ø — VD — 2-HV-64 — TE 8059 — T8059 — 2-HV-63 — 10"Ø — 11¼" x 5½" OVAL x 10"Ø — TE 8067 — T8067 — (C-9) — 18"Ø — 15

20"x18"R/A — VD — 2-EB-10 — 20"x18"R/A — VD — 2-EB-9 — 36"Ø

2-EB-157 — FD 2-HV-52 — 18"Ø — 16"Ø — PDT 8060 F — VD — 16"x16"R/A — 2-EB-25 — ELECTRICAL PENETRATION ROOM

2-EB-158 — FD 2-HV-53 — 14"Ø — VD — 16"x16"R/A — 2-EB-21 — PIPING PENETRATION ROOM

9"Ø — VD — 2-HV-62 — 22

9"Ø — VD — 2-HV-61 — 23

12"Ø

PD8060G — PDT 8060 G

EBFS PLENUM

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

ENCLOSURE BUILDING FILTRATION SYSTEM

CHANGES MADE TO THIS P&ID MUST ALSO BE REFLECTED ON THE INSERVICE INSPECTION PROGRAM DRAWING 25203-26700 SH. 26

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

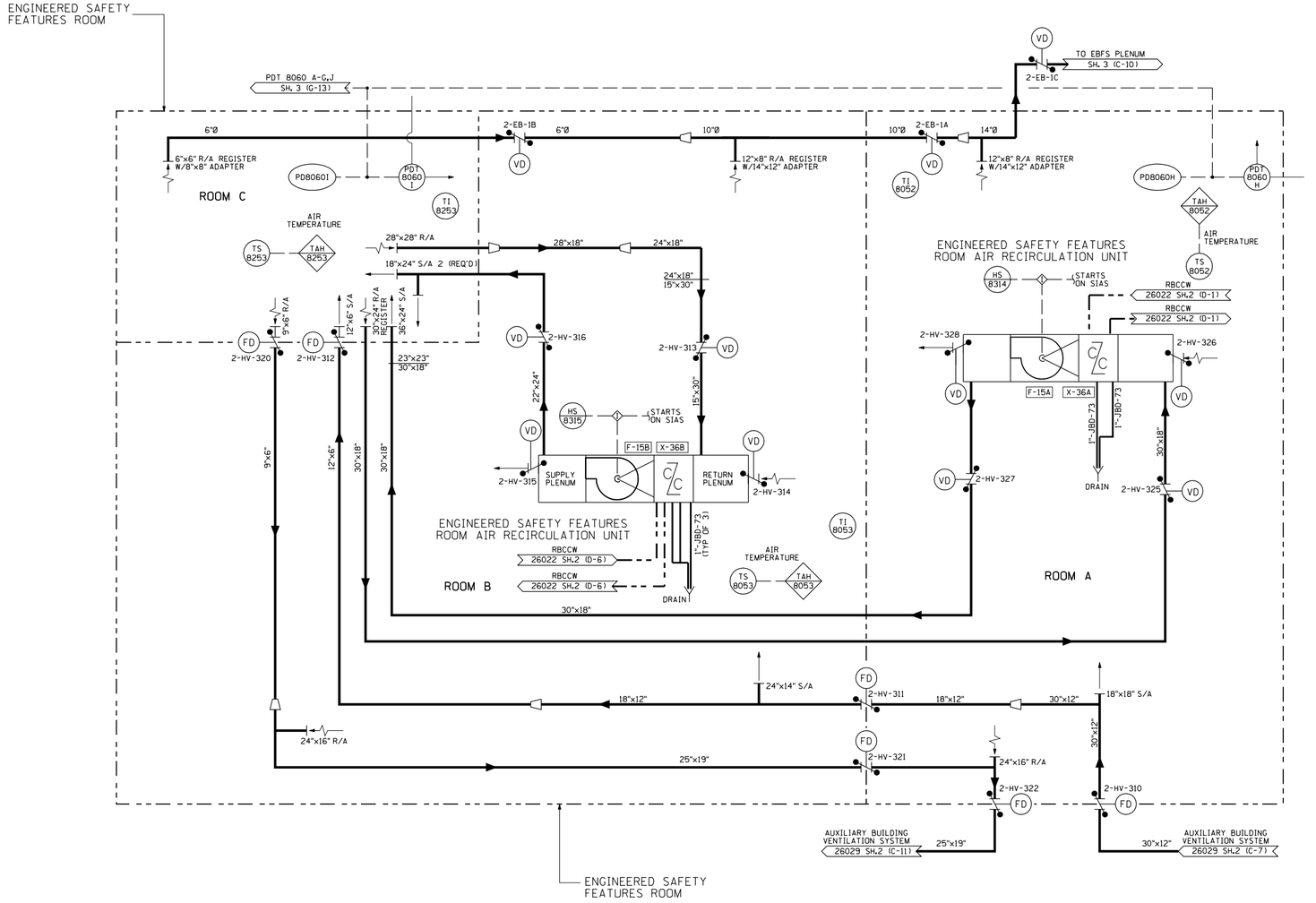
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

PIPING AND INSTRUMENTATION DIAGRAM
CONTAINMENT AND ENCLOSURE BUILDING VENTILATION

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C260283.DGN	
REVISED PER DC MP2-15-01087 & DUR 15-800446	DRAWING NO: 25203-26028	REV 26
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3



NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

REDRAWN INPART 25203-26028 SH.2 THRU REV. 16

OPERATIONS CRITICAL

Q.A.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	10	3-22-01	INCORP. DCN#DM2-00-0593-00 & DM2-00-0025-00	NJB	JTD	DLW	RJY
N/A	9	12-4-98	INC. DCN DM2-00-0213-97, REVERSED REDUCER (ZONE G-8)	GZ	AS DME	REB	REB
N/A	8	8-18-97	INC. DCN DM2-00-0636-97	MGD	AS DME	DAD for WAP	DAD for WAP
N/A	7	12-27-95	INCORP. DCN DM2-S-1288-95 PER DCR M2-S-0471-95	KSS	AS DME	JPP	JPP
84-024	6	11-8 95	INC. DCN DM2-S-0448-95 PER DCR M2-S-0403-95	LLJ	SAH RAS DME	REB FOR WAP	REB FOR WAP
N/A	5	10-12-95	INCORP. DCN DM2-S-808-95 PER DCR M2-S-392-95	RJG	AS DME	WAP	WAP

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE NUCLEAR POWER STATION UNIT NO. 2
PIPING AND INSTRUMENTATION DIAGRAM
ENGINEERED SAFETY FEATURES ROOM
AIR RECIRCULATION UNIT
WATERFORD, CT

BY	M. R. BIBISI	CHKD	S. REINHARDT	APP.	SRF	BJD	APP.	SBR
DATE	1-21-87	DATE	8-31-87	DATE	9-15-87	9-15-87	DATE	9-15-87

SCALE NONE | DWG. NO. 25203-26028 SH. 4 OF 6 | REV 10

84-024

EBFS PLENUM SHOWN ON SH. 3

OPEN ON EBFAS
CLOSE ON AEAS
IF EBFAS IS NOT PRESENT

2EB50 — CO1 — HS 8153 — ZS 8153 — M — HV 8153

FD — 2-HV-37A — 36"Ø — 2-EB-50 — 36"Ø — PLENUM — 28"Ø

HS 8074 — HY 8074 — OPENS ON EBFAS OR AEAS — 2EB51 — ZS 8074 — IA 26009 SH.5 (B-10) — FO — 27"Ø — 2-EB-51 — HV 8074

PDI 8075 — PD8075 — PDT 8075 — 3/4"OD

FAN F-25A RUNNING — HTR ON — HTR OFF — X-61 A — PDI 8075 A — PDI 8075 B — 3/4"OD

L-29A — INLET PLENUM — CPAC — X-61A

4" — FE 8068 — FS 8068 AC — FAL 8645 A — FAN F-25A RUNNING

2" — 2-EB-140C — TE 8076 — T8076 — TT 8076 — TI 8076 CO1

2-EB-140B — 2-EB-140A — 2"-HBD-411

MSD 26024 SH. 3 — 2-EB-162 — 1/2"

STARTS ON EBFAS OR AEAS — HS 8079 CO1 — IA 26009 SH.5 (B-11) — HY 8079 — 2EB52 — ZS 8079 — FO — 24"x24" — 2-EB-52 — HV 8079 — F-25A — 27"Ø

ENCLOSURE BUILDING FILTRATION FAN

CLOSES ON EBFAS OR AEAS — HS 8695 CO6 — 2EB56 CO6 — HV 8695 — M — ZS 8695 — 2-EB-56 FL — 12"Ø

2EB55 — ZS 8654 — CO6 — CLOSES ON EBFAS OR AEAS — M — 2-EB-55 FL — SPEC. CHG. — HS 8654 — HV 8654 — CO6

2-EB-54 (INSIDE PLENUM) — PLENUM 12"x4"x4' — 2-EB-43 (INSIDE PLENUM)

HS 8655 CO6 — HY 8655 — PI 8655 — L-361 — INSTRUMENT AIR SYSTEM 26009 SH. 5 (B-10)

ZS 8655 — FO — HV 8655 — 2-EB-57 C FO — 12"Ø — COND. AIR REMOVAL MAIN EXHAUST PLENUM SH. 2 (D-4)

1/2" — 2-EB-184

12"x12" — 2-EB-175 — 2-EB-176 — FAN SILENCERS

HS 9898 CO6 — HY 9898 — PCV 9896 — 3/4" O.D. — INSTRUMENT AIR SYSTEM 26009 SH. 5 (B-9)

HS 8689 CO6 — 1 1/2" — 2-EB-265 — 1 1/4" — 2-EB-187 — INTAKE PLENUM — 19"x13" (TYP) — 2-EB-188 — 1 1/4"

F-55A SJAE FAN — F-55B SJAE FAN — HS 8698 CO6 — HV 9898 — ZS 9898 — CO6

10"Ø — 10"Ø — 16"x17" — 6"Ø — 24"x24" — 2-EB-171 FC — FAN INTAKE SILENCER

FROM 2-AC-7, 2-AC-57 SH. 2 (D-1) — FD — 2-EB-131 — 36" Ø — 32"x32" — 32"x32"x98" PLENUM

FROM FUEL HANDLING SYSTEM 26029 SH.3 (G-2) — 27"Ø

HV 8143 — ZS 8143 — CO1 — 2EB60 — HS 8143 — CO1 — 2-EB-60 C FC M — 2-HBD-411

CLOSES ON EBFAS OPENS ON AEAS IF EBFAS IS NOT PRESENT — ENCLOSURE BUILDING FILTRATION SYSTEM

HS 8254 — CO1 — ZS 8254 — 2-EB-77 C FL — M — 2-EB-76 C FL — M — HV 8254 — HV 8078 — HS 8078 — ZS 8078 — CO1

12"Ø

2-EB-140D — 2-EB-140E — 2-EB-140F

HV 8062 — 2-EB-61 FC — M — HS 8062 CO1 — ZS 8062 — 2EB61 — CO1 — FAN F-25B RUNNING — FS 8069 AC — FAL 8645 B — FE 8069 — 4"

2EB72 — ZS 8650 — CO1 — CLOSES ON EBFAS OR AEAS — M — FC 2-EB-72 NC — HV 8650 — HS 8650 CO1

FD — 2-HV-37B — 36"Ø — 2-EB-40 — 36"Ø — PLENUM — 28"Ø — 27"Ø — 2-EB-41 FO

HV 8070 — ZS 8070 — CO1 — 2EB41 — OPENS ON EBFAS OR AEAS — FO — 26009 SH.5 (B-10) — HY 8070 — HS 8070 CO1

STARTS ON EBFAS OR AEAS — HS 8073 CO1 — IA 26009 SH.5 (B-11) — HY 8073 — 2EB42 — ZS 8073 — CO1 — FO — 24"x24" — 2-EB-42 — HV 8073 — F-25B — 27"Ø

ENCLOSURE BUILDING FILTRATION FAN

HV 8063 — ZS 8063 — M — 2EB40 — CO1 — HS 8063 CO1

X-61B — INLET PLENUM — CPAC — L-29B

PDI 8071 A — PDI 8071 B — 3/8"OD — HTR ON — FAN F-25B RUNNING — HTR OFF — X-61 B — PDT 8071 — PD8071 — PDI 8071 CO1

TT 8072 — TI 8072 CO1 — TE 8072 — T8072

ENCLOSURE BUILDING FILTRATION SYSTEM EQUIPMENT ROOM AUXILIARY BUILDING

FROM 2-GR-10 GASEOUS RADWASTE SYSTEM 26021 SH. 2 (C-4) — 4"-HRD-1

4" O.D. — 36"Ø — 4"

VENT — 2-AR-32 L.C. — 3/4" — SPEC. CHG.

2-EB-109 — FD — 10"Ø

TI 9897 — 2-CHW-179 — THROTTLED — 2-CHW-180 — 2-CHW-181 — 3" — 2"-HBD-448 — TO 2-CHW-134 CHILLED WATER SYSTEM 26027 SH. 2 (H-11)

3/4" O.D. — PDI 9896 — 3/4" O.D. — 2-CHW-178 — HBD HCD — 2-CHW-176 — 2-CHW-177 — 2"-HBD-447 — FROM 2-CHW-138 CHILLED WATER SYSTEM 26027 SH. 2 (F-9)

2-CHW-175 — HBD HCD — FE 9895 — TI 9896

TI 9895 — X-197 — 2-EB-267 — PDI 9897 — 2-EB-266 — TI 9894 — VD — CONDENSER AIR REMOVAL WATER BOX 26012 (H-1)

C C — 10"-JBD-17 — 2-EB-111

1" — 2-EB-178 — SPEC CHG — TO FLOOR DRAIN — LO — 2-EB-163 — SPEC CHG — HCB HBD — L-792 — VD — TURBINE GLAND SEAL DRAIN SYSTEM 26006 (K-13)

SPEC CHG — 1"-HBD-457 — 10"-HBD-205 — 2-EB-107 — SPEC. CHG.

1 1/2"-HBD-456 — 6"-JBD-19 — CONDENSER AIR REMOVAL WATER BOX 26012 (J-12)

3/4"-JBD-89 — 2-EB-167 — 2"-HBD-458 — TO FLOOR DRAIN

MILLSTONE STACK — 24"-JBD-7 — 18"-JBD-7 — L.O. 2-EB-154 — RUE 8169 — 25203-26028 SH. 6 — 16"-JBD-7 — EL. 26'-0" T.O.S. — UNIT#1 EQUIP. NUMBER FE-20-32 STACK FLOW MONITORING (SEE NOTE 3) FOR CONT. SEE 25202-26033 SH.4 (G-8) — 16"-JBD-7 — 36"-JBD-7 — 12" — EXISTING FLOOR EL. 14'-6" — 2-SSP-185 OFF-GAS PIPEWAY SUMP 26024 SH.3 (A-11) — 2"-JBD-23

T.C. — 2-EB-142 — L.C. — 1/2" — 36"-JBD-7

ABOVE GROUND
BELOW GROUND

FROM STEAM JET AIR EJECTOR DRAINS

26012 (A-8) — 3/4"-JBD-85 — 2-AR-87A
26012 (D-10) — 3/4"-JBD-86 — 2-AR-87B
26012 (E-10) — 3/4"-JBD-87 — 2-AR-87C
26012 (G-10) — 3/4"-JBD-88 — 2-AR-87D

EI 9894 — COIL DRAIN COLLECTION TANK T-104 — LS 9895 A — LS 9895 B — 2" — NP151 — HS 9895 — NP151

PI 9895 — PI 9894 — 2-EB-172 — 2-EB-173 — 1/2" — 2-EB-164 — 2-EB-174 — DRAIN — 2"-GBD-88

CONDENSATE AND FEEDWATER SYSTEM 26005 SH. 2 (E-10) — 2-EB-166 — 2-EB-165 — 1 1/2"-GBD-88

COLLECTION TANK CONDENSATE BOOSTER PUMP P-151

TURBINE BUILDING

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT #2 ABBREVIATIONS SEE 26001 SH. 1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. THE SAMPLE CONN. AND ISOLATION VALVES SHOWN MUST BE ACCESSIBLE AND APPROXIMATELY 3 FEET ABOVE THE FLOOR WITH 1 FOOT SEPERATION.
3. THE MILLSTONE STACK IS USED BY UNIT#1, 2, & 3. THE FLOW MONITORING DISCHARGE EQUIPMENT HAS BEEN GIVEN UNIT#1 EQUIPMENT NUMBERS. UNIT#1 IS RESPONSIBLE FOR MAINTAINING THEM. FOR A OVERALL VIEW SEE EACH UNIT'S P&ID LISTED BELOW:
   - UNIT#1 25202-26033 SH.4
   - UNIT#2 25203-26028 SH.5 & 6.
   - UNIT#3 25212-26948 SH.5
4. DELETED
5. DELETED

REDRAWN INPART 25203-26028 SH.2 THRU REV. 16

OPERATIONS CRITICAL

FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
PIPING AND INSTRUMENTATION DIAGRAM
CONTAINMENT AND ENCLOSURE BUILDING VENTILATION

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C26028S.DGN	
REVISED PER MSDUR 14-000121 AND DUR 14-800413	DRAWING NO: 25203-26028	REV 37
DSGN DWJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 5

TO VITAL SWITCHGEAR RM. FAN (F-52) 26027 SH.1 (D-12)

TO 4" TURB BLDG ROOF DRAIN 24019 (G-2) 2"-HBD-431 FROM X-40 F-7

2-HV-266

FROM UPPER 6.9 & 4.16KV SWGR RM EL. 56'-6" (F-52) 26027 SH.1 (D-10)

44"X36"

WEST 480V UNIT FL. EL. 36'-6" TURBINE BUILDING

2-HV-333

36" x 48" 36" x 40" 30" x 40" 30" x 32" 30" x 22" 30" x 14"

40" X 24" SA (6 PLCS)

ZE378 C01X C80

COMBINATION PENTHOUSE BACKDRAFT DAMPER

EA

HS 2142 C80 LNP START

F-142

TS 8843

FAL 8391 C06

FD

26" X 30"

24" X 30"

36" x 18" SA (TYP 8 PLCS)

2-HV-132

38" x 34"

EAST 480V. UNIT LOAD CENTER ROOM FL. EL. 36'-6"

30"X32" 30"X38" 30"X14" 30"X22" 30"X48"

FS 8850 HS 8147 TS 8850

33" Ø 36" Ø

F-26

2-HV-610

40" X 24" EA (8 PLCS.)

TS 8860 TAH 8860 C06

TI 8904 TI 8853

2-HV-497 26027 SH.3 (E-13)

14"X14"

2-HV-135

14" x 8" S.A. (TYP 3 PL)

O.A.

AUX. BLD. ROOF EL 54'-6"

VD 2-HV-730

38" x 34"

HS 8146

PDI 8146 P

ZS 8146 M HV 8146

47" x 46"

FL

2-HV-130

TE 8317 T8317

FE 8391

FS 8391

X-38 F-17

26" x 34"

2-HV-131

FD

34"X10"

14"X 10"

2-HV-200

2-HV-492

2-HV-491

SD

42"x22"

2-HV-685

36"X24" TYP (2)PLACES

20" X 12"

24" X 18"

2-HV-198

24"X 18"

30"X30"

TE 8318 T8318

SMOKE DETECTION SIGNAL STOPS FAN HS 8246

PDI 8246 X-40

C P C

51"X30"

2-HV-196

38" X 30"

35" X 34"

38" X 30"

2-HV-195

F-19

2-HV-199

20" X 12"

10" X 32"

12" X 10"

38"x24"

2-HV-192 2-HV-686 2-HV-193 2-HV-194

480V UNIT EL. 36'-6"

CONTROL ROOM EL. 36'-6"

36" x 20" (TYP 5 PL)

18" X 20"

18" X 10"

24" X 18"

2-HV-197

18" X 10"

24" X 18"

24" X 10" S.A.

14" X 8"

24" X 10" S.A.

14" X 14"

24" X 10" S.A.

14" X 8"

24" X 10" S.A.

26" X 14"

20" X 26"

16" X 12" S.A.

26" X 16"

16" X 12" S.A. (TYP 4 PLCS)

16" X 14" 14" X 12" 14" X 8"

16" X 12" S.A. (TYP 4 PLCS)

16" X 16"

16" X 12"

20" X 10"

16" X 12"

S.A. 24" X 10" (TYP 2 PLCS)

14" X 12" 14" X 8"

16" X 14"

20" X 14"

16" X 14"

TO TURB. BLDG. ROOF DRAIN K-8 2"-HBD-431

1½"-HBD-431

2"

T8319 TE 8319

26027 SH.2 (J-11)

SWITCHGEAR RM. EL. 56'-6"

CABLE VAULT FL. EL. 25'-6"

T8129 TE 8129

AUX., RADWASTE & CONTROL ROOM VENTILATION EQUIPMENT ROOM EL. 36'-6"

START ON LNP W/O SIAS

HS 8148 HS 8149

START ON LNP W/O SIAS

2-HV-637 2-HV-636

ROOF EXHAUSTERS

F-112B F-112A

PLENUM PLENUM

24"X24" 24"X24"

2-HV-139B 2-HV-139A

PLENUM

58" x 28" x 24"

24" X 20"

2-HV-140A

2-HV-145A

24"X20"

2-HV-140B

2-HV-145B

OUTSIDE YARD

FS 8391

FD

2-HV-137A

22" X 22" SEE NOTE 2

FD SD

2-HV-601A

22" X 22" R.A. (TYP 4 PLCS)

(TYP)

2-HV-601B

FD SD

SEE NOTE 2

2-HV-137B

SEE NOTE 2

2-HV-159

WEST BATTERY ROOM "B"

2-HV-138B

FD SD

TE 8436A

T8436A

2-HV-147

26" x 26"

D-9

SEE NOTE 2

2-HV-158

2-HV-138A

22" X 22"

FD SD

EAST BATTERY ROOM "A"

12"X12" SEE NOTE 3

"A" (EAST) SWITCHGEAR RM.

"B" (WEST) SWITCHGEAR RM.

12"X12" SEE NOTE 3

TE 8435 T8435

TE 8435A T8435A

2-HV-146

26" X 26"

D-4

C.S. SPEC CHG S.S.

24"X8" SA

28"X7"

14"X8"

18"X8" SA

16"X9"

12"X12" SA

18"X10" SA

12"X14"

8"X4" SA

6"X6"

27"X35"

52"x29" SA

TI 8905

2-HV-155C 2-HV-157B 2-HV-156B 2-HV-155B

8" X 8"

12" X 19"

52" X 26"

2-HV-142 22" X 22"

X-84A

26027 SH.2 (K-8)

C B P P C

F-54A

PLENUM

PDI 8872

SIAS STARTS

HS 8871 C80

24"X8" RA (4 PLCS)

24"X8" SA (3 PLCS)

18"X10"

20"X10"

24"X10"

28"X10"

27"X20"

6"X4" SA

6" X 6"

49"X27"

TE 8436 T8436

TI 8906

14" X 12"

2-HV-155A 2-HV-156A 2-HV-157A 2-HV-157C

8" X 8"

49" X 24"

11" X 21"

TI 8867

TI 8864

2-HV-141 22½" X 22"

X-84B

26027 SH.2 (K-3)

P P B C C

F-54B

PLENUM

PDI 8871

SIAS STARTS

HS 8872 C80

F-156A F-156B F-156C F-156D

NOTE 4 (TYP.)

UNIT 1 & 2 CORRIDOR EL. 14'-6"

DC SWITCHGEAR ROOMS A/C. UNITS

RC-14

HS 8998 A HS 8998 B

RJR 9373

RAH 9422

RAH 9421

RIT 8998

R8998

M FLOW CONTROL VALVE

2-HV-419

¾" O.D., CS.

¾" O.D.

¾" O.D.

2-HV-420

1" S.S.

TO EXHAUST DUCT SH.2 (J-8)

1" X ½"

2-HV-634

½" X½"

½" O.D.

F-129

HI LO FLOW

S.S.

HC 8998

RI 8998

LOCAL

FIS 8998

¼" O.D. S.S.

2-HV-632

2-HV-633

IODINE CARTRIDGE

RE 8998

2-HV-412

¾" O.D., CS.

2-HV-411

1" S.S.

FROM EXHAUST DUCT SH.2 (J-8)

1" X ½"

2-HV-635

½" X½"

½" O.D.

½" O.D.

2-HV-417

2-HV-415

¾" X½" (TYP OF 2)

2-HV-414

¾" O.D., CS

¾" O.D., CS

2-HV-416

¾" X½" (TYP OF 2)

2-HV-413

¾" X½" (TYP OF 2)

RAD MONITOR SKID RE 8998

EQUIPMENT LAYDOWN AREA EL. 14'-6"

NONRADIOACTIVE SYSTEM

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. DAMPERS HAVE 18 GAGE GALVANIZED STEEL SHIELDS INSTALLED OVER THE DAMPER TO PRECLUDE WATER FROM SPRAYING INTO THE D.C. SWITCH GEAR ROOM AND BATTERY ROOMS. DAMPERS ALSO HAVE WATER DAMS 2" HIGH INSTALLED TO PRECLUDE WATER FROM POURING INTO D.C. SWITCH GEAR ROOMS AND BATTERY ROOMS.
3. CABLE VAULT AND BATTERY ROOM FIRE/SMOKE DAMPER CLOSE ON FIRE SIGNAL FROM PANELS FLP 5,6 & 7.
4. FANS F-156A, B, C & D ARE TO BE UTILIZED FOR COMPENSATORY ACTIONS IN THE EVENT OF THE LOSS OF VENTILATION TO UNIT 2 ELECTRICAL SWITCHGEAR AREAS PER DCR M2-98035.

LEGEND:

— SUPPLY AIR

—∿— RETURN AIR

O.A. — OUTSIDE AIR

E.A. — EXHAUST AIR

A—A—A — DUCTLESS AIR

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

SUPERSEDES 25203-26029 THRU REV. 15

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

MILLSTONE POWER STATION - UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
AUX. BLDG. VENTILATION SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260291.DGN	
REVISED PER DC MP2-20-01108 & DUR 21-800175	DRAWING NO: 25203-26029	REV 29
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

MBULOGIN\jane694
13-OCT-2021 15:19
C260291.DGN

MAIN EXHAUST FAN RUNNING 26028 SH.2 (B-8)

FROM 2-HV-118 K-2

26026 SH.2 (B-2) TO PREHEAT COILS X-37A STEAM INLET TCV-9908

FROM RE-8998 SH.1 (C-8) 1" S.S.

TO RE-8998 SH.1 (B-8) 1" S.S.

O.A. 48"X60" PLENUM F-16 HS 8137 C06 PDI 8137 HV 8137 M ZS 8137 54"X54" 2-HV-101

F-16 8609 C06 EL. 36'-6"

FE 8609 FS 8609 FD 76"X42" 2-HV-102 2-HV-494 TI 9910 TE 9908 TEST PROBE CONNS. TS 9908 TE 9904 B TE 9905 B FD 84"X30" 2-HV-181

T8136 TE 8136 X-37A X-37B TE 9909

26026 SH.2 (K-3) TO REHEAT COILS X-37B STEAM INLET TCV-9909

VENTILATION ROOM SPENT POOL POOL AREA EL. 38'-6"

2-HV-650

FROM P.A.S. 26074 SH.1 (E-6) 8" Ø 2-HV-203

FROM SAMPLING SINK HOOD 26025 SH.1 (J-9) 18"X10" 8" Ø (TYP 3 PLCS) 12"X15" 18"X10" 32"X6"

48" X 18" X 12" PLENUM C72 PDI 7950 P A 10" Ø 48" X 18" X 12" PLENUM 10" Ø PDI 7930 C72

HEATING COIL CONNECTION 26026 SH.3 (B-1) TE 8233 X-56 26026 SH.3 (B-2) HS 2554 C72 SAMPLE SINK HOOD FAN 16"X4" S.A. 10"X10" CAPPED 14" Ø F-117 14"X11" 13"X11" 29"X5" 30"X12" 36"X30" 33"X15" 2-HV-182 2-HV-600 36"X30" 30"X20"

FOR A/C-4 SEE DETAIL "A" BELOW

FROM FAN F-16 SH.3 (G-11) 15"X9"

32"X30" X-57 SH.3 (G-9) 52"X30" X-58 SH.3 (G-11)

(HY-9904) 26026 SH.2 (B-9) (HY-9905) TO AUXILIARY STEAM ISOLATION

26026 SH.2 (B-12)

HS 8966 HY 8966 A/S FO ZA ZS 8966 HV 8966 2-HV-109 FC 2-HV-504 HS 8967 HY 8967 ZS 8967 HV 8967 2-HV-113 FC AUX. BLDG ENC. BLDG 36"X30" 84"X24" 60"X36" 2-HV-108 FD 54"X38" 2-HV-185 ENC. BLDG MAIN EXHAUST VENT ROOM 48"X42"

HS 8138 A/S S HY 8138 A FO TO F-16 K-10 PDIC 8138 ZS 8138 A

MAIN EXHAUST VENT ROOM RN 8999 FD 2-HV-191 54" X 30"

44" X 66" X 14'-0" PLENUM P A 2" 2" PDI 8105 L-26 PDT 8138 2-EB-147 2-EB-148 DRAIN DRAIN 2-HBD-377 2-HV-118 FO 42" X 72" TO EXHAUST PLENUM 26028 SH.2 (C-4) HV 8138 A

1" S.S.

12" X 30" FROM DRUM STORAGE AREA SH.3 (H-13)

30" X 12" FROM TOOL CRIB AREA SH.3 (D-13)

HS 8969 HY 8969 A/S FO ZS 8969 HV 8969 2-HV-144 FC HS 8968 HY 8968 A/S FO ZS 8968 HV 8968 2-HV-143 FC

36"X26" 20"X48" 30"X30" 12"X14" 12"X14" AUX. BLDG ENC. BLDG ENCLOSURE BLDG (ELEC PEN. RM.) AUX. BLDG

TO DEMINERALIZER HATCH AREA SH.3 (G-13)

RI 8999 LOCAL RJR 8999 R8999 RIT 8999 HS 8999 A HS 8999 B RAH 9422 RAH 9421 RC-14

PLENUM 9'-0" X 48" X 6'-6" 59"X24"

EL. 14'-6"

BORONOMETER ROOM LETDOWN RAD. MONITOR 4"X6" 4"X6" MCC B41 & B61 TI 6421 PURIFICATION ION EXCH. AREA SAMPLING ROOM 18"X12" COOLANT WASTE RCVR. TANKS AREA 28"X15" 28"X15" 28"X15" COOLANT WASTE MONITOR TKS. AREA 28"X15" 28"X15" 15"X12" 30"X12" 12"X30" 16"X12" 24"X12" 16"X12" 12"X16" 12"X16" 8"X12"

TRANSITION 4"x6" TO 12" x14" LOCATED OVER FLOOR DRAIN

ENC. BLDG AUX. BLDG

FROM EQUIPMENT LAYDOWN AREA SH.3 (E-10) 59"X33"

RN 8434 1" 2-HV-450 ½" O.D. LOCAL SAMPLE 1" S.S. 2-HV-121 2-HV-449 1" LOCAL SAMPLE 2-HV-123 C.S. S.S. 1" S.S.

SPEC. CHG.

M FLOW CONTROL VALVE ¾" O.D., CS. ¾" O.D., CS. 2-HV-603 ¾" O.D. F-157 2-HV-410 1" S.S. 1" X ½" 2-HV-646 ½" X ¾"

HI FLOW LO FIS 8999 ¼" O.D. S.S. 2-HV-648 2-HV-649 S.S.

HC 8999 IODINE CARTRIDGE RE 8999 PARTICULATE MOVING TAPE 2-HV-402 ¾" O.D., CS. 2-HV-401 1" S.S. 1" X ½" 2-HV-647 ½" X ¼" ¼" O.D.

¼" O.D. 2-HV-407 ¾" O.D. 2-HV-405 2-HV-404 ¾" O.D., CS. 2-HV-403 ¾" X ¼" (TYP) 2-HV-406

RAD MONITOR SKID RE 8999 MAIN EXHAUST FAN ROOM EL. 38'-6"

LOCAL RI 8434 A RAL 9421 RAH 9422 R8434A RJR 9129 RC-14

2-HV-124 ¾" O.D. ¾" O.D., C.S. 2-HV-431 2-HV-443 RE 8434 A HS 8434 D 2-HV-432 C.S. RE 8434A S.S. 1" C.S. S.S.

EL. (-)5'-0"

CONTROL AREA 26027 SH.1 (G-6) 24"X12"

FROM BA BATCH TANK AND 25 GPM EVAP. ROOMS SH.3 (E-13) 15"X12"

EL. (-)25'-0"

1" X ¾" 2-HV-444 ¾" O.D. C.S. 2-HV-434 ¾" O.D. PARTICULATE FILTER HOUSING 2-HV-125 2-HV-435 IODINE CARTRIDGE 2-HV-436 ¾" O.D. 2-HV-463 ¾" O.D. 5-10 PSIG 2-HV-127 ½" O.D. A/S 2-HV-126 1" O.D. 2-HV-641 2-HV-446 RE 8434B

FLOW CONTROL VALVE M ¾" O.D. C.S. (TYP) 2-HV-464 1" O.D. 2-HV-640 F-121 ¾" O.D. C.S.

FIS 8434 HI FLOW LO 2-HV-638 2-HV-639 HS 8434 C 2-HV-445

LOCAL RI 8434 B HS 8434 A HS 8434 B RAL 9421 RAH 9422 RIT 8434 B R8434B RJR 9373 RC-14

RAD MONITOR SKID (EL. 14'-6") RE-8434 A & B

LEGEND:
- SUPPLY AIR
- RETURN AIR
- O.A. - OUTSIDE AIR
- E.A. - EXHAUST AIR
- A—A—A - DUCTLESS AIR

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.

DEGASIFIER EFFLUENT COOLER RM. 4"X6" LETDOWN HEAT EXCHANGER RM. 4"X6" VOLUME CONTROL TANK ROOM 14"X10" EQUIPMENT LAYDOWN AREA 10"X6" 14"X12" 12"X10" BORIC ACID PUMP AND TANK ROOM 19"X13" 20"X14" 12"X12" 35"X30" 35"X25" 35"X25" 35"X5" 14"X12" 12"X12" 6"X8" 6"X8" 8"X12" 6"X8" 40"X30" 24"X24" 24"X24" 2-HV-104 FD 2-HV-500 2-HV-501 36"X30" 30"X10" 30"X10" 30"X10" 30"X10" 12"X6"

CHARGING PUMP RMS. 28"X12" 28"X12" 28"X12" 33"X12" 33"X12" 33"X12" 16"X15" 16"X14" PRE-DEGASIFIER FILTER HATCH AREA 6"X6" DEGASIFIER TANK ROOM 12"x5" DEGASIFIER PUMPS ROOM 10"x5" 26"X30" 36"X26" 36"X18" 15"X20" 15"X6" 12"X6" 7"X6" 2-HV-502 2-HV-503 30"X12" 30"X14"

HS 8133 HS 8247 22"X11" HY 8133 S A/S 26009 SH.7 (H-10) HY 8247 2HV106 * ZS 8133 HV 8133 FO FC 2-HV-106 CLOSES ON EBFAS 30"x12" 2HV107 * ZS 8247 HV 8247 2-HV-107 2-HV-310 26028 SH.4 (C-4)

HS 8249 HS 8248 HY 8249 HY 8248 2HV116 * ZS 8249 HV 8249 2-HV-116 22"x11" 2HV117 * ZS 8248 HV 8248 2-HV-117 2-HV-322 26028 SH.4 (C-6)

HI FLOW LO ¼" O.D. FIS 8997 ¼" O.D. 2-HV-645 2-HV-644 HC 8997 PARTICULATE MOVING TAPE IODINE CARTRIDGE F-128 FLOW CONTROL VALVE M 2-HV-427 ¼" O.D. 2-HV-425 2-HV-424 RE 8997 2-HV-422 2-HV-429 ¾" O.D. C.S. (TYP) 2-HV-426 ¼" X ½" (TYP) 2-HV-423 RAD MONITOR SKID RE-8997 EL. (-)25'-6" WEST PENETRATION ROOM

RJR 8997 RIT 8997 RAH 9422 RAH 9421 R8997 HS 8997 A HS 8997 B RC-14 RI 8997 LOCAL

2-HV-642 ¾" O.D. ½" O.D. 2-HV-643 ¾" O.D. ½" O.D. C.S. S.S. 1" S.S. 2-HV-430 2-HV-421 C.S. S.S. 1" S.S. 1" S.S.

EXISTING METAL ENCLOSURE MCC B61 ENCLOSURE A/C-4 TS 9921 EL. 14'-6" AIR CONDITIONING UNIT A/C-4

DETAIL "A"

OPERATIONS CRITICAL

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion®

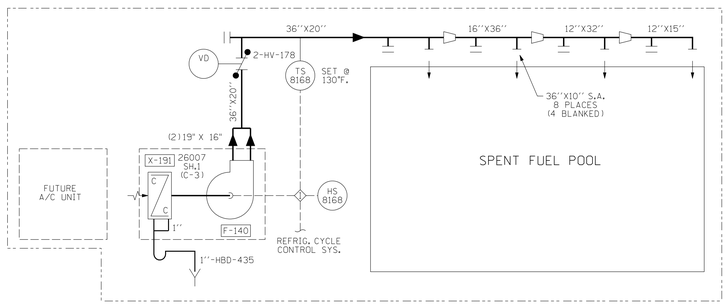
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

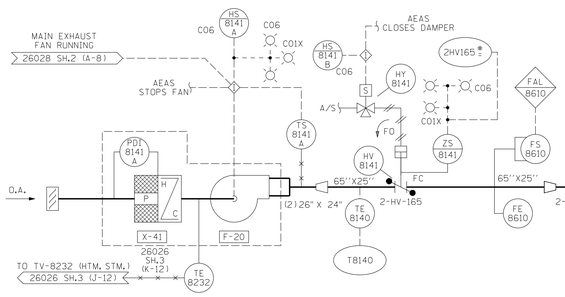
TITLE
PIPING & INSTRUMENTATION DIAGRAM
AUX. BLDG. VENTILATION SYSTEM

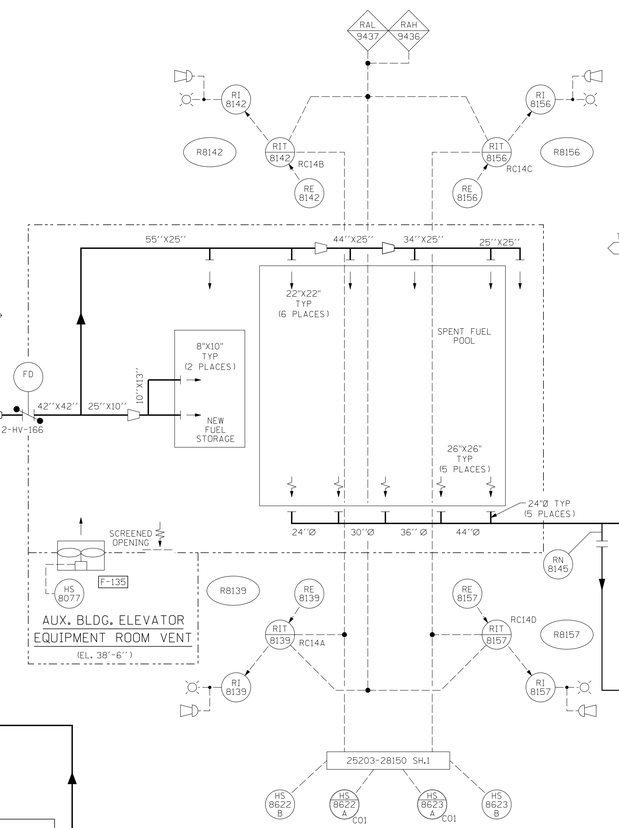
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C2602292.DGN	
REVISED PER MSDUR 15-000095 & DUR 15-800202	DRAWING NO: 25203-26029	REV 26
DSGN EAL	SCALE: NONE UNLESS OTHERWISE NOTED	SH 2

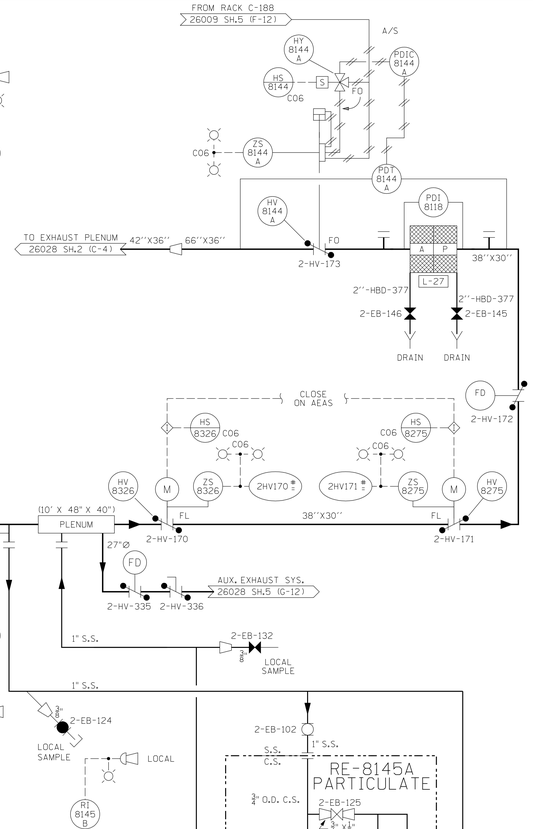
C:\ms5\Unmpoint\Cad\dm2_0029.DGN
dm2_0029

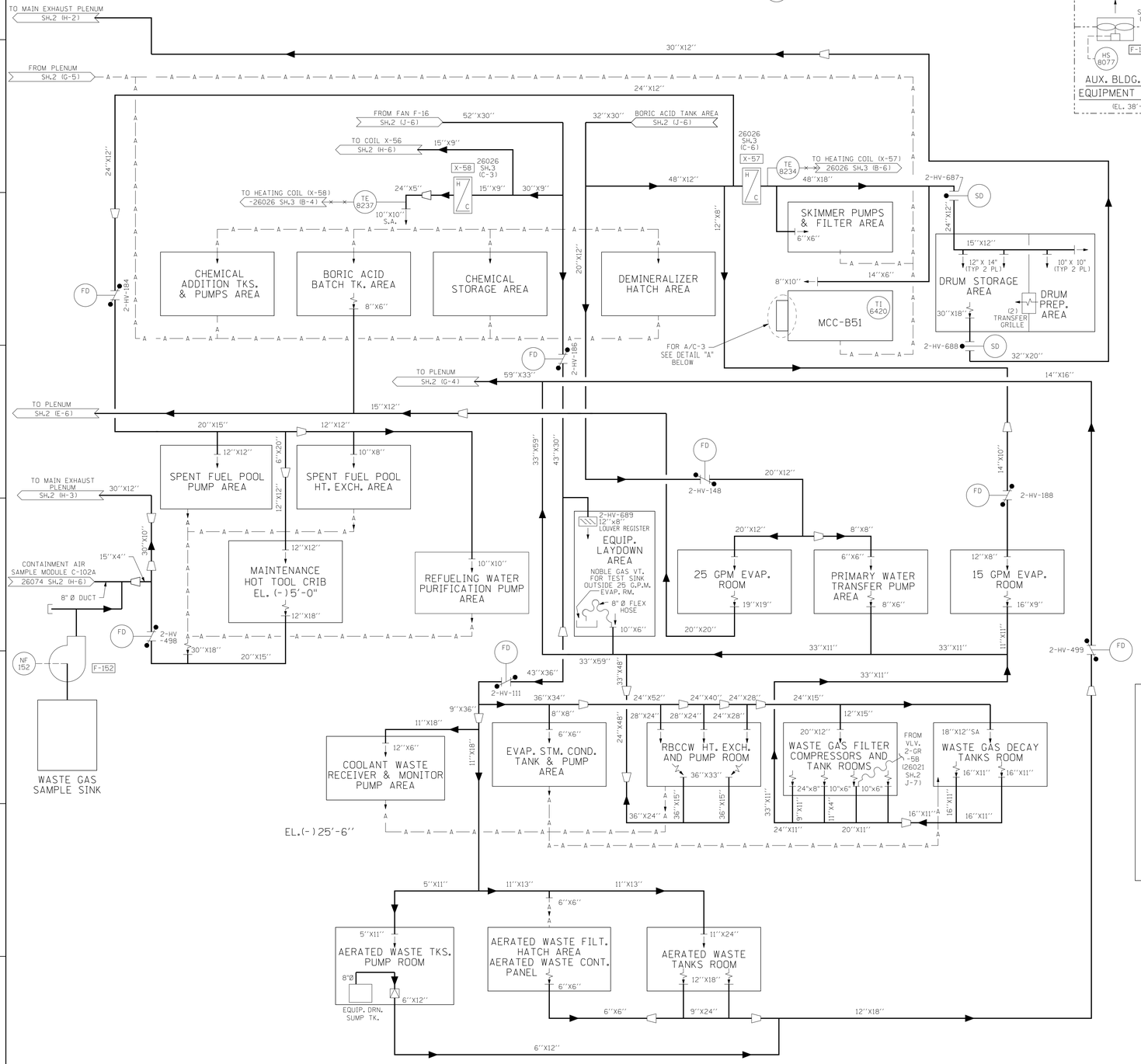
SPENT FUEL POOL AREA SUPPLEMENTAL COOLING SYSTEM

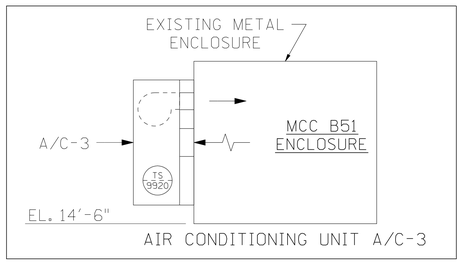












DETAIL "A"

LEGEND:
- SUPPLY AIR
- RETURN AIR
O.A. - OUTSIDE AIR
E.A. - EXHAUST AIR
A—A—A - DUCTLESS AIR

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2, & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .

SUPERSEDES 25203-26029 THRU REV. 15

OPERATIONS CRITICAL

Q.A.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

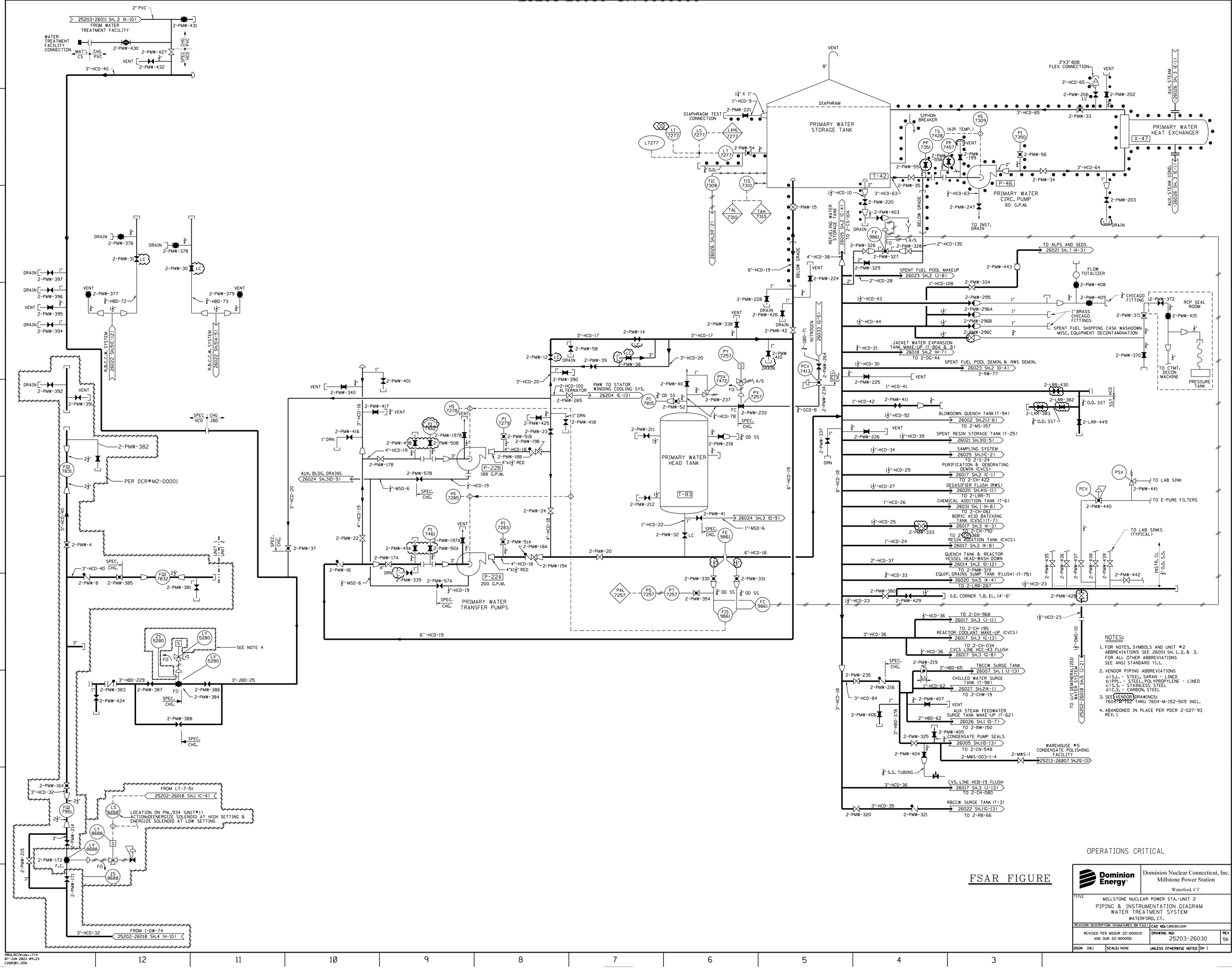
REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
19	9-10-09	INCORP. DCN DM2-00-0193-09	JCV	MP	DLW	REB
18	3-6-02	INCORP. DCN DM2-00-0697-99	JSJ	MBR	DVC	DCZ

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE POWER STATION - UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
AUX. BLDG. VENTILATION SYSTEM
WATERFORD, CT.

BY A.F.F.	CHKD. S.E.BESADE	APP. W. J. QUINLAN/JJR	APP. JGR/SBR
DATE 1-10-87	DATE 12-13-87	DATE 12-13-88	DATE 12-13-88
SCALE NONE	DWG. NO. 84-024	25203-26029 SH.3	REV 19



OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE NUCLEAR POWER STA.-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
WATER TREATMENT SYSTEM
WATERFORD, CT.

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C260301.DGN	
REVISED PER MSDUR 22-000019 AND DUR 22-800050	DRAWING NO: 25203-26030	REV 56
DSGN DWJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

MBULOGIN\dav1714
07-JUN-2022 09:23
C260301.DGN

25203-26074 SH-0000000

C-103C — POST ACCIDENT SAMPLING SYSTEM REACTOR COOLANT AUXILIARY VALVE OPERATOR PANEL

C-103D — POST ACCIDENT SAMPLING SYSTEM REACTOR COOLANT AUXILIARY VALVE OPERATOR PANEL

HPSI/LPSI CONTAINMENT SPRAY SYSTEM 26015 SH.1 (K-5)

TO R.C. SAMPLE LINE 26014 SH.2 (J-13) .375" O.D.

2-S-449 FC NOTE 2

2-S-450

2-S-503

NOTE 2 2-S-446 FC

L-736

1" X 3/8"

2-S-736

2-S-448 FC NOTE 2

2-S-447

FLUSH LINE

2-S-504

TO SAMPLE COOLER X-64 26025 SH. 1 (K-11)

LPSI/HPSI CTMT. SPRAY SAMPLE LINE

TEST POST ACCIDENT SAMPLE LINE

2-S-459

2-S-457 FC NOTE 2

2-S-456

TO AUX. BLDG. DRAINS 26025 SH.1 (J-10)

FROM RCS HOT LEG THRU SAMPLE COOLER (X-64 & X-65) 26025 SH.1 (J-12)

FROM LPSI/HPSI SAMPLE THRU SAMPLE COOLER (X-64) 26025 SH.1 (K-12)

2-S-454

2-S-455

2-S-483

2-S-556

2-S-557

INFLUENT

EFFLUENT

2-S-499 (FLF)

2-S-741

2-S-742

TO VENT'L FAN (F-117) 26029 SH. 2 (H-12)

FAN F-154

.250" O.D. N_2 SUPPLY VALVE OPER. LINES

2-S-496

2-S-494 (FB)

2-S-495 (FB)

2-GAN-259 (FS)

2-GAN-257

2-S-498 (FS)

2-S-561

2-S-497 (FS)

2-S-489 (FO)

2-S-493 (FAI)

P-159

2-S-482 (FB)

2-S-486 (FB)

2-S-491 (FB)

2-S-492

2-S-490 (FB)

2-S-487 (FAI)

2-S-488 (FO)

LIQUID SAMPLE

PH PROBE

(AREA RAD. MONITOR)

1"-WSD-48

SAMPLE ROOM FLOOR DRAIN

REACTOR COOLANT SAMPLE MODULE C-103A (SAMPLE ROOM)

AUXILIARY BUILDING

VENT 2-GAN-264

2-GAN-263

N_2 GAS FLASK

2-GAN-258

2-GAN-253

2-GAN-244

2-GAN-235

2-GAN-368

1800-100 PSIG

2-GAN-260

2-GAN-254

2-GAN-261

2-GAN-262

2-GAN-237

2-GAN-243

RUBBER HOSE (TYP 6 PL'S)

REACTOR COOLANT REMOTE OPERATOR MODULE C-103B

½" -HCD-139

PMW SUPPLY 26025 SH.1 (C-3)

2-S-505

2-S-713

T-135

30 GAL STAINLESS STEEL RESERVOIR

ACCUMULATOR

0-400 PSIG

SET AT 375 PSIG

2-S-509

2-S-712

2-S-506

L-469

2-S-508

2-S-714

1 GPM

V3 HP

P-155 FLUSH PUMP

H_2O CONNECTION (OUT)

PURGE CONNECTION (IN)

2-S-507

FLUSH MODULE

TURBINE BUILDING ELEV. 14'-6"

NOTE:
1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1.
2. ALL VALVES REFERENCED TO THIS NOTE MUST BE INSTALLED WITH THE ARROW ON THE BODY MATCHING THE ARROW ON THIS PLAN.
3. ALL LINE SIZES ARE O.D. UNLESS NOTED OTHERWISE.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD — NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

OPERATIONS CRITICAL

Q.A.

THIS DWG. SUPERSEDES 25203-26074 THRU REV.5

REVISIONS DURING CONSTRUCTION

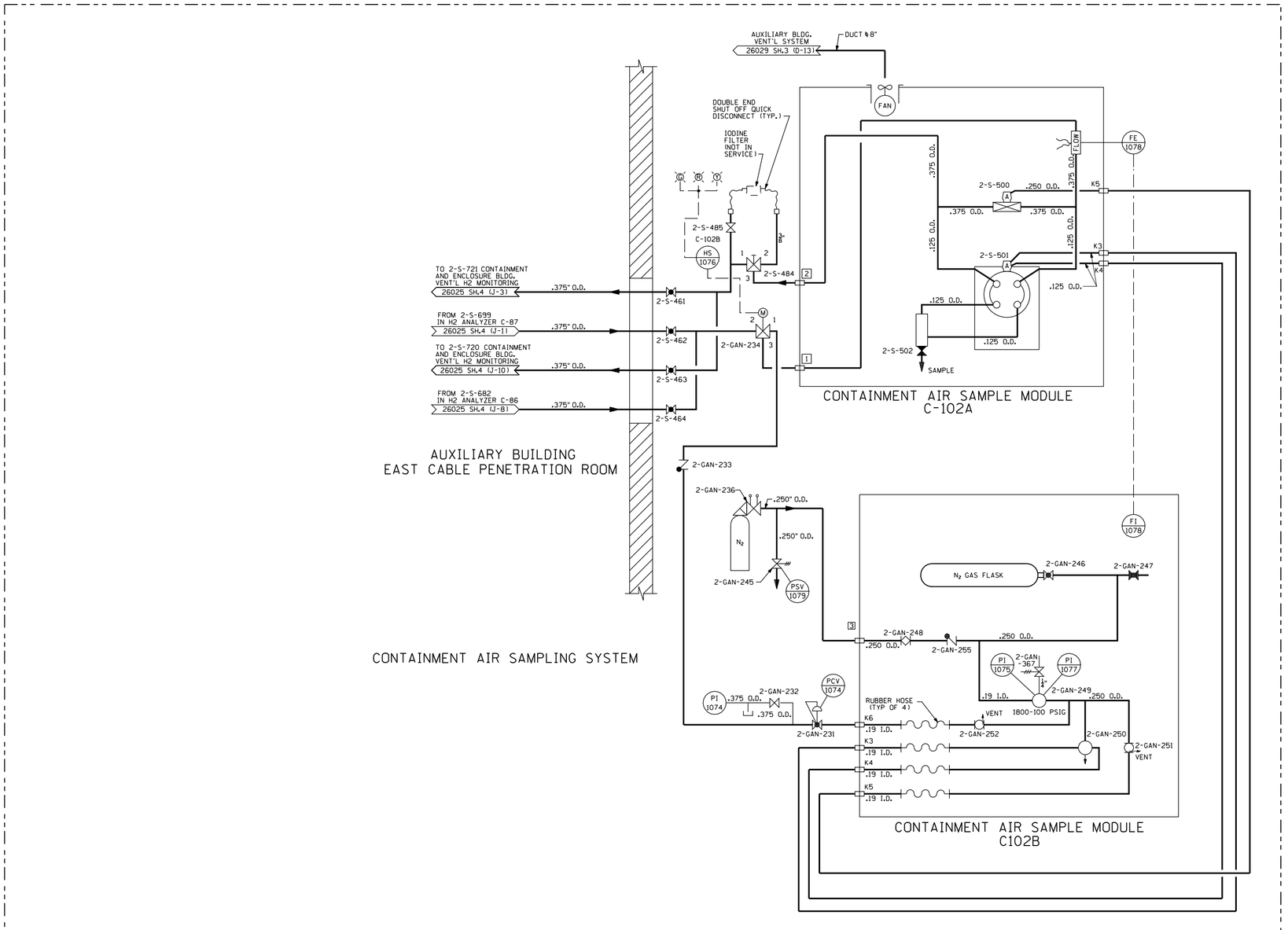
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	14	6-21-07 INC. DCN DM2-00-0115-07	JMS	DWJ	DLW	RFL
N/A	13	2-16-01 INC. DCN DM2-00-0025-00	NJB	VJG	REB	GMO
N/A	12	11-2-99 INC. DCN DM2-00-0856-99	SJM	VJG	REB	REB
N/A	11	2-5-99 INC. DCN DM2-00-1790-98	GZ	VJG	REB	REB
N/A	10	10-28-97 INC. DCN DM2-00-0874-97	MGD	VJG	REB for WAP	REB for WAP
N/A	9	12-27-96 INCORP. DCN DM2-S-1304-95 PER DCR M2-S-0474-95	KSS	VJG	JPP	JPP
84-024	8	8-23-95 INC. DCN DM2-S-0264-94 PER DCR M2-S-0270-94. UPDATED ELEC. LINE CONNECTOR POINTS PER REV. 5 NOTE 1. THIS SHEET HAS BEEN REDRAWN.	LLJ	VJG	REB for WAP	REB for WAP

Dominion — Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
POST ACCIDENT SAMPLING SYSTEM
REACTOR COOLANT SAMPLE
WATERFORD, CT

BY	CHKD'S. REINHARDT	APP. GEC	JGR	APP. SBR
RJC	DATE 9-26-86	DATE 6-1-87	DATE 6/11/87 6/5/87	DATE 6/11/87

SCALE NONE — DWG. NO. 84-024 — 25203-26074 SH.1 — REV 14



LEGEND

FO - FAIL OPEN
FS - FAIL SHUT
FI - FAIL INLINE
FB - FAIL BYPASS
FLF - FAIL LOW FLOW
FAI - FAIL AS IS

NOTES:

1. FOR NOTES, SYMBOLS AND UNIT # 2 ABBREVIATIONS SEE 26001 SH.1, 2 & 3. FOR ALL OTHER ABBREVIATIONS SEE ANSI STANDARD Y1.1 .
2. THE SYMBOLS FOR VENDOR EQUIPMENT ON THIS DRAWING, SEE DRAWINGS 25203-29429 SH.4 & 25203-29431 SH.3.

THIS DRAWING REFLECTS MODULES SHOWN ON VENDOR DRAWINGS
25203-29429 SH.2
25203-29429 SH.3
25203-29431 SH.2

OPERATIONS CRITICAL

Q.A.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
N/A	7	6-21-07	INC. DCN DM2-00-0115-07	JMS	DWJ	DLW	RFL
N/A	6	2-16-01	INC. DCN DM2-00-0025-00	NJB	VJG	REB	GMO
N/A	5	1-16-99	INC. DCN DM2-00-0474-97	GZ	XXX	REB	REB
N/A	4	10-28-97	INC. DCN DM2-00-0874-97	MGD	XXX	REB for WAP	REB for WAP
84-024	3	8-23-95	INC. DCN DM2-S-0064-94 PER DCR M2-S-0729, INSTALL ELECT. LINE TERMINATION BOX (NOTE 1). THIS SHEET WAS NEW DRAWN.	LLJ	XXX	REB for WAP	REB for WAP
NA	2	9-9-91	AS BUILT PER DCR-M2-P-75-91	[illegible]	XXX	TG	RWB
84-024	1	6/8/87	AS BUILT PER DSR-M2-S-529-87.	SEB	SBR	XXX	SBR

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT 2
PIPING & INSTRUMENTATION DIAGRAM
POST ACCIDENT SAMPLING SYSTEM
CONTAINMENT AIR SAMPLE
WATERFORD, CT

BY	CHK'D	APP.	APP.
ASK	S. REINHARDT	GEC / JGR	SBR
DATE 1-19-87	DATE 6/1/87	DATE 6/11/87 / 6/5/87	DATE 6/11/87

SCALE NONE

DWG. NO. 84-024 — 25203-26074 SH.2

REV 7

25203-27011 SH-0000000
Contains Redacted Information

25203-27012 SH-0000000
Contains Redacted Information

25203-27013 SH-0000000
Contains Redacted Information

25203-27014 SH-0000000
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25203-27015 SH-0000000
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25203-27020 SH-0000000
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25203-27021 SH-0000000
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25203-27022 SH-0000000
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25203-27023 SH-0000000
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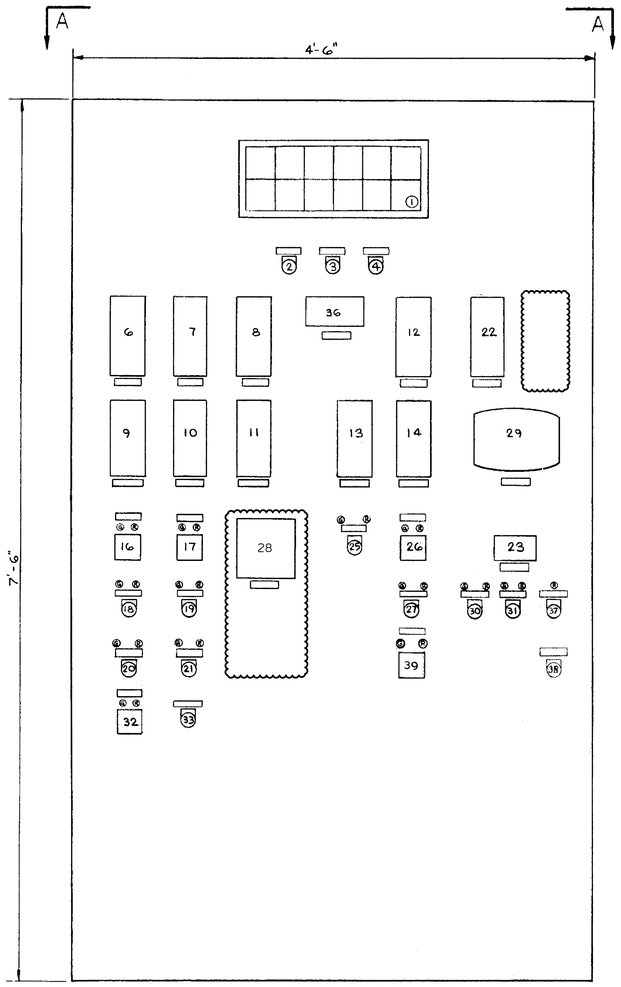
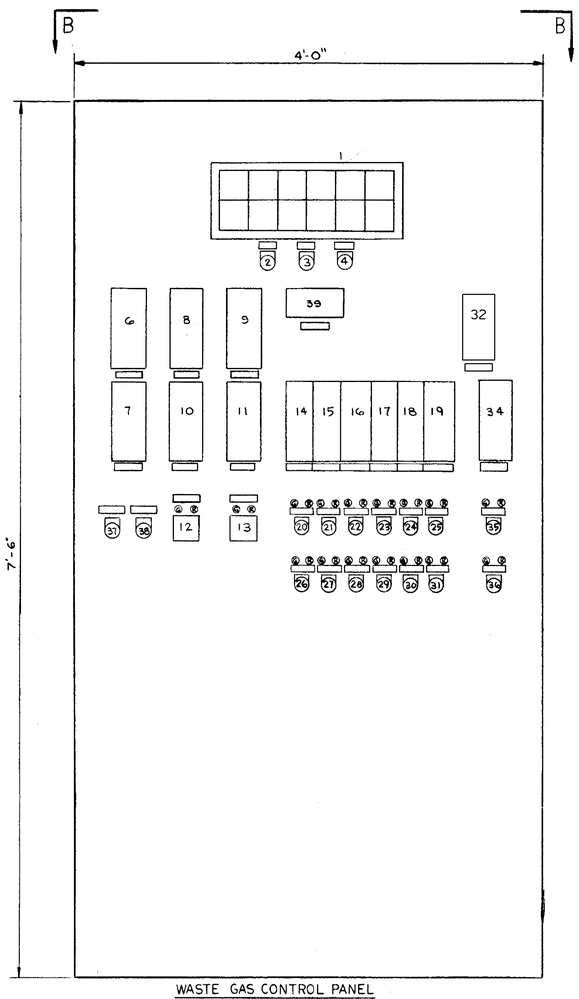
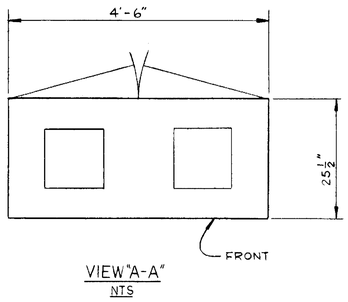


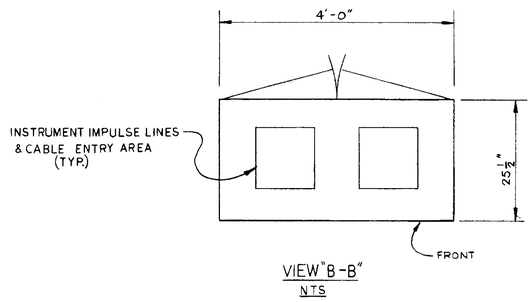
AERATED LIQUID RADWASTE CONTROL PANEL
(C60)



WASTE GAS CONTROL PANEL
(C61)



VIEW "A-A"
NTS



VIEW "B-B"
NTS

AERATED LIQUID RADWASTE CONTROL PANEL (C60)

ITEM	INSTR NO.	DESCRIPTION
1		ALARM ANNUNCIATOR
2	PB-A80	ALARM ANNUNCIATOR "ACKNOWLEDGE"
3	PB-R80	ALARM ANNUNCIATOR "RESET" PUSHBUTTON
4	PB-T80	ALARM ANNUNCIATOR "TEST" PUSHBUTTON
5		SPARE
6	TI-9100	AERATED WASTE DRAIN TANK (T-20A) TEMP. INDICATOR
7	TI-9103	AERATED WASTE DRAIN TANK (T-20B) TEMP. INDICATOR
8	TI-9109	AERATED WASTE FILTER (L-13) DISCH. TEMPERATURE INDICATOR
9	LI-9102	AERATED WASTE DRAIN TANK (T-20A) LEVEL INDICATOR
10	LI-9105	AERATED WASTE DRAIN TANK (T-20B) LEVEL INDICATOR
11	PI-9120	SPENT RESIN TANK (T-25) PRESSURE INDICATOR
12	TI-9185	AERATED WASTE MONITOR TANK (T-21) TEMP. INDICATOR
13	LI-8321	AERATED WASTE EVAPORATOR PACKAGE LEVEL INDICATOR
14	LI-8114	AERATED WASTE MONITOR TANK (T-21) LEVEL INDICATOR
15		SPARE
16	HS-9101	AERATED WASTE DRAIN TK. PUMP (P-31A) CONTROL SWITCH
17	HS-9104	AERATED WASTE DRAIN TK. PUMP (P-31B) CONTROL SWITCH
18	HS-9194	AERATED WASTE DRAIN TK. (T-20A) RECIRC. VALVE SEL. SWITCH
19	HS-9197	AERATED WASTE DRAIN TK. (T-20B) RECIRC. VALVE SEL. SWITCH
20	HS-9191	AERATED WASTE DRAIN TK. (T-20A) DISCH. VALVE SEL. SWITCH
21	HS-9193	AERATED WASTE DRAIN TK. (T-20B) DISCH. VALVE SEL. SWITCH
22	FI-3308	DEMIN T-2A INLET FLOW INDICATOR
23	FQI-9303	DEMIN T-2A INLET FLOW TOTALIZER
24		SPARE
25	HS8247	CONCENTRATES TANK HEATER CONTROL SWITCH
26	HS-8113	AERATED WASTE MONITOR TANK PUMP (P-32) CONTROL SWITCH
27	HS-8196	AERATED WASTE MONITOR TANK RECIRC. VALVE SEL. SWITCH
28	FR-9118	AERATED WASTE SYSTEM DISCHARGE FLOW RECORDER
29	FQI-9118	AERATED WASTE SYSTEM DISCHARGE FLOW TOTALIZING INTEGRATOR
30	HS-9118A	AERATED WASTE SYSTEM DISCHARGE ISOLATION VALVE SEL. SWITCH
31	HS-9118B	AERATED WASTE SYSTEM DISCHARGE ISOLATION VALVE SEL. SWITCH
32	HS-9210A	CONCENTRATES TANK PUMP P93 CONTROL SWITCH
33	HS-9201B	AERATED WASTE EVAP PKG. PUMP LOCAL-REMOTE SWITCH
34		SPARE
35		SPARE
36	HIC-9115B	SYSTEM DISCHARGE VALVE FLOW CONTROLLER (MANUAL)
37	HS-9115	AERATED WASTE SYS. SAMPLE PUMP #P-156 FOR RM9116
38	HS-9117	AERATED WASTE SYS. FLUSH VALVES RV9117A & RV9117B FOR RM9116
39	HS-3141	AERATED WASTE MONITOR TANK MIXER MOTOR MT21 CONTROL SWITCH

WASTE GAS CONTROL PANEL (C61)

ITEM	INSTR NO.	DESCRIPTION
1	PB-A81	ALARM ANNUNCIATOR
2	PB-A81	ALARM ANNUNCIATOR "ACKNOWLEDGE" PUSHBUTTON
3	PB-R81	ALARM ANNUNCIATOR "RESET" PUSHBUTTON
4	PB-T81	ALARM ANNUNCIATOR "TEST" PUSHBUTTON
5		SPARE
6	TI-8127	WASTE GAS SURGE TANK (T-18) TEMP. INDICATOR
7	PI-8128	WASTE GAS SURGE TANK (T-18) PRESSURE INDICATOR
8	TI-8084	WASTE GAS COMPRESSOR (F-1A) DISCH. TEMP. INDICATOR
9	TI-8087	WASTE GAS COMPRESSOR (F-1B) DISCH. TEMP. INDICATOR
10	PI-8085	WASTE GAS COMPRESSOR (F-1A) DISCH. PRESSURE INDICATOR
11	PI-8088	WASTE GAS COMPRESSOR (F-1B) DISCH. PRESSURE INDICATOR
12	HS-8188	WASTE GAS COMPRESSOR (F-1A) CONTROL SWITCH
13	HS-8189	WASTE GAS COMPRESSOR (F-1B) CONTROL SWITCH
14	PI-8080	WASTE GAS DECAY TANK (T-19A) PRESSURE INDICATOR
15	PI-8081	WASTE GAS DECAY TANK (T-19B) PRESSURE INDICATOR
16	PI-8241	WASTE GAS DECAY TANK (T-19C) PRESSURE INDICATOR
17	PI-8344	WASTE GAS DECAY TANK (T-19D) PRESSURE INDICATOR
18	PI-8347	WASTE GAS DECAY TANK (T-19E) PRESSURE INDICATOR
19	PI-8350	WASTE GAS DECAY TANK (T-19F) PRESSURE INDICATOR
20	HS-8079	WASTE GAS DECAY TANK (T-19A) INLET VALVE SEL. SWITCH
21	HS-8078	WASTE GAS DECAY TANK (T-19B) INLET VALVE SEL. SWITCH
22	HS-8342	WASTE GAS DECAY TANK (T-19C) INLET VALVE SEL. SWITCH
23	HS-8345	WASTE GAS DECAY TANK (T-19D) INLET VALVE SEL. SWITCH
24	HS-8348	WASTE GAS DECAY TANK (T-19E) INLET VALVE SEL. SWITCH
25	HS-8351	WASTE GAS DECAY TANK (T-19F) INLET VALVE SEL. SWITCH
26	HS-8085D	WASTE GAS DECAY TANK (T-19A) OUTLET VALVE SEL. SWITCH
27	HS-8085C	WASTE GAS DECAY TANK (T-19B) OUTLET VALVE SEL. SWITCH
28	HS-8085E	WASTE GAS DECAY TANK (T-19C) OUTLET VALVE SEL. SWITCH
29	HS-8085F	WASTE GAS DECAY TANK (T-19D) OUTLET VALVE SEL. SWITCH
30	HS-8085G	WASTE GAS DECAY TANK (T-19E) OUTLET VALVE SEL. SWITCH
31	HS-8085H	WASTE GAS DECAY TANK (T-19F) OUTLET VALVE SEL. SWITCH
32	FIT-9097	WASTE GAS DISCHARGE FLOW SCFH
34	PIC-9099	WASTE GAS DECAY TANK OUTLET PRESSURE IND/CONTROLLER
35	HS-8083A	WASTE GAS SYSTEM DISCH. ISOLATION VALVE SEL. SWITCH
36	HS-8083B	WASTE GAS SYSTEM DISCH. ISOLATION VALVE SEL. SWITCH
37	HS-8188R	WASTE GAS COMPRESSOR F-1A RESET PUSHBUTTON
38	HS-8189R	WASTE GAS COMPRESSOR F-1B RESET PUSHBUTTON
39	HIC-8095B	SYSTEM DISCHARGE VALVE FLOW CONTROLLER (MANUAL)

NOTES:

1. FOR LAYOUT AND CONSTRUCTION DETAILS, REFER TO BECHTEL DWG. 7604-M420-B3(C60) & 7604-M420-BQ(C61).
2. FOR BILL OF MATERIAL SEE BECHTEL DWG. 7604-M420-BM-C60 OR C61.
3. THE ABOVE NUMBERING SEQUENCE APPLIES ONLY TO THE ITEMS ON THIS DWG.
4. FOR NAMEPLATE ENGRAVING SCHEDULE SEE BECHTEL DWG. 7604-M-420-B4 (C60) & 7604-M420-B7 (C61). 29097, SH.81A, 81B. 29097, SH.84A
5. FOR WOLFE & MANN FRONT VIEW & SECTIONS, SEE 29097, SH. 74 FOR C63 & 29097, SH.83 FOR C61.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
LOCAL CONTROL PANELS
(C60) & (C61)

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C28006L.MYB	
REVISED PER DC MP2-21-01038 & DUR 23-800022	DRAWING NO: 25203-28006	REV 5
DSGN JSJ	SCALE: N/A	UNLESS OTHERWISE NOTED SH 1

MBLOGIN\scottsB4
6-MAR-2023 09:15
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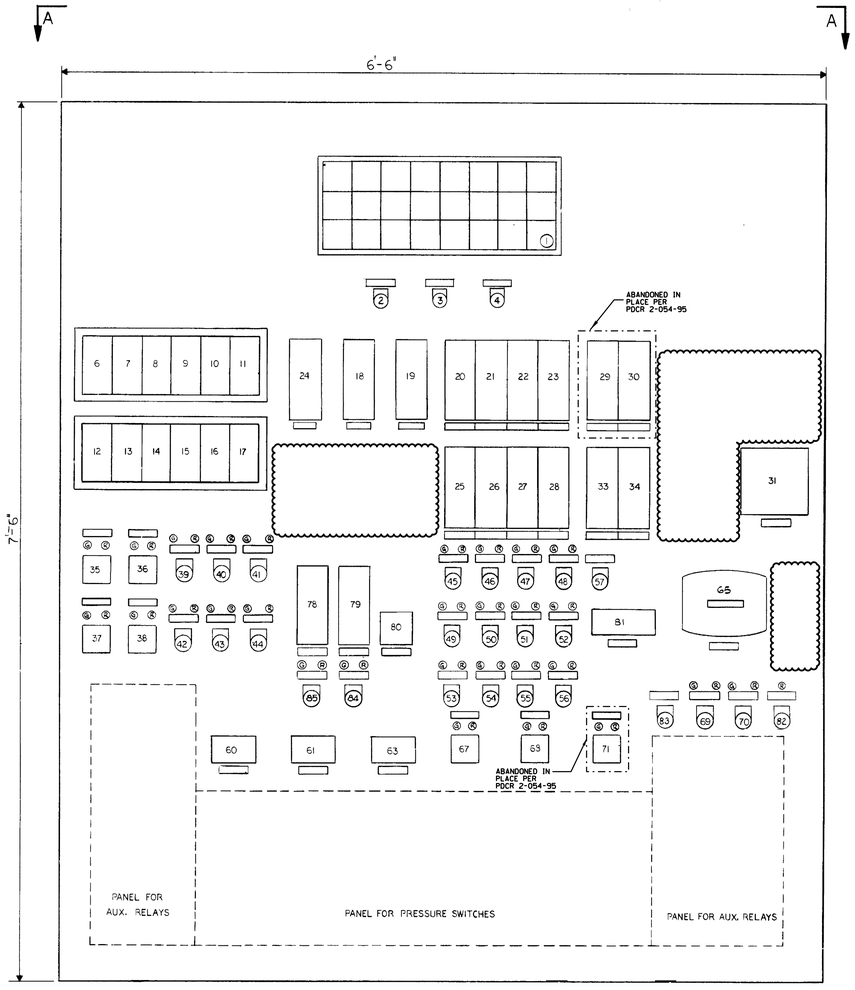
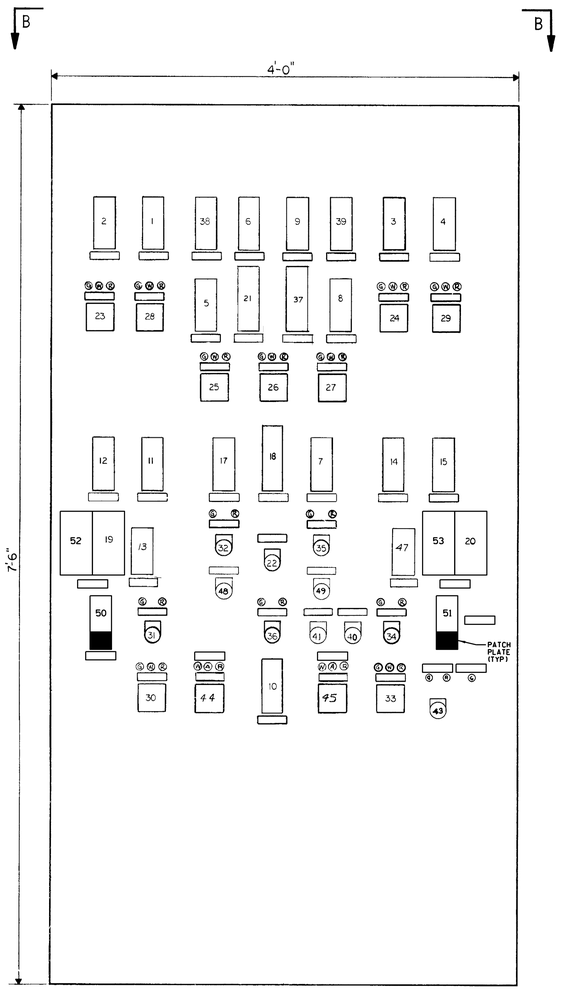
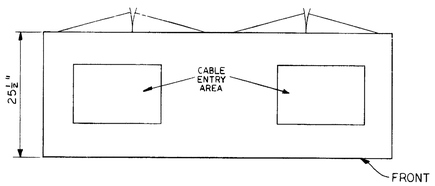


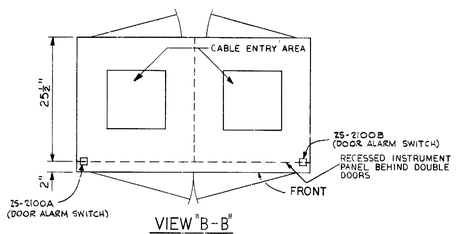
CLEAN LIQUID RADWASTE CONTROL PANEL
(C63)



HOT SHUTDOWN CONTROL PANEL
(C21)



VIEW "A-A"
NTS



VIEW "B-B"
NTS

NOTES

1. FOR LAYOUT AND CONSTRUCTION DETAILS, REFER TO BECHTEL DWG 7604-M420-77 (C63) 29097 SH.74 & 7604-M420-80 (C21) 29097 SH.77.
2. FOR BILL OF MATERIAL SEE BECHTEL DWG 7604-M420-BM-C63 OR C21
3. THE ABOVE NUMBERING SEQUENCE APPLIES ONLY TO THE ITEMS ON THIS DWG
4. FOR NAMEPLATE ENGRAVING SCHEDULE SEE BECHTEL DWG. 7604-M420-78 (C63) & 7604-M420-81 (C21) 29097, SH 75A & 29097, SH 78A, 78B, 1B, C
5. FOR WOLFE & MANN FRONT VIEW & SECTIONS SEE 29097, SH 74 FOR C63 & 29097, SH.77 FOR C21

CLEAN LIQUID RADWASTE CONTROL PANEL (C63)

ITEM	INSTR NO.	DESCRIPTION
1		ALARM ANNUNCIATOR
2	PB-A83	ALARM ANNUNCIATOR "ACKNOWLEDGE" PUSHBUTTON
3	PB-R84	ALARM ANNUNCIATOR "RESET" PUSHBUTTON
4	PB-T83	ALARM ANNUNCIATOR "TEST" PUSHBUTTON
5		SPARE
6	TIC-9089	DEGASIFIER TANK PRESSURE INDICATOR/CONTROLLER
7	TIC-9286	DEGASIFIER TANK TEMPERATURE INDICATOR/CONTROLLER
8	LIC-9005	DEGASIFIER TANK LEVEL INDICATOR/CONTROLLER
9	TI-9300	DEGASIFIER TEMPERATURE INDICATOR
10	MIC-9310	DEGASIFIER NITROGEN INLET MANUAL LOADING STATION
11		SPARE
12	TIC-9296	DEGASIFIER PREHEATER TEMPERATURE INDICATOR/CONTROLLER
13	FI-9311	DEGASIFIER INLET FLOW INDICATOR
14	TI-9282	DEGASIFIER INLET TEMPERATURE INDICATOR
15	MIC-9309	RBCCW TO DEGASIFIER VENT COND. M/A STATION
16	MIC-9305	DEGASIFIER INLET MANUAL LOADING STATION
17		SPARE
18	TI-9019	DEGASIFIER PUMP TO PRIMARY DEMIN. TEMPERATURE INDICATOR
19	TI-9035	SECONDARY DEMIN. INLET TEMPERATURE INDICATOR
20	PI-9091	COOLANT WASTE RECEIVER TK. (T-14A) PRESSURE INDICATOR
21	PI-9092	COOLANT WASTE RECEIVER TK. (T-14B) PRESSURE INDICATOR
22	PI-9045	COOLANT WASTE MONITOR TK. (T-15A) PRESSURE INDICATOR
23	PI-9046	COOLANT WASTE MONITOR TK. (T-15B) PRESSURE INDICATOR
24	PI-9207	EQUIPMENT DRAIN TANK PRESSURE INDICATOR
25	LI-9988	COOLANT WASTE RECEIVER TK. (T-14A) LEVEL INDICATOR
26	LI-9989	COOLANT WASTE RECEIVER TK. (T-14B) LEVEL INDICATOR
27	LI-9990	COOLANT WASTE MONITOR TK. (T-15A) LEVEL INDICATOR
28	LI-9991	COOLANT WASTE MONITOR TK. (T-15B) LEVEL INDICATOR
29	LI-9241	RECOVERED BORIC ACID TK. (T-61A) LEVEL INDICATOR
30	LI-9243	RECOVERED BORIC ACID TK. (T-61B) LEVEL INDICATOR
31	FR-9050	SYSTEM DISCH. FLOW RECORDER (MULTI-CHANNEL)
32		SPARE
33	FI-9034	SECONDARY DEMIN INLET FLOW INDICATOR T23A
34	FI-9277	SECONDARY DEMIN INLET FLOW INDICATOR T23B
35	HS-9001	DEGASIFIER PUMP P-18A CONTROL SWITCH
36	HS-9203	DEGASIFIER PUMP P-18B CONTROL SWITCH
37	HS-9208	EQUIPMENT DRAIN TANK PUMP (P-94A) CONTROL SWITCH
38	HS-9211	EQUIPMENT DRAIN TANK PUMP (P-94B) CONTROL SWITCH
39	HS-9052A	PRIMARY DEMIN. (T-22A) INLET VALVE SEL. SWITCH
40	HS-9052B	PRIMARY DEMIN. (T-22B) INLET VALVE SEL. SWITCH
41	HS-9008	RADWASTE BYPASS VALVE SEL. SWITCH
42	HS-9052C	DEGASIFIER EFFLUENT BYPASS VALVE SEL. SWITCH
43	HS-9052D	FUTURE PRIMARY DEMIN. INLET VALVE SEL. SWITCH
44	HS-8069	PRIMARY DEMIN. TO COOL WASTE MONITOR TKS. VLV. SEL. SWITCH
45	HS-9184	COOLANT WASTE RECEIVER TK. (T-14A) INLET VALVE SEL. SWITCH
46	HS-9185	COOLANT WASTE RECEIVER TK. (T-14B) INLET VALVE SEL. SWITCH
47	HS-9186	COOLANT WASTE MONITOR TK. (T-15A) INLET VALVE SEL. SWITCH
48	HS-9187	COOLANT WASTE MONITOR TK. (T-15B) INLET VALVE SEL. SWITCH
49	HS-9027	COOLANT WASTE RECEIVER TK. (T-14A) DISCH. VALVE SEL. SWITCH
50	HS-9028	COOLANT WASTE RECEIVER TK. (T-14B) DISCH. VALVE SEL. SWITCH
51	HS-9040	COOLANT WASTE MONITOR TK. (T-15A) DISCH. VALVE SEL. SWITCH
52	HS-9041	COOLANT WASTE MONITOR TK. (T-15B) DISCH. VALVE SEL. SWITCH
53	HS-9222	RADWASTE COOLANT RECIRCULATION VALVE SEL. SWITCH
54	HS-9223	RADWASTE COOLANT RECIRCULATION VALVE SEL. SWITCH
55	HS-9224	RADWASTE COOLANT RECIRCULATION VALVE SEL. SWITCH
56	HS-9225	RADWASTE COOLANT RECIRCULATION VALVE SEL. SWITCH
57	HS-9053	COOLANT WASTE MONITOR TANKS RECIRC. PERMISSIVE SEL. SWITCH
58		
59		SPARE
60	FQI-9075	PRIMARY DEMIN. (T-22A) INLET FLOW TOTALIZING INTEGRATOR
61	FQI-9074	PRIMARY DEMIN. (T-22B) INLET FLOW TOTALIZING INTEGRATOR
62		SPARE
63	FQI-8034	SECONDARY DEMIN. (T-23) INLET FLOW TOTALIZING INTEGRATOR
64		SPARE
65	FQI-8050	CLEAN LIQUID RADWASTE SYSTEM DISCH FLOW TOTALIZING INTEGRATOR
66		SPARE
67	HS-9229	COOLANT WASTE RECEIVER TKS. PUMP (P-20) CONTROL SWITCH
68	HS-9233	COOLANT WASTE MONITOR TKS. PUMP (P-30) CONTROL SWITCH
69	HS-9049A	CLEAN LIQUID RADWASTE SYSTEM DISCH ISOLATION VLV. SEL. SWITCH
70	HS-9049B	CLEAN LIQUID RADWASTE SYSTEM DISCH ISOLATION VLV SEL. SWITCH
71	HS-9233	RECOVERED BORIC ACID STORAGE TKS PUMP CONTROL SWITCH
72		SPARE
73		SPARE
74		SPARE
75		SPARE
76		SPARE
77		SPARE
78	FI-9075	PRIMARY DEMIN T-22A INLET FLOW INDICATOR
79	FI-9074	PRIMARY DEMIN T-22B INLET FLOW INDICATOR
80	FQI-9277	SECONDARY DEMIN T23B INLET FLOW TOTALIZING INTEGRATOR
81	MIC-9049S	SYSTEM DISCHARGE ISOLATION VALVE MANUAL FLOW CONTROLLER
82	HS-9051	CLEAN LIQUID WASTE SM. SAMPLE PUMP ZP-13T FOR RM9049A,B
83	HS-9050	CLEAN LIQUID WASTE SYS FLUSH VALVES RV9050A (RV9050B FOR RM9049
84	HS-41A06	COOLANT WASTE MONITOR TANK MIXER MOTOR MT15B CONTROL SW.
85	HS-41A04	COOLANT WASTE MONITOR TANK MIXER MOTOR MT15A CONTROL SW.

HOT SHUTDOWN CONTROL PANEL (C21)

ITEM	INSTR. NO.	DESCRIPTION
1	TI-115	LOOP #1 COLD LEG TEMPERATURE INDICATOR
2	JI-001B	WIDE RANGE NUCLEAR INSTRUMENT INDICATOR
3	TI-125	LOOP #2 COLD LEG TEMPERATURE INDICATOR
4	JI-002B	WIDE RANGE NUCLEAR INSTRUMENT INDICATOR
5	LI-110X	PRESSURIZER LEVEL INDICATOR
6	PI-102AB	PRESSURIZER PRESSURE INDICATOR
7	TI-224B	LETDOWN TEMPERATURE INDICATOR
8	LI-110Y	PRESSURIZER LEVEL INDICATOR
9	PI-102BB	PRESSURIZER PRESSURE INDICATOR
10	LI-5282-1	CONDENSATE STORAGE TANK LEVEL INDICATOR
11	LI-1113 A2	STEAM GEN #1 LEVEL INDICATOR
12	PI-4223	STEAM GEN. #1 DUMP TO ATMOS. PRESSURE INDICATOR
13	FI-5271A-2	AUX FEEDWATER PUMP TO STM. GEN. #1 FLOW INDICATOR
14	LI-1123B2	STEAM GEN. #2 LEVEL INDICATOR
15	PI-4224	STEAM GEN. #2 DUMP TO ATMOS. PRESSURE INDICATOR
16		
17	PI-201B	LETDOWN HX. DISCH. PRESSURE INDICATOR
18	HIC-110 P&Q	LETDOWN MANUAL CONTROLLER STATION
19	HIC-4223	STEAM GEN. #1 DUMP TO ATMOS. M/A STATION
20	HIC-4224	STEAM GEN #2 DUMP TO ATMOS. M/A STATION
21	HIC-100 E & F	PRESSURIZER SPRAY M/A STATION
22	HS-4192B	AUX. STEAM FEEDWATER TURBINE SPEED CONTROL SWITCH
23	HS-100-4B	PRESSURIZER BACK-UP HEATER CONTROL SWITCH
24	HS-100-5B	PRESSURIZER BACK-UP HEATER CONTROL SWITCH
25	HS-2003B	BORIC ACID CHARGING PUMP (P-19A) CONTROL SWITCH
26	HS-2004B	BORIC ACID CHARGING PUMP (P-19B) CONTROL SWITCH
27	HS-2005B	BORIC ACID CHARGING PUMP (P-19C) CONTROL SWITCH
28	HS-100-6B	PRESSURIZER BACK-UP HEATER CONTROL SWITCH
29	HS-100-7B	PRESSURIZER BACK-UP HEATER CONTROL SWITCH
30	HS-5281B	STM GEN #1 F.W. PUMP (P-9A) CONTROL SWITCH
31	HS-5278B	AUX. F.W. PUMP TO STM. GEN. #1 EMERG OVERIDE/RESET SEL. SW
32	HS-4191B	STEAM GEN. #1 AUX F.W. TURBINE STOP VALVE SEL. SWITCH
33	HS-5288B	STM. GEN. #2 F.W. PUMP (P-9B) CONTROL SWITCH
34	HS-5279B	AUX. F.W. PUMP TO STM. GEN. #2 EMERG OVERIDE/RESET SEL. SW
35	HS-4188B	STM. GEN. #2 AUX F.W. TURBINE STOP VALVE SEL. SWITCH
36	HS-5218B	STM. GEN. AUX F.W TURB ENDS VALVE SEL. SWITCH
37	MIC-100F2	PRESSURIZER SPRAY M/A STATION
38	PT-103-1B	PRESSURIZER PRESSURE INDICATOR
39	PT-103B	PRESSURIZER PRESSURE INDICATOR
40	CS-110	LETDOWN FLOW CONTROL 60GPM (CS) SELECTOR SWITCH
41	HS-110-1B	LETDOWN FLOW CONTROL VALVE (CH110Q, BOTH, CH110P) SELECTOR SW.
42		
43	HS-4188C	AUX FW TURBINE STOP VALVE SELECTOR SWITCH
44	HS-5278C	AUTO AUX F.W. L.O. SELECTOR SW FACE
45	HS-4192D	AUTO AUX F.W. L.O. SELECTOR SW. FACE
46		
47	FI-5278B-2	AUX FEEDWATER PUMP TO STM GEN #2 FLOW INDICATOR
48	HS-5276C	AUX FEEDWATER TO STM GEN #1 CONT TRANSFER SELECTOR SW
49	HS-5279C	AUX FEEDWATER TO STM GEN #2 CONT TRANSFER SELECTOR SW
50	HIC-5276B	AUX FEEDWATER TO STM GEN #1 HAND INDICATING VALVE CONTROLER
51	HIC-5279B	AUX FEEDWATER TO STM GEN #2 HAND INDICATING VALVE CONTROLER
52		BLANK
53		BLANK

M/L 79-203 ITEM E54
M/L 79-203 ITEM E54
M/L 79203- E21

FSAR FIGURE
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion Energy
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
LOCAL CONTROL PANELS (C63) & (C21)

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C280062.MYB	
REVISED PER DC MP2-21-01038 & DUR 23-800022	DRAWING NO: 25203-28006	REV 12
DSGN JSJ	SCALE: N/A	UNLESS OTHERWISE NOTED SH 2

MBULOGIN\scottB4
6-MAR-2023 09:15
C280062.MYB

$FILE$
$DATE$

TEST SIGNAL INJECTION POINT

P-102A
CH."A" BYPASS HS-102A
BISTABLE 2397 psia ↑

P-102B
CH."B" BYPASS HS-102B
BISTABLE 2397 psia ↑

P-102C
CH."C" BYPASS HS-102C
BISTABLE 2397 psia ↑

P-102D
CH."D" BYPASS HS-102D
BISTABLE 2397 psia ↑

LOGIC MATRIX (AB, AC, AD, BC, BD, CD)
2/4

Z1

ENERGIZE 10 SEC TIMER 62A/DSS
(25203-32012 SH.44)

R.R.S. CH."X" (U+L/2)
BISTABLE 20% RX PWR. ↑

CH.Z1 BLOCK HS-4188E C05
CH.Z1 BLOCK HS-4188C C21

ENERGIZE 86Z1A/86Z1B (25203-28105 SH.23)

ENERGIZE 94A DSS
(25203-32012 SH.48)

ENERGIZE 94X MG-1
(25203-32012 SH.48A)

DE-ENERGIZE MG-1 CONTACTOR (25203-39055 SH.15A)

ENERGIZE 62/Z1 (25203-28105 SH.23)

DIVERSE RX TRIP ACTUATED

ENERGIZE 62/Z2 (25203-28105 SH.23)

Z2

ENERGIZE 94B DSS
(25203-32012 SH.48)

ENERGIZE 94X MG-2
(25203-32012 SH.48A)

DE-ENERGIZE MG-2 CONTACTOR (25203-39055 SH.15B)

BISTABLE 20% RX PWR. ↑
R.R.S. CH."Y" (U+L/2)
CH.Z2 BLOCK HS-4188D C21
CH.Z2 BLOCK HS-4188F C05

ENERGIZE 10 SEC TIMER 62B/DSS
(25203-32012 SH.45)

ENERGIZE 86Z2A/86Z2B (25203-28105 SH.23)

HS-102A, B, C, D
NORMAL BYPASS (C100)

HS-4188C, D, E, F
RESET START (C05) (C21)
PULL TO LOCK

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
PO# 0203 1927	4	11-20-01	INCORPORATE DCN'S DM2-00-1075-97 DM2-00-0025-00	AGC	GHH	TP	CAM
	3	9/25/99	INCORP. DCN DM2-00-1025-98	MEL	WBZ	JPP	JPP
	2	7/31/98	INCORP. DCN DM2-00-1527-97	WBZ	RTM	JPP	JPP

Q.A. CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE MILLSTONE UNIT 2
LOGIC DIAGRAM - ANTICIPATED TRANSIENT WITHOUT SCRAM (A.T.W.S.)
WATERFORD, CT.

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY	CHKD.	APP.	APP.
WBZ	MEL	JTK	DWC
DATE 10-25-88	DATE 11-16-88	DATE 11-17-88	DATE 11-18-88
SCALE N/A	DWG. NO. 80-082	25203-28105 SH.36	

START:

28129 SH 30

VENT VLV 8138A NOT CLOSED

(6) FS-8137 TWO EXHAUST FANS RUNNING

HS8137 ''START''

MOTOR OVERLOAD TRIP RESET

FAN F-16 START

EG90

C06 R

DAMPER HV8137 OPEN

ZS8137

C06 G

POWER AVAILABLE

STOP:

(A)

MOTOR OVERLOAD TRIP

ZS8138A DAMPER HV8138A ''CLOSED''

HS8137 ''STOP''

(6) MAIN EXHAUST FAN LOW FLOW FS-8137

FAN F-16 STOP

C06 G

DAMPER HV8137 CLOSE

ZS 8137

C06 R

POWER AVAILABLE

ALARMS:

(A) FAN OVERLOAD TRIP

ZAH E806

HS8137 (C06)

STOP START

MAINTAINED CONTACTS

CAD

NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

40 H.P. 480V

Q.A.

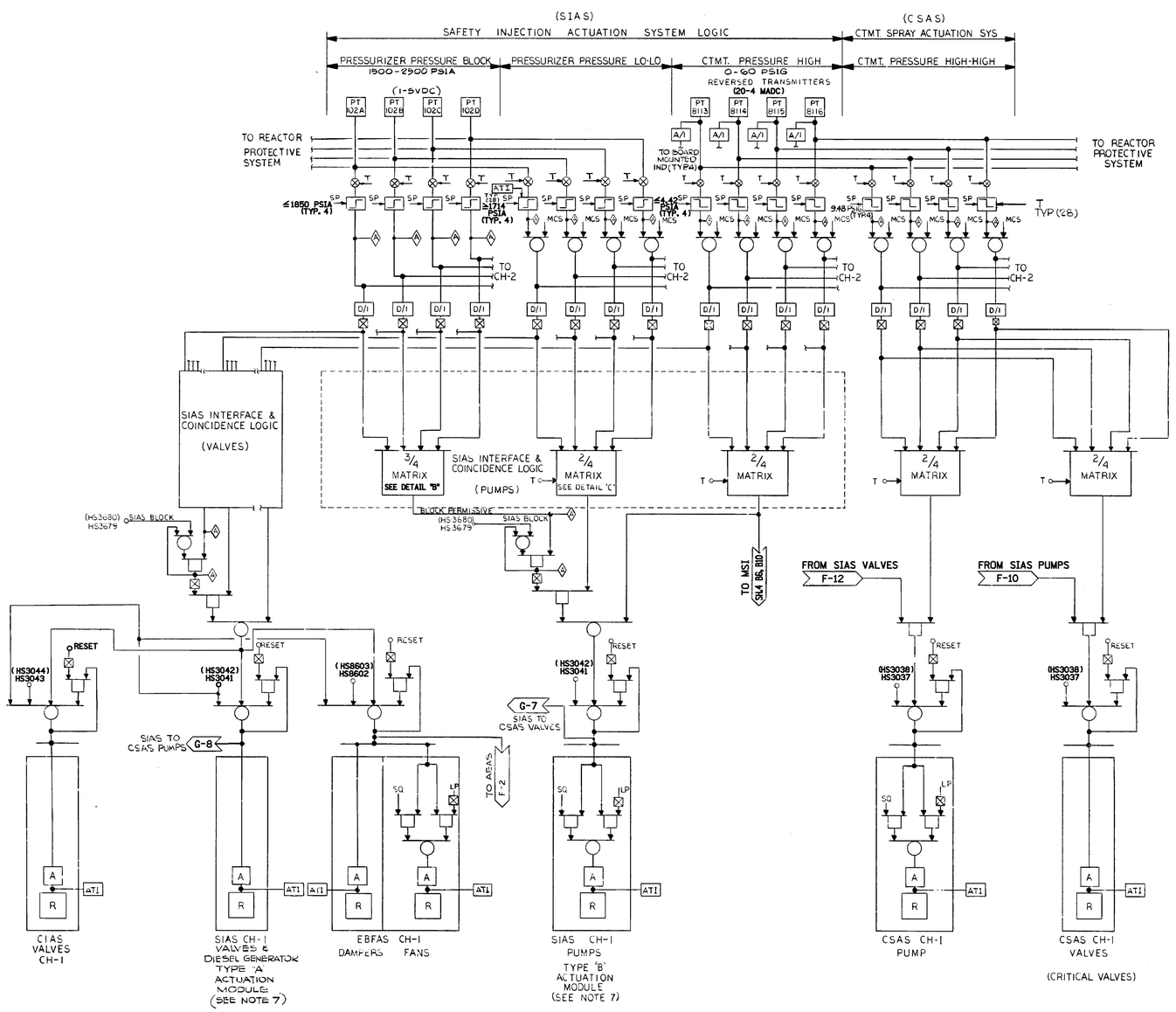
MF	NO.	DATE	REVISIONS	BY	CHK.	CORR	APP.
	8	11-28-01	INCORPORATE DCN DM2-00-0025-00	GHH	RJH	TP	CAM
—	7	7/11/83	AS BUILT PER DCR-M2-P-087-82 & PDCR-2-75-82	[illegible]	[illegible]	[illegible]	[illegible]

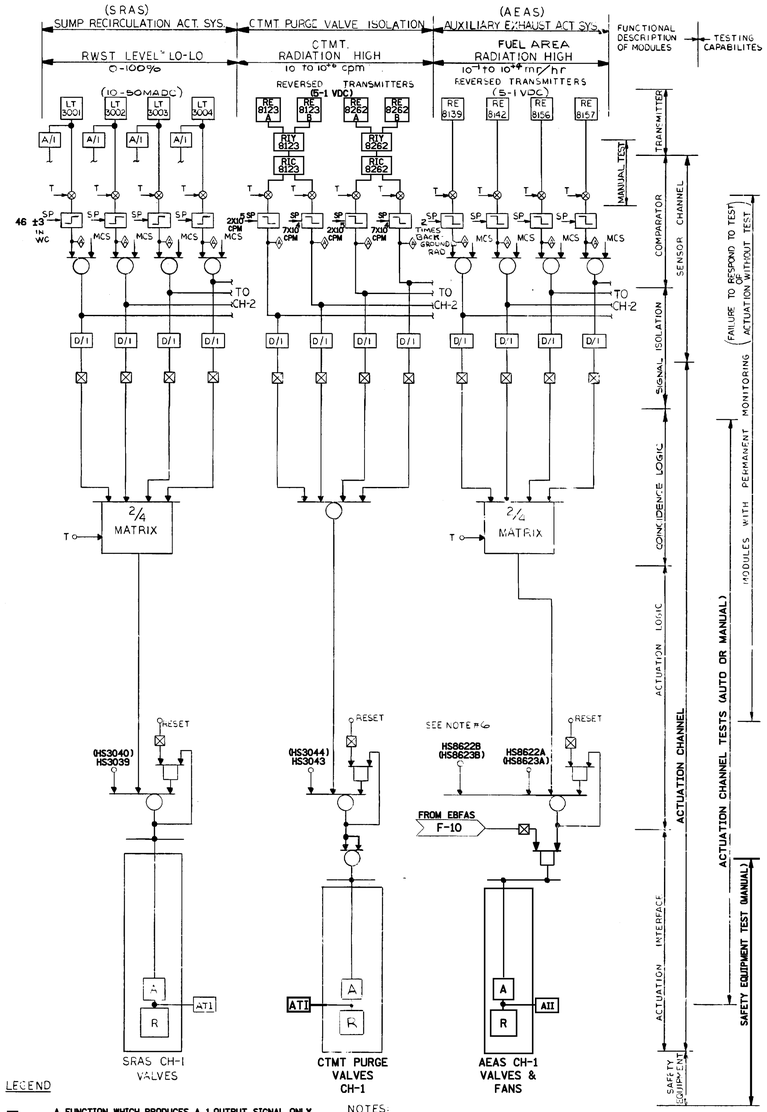
BECHTEL CORPORATION | JOB 7604
Gaithersburg, Md.
for THE MILLSTONE POINT COMPANY

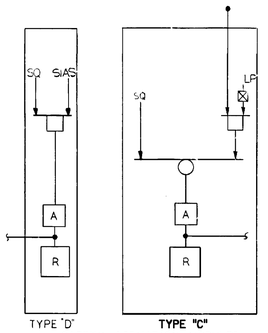
NORTHEAST UTILITIES SERVICE COMPANY
MILLSTONE NUCLEAR POWER STATION
UNIT NO. 2

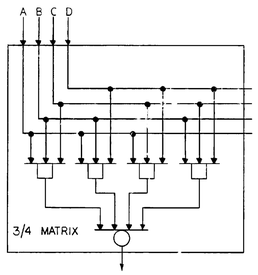
LOGIC DIAGRAM
RADWASTE AREA SUPPLY FAN

dwg. no. 25203-28129 | sh. 22

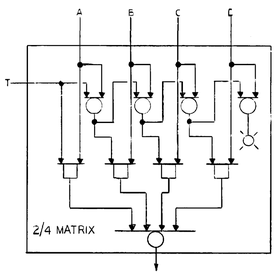








DETAIL "B"



DETAIL "C"

LEGEND

- ⌐ — A FUNCTION WHICH PRODUCES A 1 OUTPUT SIGNAL ONLY WHEN THE INPUT SIGNAL IS BELOW A PRESET VALUE.
- ¬ — A FUNCTION WHICH PRODUCES A 1 OUTPUT SIGNAL ONLY WHEN THE INPUT SIGNAL IS ABOVE A PRESET VALUE.
- A/I — ANALOG ISOLATION
- D/I — DIGITAL ISOLATION
- T — TEST SIGNAL INJECTION POINT (ANALOG)
- MCS — MATRIX CONVERSION SIGNAL KEY OPERATED (SEE NOTE 5)
- ATI — AUTO TEST INSTRUMENT
- SP — SET POINT
- LP — LOSS OF POWER
- ◇ — MAIN CONTROL BOARD ANNUNCIATOR
- SQ — SEQUENCER INPUT
- A — DRIVER AMPLIFIER
- R — RELAY

NOTES:

(1) ACTUATION CHANNEL-1 SHOWN-IDENTICAL FOR ACTUATION CHANNEL 2.

(2) FAIL-SAFE PORTION OF THE LOGIC ASSUMES THAT:
LOGIC "1" CORRESPONDS TO HIGH LEVEL SIGNAL.
LOGIC "0" CORRESPONDS TO ZERO SIGNAL.

(3) FOR LOGIC SYMBOL LEGEND SEE DRAWING No. 25203-28101 SH.2 &3.

(4) HAND SWITCH INSTRUMENT NUMBERS SHOWN IN PARENTHESES, ( ), APPLY TO CHANNEL 2; ALL OTHER HAND SWITCH INSTRUMENT NUMBERS APPLY TO CHANNEL 1. SWITCHES IDENTIFIED BY INSTR. No. ARE LOCATED ON UNIT CONTROL PANEL (C01).

(5) MC SIGNAL APPLIED TO THE MATRICES OR LOGIC MODULES SHALL CONVERT THEM AS FOLLOWS:
2/4 LOGIC TO 2/3
NO MORE THAN ONE SENSOR CHANNEL CAN BE BYPASSED IN EACH PARAMETER.

(6) PUSH BUTTONS LOCATED NEAR REFUELING MACHINE

(7) THE ESAS COMPRISES 4 TYPES OF ACTUATION INTERFACE MODULES:
TYPE 'A' FOR VALVE ACTUATION (NO SEQUENCING REQUIRED ON LOSS OF POWER).
TYPE 'B' FOR EQUIPMENT ACTUATION WITH SEQUENCING SAFETY ACTUATION SIGNAL AND LOSS OF POWER.
TYPE 'C' FOR EQUIPMENT ACTUATION AS PER TYPE B WITH ADDITIONAL SEQUENCING ON LOSS OF POWER ONLY.
TYPE 'D' FOR EQUIPMENT ACTUATION WITH SEQUENCING ON LOSS OF POWER ONLY. EXCEPT ON SIAS.
TYPE 'S' FOR EQUIPMENT ACTUATION WITH SEQUENCING ON LOSS OF POWER ONLY

(8) "0" INPUT TO ACTUATION MODULE MEANS NO SAFETY INITIATION.
"1" INPUT TO ACTUATION MODULE MEANS SAFETY INITIATION.

FOR SUMMARY TABLE SEE 25203-28150 SH. 2.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Q.A.

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REDRAWN FROM SAME NNECO DWG. No. REV.4

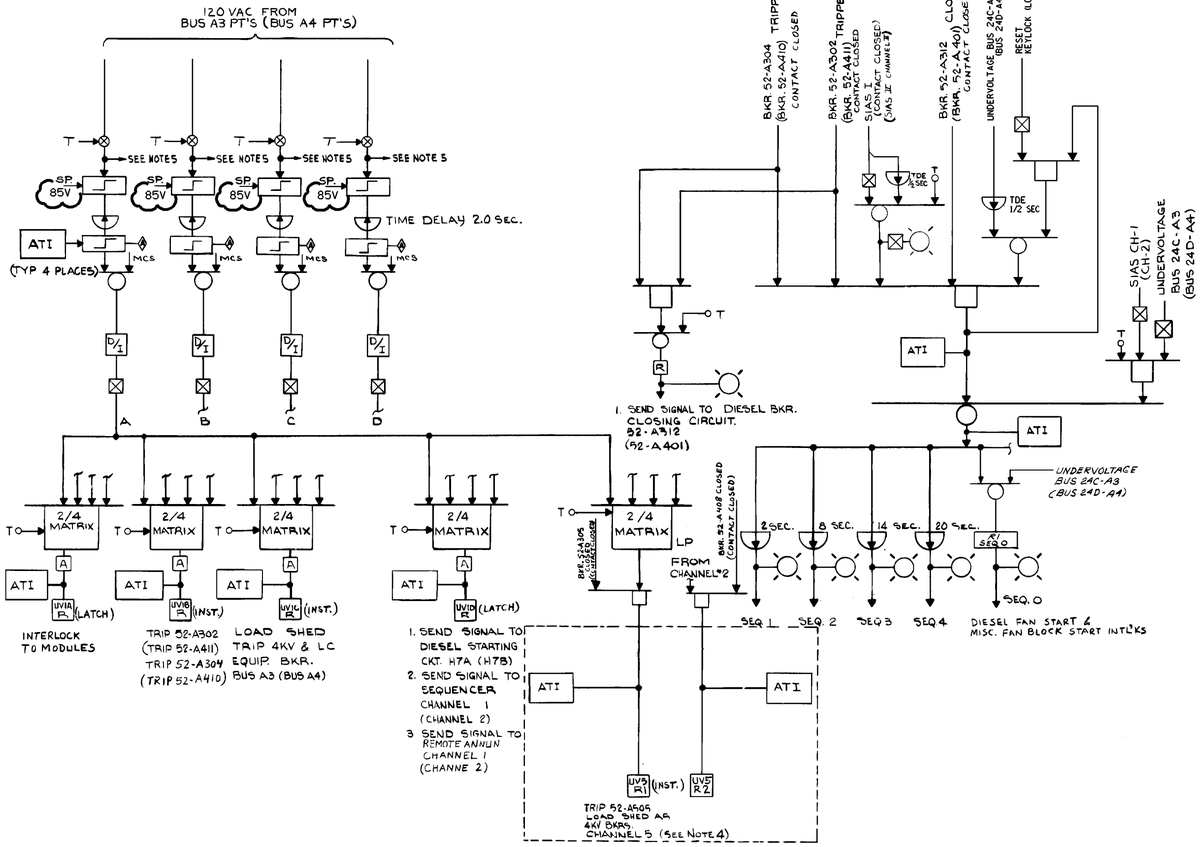
REVISIONS DURING CONSTRUCTION

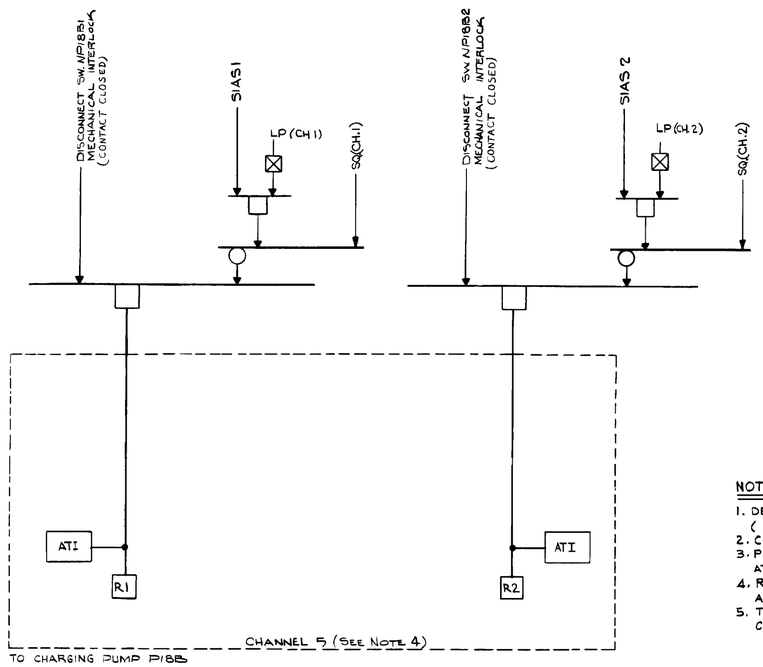
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
9	4-2-05	INCORP. DCN DM2-00-0024-04 & DM2-01-0024-04	JSJ	WBZ	—	DLW
8	12-1-04	INCORP. DCN DM2-00-0025-04 DM2-01-0025-04	DLW	WBZ	-	DED
PO# 0203 1927 7	2-4-02	INCORP. DCN DM2-00-0025-00	AGC	GHH	TP	CAM
6	8/4/99	INCORP. DCN's DM2-S-0010-96, 00-0687-98, 00-1025-98	WBZ	JSJ	JPP	JPP
92-067 5	10/20/95	INCORP. DCN DM2-S-0655-93 PER DCR M2-S-2428-93	JSJ	MEL RTM	JPP	JPP

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE: MILLSTONE UNIT No.2
ENGINEERED SAFETY LOGIC
WATERFORD, CT.

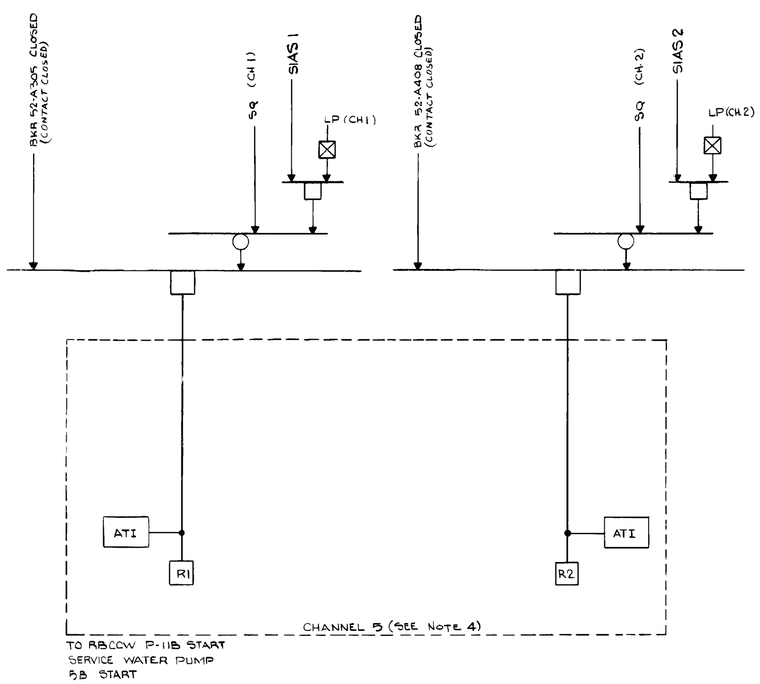
BY	CHKD.	APP.	APP.
JSJ			
DATE 6-19-95	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-28150 SH.1		
N/A			

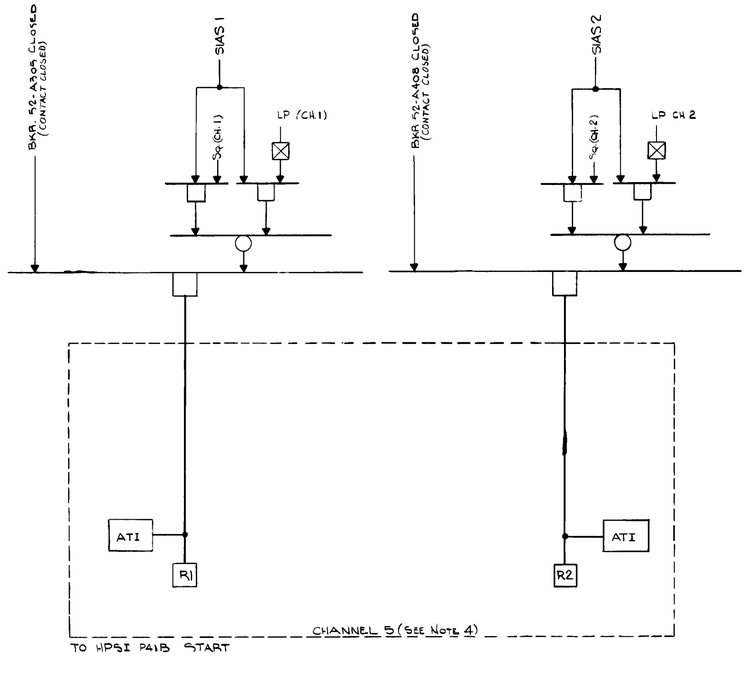


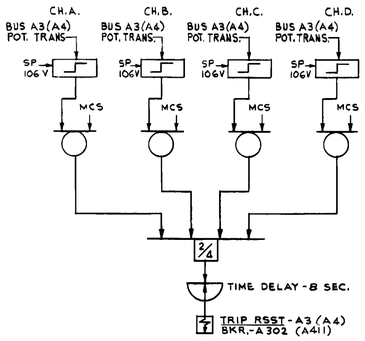


NOTES:

1. DESIGNATIONS APPEARING IN PARENTHESIS, ( ), APPLY TO CHANNEL 2.
2. CHANNEL 2 IS SIMILAR TO CHANNEL 1.
3. PUSHBUTTON WITH COVER TO BE LOCATED AT MAIN CONTROL BOARD C08.
4. R1 & R2 CONTACTS PARALLELED TO ACTUATE THE SAFEGUARD EQUIPMENT.
5. TO RSST UV LOGIC SENSING CIRCUIT.







FSAR FIGURE

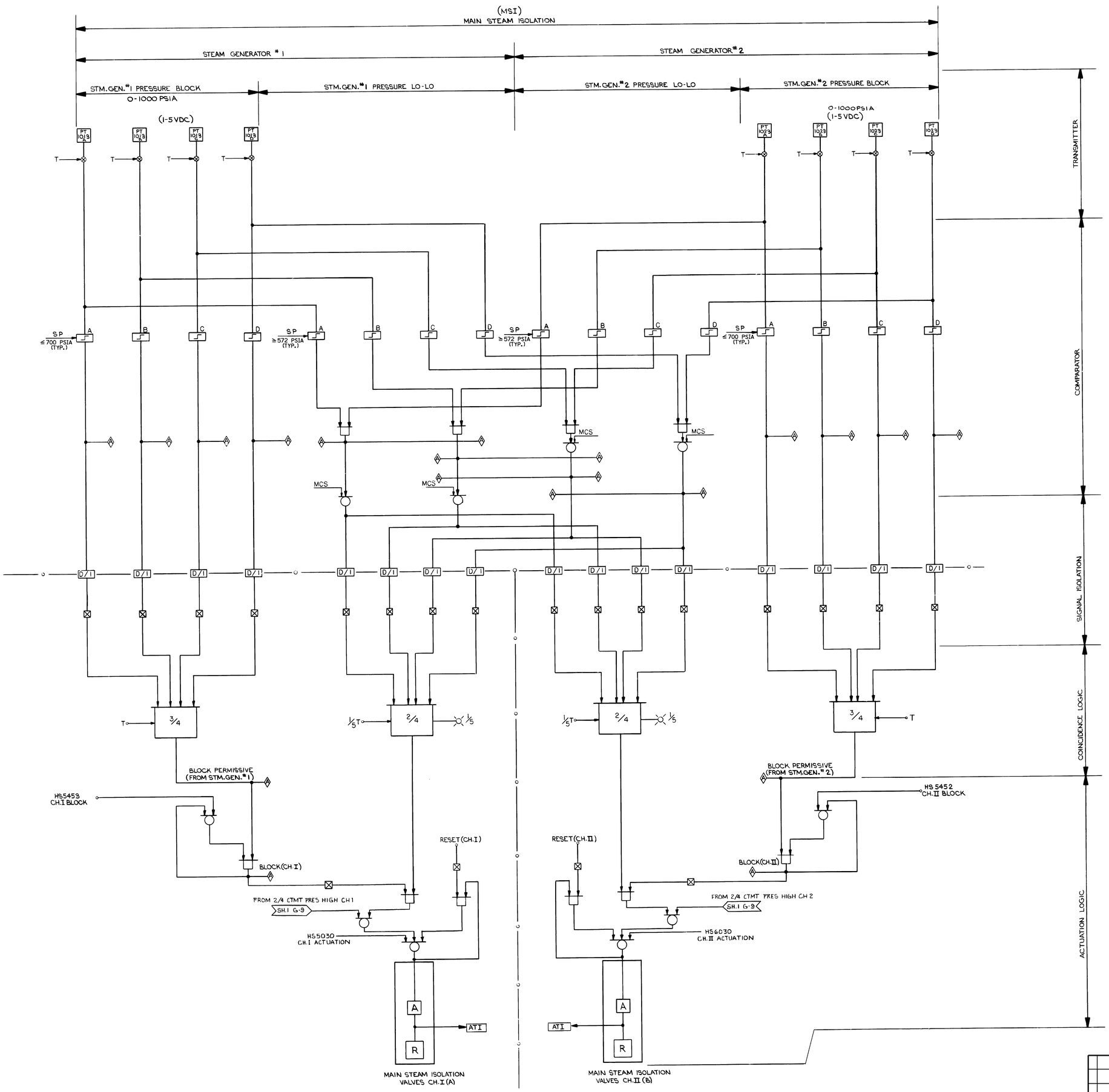
SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: ENGINEERED SAFETY LOGIC SEQUENCER & CHANNEL 5 OUTPUT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C28150JW8	
REVISED PER MSDUR 12-800141 & DUR 12-800381	DRAWING NO: 25203-28150	REV 6
DSGN DLW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3



FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Q.A.

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REDRAWN FROM SAME NNECO DWG. No. REV.2

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
PDR 0203 1527	4	11-05-01 INCORP. DCN DM2-00-0025-00	AGC	GHH	TP	CAM
	3	9/25/99 INCORP. DCN DM2-00-1025-98	WBZ	JSJ	JPP	JPP

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY LOGIC
WATERFORD, CT.

BY	CHKD.	APP.	APP.
WBZ/JSJ	JSJ	JPP	JPP
DATE 7-20-99	DATE 8/30/99	DATE 9/25/99	DATE 9/25/99

SCALE N/A

DWG. NO. 25203-28150 SH.4

N/A

EQUIPMENT NO.	DESCRIPTION	ACTUATION MODULE TYPE (NOTE 10)	ACTUATION MODULE NO. (NOTE 12)	ACTUATION GROUP (NOTE 12)	RELAY NO. (NOTE 12)	INCIDENT & ACTUATION CHANNEL																								DG SEQUENCE						P & ID	SCHEMATIC	NOTES
						SIAS				CSAS			CIAS			SRAS			EBFAS			AEAS			MSI			CPVIS			AFAIS			LOAD SHED			INCIDENT AND LOSS OF POWER		LOSS OF POWER ONLY					
						1	2	5	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	5	1	2	1	2	F			
A302	4KV MAIN FEEDER BKR. 52-A302	—	AM528	UV 2A	K528A, K540																													●							—	32002 SH.1	
A304	4KV TIE FEEDER BKR. 52-A304 TO BUS 24A	—	AM528	UV 2A	K528A																													●							—	32002 SH.11	
A305	4KV TIE FEEDER BKR. 52-A305 TO BUS 24E	—	—	SIAS 1A	K500																																				—	32002 SH.13	8
A312	4KV DIESEL GENERATOR FEEDER BKR. 52-312	—	—	SEQ501	K534A																													●							—	32041 SH.1	9
A401	4KV DIESEL GENERATOR FEEDER BKR. 52-401	—	—	SEQ601	K634A																														●						—	32041 SH.2	9
A408	4KV TIE FEEDER BKR. 52-A408 TO BUS 24E	—	—	SIAS 1B	K600																																				—	32002 SH.14	8
A410	4KV TIE FEEDER BKR. 52-A410 TO BUS 24B	—	AM628	UV 2B	K628A																														●						—	32002 SH.12	
A411	4KV MAIN FEEDER BKR. 52-A411	—	AM628	UV 2B	K628A, K640																														●						—	32002 SH.2	
A505	4KV MAIN FEEDER BKR. 52-A505	—	AM531, AM631	UV 5A, UV 5B	KS531A, KS631A																															●					—	32002 SH.15	
F13A	"A" CEDM COOLING FAN	D	AM522	SIAS 9A, SEQ501	K522A, K534H, K534J	●			I																														4, O		ST	26028 SH.1	32022 SH.5	
F13B	"B " CEDM COOLING FAN	D	AM622	SIAS 9B, SEQ601	K622A, K634H, K634L		●		I																															4, O		ST	26028 SH.1	32022 SH.6	
F13C	"C" CEDM COOLING FAN	D	AM522	SIAS 9A, SEQ501	K522A, K534H, K534J	●			I																														4, O		ST	26028 SH.1	32022 SH.7	
F14A	"A" CONTAINMENT AIR RECIRC. COOLER FAN	C	AM515	SIAS 1A, SEQ501	K515A, K534C	●			SH																													1, O		1, O		SH	26028 SH.1	32022 SH.1	
F14B	"B" CONTAINMENT AIR RECIRC. COOLER FAN	C	AM615	SIAS 1B, SEQ601	K615A, K634C		●		SH																														1, O		1, O		SH	26028 SH.1	32022 SH.3	
F14C	"C" CONTAINMENT AIR RECIRC. COOLER FAN	C	AM515	SIAS 1A, SEQ501	K515A, K534C	●			SH																														1, O		1, O		SH	26028 SH.1	32022 SH.2	
F14D	"D" CONTAINMENT AIR RECIRC. COOLER FAN	C	AM615	SIAS 1B, SEQ601	K615A, K634C		●		SH																														1, O		1, O		SH	26028 SH.1	32022 SH.4	
F15A	"A" ENGINEERED SAFEGUARD ROOM AIR RECIRC. FAN	B	AM520	SIAS 7A	K520A	●			ST																														4					26028 SH.4	32022 SH.17	
F15B	"B" ENGINEERED SAFEGUARD ROOM AIR RECIRC. FAN	B	AM620	SIAS 7B	K620A		●		ST																															4					26028 SH.4	32022 SH.18	
F20	FUEL HANDLING AREA HEATING AND VENTILATION FAN	A	AM513	AEAS 1A	K513D																	●		SP																		26029 SH.3	32021 SH.6	
F21A	CONTROL ROOM SUPPLY FAN "A"	S	—	SEQ501	K534B, K534H																																			4, O			ST	26027 SH.3	32023 SH.1	
F21B	CONTROL ROOM SUPPLY FAN "B"	S	—	SEQ601	K634B, K634H																																				4, O		ST	26027 SH.3	32023 SH.2	
F23	CONTAINMENT PURGE SUPPLY FAN	A	AM602	CIAS1B	K602A									●	SP																											26028 SH.1	32022 SH.10	
F25A	"A" ENCLOSURE BUILDING FILTRATION FAN	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508C, K513C, K534F														●		ST	●		ST														3					26028 SH.5	32022 SH.21	4
F25B	"B" ENCLOSURE BUILDING FILTRATION FAN	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608C, K613C, K634F															●	ST		●	ST															3					26028 SH.5	32022 SH.22	4
F31A	"A" CONTROL ROOM EXHAUST FAN	S	—	SEQ501	K534C, K534G																																				4, O			ST	26027 SH.3	32023 SH.3	
F31B	"B" CONTROL ROOM EXHAUST FAN	S	—	SEQ601	K634C, K634G																																				4, O			ST	26027 SH.3	32023 SH.4	
F32A	"A" CONTROL ROOM FILTER FAN	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		ST	●		ST														4					26027 SH.3	32023 SH.5	4
F32B	"B" CONTROL ROOM FILTER FAN	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K634K															●	ST		●	ST															4					26027 SH.3	32023 SH.6	4
F34A	MAIN EXHAUST FAN	A	AM504	CIAS 3A	K504A								●		SP																											26028 SH. 2	32022 SH.11 & 105	7, 15
F34B	MAIN EXHAUST FAN	A	AM504	CIAS 3A	K504A								●		SP																											26028 SH. 2	32022 SH.12 & 105	7, 15
F34C	MAIN EXHAUST FAN	A	AM504	CIAS 3A	K504A								●		SP																											26028 SH. 2	32022 SH.13 & 105	7, 15
F38A	"A" DIESEL GENERATOR VENT FAN	S	—	SEQ501	K534C, K534H																																				4, O			ST	26027 SH.1	32023 SH.7	
F38B	"B" DIESEL GENERATOR VENT FAN	S	—	SEQ601	K634C, K634H																																					4, O		ST	26027 SH.1	32023 SH.8	
F39A	CONTAINMENT RADIATION MONITORING FAN	A	AM502	CIAS 1A	K502C								●		SP																											26028 SH.2	32022 SH.14C	7
F39B	CONTAINMENT RADIATION MONITORING FAN	A	AM602	CIAS 1B	K602C									●	SP																											26028 SH.2	32022 SH.15C	7
F51	WEST VITAL SWITCHGEAR ROOM COOLING FAN	S	—	SEQ501	K534G, K534I																																				4, O			ST	26027 SH.1	32023 SH.42	
F52	EAST VITAL SWITCHGEAR ROOM COOLING FAN	S	—	SEQ601	K634G																																					4		ST	26027 SH.1	32023 SH.43	7
F54A	"A" DC VITAL SWITCHGEAR ROOM COOLING FAN	C	AM518	SIAS 5A	K535B	●			ST																														4		4		ST	26029 SH.1	32021 SH.19	
F54B	"B" DC VITAL SWITCHGEAR ROOM COOLING FAN	C	AM618	SIAS 5B	K635B		●		ST																															4		4		ST	26029 SH.1	32021 SH.20	
F112A	BATTERY ROOM VENT FAN A	D	AM514	SIAS 2A, SEQ501	K514A, K534G, K534J	●			I																																4, O			ST	26029 SH.1	32021 SH.1	
F112B	BATTERY ROOM VENT FAN B	D	AM614	SIAS 2B, SEQ601	K614A, K634G, K634L		●		I																																	4, O		ST	26029 SH.1	32021 SH.2	
F133	UPPER VITAL SWITCHGEAR ROOM COOLING FAN	S	—	SEQ601	K634H																																					4		ST	26027 SH.1	32023 SH.70	
F134	LOWER VITAL SWITCHGEAR ROOM COOLING FAN	S	—	SEQ501	K534K																																				4, O			ST	26027 SH.1	32023 SH.71	
F142	EAST VITAL SWITCHGEAR ROOM EXHAUST FAN	S	—	SEQ601	K634G																																					4		ST	26029 SH.1	32023 SH.80	
H5A	"A" STEAM GENERATOR FEED PUMP TURBINE	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	SP														26005 SH.2, 26002 SH.3		32012 SH.7 & 50	7, 11
H5B	"B" STEAM GENERATOR FEED PUMP TURBINE	A	AM532, AM632	MSI 1B, MSI 1A	K625B, K525B																				●	●	SP														26005 SH.2, 26002 SH.3		32012 SH.9 & 49	7, 11
H7A	"A" DIESEL GENERATOR	A	AM524, AM530	SIAS 11A, UV 4A	K524C, K530A	●			ST																														●			ST	26018 SH.2 & 5	32041 SH.3 & 6	
H7B	"B" DIESEL GENERATOR	A	AM624, AM630	SIAS 11B, UV 4B	K624C, K630A		●		ST																															●		ST	26018 SH.3 & 5	32041 SH.15 & 18	
L046A	MAG JACK M-G SET No.1	—	AM529	UV 3A	K529B																													●							—	32003 SH.30	
L046B	MAG JACK M-G SET No.2	—	AM629	UV 3B	K629B																														●						—	32003 SH.40	
L105	PROPORTIONAL HEATER GR1	—	AM529	UV 3A	K529B																													●							26014 SH.2	32007 SH.5	
L106	PROPORTIONAL HEATER GR2	—	AM629	UV 3B	K629B																														●						26014 SH.2	32007 SH.10	
P5A	SERVICE WATER PUMP "A"	C	AM515, AM529	SIAS 1A, UV 3A	K515C, K529A	●			ST																										●			1		1		ST	26008 SH.2	32013 SH.5	
P5B	SERVICE WATER PUMP "B"	C	AM515, AM531, AM615, AM631	SIAS 1A, SIAS 1B, UV 5A, UV 5B	K515B, K531A, K615B, K631A			●	ST																										●			1	1	1	1	ST	26008 SH.2	32013 SH.6	
P5C	SERVICE WATER PUMP "C"	C	AM615, AM629	SIAS 1B, UV 3B,	K615C, K629A,		●		ST																										●				1		1	ST	26008 SH.2	32013 SH.7	
P9A	STEAM GENERATOR No.1 AFW PUMP	—	AM529	UV 3A, SEQ501	K529B, K534I, K534L																							●		ST	●			4, O		4, O		P	26005 SH.3	32012 SH.11	
P9B	STEAM GENERATOR No.2 AFW PUMP	—	AM629	UV 3B, SEQ601	K629B, K634L, K634M																								●	ST		●			4, O		4, O		P	26005 SH.3	32012 SH.12	
P11A	RBCCW PUMP "A"	C	AM516, AM529	SIAS 3A, UV 3A,	K516A, K529A,	●			ST																										●			2		2		ST	26022 SH.1	32015 SH.1	
P11B	RBCCW PUMP "B"	C	AM516, AM531, AM616, AM631	SIAS 3A, SIAS 3B, UV 5A, UV 5B	K516B, K531A, K616B, K631A				● ST																											●	2	2	2	2	ST	26022 SH.1	32015 SH.2	
P11C	RBCCW PUMP "C"	C	AM616, AM629	SIAS 3B, UV 3B,	K616A, K629A,		●		ST																										●				2		2	ST	26022 SH.1	32015 SH.3	
P18A	CHARGING PUMP "A"	C	AM516	SIAS 3A	K516A, K534B	●			ST																													2, O		2, O		ST	26017 SH.1	32009 SH.40	
P18B	CHARGING PUMP "B"	C	AM516, AM616	SIAS 3A, SIAS 3B, SEQ501, SEQ601	K516C, K534J, K534E, K616C, K634J, K634E			●	ST																													2, O	2, O	2, O	2, O	ST	26017 SH.1	32009 SH.41 & 42	
P18C	CHARGING PUMP "C"	C	AM616	SIAS 3B	K616A, K634B		●		ST																															2, O		2, O	ST	26017 SH.1	32009 SH.43	
P19A	BORIC ACID PUMP "A"	C	AM625	SIAS 12B, SEQ601	K618A, K634J		●		ST																															2, O		2, O	ST	26017 SH.3	32009 SH.4	
P19B	BORIC ACID PUMP "B"	C	AM625	SIAS 12B, SEQ601	K618A, K634J		●		ST																															2, O		2, O	ST	26017 SH.3	32009 SH.5	
P33A	CONTAINMENT SUMP PUMP "A"	—	AM506	CIAS 5A	K506B								●		SP																											26024 SH.1	32030 SH.1 & 3	5
P33B	CONTAINMENT SUMP PUMP "B"	—	AM607	CIAS 6B	K607B									●	SP																											26024 SH.1	32030 SH.2 & 4	5
P41A	HPSI PUMP "A"	B	AM514, AM529	SIAS 2A, UV 3A	K514A, K529A	●			ST																										●			1					26015 SH.2	32008 SH.1	
P41B	HPSI PUMP "B"	B	AM514, AM531 AM614, AM631	SIAS 2A, SIAS 2B, UV 5A, UV 5B	K514B, K531A, K614B, K631A			●	ST																											●	1	1					26015 SH.2	32008 SH.3	
P41C	HPSI PUMP "C"	B	AM614, AM629	SIAS 2B, UV 3B	K614A, K629A		●		ST																										●				1					26015 SH.2	32008 SH.2	
P42A	LPSI PUMP "A"	B	AM517, AM512, AM529	SIAS4A, UV 3A, SRAS 2A	K517A, K512A, K529A	●			ST									SP																●			3					26015 SH.1	32008 SH.4	
P42B	LPSI PUMP "B"	B	AM612, AM617, AM629	SRAS 2B, SIAS 4B, UV 3B	K612A, K617A, K629A		●		ST										SP																●			3					26015 SH.1	32008 SH.5	

LEGEND:

F - FUNCTION
ST - START
SH - START HALF SPEED
SP - STOP
O - OPEN
C - CLOSE
T - TRIP
I - INHIBIT
P - PERMISSIVE FOR START

NOTES:

1. ACTUATED BY HIGH AIRBORNE GASEOUS OR PARTICULATE RADIOACTIVITY IN CONTAINMENT.
2. ACTUATED BY SEPARATE AND REDUNDANT SOLENOID PILOTS.
3. TYPE "S" ACTUATION MODULE IS A DIRECT SEQUENCER OUTPUT PROVIDING ACTUATION SIGNALS ON ANY LOSS OF POWER SEQUENCE.
4. EBFAS SIGNAL TO BE SEQUENCED ONLY AND INITIATED BY "B" MODULE. AEAS TO BE "A" MODULE ONLY.
5. ACTUATED BY AUXILIARY RELAY FROM ISOLATION VALVES.
6. ALSO ACTUATED BY CONTAINMENT AREA HIGH RADIATION.
7. ACTUATED BY AUXILIARY CONTACT.
8. TIE BREAKER TO EMERGENCY BUS 24E, PERMISSIVE.
9. DIESEL GENERATOR BREAKER CLOSE PERMISSIVE.
10. THE ESAS COMPRISES 5 TYPES OF ACTUATION INTERFACE MODULES:
    - TYPE "A" FOR EQUIPMENT ACTUATION (NO SEQUENCING REQUIRED ON LOSS OF POWER).
    - TYPE "B" FOR EQUIPMENT ACTUATION WITH SEQUENCING SAFETY ACTUATION SIGNAL AND LOSS OF POWER.
    - TYPE "C" FOR EQUIPMENT ACTUATION AS PER TYPE "B" WITH ADDITIONAL SEQUENCING ON LOSS OF POWER ONLY.
    - TYPE "D" FOR EQUIPMENT ACTUATION WITH SEQUENCING ON LOSS OF POWER ONLY, EXCEPT ON SIAS.
    - TYPE "S" FOR EQUIPMENT ACTUATION WITH SEQUENCING ON LOSS OF POWER ONLY.
11. EQUIPMENT NOT TO BE TESTED VIA ENGINEERED SAFETY FEATURE ACTUATION SYSTEM DURING POWER OPERATION.
12. SEE DRAWINGS 39047 SHEETS 9 THRU 16.
13. ACTUATED TO CLOSE VALVE BY AUXILIARY FEEDWATER ACTUATION INITIATION AND CONTAINMENT ISOLATION.
14. EQUIPMENT LISTED ON THIS DRAWING AFFECTS TECHNICAL REQUIREMENTS MANUAL SECTION 11.4. THE TRM SHOULD BE REVIEWED FOR IMPACT WHEN MAKING CHANGES TO THIS DRAWING.
15. NOT A CREDITED ESF ACTUATION. FOR ADDITIONAL INFORMATION GO TO FSAR.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

OPERATIONS CRITICAL

REDRAWN IN PART FROM NNECO DWG. NO. 25203-28150 SH.2 REV.13

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY LOGIC
ACTUATED EQUIPMENT TABULATION

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: c:\fms32\tmpplot\C2815024.DGN	
REVISED PER MSDUR 14-000100 & DUR 14-800316	DRAWING NO: 25203-28150	REV 8
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH 2A

EQUIPMENT NO.	DESCRIPTION	ACTUATION MODULE TYPE (NOTE 10)	ACTUATION MODULE NO. (NOTE 12)	ACTUATION GROUP (NOTE 12)	RELAY NO. (NOTE 12)	INCIDENT & ACTUATION CHANNEL: SIAS 1	SIAS 2	SIAS 5	SIAS F	CSAS 1	CSAS 2	CSAS F	CIAS 1	CIAS 2	CIAS F	SRAS 1	SRAS 2	SRAS F	EBFAS 1	EBFAS 2	EBFAS F	AEAS 1	AEAS 2	AEAS F	MSI 1	MSI 2	MSI F	CPVIS 1	CPVIS 2	CPVIS F	AFAIS 1	AFAIS 2	AFAIS F	LOAD SHED 1	LOAD SHED 2	LOAD SHED 5	DG SEQUENCE: INCIDENT AND LOSS OF POWER 1	INCIDENT AND LOSS OF POWER 2	LOSS OF POWER ONLY 1	LOSS OF POWER ONLY 2	LOSS OF POWER ONLY F	P & ID	SCHEMATIC	NOTES
P43A	CONTAINMENT SPRAY PUMP "A"	B	AM509, AM529	CSAS 1A, UV 3A	K509A, K529B					●		ST																				●			3						26015 SH.1	32008 SH.6	
P43B	CONTAINMENT SPRAY PUMP "B"	B	AM609, AM629	CSAS 1B, UV 3B	K609A, K629B						●	ST																				●				3					26015 SH.1	32008 SH.7	
P122A	VITAL SWITCHGEAR ROOM CHILLED WATER PUMP "A"	C	AM518	SIAS 5A	K535A	●			ST																										4			4		ST	26027 SH.2	32023 SH.44	
P122B	VITAL SWITCHGEAR ROOM CHILLED WATER PUMP "B"	C	AM618	SIAS 5B	K635A		●		ST																											4			4	ST	26027 SH.2	32023 SH.45	
X61A	"A" ENCLOSURE BUILDING FILTRATION SYSTEM DEHUMID. HEATER	B, A	AM508, AM513	EBFAS 1A, AEAS 1A	K508C, K513C														●		ST	●		ST											3						26028 SH.5	32022 SH.41	7
X61B	"B" ENCLOSURE BUILDING FILTRATION SYSTEM DEHUMID. HEATER	B, A	AM608, AM613	EBFAS 1B, AEAS 1B	K608C, K613C															●	ST		●	ST												3					26028 SH.5	32022 SH.42	7
X169A	"A" DC SWITCHGEAR ROOM CHILLER	C	AM518	SIAS 5A	K535A	●			ST																										4			4		ST	26027 SH.2	32023 SH.58	7
X169B	"B" DC SWITCHGEAR ROOM CHILLER	C	AM618	SIAS 5B	K635A		●		ST																											4			4	ST	26027 SH.2	32023 SH.59	7
2-AC-1	PURGE INLET SHUTOFF DAMPER HV-8050	A	AM507	CIAS 6A	K507A								●		C																											26028 SH.1	32022 SH.29	
2-AC-3	ENCLOSURE BUILDING PURGE SHUTOFF DAMPER HV-8081	A	AM502	CIAS 1A	K502A								●		O																											26028 SH.1	32022 SH.46	
2-AC-4	CONTAINMENT PURGE INLET DAMPER HV-8080	A	AM501	PURGE VLV 1A	K501A																							●		C											26028 SH.1	32022 SH.45	1, 11
2-AC-5	CONTAINMENT PURGE OUTLET SHUTOFF DAMPER HV-8082	A	AM601	PURGE VLV 1B	K601A																								●	C										26028 SH.1	32022 SH.28	1, 11
2-AC-6	CONTAINMENT PURGE OUTLET DAMPER HV-8125	A	AM601	PURGE VLV 1B	K601A																								●	C										26028 SH.1	32022 SH.44	1, 11
2-AC-7	CONTAINMENT PURGE OUTLET DAMPER HV-8126	A	AM501	PURGE VLV 1A	K501A																							●		C											26028 SH.1	32022 SH.47	1, 11
2-AC-8	ENCLOSURE BUILDING PURGE OUTLET DAMPER HV-8127	A	AM502	CIAS 1A	K502B								●		O																											26028 SH.2	32022 SH.48	
2-AC-11	CONTAINMENT & ENCLOSURE BUILDING PURGE OUTLET DAMPER HV-8128	A	AM602	CIAS 1B	K602D									●	C																											26028 SH.2	32022 SH.36	
2-AC-12	CONTAINMENT AIR SAMPLE VALVE HV-8122	A	AM602	CIAS 1B	K602C									●	C																											26028 SH.2	32022 SH.27, 104	7
2-AC-15	CONTAINMENT AIR SAMPLE VALVE HV-8124	A	AM502	CIAS 1A	K502D								●		C																											26028 SH.2	32022 SH.43	
2-AC-20	CONTAINMENT AIR SAMPLE VALVE HV-8656	A	AM602	CIAS 1B	K602D									●	C																											26028 SH.2	32022 SH.73	
2-AC-47	CONTAINMENT AIR SAMPLE VALVE HV-8121	A	AM502	CIAS 1A	K502C								●		C																											26028 SH.2	32022 SH.26, 104	7
2-AC-527	CONTAINMENT SAMPLE FAN F39B ISOLATION VALVE HV-8262AB1	A	AM602	CIAS 1B	K602C									●	SP																											26028 SH.2	32022 SH.15C	7
2-AC-528	CONTAINMENT SAMPLE FAN F39B ISOLATION VALVE HV-8262AB2	A	AM602	CIAS 1B	K602C									●	SP																											26028 SH.2	32022 SH.15C	7
2-AC-529	CONTAINMENT SAMPLE FAN F39A ISOLATION VALVE HV-8123AB1	A	AM502	CIAS 1A	K502C								●		SP																											26028 SH.2	32022 SH.14C	7
2-AC-530	CONTAINMENT SAMPLE FAN F39A ISOLATION VALVE HV-8123AB2	A	AM502	CIAS 1A	K502C								●		SP																											26028 SH.2	32022 SH.14C	7
2-CH-089	REG. HEAT EXCH. DISCHARGE TO LETDOWN HX HV-2525	A	AM604	CIAS 3B	K604A									●	C																											26017 SH.2	32009 SH.39	11
2-CH-196	VCT BYPASS VALVE	C	AM515	SIAS 1A	K515C	●			C																										1			1		C	26017 SH.3	32009 SH.54	
2-CH-198	R.C.P. BLEED-OFF	A	AM505	CIAS 4A	K505A								●		C																											26017 SH.2	32009 SH.30	11
2-CH-501	VCT DISCHARGE VALVE	A	AM523	SIAS 10A	K523C	●			C																																	26017 SH.1	32009 SH.6	11
2-CH-505	RCP CONTROLLED BLEED-OFF ISOLATION VALVE	A	AM505	CIAS 4A	K505A								●		C																											26017 SH.2	32009 SH.30	11
2-CH-506	RCP CONTROLLED BLEED-OFF ISOLATION VALVE	A	AM604	CIAS 3B	K604B									●	C																											26017 SH.2	32009 SH.31	11
2-CH-508	"B" BORIC ACID STORAGE TANK OUTLET GRAVITY FEED VALVE	A	AM519	SIAS 6A	K519A, K519B	●			O																																	26017 SH.3	32009 SH.8	
2-CH-509	"A" BORIC ACID STORAGE TANK OUTLET GRAVITY FEED VALVE	A	AM519	SIAS 6A	K519A, K519B	●			O																																	26017 SH.3	32009 SH.9	
2-CH-510	BORIC ACID PUMP "A" RECIRC. VALVE	A	AM619	SIAS 6B	K619A		●		C																																	26017 SH.3	32009 SH.24	
2-CH-511	BORIC ACID PUMP "B" RECIRC. VALVE	A	AM619	SIAS 6B	K619A		●		C																																	26017 SH.3	32009 SH.25	
2-CH-512	VOLUME CONTROL MAKEUP VALVE	A	AM519	SIAS 6A	K519B	●			C																																	26017 SH.3	32009 SH.21	
2-CH-514	BORIC ACID TO CHARG. LINE VALVE	C	AM615	SIAS 1B	K615D		●		O																												1			1	O	26017 SH.3	32009 SH.10	
2-CH-515	REACTOR COOLANT LETDOWN VALVE	A	AM523	SIAS 10A	K523B	●			C																																	26017 SH.2	32009 SH.33	11
2-CH-516	REACTOR COOLANT LETDOWN VALVE	A	AM604	CIAS 3B	K604A									●	C																											26017 SH.2	32009 SH.34	11
2-CHW-3	CHILLED WATER X84A SUPPLY CONTROL VALVE HV-8846	C	AM518	SIAS 5A	K535A	●			O																										4			4		O	26027 SH.2	32023 SH.47	7
2-CHW-11	CHILLED WATER SUPPLY HEADER CROSS TIE VALVE HV-8847	C	AM518	SIAS 5A	K535A	●			C																										4			4		C	26027 SH.2	32023 SH.47	7
2-CHW-12	CHILLED WATER SUPPLY HEADER CROSS TIE VALVE HV-8854	C	AM618	SIAS 5B	K635A		●		C																											4			4	C	26027 SH.2	32023 SH.48	7
2-CHW-13	CHILLED WATER RETURN HEADER CROSS TIE VALVE HV-8850	C	AM518	SIAS 5A	K535A	●			C																										4			4		C	26027 SH.2	32023 SH.47	7
2-CHW-14	CHILLED WATER RETURN HEADER CROSS TIE VALVE HV-8855	C	AM618	SIAS 5B	K635A		●		C																											4			4	C	26027 SH.2	32023 SH.48	7
2-CHW-33	CHILLED WATER X84B SUPPLY CONTROL VALVE HV-8853	C	AM618	SIAS 5B	K635A		●		O																											4			4	O	26027 SH.2	32023 SH.48	7
2-CS-4.1A	CONTAINMENT SPRAY CONTROL VALVE HV-3021	A	AM510	CSAS 2A	K510A					●		O																															26015 SH.1	32008 SH.29	
2-CS-4.1B	CONTAINMENT SPRAY CONTROL VALVE HV-3022	A	AM610	CSAS 2B	K610A						●	O																															26015 SH.1	32008 SH.30	
2-CS-16.1A	CONTAINMENT SUMP RECIRC. STOP VALVE HV-3008	A	AM511	SRAS 1A	K511B											●		O																									26015 SH.2	32008 SH.25	
2-CS-16.1B	CONTAINMENT SUMP RECIRC. STOP VALVE HV-3009	A	AM611	SRAS 1B	K611B												●	O																									26015 SH.2	32008 SH.26	
2-EB-40	ENCLOSURE BUILDING PLENUM ISOLATION DAMPER HV-8063	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613B, K608B															●	O		●	C																			26028 SH.5	32022 SH.53	
2-EB-41	ENCLOSURE BUILDING VENT. SUCTION ISOLATION DAMPER HV-8070	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613C, K608C															●	O		●	O																			26028 SH.5	32022 SH.50	
2-EB-42	E.B. FILT. DISCHARGE HV-8073	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613C, K608C															●	O		●	O																			26028 SH.5	32022 SH.49	7
2-EB-50	ENCLOSURE BUILDING PLENUM ISOLATION DAMPER HV-8153	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513B, K508B														●		O	●		C																			26028 SH.5	32022 SH.37	
2-EB-51	ENCLOSURE BUILDING VENT. SUCTION ISOLATION DAMPER HV-8074	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513C, K508B														●		O	●		O																			26028 SH.5	32022 SH.35	
2-EB-52	E.B. FILT. DISCHARGE HV-8079	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513C, K508C														●		O	●		O																			26028 SH.5	32022 SH.34	7
2-EB-55	STEAM JET AIR INJECTOR MOV HV-8654	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613E, K608E															●	C		●	C																			26028 SH.5	32022 SH.65	
2-EB-56	STEAM JET AIR INJECTOR MOV HV-8695	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513E, K508E														●		C	●		C																			26028 SH.5	32022 SH.66	
2-EB-60	FUEL HANDLING AREA VENT. TO PLENUM HV-8143	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513A, K508D														●		C	●		O																			26028 SH.5	32022 SH.52	
2-EB-61	FUEL HANDLING AREA VENT. TO PLENUM HV-8062	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613A, K608D															●	C		●	O																			26028 SH.5	32022 SH.39	
2-EB-72	CONTAINMENT CLEAN-UP DAMPER HV-8650	A	AM613, AM608	AEAS 1B, EBFAS 1B	K613B, K608D															●	C		●	C																			26028 SH.5	32022 SH.67	
2-EB-73	CONTAINMENT CLEAN-UP DAMPER HV-8651	A	AM513, AM508	AEAS 1A, EBFAS 1A	K513B, K508D														●		C	●		C																			26028 SH.5	32022 SH.68	
2-EB-88	CONTAINMENT H_2 MONITORING SAMPLE ISOLATION VALVE HV-8150	A	AM504	CIAS 3A	K504B								●		C																											26028 SH.2	32022 SH.30	
2-EB-89	CONTAINMENT H_2 MONITORING SAMPLE ISOLATION VALVE HV-8151	A	AM605	CIAS 4B	K605B									●	C																											26028 SH.2	32022 SH.31	
2-EB-91	H_2 PURGE ISOLATION VALVE HV-8380	A	AM605	CIAS 4B	K605B									●	C																											26028 SH.3	28500 SH.863 32022 SH.62	6, 7
2-EB-92	H_2 PURGE ISOLATION VALVE HV-8379	A	AM504	CIAS 3A	K504A								●		C																											26028 SH.3	28500 SH.862 32022 SH.59, 61	6, 7

NOTE:
FOR NOTES AND LEGEND SEE DWG. 28150 SH.2A.

OPERATIONS CRITICAL

Q.A.

REVISIONS DURING CONSTRUCTION

REDRAWN IN PART FROM NNECO DWG. NO. 25203-28150 SH.2 REV.13

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
6	2-28-05	INCORP. DCN DM2-00-0024-04	JCV	WBZ	-	DLW
5	11-24-04	INCORP. DCN DM2-00-0025-04	DLW	WBZ	-	DED
4	3-9-02	INCORP. DCN DM2-00-0113-01	JSJ	DLW	N/A	DCZ

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY LOGIC
ACTUATED EQUIPMENT TABULATION
WATERFORD, CT.

BY	CHKD.	APP.	APP.
WBZ	—	JPP	JPP
DATE 4-18-98	DATE —	DATE 3/8/99	DATE 3/8/99

SCALE N/A
DWG. NO. 25203-28150 SH.2B

EQUIPMENT NO.	DESCRIPTION	ACTUATION MODULE TYPE (NOTE 10)	ACTUATION MODULE NO. (NOTE 12)	ACTUATION GROUP (NOTE 12)	RELAY NO. (NOTE 12)	INCIDENT & ACTUATION CHANNEL: SIAS 1	SIAS 2	SIAS 5	SIAS F	CSAS 1	CSAS 2	CSAS F	CIAS 1	CIAS 2	CIAS F	SRAS 1	SRAS 2	SRAS F	EBFAS 1	EBFAS 2	EBFAS F	AEAS 1	AEAS 2	AEAS F	MSI 1	MSI 2	MSI F	CPVIS 1	CPVIS 2	CPVIS F	AFAIS 1	AFAIS 2	AFAIS F	LOAD SHED 1	LOAD SHED 2	LOAD SHED 5	DG SEQUENCE: INCIDENT AND LOSS OF POWER 1	INCIDENT AND LOSS OF POWER 2	LOSS OF POWER ONLY 1	LOSS OF POWER ONLY 2	LOSS OF POWER ONLY F	P & ID	SCHEMATIC	NOTES
2-EB-99	H_2 PURGE ISOLATION VALVE HV-8377	A	AM504	CIAS 3A	K504A								●		C																											26028 SH.3	28500 SH.862 32022 SH.59	6, 7
2-EB-100	H_2 PURGE ISOLATION VALVE HV-8378	A	AM605	CIAS 4B	K605B									●	C																											26028 SH.3	28500 SH.863 32022 SH.60, 62	6, 7
2-FW-5A	STEAM GENERATOR #1 FEEDWATER ISOLATION VALVE HV-5419	A	AM526	MSI 1A	K526B																				●		C															26005 SH.2	32012 SH.25	11
2-FW-5B	STEAM GENERATOR #2 FEEDWATER ISOLATION VALVE HV-5420	A	AM626	MSI 1B	K626B																					●	C															26005 SH.2	32012 SH.26	11
2-FW-38A	STEAM GENERATOR #1 FEED PUMP 1A DISCHARGE VALVE HV-5245	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-38B	STEAM GENERATOR #2 FEED PUMP 1B DISCHARGE VALVE HV-5247	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-41A	STEAM GENERATOR #1 FEEDWATER REG. BYPASS VALVE HV-5215	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-41B	STEAM GENERATOR #2 FEEDWATER REG. BYPASS VALVE HV-5216	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-42A	FEEDWATER BLOCK VALVE TO STEAM GENERATOR #1 HV-5263	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-42B	FEEDWATER BLOCK VALVE TO STEAM GENERATOR #2 HV-5264	A	AM532, AM632	MSI 1A, MSI 1B	K525B, K625B																				●	●	C															26005 SH.2	32012 SH.49, 50	7, 11
2-FW-43A	STEAM GENERATOR #1 AFW REG. VALVE HV-5276	——																													●	●	O								26005 SH.3	32012 SH.21	
2-FW-43B	STEAM GENERATOR #2 AFW REG. VALVE HV-5279	——																													●	●	O								26005 SH.3	32012 SH.22	
2-FW-51A	STEAM GENERATOR #1 FEEDWATER REG. VALVE FV-5268	A	AM532, AM526, AM632	MSI 1A, MSI 1B	K525B, K526B, K625B																				●	●	C															26005 SH.2	32012 SH.31A, 50	7, 11
2-FW-51B	STEAM GENERATOR #2 FEEDWATER REG. VALVE FV-5269	A	AM532, AM626, AM632	MSI 1B, MSI 1A	K625B, K626B, K525B																				●	●	C															26005 SH.2	32012 SH.32A, 49	7, 11
2-GR-11.1	WASTE GAS SURGE TANK INLET VALVE HV-9125	A	AM607	CIAS 6B	K607A									●	C																											26021 SH.2	32026 SH.1	
2-GR-11.2	WASTE GAS SURGE TANK INLET VALVE HV-9126	A	AM506	CIAS 5A	K506A								●		C																											26021 SH.2	32026 SH.2	
2-HV-106	E.S. ROOM AIR SUPPLY VALVE HV-8133	A	AM608	EBFAS 1B	K608A															●	C																					26029 SH.2	32021 SH.13	
2-HV-107	E.S. ROOM AIR SUPPLY VALVE HV-8247	A	AM508	EBFAS 1A	K508A														●		C																					26029 SH.2	32021 SH.12	
2-HV-116	E.S. ROOM AIR SUPPLY VALVE HV-8249	A	AM508	EBFAS 1A	K508A														●		C																					26029 SH.2	32021 SH.14	
2-HV-117	E.S. ROOM AIR SUPPLY VALVE HV-8248	A	AM608	EBFAS 1B	K608A															●	C																					26029 SH.2	32021 SH.15	
2-HV-165	FUEL HANDLING DAMPER HV-8141	A	AM613	AEAS 1B	K613D																		●	C																		26029 SH.3	32021 SH.8	
2-HV-170	SPENT FUEL ISOLATION DAMPER HV-8326	A	AM613	AEAS 1B	K613A, K613D																		●	C																		26029 SH.3	32021 SH.10	
2-HV-171	FUEL HANDLING SYSTEM DISCHARGE DAMPER HV-8275	A	AM513	AEAS 1A	K513A, K513D																																					26029 SH.3	32021 SH.9	
2-HV-202	CONTROL ROOM AIR CONDITIONING SYSTEM HV-8361	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		C	●		C																	4					26027 SH.3	32023 SH.37	4, 7
2-HV-203A	CONTROL ROOM SUPPLY FAN F21A DAMPER HV-8009	S	—	SEQ501	K534B, K534H																																					4, 0		0	26027 SH.3	32023 SH.1	7
2-HV-203B	CONTROL ROOM SUPPLY FAN F21B DAMPER HV-8010	S	—	SEQ601	K634B, K634H																																						4, 0	0	26027 SH.3	32023 SH.2	7
2-HV-206A	CONTROL ROOM EXHAUST FAN F31A DAMPER HV-8001	S	—	SEQ501	K534C, K534G																																					4, 0		0	26027 SH.3	32023 SH.3	7
2-HV-206B	CONTROL ROOM EXHAUST FAN F31B DAMPER HV-8000	S	—	SEQ601	K634C, K634G																																						4, 0	0	26027 SH.3	32023 SH.4	7
2-HV-207	CONTROL ROOM AIR CONDITIONING SYSTEM HV-8002	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		C	●		C																	4					26027 SH.3	32023 SH.37	4, 7
2-HV-208	CONTROL ROOM AIR CONDITIONING SYSTEM TV-8003B	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		C	●		C																	4					26027 SH.3	32023 SH.37	4, 7
2-HV-210	CONTROL ROOM AIR CONDITIONING SYSTEM HV-8005	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K634K															●	O		●	O																	4					26027 SH.3	32023 SH.37, 39	4, 7
2-HV-211	CONTROL ROOM AIR CONDITIONING SYSTEM HV-8004	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		C	●		C																	4					26027 SH.3	32023 SH.37A	4, 7
2-HV-212A	CONTROL ROOM FILTER FAN F-32A DAMPER HV-8006	B, A	AM508, AM513	EBFAS 1A, AEAS 1A, SEQ501	K508E, K513E, K534K														●		O	●		O																	4					26027 SH.3	32023 SH.5	4, 7
2-HV-212B	CONTROL ROOM FILTER FAN F-32B DAMPER HV-8007	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K534K															●	O		●	O																	4					26027 SH.3	32023 SH.6	4, 7
2-HV-255A	D/G VENT FAN F38A DAMPER HV-8296	S	—	SEQ501	K534C, K534H																																					4, 0		0	26027 SH.1	32023 SH.7	7
2-HV-255B	D/G VENT FAN F38B DAMPER HV-8298	S	—	SEQ601	K634C, K634H																																						4, 0	0	26027 SH.1	32023 SH.8	7
2-HV-274	VITAL SWITCHGEAR ROOM FAN F52 DAMPER HV-8843	S	—	SEQ601	K634C, K634G																																						4	0	26027 SH.1	32023 SH.43	7
2-HV-495	CONTROL ROOM AIR CONDITIONING SYSTEM	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K534K															●	C		●	C																	4					26027 SH.3	32023 SH.39	4, 7
2-HV-496	CONTROL ROOM AIR CONDITIONING SYSTEM	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K534K															●	C		●	C																	4					26027 SH.3	32023 SH.39	4, 7
2-HV-497	CONTROL ROOM AIR CONDITIONING SYSTEM	B, A	AM608, AM613	EBFAS 1B, AEAS 1B, SEQ601	K608E, K613E, K534K															●	C		●	C																	4					26027 SH.3	32023 SH.39A	4, 7
2-LRR-43.1	PDT PUMP DISCHARGE VALVE HV-9015	A	AM607	CIAS 6B	K607A									●	C																											26020 SH.5	32025 SH.10	
2-LRR-43.2	PDT PUMP DISCHARGE VALVE HV-9016	A	AM506	CIAS 5A	K506A								●		C																											26020 SH.5	32025 SH.11	
2-LRR-61.1	PDT SAMPLE SHUTOFF VALVE HV-9230	A	AM606	CIAS 5B	K606A									●	C																											26020 SH.5	32025 SH.17	
2-MS-64A	STEAM GENERATOR #1 ISOLATION VALVE HV-4217	A	AM532, AM632	MSI 1A, MSI 1B	K525A, K625A																				●	●	C															26002 SH.1	32020 SH.18, 19	2, 11
2-MS-64B	STEAM GENERATOR #2 ISOLATION VALVE HV-4221	A	AM532, AM632	MSI 1A, MSI 1B	K525A, K625A																				●	●	C															26002 SH.1	32020 SH.21, 20	2, 11
2-MS-65A	STEAM GENERATOR #1 ISOLATION VALVE BYPASS HV-4218	A	AM526	MSI 1A	K526A																				●		C															26002 SH.1	32020 SH.1	11
2-MS-65B	STEAM GENERATOR #2 ISOLATION VALVE BYPASS HV-4222	A	AM626	MSI 1B	K626A																					●	C															26002 SH.1	32020 SH.2	11
2-MS-191A	STEAM GENERATOR #1 SAMPLE LINE VALVE HV-4250	A	AM503	CIAS 2A	K503A								●		C																											26002 SH.2	32020 SH.11	
2-MS-191B	STEAM GENERATOR #2 SAMPLE LINE VALVE HV-4251	A	AM603	CIAS 2B	K603A									●	C																											26002 SH.2	32020 SH.12	
2-MS-220A	STEAM GENERATOR #1 BLOWDOWN VALVE HV-4246	A	AM503	CIAS 2A	K503A								●		C																●		C								26002 SH.2	32020 SH.14	13
2-MS-220B	STEAM GENERATOR #2 BLOWDOWN VALVE HV-4248	A	AM603	CIAS 2B	K603A									●	C																	●	C								26002 SH.2	32020 SH.15	13
2-MS-265B	MAIN STEAM LEG LOW PT. DRAIN HV-4193	A	AM526	MSI 1A	K526B																				●		C															26002 SH.1	32020 SH.45	
2-MS-266B	MAIN STEAM LEG LOW PT. DRAIN HV-4209	A	AM626	MSI 1B	K626B																					●	C															26002 SH.1	32020 SH.46	
2-PMW-43	QUENCH TANK MAKEUP VALVE HV-7311	A	AM506	CIAS 5A	K506B								●		C																											26014 SH.2	32007 SH.31	
2-RB-8.1A	SFP HX A-RBCCW OUTLET HV-6315	A	AM524	SIAS 1A	K524B	●			C																																26022 SH.2	32015 SH.12	
2-RB-8.1B	SFP HX A-RBCCW OUTLET HV-6316	A	AM624	SIAS 1B	K624B		●		C																																26022 SH.2	32015 SH.13	
2-RB-13.1A	SHUTDOWN HEAT EXCHANGER A CW HV-6050	A	AM511	SRAS 1A	K511A											●		O																								26022 SH.2	32015 SH.10	
2-RB-13.1B	SHUTDOWN HEAT EXCHANGER B CW HV-6055	A	AM611	SRAS 1B	K611A												●	O																								26022 SH.2	32015 SH.11	
2-RB-28.3A	CONTAINMENT AIR RECIRC. COOL. A OUTLET HV-6080	A	AM524	SIAS 1A	K524B	●			O																																26022 SH.5	32015 SH.31	
2-RB-28.3B	CONTAINMENT AIR RECIRC. COOL. B OUTLET HV-6084	A	AM624	SIAS 1B	K624B		●		O																																26022 SH.5	32015 SH.33	
2-RB-28.3C	CONTAINMENT AIR RECIRC. COOL. C OUTLET HV-6088	A	AM524	SIAS 1A	K524A	●			O																																26022 SH.5	32015 SH.35	
2-RB-28.3D	CONTAINMENT AIR RECIRC. COOL. D OUTLET HV-6092	A	AM624	SIAS 1B	K624A		●		O																																26022 SH.5	32015 SH.37	
2-RB-68.1A	ES ROOM RBCCW AIR COOLER HV-6731	A	AM524	SIAS 1A	K524A	●			O																																26002 SH.2	32015 SH.8	
2-RB-68.1B	ES ROOM RBCCW AIR COOLER HV-6735	A	AM624	SIAS 1B	K624A		●		O																																26002 SH.2	32015 SH.9	
2-RB-210	DEGASIFIER EFFL. COOL. LETDOWN SHUTOFF HV-6739	A	AM624	SIAS 1B	K624C		●		C																																26022 SH.5	32015 SH.38	
2-RB-402	LETDOWN HEAT EXCHANGER OUTLET	A	AM604	CIAS 3B	K604A									●	C																											26022 SH.5	32009 SH.39	11

NOTE:
FOR NOTES AND LEGEND SEE DWG. 28150 SH.2A.

OPERATIONS CRITICAL

Q.A.

REDRAWN IN PART FROM NNECO DWG. NO. 25203-28150 SH.2 REV.13

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
2	1-14-00	INCORP. DCN DM2-00-0025-00	JSJ	REB	JPP	JPP
1	3/8/99	INCORP. DCN DM2-00-0678-98	WBZ	RTM	JPP	JPP

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION
WATERFORD, CT.

BY	CHKD.	APP.	APP.
WBZ	——	JPP	JPP
DATE 4-18-98	DATE ——	DATE 3/8/99	DATE 3/8/99

SCALE N/A
DWG. NO. 25203-28150 SH.2C

EQUIPMENT NO.	DESCRIPTION	ACTUATION MODULE TYPE (NOTE 10)	ACTUATION MODULE NO. (NOTE 12)	ACTUATION GROUP (NOTE 12)	RELAY NO. (NOTE 12)	INCIDENT & ACTUATION CHANNEL																														DG SEQUENCE						P & ID	SCHEMATIC	NOTES
						SIAS				CSAS			CIAS			SRAS			EBFAS			AEAS			MSI			CPVIS			AFAIS			LOAD SHED			INCIDENT AND LOSS OF POWER		LOSS OF POWER ONLY					
						1	2	5	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	F	1	2	5	1	2	1	2	F			
2-RC-001	R.C. HOT LEG SAMPLING VALVE HV-1064	A	AM606	CIAS 5B	K606A								●		C																										26014 SH.1	32007 SH.30	
2-RC-002	PRESSURIZER SURGE LINE SAMPLING VALVE HV-1062	A	AM606	CIAS 5B	K606B								●		C																										26014 SH.1	32007 SH.29	
2-RC-003	PRESSURIZER STEAM SAMPLING VALVE HV-1060	A	AM606	CIAS 5B	K606B								●		C																										26014 SH.2	32007 SH.28	
2-RC-45	R.C. SAMPLING ISOLATION VALVE HV-7690	A	AM507	CIAS 6A	K507A								●		C																										26014 SH.2	32027 SH.3	
2-SI-312	N_2 TO SI TANKS SHUT-OFF VALVE HV-7312	A	AM607	CIAS 6B	K607B									●	C																										26015 SH.3	32008 SH.49	
2-SI-614	#1 SI TANK OUTLET VALVE	A	AM523	SIAS 10A	K523B, K523C	●			O																															26015 SH.3	32008 SH.12	11
2-SI-615	LPSI HEADER TO LOOP 1A INJECTION VALVE	A	AM521	SIAS 8A	K521A, K521E	●			O																																26015 SH.1	32008 SH.21	
2-SI-616	"B" HPSI HEADER TO LOOP 1A INJECTION VALVE	A	AM621	SIAS 8B	K621B, K621D		●		O																																26015 SH.2	32008 SH.13	
2-SI-617	"A" HPSI HEADER TO LOOP 1A INJECTION VALVE	A	AM521	SIAS 8A	K521B, K521D	●			O																																26015 SH.2	32008 SH.17	
2-SI-618	LOOP 1A CHECK LEAKOFF DRAIN VALVE	A	AM522	SIAS 9A	K522A	●			C																																26015 SH.3	32008 SH.56	
2-SI-624	#2 SI TANK OUTLET VALVE	A	AM523	SIAS 10A	K523D	●			O																																26015 SH.3	32008 SH.38	11
2-SI-625	LPSI HEADER TO LOOP 1B INJECTION VALVE	A	AM521	SIAS 8A	K521A, K521E	●			O																																26015 SH.1	32008 SH.22	
2-SI-626	"B" HPSI HEADER TO LOOP 1B INJECTION VALVE	A	AM621	SIAS 8B	K621B, K621D		●		O																																26015 SH.2	32008 SH.14	
2-SI-627	"A" HPSI HEADER TO LOOP 1B INJECTION VALVE	A	AM521	SIAS 8A	K521B, K521D	●			O																																26015 SH.2	32008 SH.18	
2-SI-628	LOOP 1B CHECK LEAKOFF DRAIN VALVE	A	AM522	SIAS 9A	K522A	●			C																																26015 SH.3	32008 SH.57	
2-SI-634	#3 SI TANK OUTLET VALVE	A	AM623	SIAS 10B	K623B		●		O																																26015 SH.3	32008 SH.39	11
2-SI-635	LPSI HEADER TO LOOP 2A INJECTION VALVE	A	AM621	SIAS 8B	K621A, K621E		●		O																																26015 SH.1	32008 SH.23	
2-SI-636	"B" HPSI HEADER TO LOOP 2A INJECTION VALVE	A	AM621	SIAS 8B	K621C, K621D		●		O																																26015 SH.2	32008 SH.15	
2-SI-637	"A" HPSI HEADER TO LOOP 2A INJECTION VALVE	A	AM521	SIAS 8A	K521C, K521D	●			O																																26015 SH.2	32008 SH.19	
2-SI-638	LOOP 2A CHECK LEAKOFF DRAIN VALVE	A	AM622	SIAS 9B	K622A		●		C																																26015 SH.3	32008 SH.58	
2-SI-644	#4 SI TANK OUTLET VALVE	A	AM623	SIAS 10B	K623C		●		O																																26015 SH.3	32008 SH.40	11
2-SI-645	LPSI HEADER TO LOOP 2B INJECTION VALVE	A	AM621	SIAS 8B	K621A, K621E		●		O																																26015 SH.1	32008 SH.24	
2-SI-646	"B" HPSI HEADER TO LOOP 2B INJECTION VALVE	A	AM621	SIAS 8B	K621C, K621D		●		O																																26015 SH.2	32008 SH.16	
2-SI-647	"A" HPSI HEADER TO LOOP 2B INJECTION VALVE	A	AM521	SIAS 8A	K521C, K521D	●			O																																26015 SH.2	32008 SH.20	
2-SI-648	LOOP 2B CHECK LEAKOFF DRAIN VALVE	A	AM622	SIAS 9B	K622A		●		C																																26015 SH.3	32008 SH.59	
2-SI-659	SI SUMP RECIRC. HEADER SHUT-OFF VALVE	A	AM511	SRAS 1A	K511A											●		C																							26015 SH.1	32008 SH.54	
2-SI-660	SI SUMP RECIRC. HEADER SHUT-OFF VALVE	A	AM611	SRAS 1B	K611A												●	C																							26015 SH.1	32008 SH.55	
2-SSP-16.1	CONTAINMENT DRAIN SUMP ISOLATION VALVE HV-9151	A	AM607	CIAS 6B	K607B									●	C																										26024 SH.1	32030 SH.4	
2-SSP-16.2	CONTAINMENT DRAIN SUMP ISOLATION VALVE HV-9150	A	AM506	CIAS 5A	K506B								●		C																										26024 SH.2	32030 SH.3	
2-SW-3.2A	TBCCW HEAT EXCHANGER VALVE HV6438	A, S	AM623	SIAS 10B, SEQ601	K623A, K634F		●		C																													3		3	C	26008 SH.2	32013 SH.19	11
2-SW-3.2B	TBCCW HEAT EXCHANGER VALVE HV6439	A, S	AM523	SIAS 10A, SEQ501	K523A, K534F	●			C																												3		3	C	26008 SH.2	32013 SH.20	11
2-SW-8.1A	RBCCW HEAT EXCHANGER VALVE TV-6308	A	AM523	SIAS 10A	K523A	●			O																																26008 SH.2	32013 SH.40	11
2-SW-8.1B	RBCCW HEAT EXCHANGER VALVE TV-6307	A	AM523, AM623	SIAS 10A, SIAS 10B	K523D, K623D	●	●		O																																26008 SH.2	32013 SH.41, 43	2, 11
2-SW-8.1C	RBCCW HEAT EXCHANGER VALVE TV-6306	A	AM623	SIAS 10B	K623A		●		O																																26008 SH.2	32013 SH.42	11

NOTE:
FOR NOTES AND LEGEND SEE DWG. 28150 SH.2A.

OPERATIONS CRITICAL

Q.A.

REVISIONS DURING CONSTRUCTION

REDRAWN IN PART FROM NNECO DWG. NO. 25203-28150 SH.2 REV.13

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

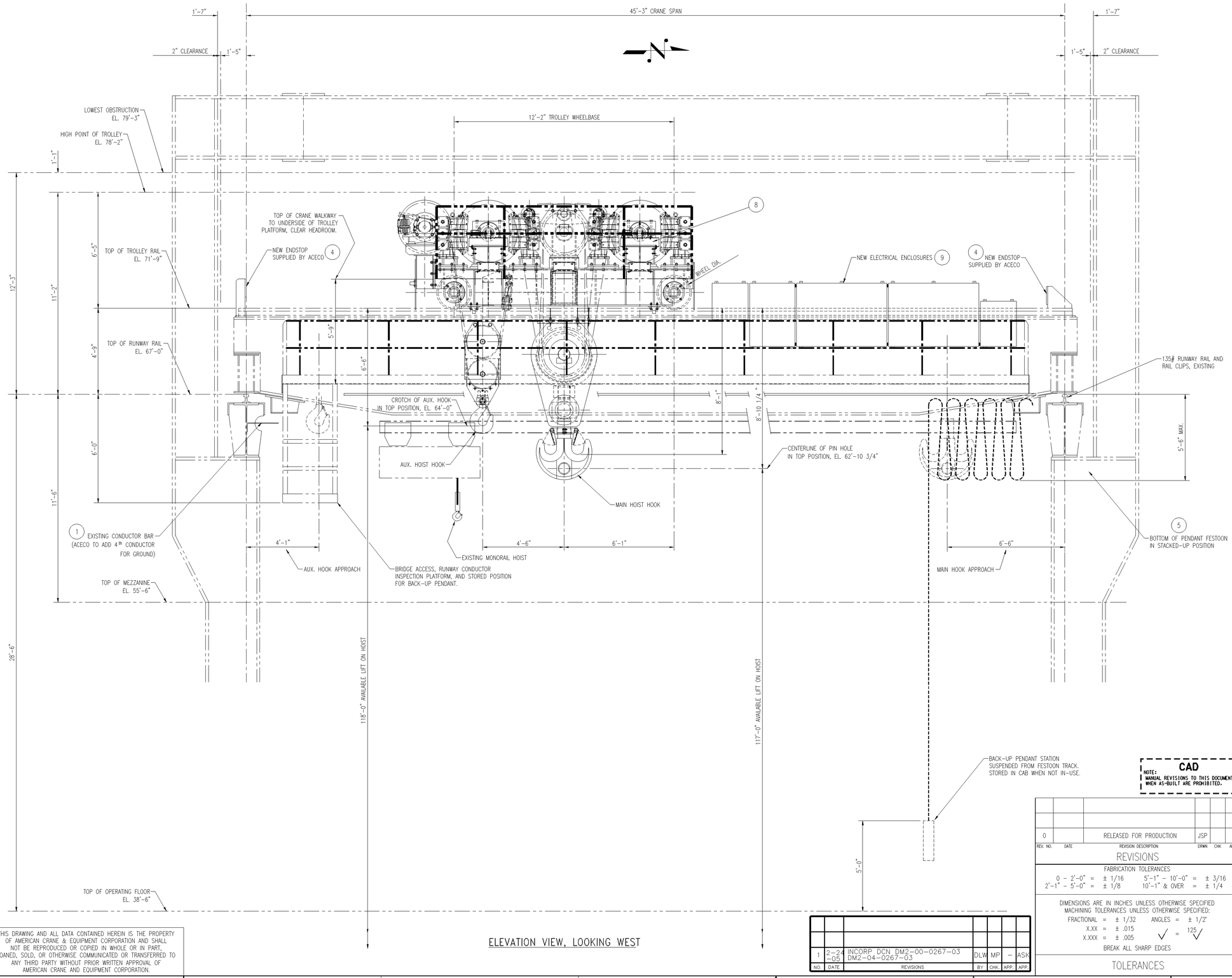
FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
2	1-14-00	INCORP. DCN DM2-00-0025-00	JSJ	REB	JPP	JPP
1	3/8/99	INCORP. DCN DM2-00-0678-98	WBZ	RTM	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION
WATERFORD, CT.

BY	WBZ	CHK'D.	—	APP.	JPP
DATE	4-18-98	DATE	—	DATE	3/8/99
SCALE	N/A	DWG. NO.	25203-28150 SH.2D		
	N/A				



ELEVATION VIEW, LOOKING WEST

CRANE DATA

TROLLEY NO.	2762
CAPACITIES	MAIN HOIST: DRL = MCL = 125 TONS AUX. HOIST: DRL = MCL = 15 TONS TROLLEY: DRL = MCL = 125 TONS BRIDGE: DRL & MCL TO BE DETERMINED AFTER FURTHER ANALYSIS
MAIN HOIST SPEED	0.25–5 F.P.M. VARIABLE (CRITICAL LOAD) 0.25–7.5 F.P.M. VARIABLE (NON CRITICAL LOAD)
AUX. HOIST SPEED	0.25–20 F.P.M. VARIABLE (CRITICAL LOAD) 0.25–31 F.P.M. VARIABLE (NON CRITICAL LOAD)
TROLLEY SPEED	1–40 F.P.M. VARIABLE
BRIDGE SPEED	1.25–50 F.P.M. VARIABLE
TROLLEY WEIGHT	110,000 LBS. (TO BE CONFIRMED)
BRIDGE WEIGHT (UPGRADED)	85,000 LBS. (TO BE CONFIRMED)
MAXIMUM BRIDGE WHEEL LOAD (WITHOUT IMPACT)	88,100 LBS. @ 125 TON CAPACITY (TO BE CONFIRMED)
APPLICATION	NUREG 0554, CMAA CLASS A, INDOORS REFER TO DIR–20365–ME–001 FOR OTHER DATA & ENVIRONMENTAL CONDITIONS
MONORAIL HOIST DATA – EXISTING	RATED CAPACITY: 5 TONS LIFT: 85 FT. HOIST SPEED: 30 F.P.M. MAX, 5 STEP TROLLEY SPEED: 100 F.P.M. MAX, 5 STEP
POWER SUPPLY	460–3–60 AC
OPERATOR CONTROLS	RADIO REMOTE CONTROL WITH BACK–UP PENDANT STATION
CONTROL ENCLOSURE	NEMA 4 (FOR NEW ENCLOSURES)
RUNWAY COLLECTORS	CONDUCTOR BAR – 3 EXISTING ACECO TO ADD 4TH FOR GROUND
RUNWAY LENGTH	200 FT, APPROX.
SURFACE PREPARATION	SSPC–SP6
PAINT PRIMER:	2.5–6.0 MILS DFT, KEELER & LONG KL3200, WHITE
PAINT FINISH:	1.5–2.5 MILS DFT, KEELER & LONG KLP10445, DEEP YELLOW REFER TO REP–20365–016 FOR FURTHER DETAILS

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NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
1	2–24–05	INCORP DCN DM2–00–0267–03 DM2–04–0267–03	DLW	MP	–	ASK

REV. NO.	DATE	REVISION DESCRIPTION	DRWN	CHK	APP
0		RELEASED FOR PRODUCTION	JSP		

REVISIONS

FABRICATION TOLERANCES

0 – 2'–0" = ± 1/16 5'–1" – 10'–0" = ± 3/16
2'–1" – 5'–0" = ± 1/8 10'–1" & OVER = ± 1/4

DIMENSIONS ARE IN INCHES UNLESS OTHERWISE SPECIFIED
MACHINING TOLERANCES UNLESS OTHERWISE SPECIFIED:
FRACTIONAL = ± 1/32 ANGLES = ± 1/2°
X.XX = ± .015 √ = 125 √
X.XXX = ± .005
BREAK ALL SHARP EDGES

TOLERANCES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

25203–29022 SH 169 UNIT #2

NON Q.A.

ISSUE STAMP

STRUCTURAL ENGR.	DATE
MECHANICAL ENGR.	DATE
ELECTRICAL ENGR.	DATE
QUALITY ASSURANCE	DATE
RADIATION ENGR.	DATE
ENGR. MANAGER	DATE
NEXT ASSEMBLY	NA

DOMINION NUCLEAR CONNECTICUT
MILLSTONE POWER STATION
P.O. NUMBER: 03005608

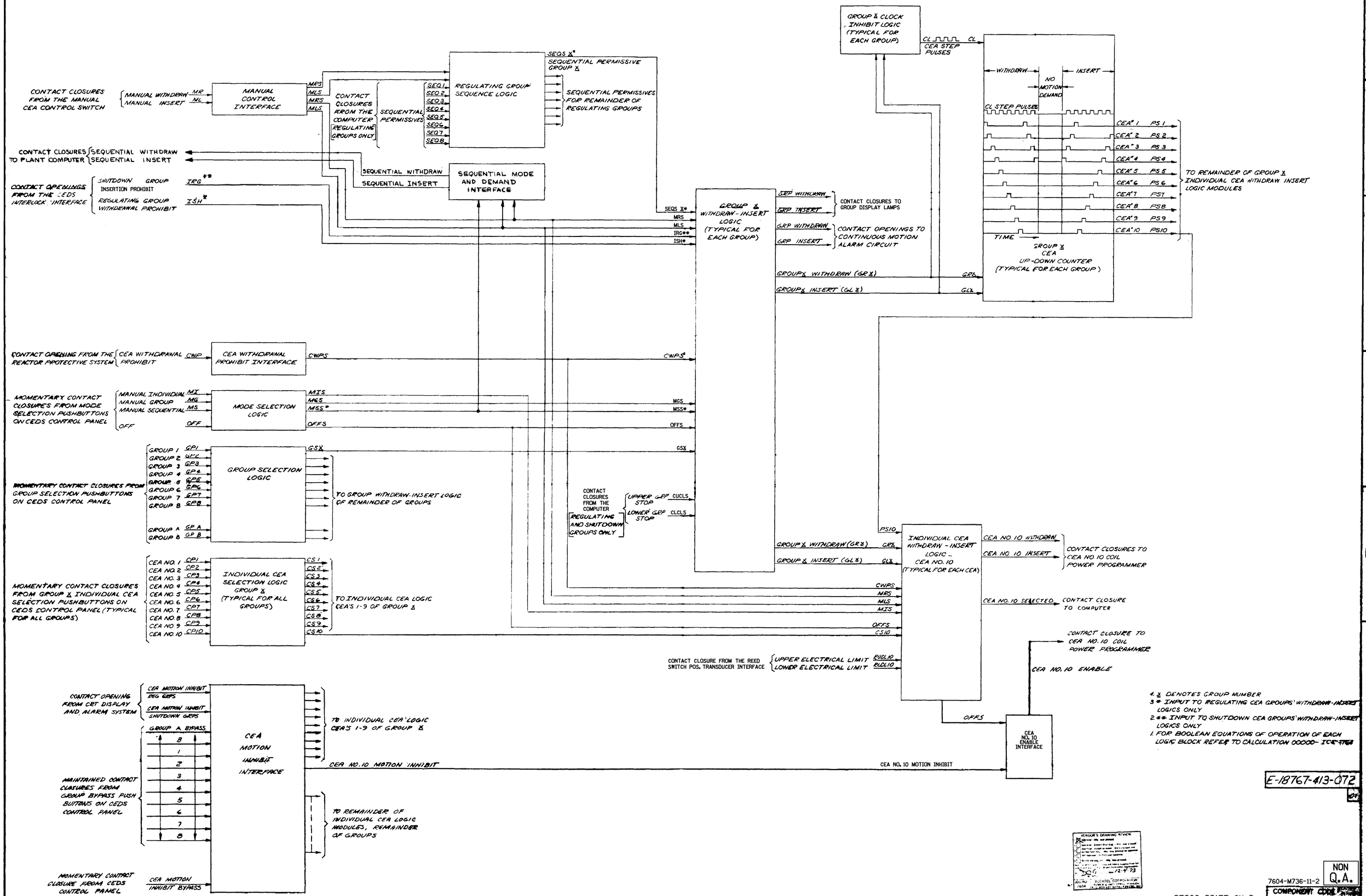
AMERICAN CRANE & EQUIPMENT CORPORATION
DOUGLASSVILLE, PA 19518 U.S.A.
PHONE: 610–385–6061 FAX: 610–385–3191

TITLE
SINGLE FAILURE PROOF TROLLEY NO. 2762
GENERAL ARRANGEMENT, ELEVATION VIEW

PROJECT
UPGRADE OF UNIT 2 (MP2) SPENT FUEL CASK CRANE TO SINGLE FAILURE PROOF

DRAWN BY	DATE	SCALE
J. PENTZ	08/09/02	3/8" = 1' – 0"
CHECKED BY	DATE	ELECTRONIC FILE
		20365001R0.DWG
APPROVED BY	DATE	ACECO WORK ORDER
		20365

DRAWING NUMBER D–20365–001 REV. 0



1. X DENOTES GROUP NUMBER
3 * INPUT TO REGULATING CEA GROUPS' WITHDRAW-INSERT LOGICS ONLY
2 ** INPUT TO SHUTDOWN CEA GROUPS' WITHDRAW-INSERT LOGICS ONLY
1. FOR BOOLEAN EQUATIONS OF OPERATION OF EACH LOGIC BLOCK REFER TO CALCULATION 00000- ICE-1767

E-18767-413-072

D7

VENDOR'S DRAWING REVIEW
APPROVED
DATE 12-4-73
BUCHTEL CORPORATION

NON Q.A.

7604-M736-11-2

COMPONENT CODE 21-2531-8422

25203-29155 SH 9

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2

NORTHEAST UTILITIES

CAD NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
3	11-16-04	INCORP DCN DM2-00-0025-00, DM2-00-0959-98	DLW	WBZ	-	DED

REV.	DESCRIPTION	ST & DATE	CHK. & DATE	ENG. APPD

REV.	DESCRIPTION	ARCH APPD	ST & DATE	CHK. & DATE	ENG. APPD
D7	ADDED CEA MOTION INHIBIT (BYPASS) REVISED SIGNAL NAMES TO CONFORM WITH C-E DRAWINGS, ADDED FSAR MODE (DEM - SUPPLY) UPGRADED TO APPROVED DWG PER REV. REQ. # 18767-144	1/1	3/72	FA	4-25	H.S.

DRAWN BY: Leo Calkins 5-6-71
CHECKER: H.J. St. Jean 5-11-71
APPROVED: E.M. Anson 5-11-71

CEDS FUNCTIONAL BLOCK DIAGRAM

E-18767-413-072

INPUTS FROM PLANT VARIABLES (SEE TABLE)

Ⓐ CHANNELS | Ⓑ CHANNELS | Ⓒ CHANNELS | Ⓓ CHANNELS

1 2 3 4 5 6 7 8 (N)

TRIP UNIT RELAYS (SEE NOTE 1)

TO SW-AB TO SW-AC TO SW-AD

CHANNEL TRIP SELECT S(AB)-2

SW-AB

PS5 dc POWER SUPPLY — MATRIX LOGIC A-B — PS15 dc POWER SUPPLY

FU FU

120VAC VITAL VA10 BKR.11 30022 SH.10LA

120VAC VITAL VA30 BKR. 11 30022 SH. 11LA

A1-1 A2-1 A3-1 A4-1 A5-1 A6-1 A7-1 A8-1 A(N)-1

B1-1 B2-1 B3-1 B4-1 B5-1 B6-1 B7-1 B8-1 B(N)-1

AB1 AB2 AB3 AB4

"HOLD" "DROP OUT"

MATRIX RELAY HOLD PB(AB)-1

MATRIX RELAY TRIP SELECT

S(AB)-1

TEST & POWER SUPPLY PS17

TO SW-AB TO SW-BC TO SW-BD

SW-AC

PS7 dc POWER SUPPLY — MATRIX LOGIC A-C — PS8 dc POWER SUPPLY

120VAC VITAL VA10 BKR.11 30022 SH.10LA

120VAC VITAL VA30 BKR.11 30022 SH.11LA

C1-1 A1-2 C2-1 A2-2 C3-1 A3-2 C4-1 A4-2 C5-1 A5-2 C6-1 A6-2 C7-1 A7-2 C8-1 A8-2 C(N)-1 A(N)-2

AC1 AC2 AC3 AC4

"HOLD" "DROP OUT"

SW-AD

PS9 dc POWER SUPPLY — MATRIX LOGIC A-D — PS10 dc POWER SUPPLY

120VAC VITAL VA10 BKR.11 30022 SH.10LA

120VAC VITAL VA40 BKR.9 30022 SH.13JA

D1-1 A1-3 D2-1 A2-3 D3-1 A3-3 D4-1 A4-3 D5-1 A5-3 D6-1 A6-3 D7-1 A7-3 D8-1 A8-3 D(N)-1 A(N)-3

AD1 AD2 AD3 AD4

"HOLD" "DROP OUT"

TO SW-AC TO SW-BC

SW-BC

PS11 dc POWER SUPPLY — MATRIX LOGIC B-C — PS12 dc POWER SUPPLY

120VAC (VITAL) VA25 BKR. 02 30022 SH. 37BA

120VAC VITAL VA30 BKR.11 30022 SH.11LA

C1-2 B1-2 C2-2 B2-2 C3-2 B3-2 C4-2 B4-2 C5-2 B5-2 C6-2 B6-2 C7-2 B7-2 C8-2 B8-2 C(N)-2 B(N)-2

BC1 BC2 BC3 BC4

"HOLD" "DROP OUT"

TO SW-CD

SW-BD

PS13 dc POWER SUPPLY — MATRIX LOGIC B-D — PS14 dc POWER SUPPLY

120VAC (VITAL) VA25 BKR. 02 30022 SH. 37BA

120VAC VITAL VA40 BKR.9 30022 SH.13JA

D1-2 B1-3 D2-2 B2-3 D3-2 B3-3 D4-2 B4-3 D5-2 B5-3 D6-2 B6-3 D7-2 B7-3 D8-2 B8-3 D(N)-2 B(N)-3

BD1 BD2 BD3 BD4

"HOLD" "DROP OUT"

TO SW-AD TO SW-BD TO SW-CD

SW-CD

PS6 dc POWER SUPPLY — MATRIX LOGIC C-D — PS16 dc POWER SUPPLY

120VAC (VITAL) VA25 BKR. 02 30022 SH. 37BA

120VAC VITAL VA40 BKR. 13JA 30022 SH. 13JA

C1-3 D1-3 C2-3 D2-3 C3-3 D3-3 C4-3 D4-3 C5-3 D5-3 C6-3 D6-3 C7-3 D7-3 C8-3 D8-3 C(N)-3 D(N)-3

CD1 CD2 CD3 CD4

"HOLD" "DROP OUT"

MG-2 240 V, 3Ø, 60Hz

480V, 3Ø, 60Hz BUS #2

MOTOR GEN

MCB-2

SYNCHROSCOPE

MG-1 240 V, 3Ø, 60Hz

480V, 3Ø, 60Hz BUS #1

GEN MOTOR

MCB-1

M2 TRIP SYNCHRONIZER #1 #2 OFF SYNCHRONIZER TRIP M1

BUS DV20 125 Vd-c

BUS DV30 125 Vd-c

120VAC (VITAL) VA25 BKR. 02 (30022 SH. 37B)

120VAC (VITAL) VA30 BKR. 11 (30022 SH. 11L)

IT-2 IT-3

AB2 AC2 AD2 BC2 BD2 CD2

MANUAL TRIP #1

K2 K3

UNDERVOLTAGE TEST SWITCH CONTACTS

SHUNT TRIP COIL CLOSE COIL TCB 9

BUS TIE

120VAC VITAL VA10 BKR. 11 (30022 SH. 10L)

120VAC VITAL VA40 BKR. 9 (30022 SH. 13J)

IT-1 IT-4

AB1 AC1 AD1 BC1 BD1 CD1

K1 K4

UNDER VOLTAGE TRIP COILS SHUNT TRIP COILS CLOSE COILS

TCB 6 TCB 8 TCB 5 TCB 1 TCB 3 TCB 7 TCB 4 TCB 2

TRIP CIRCUIT BREAKERS

CURRENT MONITORING

MANUAL TRIP #2

TEST

CEDM POWER SUPPLIES | UNDER VOLTAGE RELAY | UNDER VOLTAGE RELAY | UNDER VOLTAGE RELAY | UNDER VOLTAGE RELAY | CEDM POWER SUPPLIES

AUXILIARY FUNCTIONS

TABLE 1	
	TRIP INPUTS
1	VARIABLE HIGH POWER TRIP
2	LOW REACTOR COOLANT PUMP SPEED
3	LOW FLOW, REACTOR COOLANT
4	LOW WATER LEVEL, STEAM GENERATORS 1&2
5	LOW PRESSURE, STEAM GENERATORS 1&2
6	HIGH PRESSURE, PRESSURIZER
7	THERMAL MARGIN/LOW PRESSURE
8	TURBINE TRIP
9	HIGH CONTAINMENT PRESSURE
10	SPARE
11	LOCAL POWER DENSITY
12	SPARE

2. ALL CONTACTS SHOWN FOR RELAYS IN DE-ENERGIZED CONDITION.

1. EACH TRIP-UNIT HAS FIVE (5) RELAYS, THREE (3) AS SHOWN FOR TRIPPING LOGIC MATRICES A-B THRU C-D, ONE (1) FOR REMOTE PRE-TRIP ALARM ANNUNCIATION, ONE (1) FOR REMOTE TRIP ALARM ANNUNCIATION AND FOR SEQUENCE-OF-EVENTS RECORDING

E-13767-411-003

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT No. 2

REACTOR PROTECTION SYSTEM FUNCTIONAL DIAGRAM

REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER MSDUR 17-000126 & DUR 17-800266

CAD NO: C29193LHYB

DRAWING NO: 25203-29193

REV 12

DSGN DLW | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 1

POWER RANGE SAFETY CHANNELS

POWER RANGE CONTROL CHANNELS

CHANNEL #5 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-005-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — CONNECTORS — PENETRATIONS — CONNECTORS — CONNECTORS — J4 J6 J5

CHANNEL #6 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-006-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — CONNECTORS — CONNECTORS — CONNECTORS — J4 J6 J5

CHANNEL #7 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-007-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — CONNECTORS — PENETRATIONS — CONNECTORS — CONNECTORS — J4 J6 J5

CHANNEL #8 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-008-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — CONNECTORS — CONNECTORS — J4 J6 J5

CHANNEL #9 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-009-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — PENETRATIONS — CONNECTORS — CONNECTORS

CHANNEL #10 — A/B DUAL-SECTION UNCOMPENSATED IONIZATION CHAMBER DETECTOR ASS'Y NI-010-OFD1 — THREE (3) COAX. CABLES — CONNECTORS — THREE (3) COAX. CABLES — PENETRATIONS — CONNECTORS — CONNECTORS

CONTAINMENT WALL

POWER RANGE SAFETY CHANNEL DRAWER ASSEMBLY CHANNEL #9 NI-009-DA1

POWER RANGE SAFETY CHANNEL DRAWER A (#5)
- 39629 SH.1-3 DRAWER OUTLINE
- 39629 SH.11-14 DRAWER BLOCK DIAG.
- 39629 SH.15-17 DRAWER SCHEMATICS
- 39629 SH.20-24 DRAWER PARTS LIST
- J1 AC POWER
- J2 RPS OUTPUTS (39629 SH.4)
- J3 RPS OUTPUTS (39629 SH.4)
- J7 OUTPUT MONITOR

POWER RANGE SAFETY CHANNEL DRAWER B (#6)
- 39629 SH.1-3 DRAWER OUTLINE
- 39629 SH.11-14 DRAWER BLOCK DIAG.
- 39629 SH.15-17 DRAWER SCHEMATICS
- 39629 SH.20-24 DRAWER PARTS LIST
- J1 AC POWER
- J2 RPS OUTPUTS (39629 SH.4)
- J3 RPS OUTPUTS (39629 SH.4)
- J7 OUTPUT MONITOR

POWER RANGE SAFETY CHANNEL DRAWER C (#7)
- 39629 SH.1-3 DRAWER OUTLINE
- 39629 SH.11-14 DRAWER BLOCK DIAG.
- 39629 SH.15-17 DRAWER SCHEMATICS
- 39629 SH.20-24 DRAWER PARTS LIST
- J1 AC POWER
- J2 RPS OUTPUTS (39629 SH.4)
- J3 RPS OUTPUTS (39629 SH.4)
- J7 OUTPUT MONITOR

POWER RANGE SAFETY CHANNEL DRAWER D (#8)
- 39629 SH.1-3 DRAWER OUTLINE
- 39629 SH.11-14 DRAWER BLOCK DIAG.
- 39629 SH.15-17 DRAWER SCHEMATICS
- 39629 SH.20-24 DRAWER PARTS LIST
- J1 AC POWER
- J2 RPS OUTPUTS (39629 SH.4)
- J3 RPS OUTPUTS (39629 SH.4)
- J7 OUTPUT MONITOR

LINEAR AMPLIFIER — CALIBRATION & TEST — H.V. POWER SUPPLY — LINEAR AMPLIFIER

DATA PROCESSING COMPUTER OUTPUT

SUB CHANNEL DEVIATION SIGNAL OUTPUT — SUB CHANNEL DEVIATION MEASUREMENT

COMPARATOR HI CH. DEVIATION ALARM OUTPUT — CHANNEL POWER LEVEL HI ALARM SETPOINT — CHANNEL COMPARATOR — CHANNEL COMPARATOR — SUMMER CONTROL SWITCH

COMPARATOR AVERAGER INPUT — SUMMER GAIN OUTPUT — SUMMER GAIN INPUT — POWER SUMMER

OUTPUT TO COMPARATOR AVERAGER

AUXILIARY SIGNAL OUTPUT — TIME DELAY — ROD DROP BISTABLE

DATA PROCESSING COMPUTER OUTPUTS

TO DRAWER SUBASSEMBLIES

NI-008-BS1 FLUX BISTABLE

NI-008-BS2 H.V. BISTABLE

LOW VOLTAGE DC POWER SUPPLY

OUTPUT TO REACTOR PROTECTIVE SYSTEM

BISTABLE RELAY CONTACT OUTPUTS

- DATA PROCESSING COMPUTER OUTPUT
- COMPARATOR HI-HI CHANNEL DEVIATION ALARM OUTPUT. HI HI ALARM SETPOINT
- RECORDER OUTPUT.
- REMOTE METER OUTPUT "% POWER LEVEL".
- ROD DROP CIRCUIT.
- ROD DROP BISTABLE REMOTE ALARM OUTPUT
- 120 Va.c. INPUT FROM INSTRUMENT POWER BUS

POWER RANGE CONTROL CHANNEL DRAWER ASSEMBLY CHANNEL #10 (SEE NOTE 2)

NI-010-DA1

NI-010-BS1,2

4. CH = CHANNEL
   NI = NUCLEAR INSTRUMENTATION
   BS = BISTABLE
   DA = DRAWER ASS'Y
   OFD = OUT OF CORE FLUX DETECTOR
   PA = PREAMPLIFIER
   REC = RECORDER
   RM = REMOTE METER
   CA = COMPARATOR AVERAGER
   HV = HIGH VOLTAGE
   CPS = COUNTS PER SECOND
3. ALL DRAWING DESIGNATIONS ACCORDING TO C.E. STANDARDS UNLESS OTHERWISE SPECIFIED.
2. DRAWER ASSEMBLIES DESIGNATED BY THIS NOTE & IDENTICAL TO DRAWER ASSEMBLY FOR CHANNEL #9.

TYPICAL CHANNEL FLUX COVERAGE

SOURCE LEVEL = 0.1 nv

DETECTOR RANGES TYPICAL — FISSION CHAMBER — FISSION CHAMBER — UIC

THERMAL NEUTRON FLUX AT DETECTORS (nv) — 10^0 10^1 10^2 10^3 10^4 10^5 10^6 10^7 10^8 10^9 2×10^9 nv

FROM BUFFERED SUMMED OUTPUTS OF CHANNELS #5 THRU 8

CH 5 CH 6 CH 7 CH 8

COMPARATOR AVERAGER NI-CA1 — 120 Va.c. INPUT FROM INSTRUMENT POWER BUS

CH 5 CH 6 CH 7 CH 8

TO CHANNEL COMPARATORS OF POWER RANGE SAFETY CHANNELS #5 THRU 8.

Q = MAXIMUM OF POWER RANGE SAFETY CHANNEL POWER OR CALCULATED AT POWER Q_{TE} = CALCULATED VARIABLE OVERPOWER TRIP SET POINT

CH-5 Q_{TR} Q — CH-6 Q_{TR} Q — CH-7 Q_{TE} Q — CH-8 Q_{TE} Q — CH 9 % POWER — CH 10 % POWER

INDICATOR JI-005-RM1 — INDICATOR JI-006-RM1 — INDICATOR JI-007-RM1 — INDICATOR JI-008-RM1 — INDICATOR JI-009-RM1 — INDICATOR JI-010-RM1

POWER LEVEL RESET LIGHTED PUSHBUTTON (IDENTICAL FOR CH-5 THRU 8)

REMOTE INDICATOR ARRANGEMENT CH5 THRU CH 10

Q.A.

BECHTEL DWG. No. 7604-M020-17-1
C.E. DWG. No. E-18767-411-103

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NOTE: "FSAR FIGURE-SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES"

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
7	12-4-06	INCORP. DCN DM2-00-0243-05	JSJ	MEL	-	DLW
PDR 0203 1927 6	3/12/02	INCORP. DCN'S DM2-00-0831-99 DM2-DM2-00-0108-00, DM2-00-0109-00 DM2-DM2-00102-00	AGC	GHH	TP	CAM
5	4-7-00	INCORP. DCN DM2-00-0025-00	JMM	JKY	JPP	JPP
4	9/25/99	INCORP. DCN DM2-00-1072-97	WBZ	JSJ	JPP	JPP

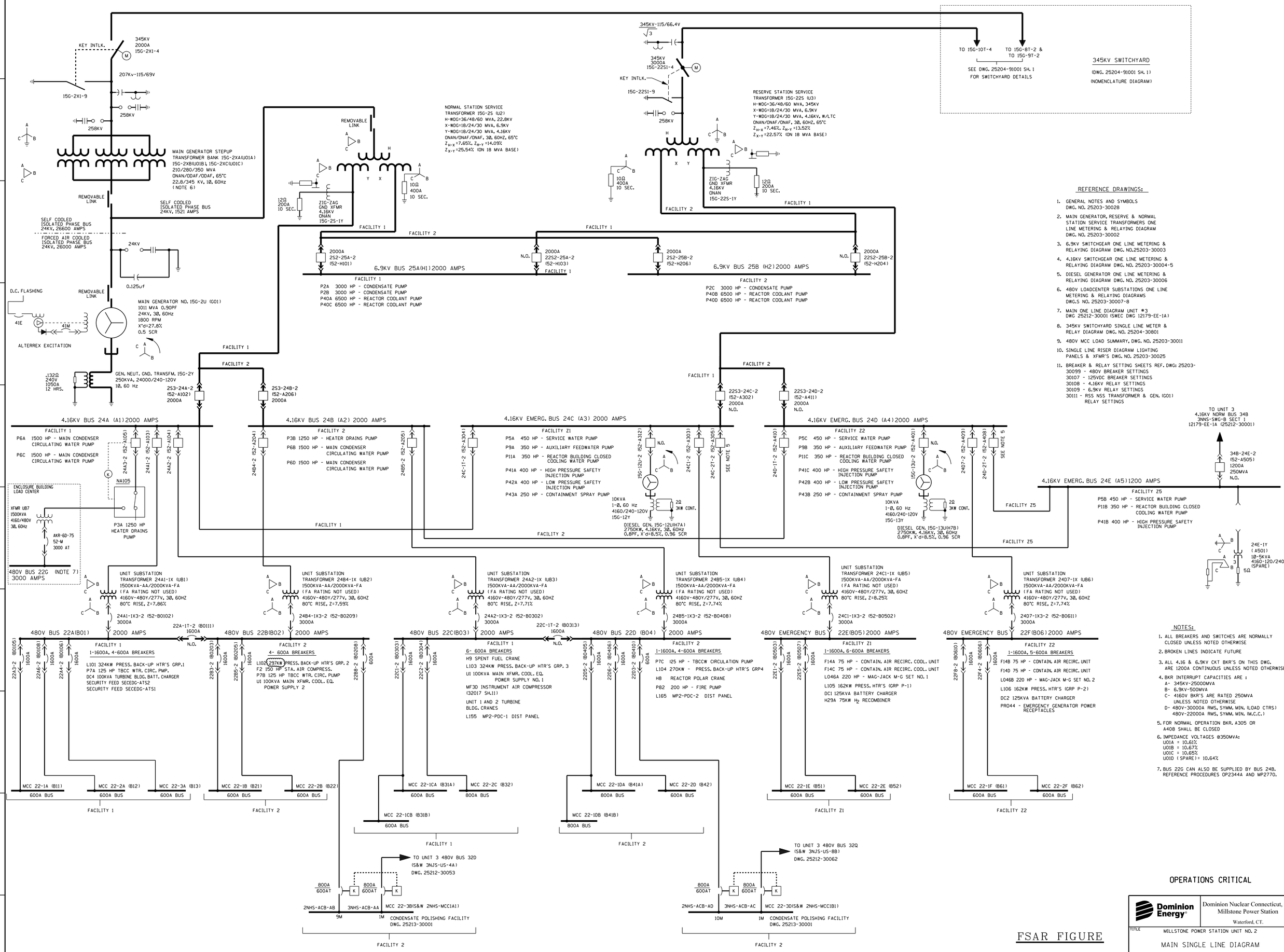
REVISIONS DURING CONSTRUCTION

Dominion — Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: NEUTRON FLUX MONITORING SYSTEM POWER RANGE CHANNELS

BY	CHK'D.	APP.	APP.
WBZ/JSJ	JSJ	JPP	JPP
DATE 7-21-99	DATE 8/31/99	DATE 9/25/99	DATE 9/25/99

SCALE N/A — DWG. NO. 25203-29198 SH.14

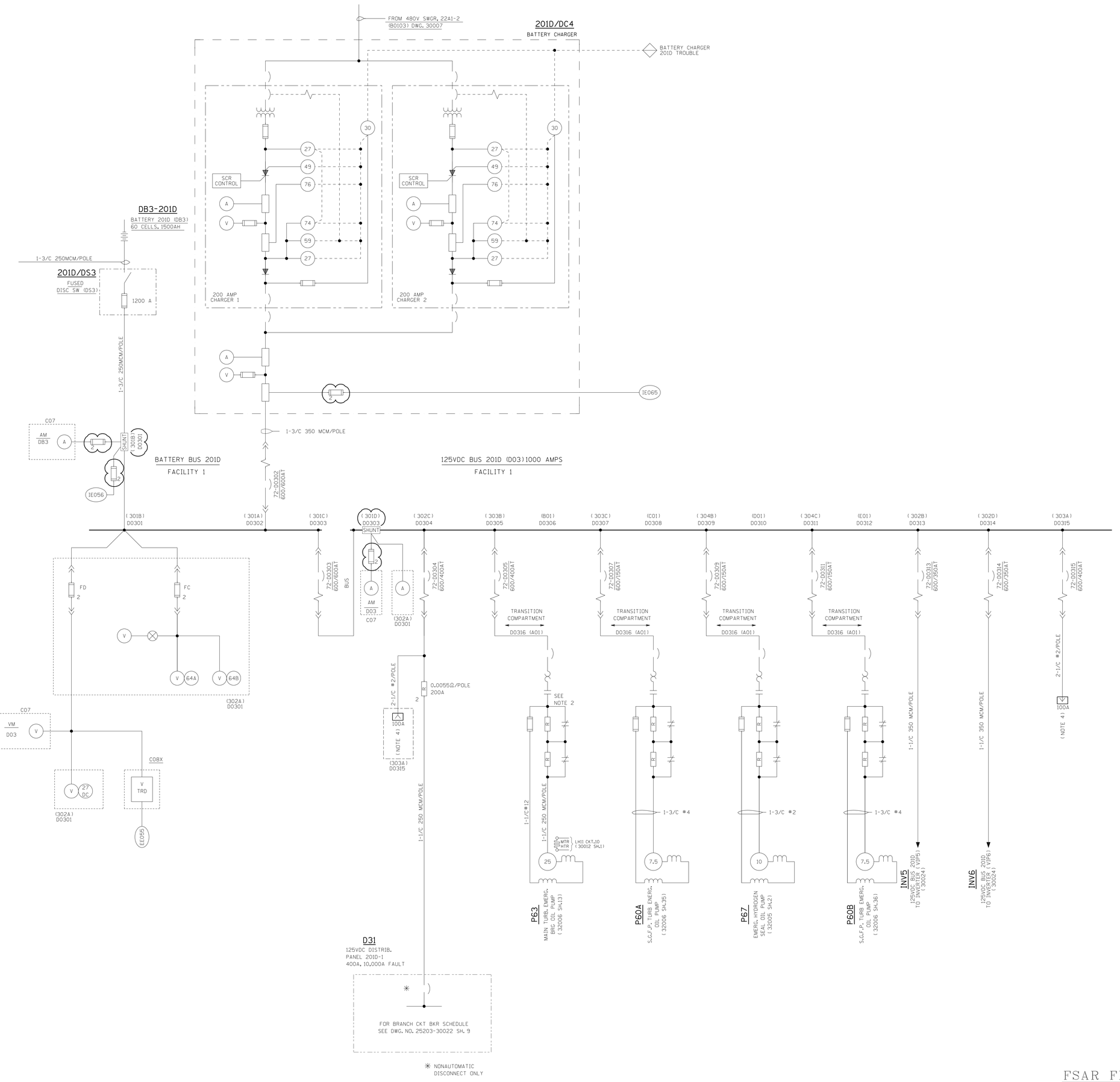


OPERATIONS CRITICAL

FSAR FIGURE

Dominion Energy®	Dominion Nuclear Connecticut, Inc. Millstone Power Station, Waterford, CT.
TITLE	MILLSTONE POWER STATION UNIT NO. 2
	MAIN SINGLE LINE DIAGRAM
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C30001.DGN
REVISED PER DC MP2-23-01071 & DUR 23-80173	DRAWING NO: 25203-30001 REV 54
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED SH 0

Authenticated By: MBLOGIN\scott84
21-JUN-2023 13:37
C30001.DGN



LOAD SUMMARY

125VDC BUS 201D (D03)	
BREAKER/CMPMT	SERVICE
D0301	BATTERY 201D TO BATTERY BUS 201D
D0302	BATTERY CHARGER DC4 TO BATTERY BUS 201D
D0303	BATTERY BUS 201D TO 125VDC BUS 201D
D0304	125VDC BUS 201D TO 125VDC DIST. PANEL D31
D0305	MAIN TURBINE EMERGENCY BEARING OIL PUMP, P-63
D0307	"A" SGFP EMERGENCY OIL PUMP, P-60A
D0309	EMERGENCY HYDROGEN SEAL OIL PUMP, P-67
D0311	"B" SGFP EMERGENCY OIL PUMP, P-60B
D0313	125VDC BUS 201D TO INVERTER INV5
D0314	125VDC BUS 201D TO INVERTER INV6
D0315	125VDC BUS 201D JUMPER BREAKER

NOTES

1. FOR STANDARD NOTES, SYMBOLS SEE DRAWING 25203-30028
2. MOTOR STARTER IS FURNISHED BY GE UNDER PO M-915
3. FOR LOAD CENTER CIRCUIT BREAKER SETTING SHEETS, SEE DRAWING 30107 SERIES
4. QUICK-DISCONNECT CONNECTOR FOR ALTERNATE POWER FEED DURING BREAKER MAINTENANCE.

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION UNIT NO. 2
SINGLE LINE DIAGRAM
125VDC SYSTEM - TURBINE BATTERY

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C:\fms32\tmpplot\C30023.DGN	
REVISED PER DC MP2-14-01007 & DUR 14-800121	DRAWING NO: 25203-30023	REV 9
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 0

125VDC BUS 201A (D01) 2000A

125VDC BUS 201B (D02)-2000A

B0506 FROM 480V SWGR. 22E4-2 (B0506) DWG. 30008 — 3-1/C 250MCM

201A/DC1 BATTERY CHARGER

NONAUTO NONAUTO

201A-2 AC INPUT (65KAIC (NOTE 3)) — 400 AMP CHARGER No.2 — 201A-2 DC OUTPUT (22KAIC (NOTE 3))

201A-2 AC INPUT TO CHARGER — 201A-1 AC INPUT TO CHARGER — TERMINATION CABINET — 201A-2 DC OUTPUT TO BATT BUS BKR. D0102 — 201A-1 DC OUTPUT TO BATT BUS BKR. D0102

201A-1 AC INPUT (65KAIC (NOTE 3)) — 400 AMP CHARGER No.1 — 201A-1 DC OUTPUT (22KAIC (NOTE 3))

NONAUTO NONAUTO

SHUNT — A 0-1000A — V 100-150V — IE060

AMETEK FOR BATTERY CHARGER ONE-LINE DIAGRAM SEE 25203-39361 SH.12

3-1/C 500MCM PER POLE

B0607 FROM 480V SWGR. 22F5-2 DWG. 30008 — 3-1/C 250MCM

201B/DC2 BATTERY CHARGER

201B-2 AC INPUT (65KAIC (NOTE 3)) — 400 AMP CHARGER No.2 — 201B-2 DC OUTPUT (22KAIC (NOTE 3))

201B-2 AC INPUT TO CHARGER — 201B-1 AC INPUT TO CHARGER — TERMINATION CABINET — 201B-2 DC OUTPUT TO BATT BUS BKR. D0202 — 201B-1 DC OUTPUT TO BATT BUS BKR. D0202

201B-1 AC INPUT (65KAIC (NOTE 3)) — 400 AMP CHARGER No.1 — 201B-1 DC OUTPUT (22KAIC (NOTE 3))

NONAUTO NONAUTO

SHUNT — A 0-1000A — V 100-150V — IE064

AMETEK FOR BATTERY CHARGER ONE-LINE DIAGRAM SEE 25203-39361 SH.12

3-1/C 500MCM PER POLE

30053 SH. 1A SEE DWG. FOR INTERNALS CIRCUIT. DETECTION OPEN POLE

DB1-201A BATTERY 201A 60 CELL, 2320AH — 4-1/C, 500MCM PER POLE

DS1 — 2000A — FUSED DISC SWITCH (DS1) — 4-1/C 500MCM PER POLE

DB2-201B BATTERY 201B 60 CELL, 2320AH — 4-1/C 500MCM PER POLE

DS2 — 2000A — FUSED DISC SWITCH (DS2) — 4-1/C 500MCM PER POLE

OPEN POLE DETECTION CIRCUIT. FOR INTERNALS SEE DWG. 30053 SH. 3A

AM DB1 — A — SHUNT (D0103) — COB — IE052

AM DB2 — A — SHUNT (D0203) — COB — IE054

72-D0102 1600/800AT

KEY INTLK

(NOTE 6) 125VDC BUS 201A (D01) 2000A

101C (D0103) — 102B (D0105) — 102C (D0109) — 102C (D0106) — 103C (D0107) — 104C (D0110) — 103B (D0108) — 103A (D0111) — 104B (D0104)

(NOTES 5 & 6) — KEY INTLK

(NOTE 6) 125VDC BUS 201B (D02)-2000A

204B (D0204) — 203B (D0208) — 203A (D0211) — 201C (D0209) — 203C (D0207) — 202C (D0206) — 204C (D0210) — 202B (D0205) — 201C (D0203) — KEY INTLK — 202A (D0202) — 201B (D0203) — 201A (D0201)

BATTERY BUS 201A — BATTERY BUS 201B

101A (D0101) — 102A (D0102) — 101B (D0103)

72-D0103 1600/1200AT — 72-D0105 1600/3500AT — 72-D0109 1600/3500AT — 72-D0106 1600/3500AT — 72-D0107 1600/3500AT — 72-D0110 1600/3500AT — 72-D0108 1600/3500AT — 72-D0111 1600/3500AT (NOTE 7) — 72-D0104 1600/800AT — 72-D0204 1600/800AT — 72-D0208 1600/3500AT — 72-D0211 2000/3500AT (NOTE 7) — 72-D0209 1600/3500AT — 72-D0207 1600/3500AT — 72-D0206 1600/3500AT — 72-D0210 1600/3500AT — 72-D0205 1600/3500AT — 72-D0203 1600/1200AT

AM DD1 — COB — A — A

V TDR — COBX — EE051 — VM DD1 — COB — V

27 DC — V — 64B — V — 64A — V

FACILITY Z1 — FACILITY Z5 — FACILITY Z2

3-1/C, 500MCM/POLE — 2-1/C #2/POLE — 100A

0.0077Ω PER POLE 200A — 1-1/C 250MCM/POLE — 1-1/C #4/0/POLE — 1-1/C, #4/0/POLE NJ — 1-1/C, 500MCM/POLE

(NOTE 4)

DV10

125VDC VITAL INSTR. PANEL 201A-1V (DV10) 400A, 10,000A FAULT

DV30

125VDC VITAL INSTR. PANEL 201A-2V 225A, 10000A FAULT

DV20

125VDC VITAL INST. PANEL 201B-1V 400A, 10,000A FAULT

DV40

125VDC VITAL INSTR. PANEL 201B-2V 225A, 10,000A FAULT

FACILITY Z1 — FACILITY Z3 — FACILITY Z2 — FACILITY Z4

FOR BRANCH CKT BKR SCHEDULE SEE DWG. NO. 25203-30022 SH. 1 / SH. 2 / SH. 3 / SH. 4

D11

125VDC DISTRIB. PANEL 201A-1 (D11) 225A, 10,000A FAULT

D12

125VDC DISTRIB. PANEL 201A-2 225A, 10,000A FAULT

D21

125VDC DISTRIB. PANEL 201B-1 225A, 10,000A FAULT

D22

125VDC DISTRIB. PANEL 201B-2 225A, 10,000A FAULT

FACILITY 1 — FACILITY 2

FOR BRANCH CKT BKR SCHEDULE SEE DWG. NO. 25203-30022 SH. 5 / SH. 6 / SH. 7 / SH. 8

INV1

INVERTER (VIP1) VITAL 125VDC,120VAC, 1Ø, 10KVA

INV3

INVERTER (VIP3) VITAL 125VDC,120VAC, 1Ø, 10KVA

INV2

INVERTER (VIP2) VITAL 125VDC,120VAC, 1Ø, 10KVA

INV4

INVERTER (VIP4) VITAL 125VDC, 120VAC, 1Ø, 10KVA

FILTER — B1 200A (DC INPUT) — FOR INVERTER SCHEMATIC SEE 25203-39391 SH.3 / SH.11 — B2 125A (INVERTER OUTPUT)

BYPASS SOURCE (AC INPUT) — B4 150A — STATIC SWITCH VS1 / VS2 / VS3 / VS4 — MANUAL BYPASS SWITCH — (NORMAL OPERATION) — (BYPASS TO LOAD)

200A

INV5

INVERTER (VIP5) NON-VITAL 125VDC, 120VAC, 1Ø, 10KVA

ALT. AC INPUT — 200A — B4 150A — STATIC SWITCH VS1 — MANUAL BYPASS SWITCH — (NORMAL OPERATION) — (BYPASS TO LOAD)

D0313 FROM BUS 201D TO DWG. 25203-30023 — DC INPUT — FILTER B1 200A — (DC INPUT) — FOR INVERTER SCHEMATIC SEE 25203-39391 SH.3 — B2 125A (INVERTER OUTPUT) — 2-1/C #4/0

S8 REMOTE SYNC ENABLE (INSIDE CABINET)

2-1/C #4/0 — 2-1/C #2/0 — 2/C #10 (SYNC.)

VA10

120VAC VITAL INSTR. PANEL VIAC-1 (VA10) 200A, 10,000A FAULT

VA30

120VAC VITAL INSTR. PANEL VIAC-3 200A, 10,000A FAULT

FACILITY Z1 — FACILITY Z3

FOR BRANCH CKT BKR SCHEDULE SEE DWG. NO. 25203-30022 SH. 10 / SH. 11

VA20

120VAC VITAL INSTR. PANEL VIAC-2 (VA20) 200A, 10,000A FAULT

VA25

120VAC VITAL INSTR. PANEL VIAC-2A (VA25) 100A, 10,000A FAULT

VA40

120VAC VITAL INSTR. PANEL VIAC-4 200A, 10,000A FAULT

FACILITY Z2 — FACILITY Z4

FOR BRANCH CKT BKR SCHEDULE SEE DWG. NO. 25203-30022 SH. 12 / SH. 13

FOR BRANCH CKT BKR SCHEDULE SEE DWG. NO 25203-30022 SH.37

UIS02

ISOLATING XFMR. 120-120VAC 1Ø, 60HZ, 15KVA

UIS01

ISOLATING XFMR. 120-120VAC 1Ø, 60HZ, 15KVA

B51

FROM 480V MCC 22-1E BKR. NO. B5125 (C/D 31023 SH. 44) (ADMINISTRATIVELY KEY CONTROLLED) — 3-1/C#3/0

B32

FROM 480V MCC 22-2C BKR. No. B3246 C/D 31018 SH.15 — 3-1/C#3/0

B61

FROM 480V MCC 22-1F BKR. NO. B6121 (C/D 31025 SH. 46) (ADMINISTRATIVELY KEY CONTROLLED) — 3-1/C#3/0

B41A

FROM 480V MCC 22-1DA BKR. No. B41A54 C/D 31020 SH.30 — 3-1/C#3/0

B62

FROM 480V MCC 22-2F BKR. No. B6242 (C/D 31026 SH.23) — 3-1/C #4

B52

FROM 480V MCC 22-2E BKR. No. B5237 (C/D 31024 SH.22) — 3-1/C #4

UPS-VR11

DB-UPS-VR11 60 CELL, 572AH — 4-1/C 350MCM

CB1 125A — BATTERY CHARGER 410A — CB2 600A — 1-7/C #14 — TO RC22 (DWG 31078 SH.8)

INV — CB3 175A — S/S — BYPASS INPUT — CB4 175A

4-1/C#4/0 — 4-1/C#4/0

VS-UPS-VR11

BYPASS SWITCH — NORMAL SOURCE — BYPASS SOURCE OUTPUT — AC OUTPUT — BYPASS SOURCE

UPS-VR21

DB-UPS-VR21 60 CELL, 572AH — 4-1/C 350MCM

CB1 125A — BATTERY CHARGER 410A — CB2 600A — 1-7/C #14 — TO RC22 (DWG 31078 SH.8)

CB3 175A — INV — S/S — CB4 175A

4-1/C#4/0 — 4-1/C#4/0

VS-UPS-VR21

BYPASS SWITCH — BYPASS SOURCE OUTPUT — NORMAL SOURCE — BYPASS SOURCE — AC OUTPUT

NVR11

DISCONNECT SWITCH — 200A — 4-1/C#4/0

NVR21

DISCONNECT SWITCH — 200A — 4-1/C#4/0

UAC3

REG. XFMR. 480/208-120VAC 30,60HZ, 45KVA — 3-1/C #4

UAC4

REG. XFMR. UAC4 480/208-120VAC 30,60HZ, 45KVA

VR11

120V REGULATED A.C. INSTR. PNL. IAC-1 — FACILITY 1

27 — 27N — 225A — 150A INVERTER INV3 (ALT SOURCE) VR11 BKR.45 — 20A INVERTER SYNC. SOURCE (VIP5) VR11 BKR.52 — GD — FOR BRANCH CKT. BKR. SCHEDULE SEE DWG. NO. 25203-30022 SH. 14 TO 17

VR21

120V REGULATED A.C. INSTR. PNL. IAC-2 — FACILITY 2

27 — 27N — 225A — 20A INVERTER SYNC. SOURCE (VIP6) VR21 BKR.9 — 150A INVERTER INV4 (ALT SOURCE) VR21 BKR.45 — GD — FOR BRANCH CKT. BKR. SCHEDULE SEE DWG. NO. 25203-30022 SH. 20 TO 23

PR045

NOTE 9 — 100A

PR046

NOTE 9 — 100A

INV6

INVERTER (VIP6) NON-VITAL 125VDC, 120VAC, 1Ø, 10KVA

(INVERTER OUTPUT) — B2 125A — FOR INVERTER SCHEMATIC SEE 25203-39391 SH.11 — (DC INPUT) — B1 200A FILTER — DC INPUT

S8 REMOTE SYNC ENABLE (INSIDE CABINET)

D0314

FROM BUS 201D DWG. 25203-30023

✱- NONAUTOMATIC DISCONNECT ONLY.

REFERENCE DWGS.:

25203-39323 SH. 5 EMERGENCY SYSTEM ONE LINE DIAGRAM

LOAD SUMMARY

125VDC BUS 201A (D01)

BREAKER/CMPMT	SERVICE
D0101	BATTERY 201A TO BATTERY BUS 201A
D0102	BATTERY CHARGER DC1 TO BATTERY BUS 201A
D0103	BATTERY BUS 201A TO 125VDC BUS 201A
D0104	125VDC BUS 201A TO 125VDC BUS 201B BUS TIE
D0105	125VDC BUS 201A TO 125VDC INST. PANEL DV10
D0106	125VDC BUS 201A TO INVERTER INV1
D0107	125VDC BUS 201A TO INVERTER INV3
D0108	125VDC BUS 201A TO 125VDC INST. PANEL DV30
D0109	125VDC BUS 201A TO 125VDC DIST. PANEL D11
D0110	125VDC BUS 201A TO 125VDC DIST. PANEL D12
D0111	125VDC BUS 201A JUMPER BREAKER

125VDC BUS 201B (D02)

BREAKER/CMPMT	SERVICE
D0201	BATTERY 201B TO BATTERY BUS 201B
D0202	BATTERY CHARGER DC2 TO BATTERY BUS 201B
D0203	BATTERY BUS 201B TO 125VDC BUS 201B
D0204	125VDC BUS 201B TO 125VDC BUS 201A BUS TIE
D0205	125VDC BUS 201B TO 125VDC INST. PANEL DV40
D0206	125VDC BUS 201B TO INVERTER INV4
D0207	125VDC BUS 201B TO INVERTER INV2
D0208	125VDC BUS 201B TO 125VDC INST. PANEL DV20
D0209	125VDC BUS 201B TO 125VDC DIST. PANEL D21
D0210	125VDC BUS 201B TO 125VDC DIST. PANEL D22
D0211	125VDC BUS 201B JUMPER BREAKER

NOTES

1. FOR GENERAL NOTES & SYMBOLS SEE DWG. 25203-30028
2. FOR LOAD CENTER CIRCUIT BREAKER SETTING SHEETS SEE DWG. 30107 SERIES
3. THE BREAKER ADJUSTABLE INSTANTANEOUS TRIP UNIT SHALL BE SET IN THE HIGH POSITION
4. QUICK-DISCONNECT CONNECTOR FOR ALTERNATE POWER FEED DURING BREAKER MAINTENANCE.
5. THE LINE SIDE BUS BAR FOR CUBICLE 103A (D0111) IS CONNECTED TO THE MAIN BUS WITH 250 MCM CABLES, ONE PER POLE.
6. BREAKERS D0105, D0111 AND D0208, D0211 LOAD SIDE BUS BARS HAVE CONNECTION PLATES WITH MOUNTING HOLES TO ACCOMODATE 250 MCM LUGS, ONE PER POLE, TO BE USED FOR TEMP. POWER DURING MAINTENANCE.
7. BREAKER SETTINGS TO BE SET TO MATCH THE BREAKER BEING REMOVED FOR MAINTENANCE.
8. DELETED
9. 120VAC PORTABLE EMERGENCY GENERATOR POWER RECEPTACLE

OPERATIONS CRITICAL

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

SINGLE LINE DIAGRAM
125VDC EMERGENCY & 120VAC VITAL SYSTEMS

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C30024.DGN

REVISED PER DC MP2-21-01158 & DUR 23-800082 | DRAWING NO: 25203-30024 | REV 50

DSGN JSJ | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 0

Authenticated By: MBULOGIN\jome694
17-APR-2023 09:15
C30024.DGN

UNIT PARALLEL SW. TURNED TO PARALLEL POSITION

SETS GOVERNOR TO DROOP MODE.

SETS VOLTAGE ADJUSTER TO DROOP MODE.

SETS GOVERNOR TO ISOCHRONOUS MODE

SETS VOLTAGE REGULATOR TO AUTO MODE

LOSS OF PLANT POWER

SIAS 1

LOSS OF CONTROL DC POWER

SELECTOR SW IS IN LOCAL POSITION (KEY LOCKED)(43-MSS) LOCAL PUSH BUTTON START (OFF 180 DEG) (PB-01)

CONT'L. RM. EMERG. RELAY LOSS P.B. ACTUATED 2000M DIESEL 2000M

SELECTOR SW. IS IN REMOTE POSITION (KEY LOCKED)(43-MSS)

REMOTE CONT'L SW. START POSITION (DGH7A)

OPS 1 — LUBE OIL PRESSURE LOW

OPS 2 — LUBE OIL PRESSURE LOW

OPS 3

JACKET COOLANT TEMPERATURE HIGH (CTH)

CRANKCASE PRESSURE HIGH (CCP)

FUEL OIL PRESSURE LOW (FPL)A

LUBE OIL TEMPERATURE HIGH (OTH)

DC CONTROL POWER FAILURE (CR)

ENGINE OVERSPEED (EOS)

JACKET COOLANT PRESSURE LOW (CPL)A

ENGINE SPEED 810 RPM

DIESEL ENGINE SLOW START SW IN NORM POS

MAINT. SW. IS IN NORMAL POS. (43-MSS3)

OIL PRESS. ESTABLISHED (OPS)

DIESEL AT 810 R.P.M.

DIESEL AT 2ØØ R.P.M. (LSR)

LOCAL START PUSH-BUTTON ACTUATED (HS-6ØØ8)

START PUSHBUTTON AT ALCCS ACTUATED

LOSS OF PLANT POWER

DIESEL GENERATOR SEQUENCER ACTUATED

AND

4 START RELAY

DE-ENERGIZE TO OPEN JACKET COOLANT VALVE

STARTS DIESEL GENERATOR VENT FAN MF-38A

ALARM PUSHBUTTON RESET (AT ENGINE PANEL)

ESR

ENGINE OVERSPEED RESET (MECHANICAL)

ENGINE OVERSPEED 116% FULL R.P.M (EOS)

86 15G-12U-2 ENERGIZED

LUBE OIL PRESS. LOW

OPS1 OPS2 OPS3

2/3

SEE NOTE 2

TDDO 12 SEC.

SFR

2-REMOTE EMERGENCY STOP PUSHBUTTONS (PB-CR1, PB-1542) (PB-CR2, PB-1543) LOCAL POSITION (43-MSS)

2-LOCAL EMERGENCY STOP PUSHBUTTONS ACTUATED (PB-ØØ1 & PB-Ø2)

LOSS OF PLANT POWER

SIAS 1

LOCAL START PUSH BUTTON (PB-15)

LOCAL/REMOTE SW. IS IN LOCAL (43-MSS)

REMOTE START SWITCH IS IN START (DG-H7A)

MAINTANCE SWITCH IS IN MAINTENANCE POSITION (43-MSS3)

LOCAL STOP PUSH BUTTON ACTUATED AT ENGINE PANEL (PB-2)

NORMAL/MAINTENANCE SWITCH IS IN MAINTENANCE POSITION (43-MSS3)

LOCAL REMOTE SEL. SW IS IN LOCAL POS.(43-MSS)

LOCAL STOP PUSH BUTTON ACTUATED (PB-14)

LOCAL REMOTE SEL. SW IS IN REMOTE POS. (43-MSS)

REMOTE CONT. SW. IS IN STOP POS. (DG-H7A)

ALR COMMON ALARM RELAY

TDDO 60 SEC

ENERGIZE GOVERNOR SOL. TO STOP ENGINE

86 LOCKOUT ENERGIZED

LOCAL REMOTE SEL. SW IS IN LOCAL POS.(43-MSS)

LOCAL STOP PB. ACTUATED (PB-14)

LOCAL FIELD SHUTDOWN PUSH BUTTONS NOT ACTUATED (PB-13) (PB-10)

SHUTDOWN RELAY

SH

REMOTE CONT. SW. IS IN STOP POS. (DG-H7A)

LOCAL REMOTE SEL. SW. IS IN REMOTE POSITION (43-MSS)

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

ENGINE SPEED ENGINE PANEL ENGINE SPEED 810 R.P.M.

OIL PRESS. ESTABLISHED (OPS)

LOSS OF DC CONTROL POWER

LOSS OF DC CONTROL POWER

LOCAL REMOTE SW. IS IN LOCAL POS. (43-MSS)

MAINTENANCE SW. IS IN MAINTENANCE. (43-MSS3)

LOCAL START PUSH-BUTTON ACTUATED AT ENGINE PANEL (PB-1)

DE-ENERGIZE TO OPEN AIR SUPPLY VALVE (AS1)

DE-ENERGIZE TO OPEN AIR SUPPLY VALVE (AS2)

DE-ENERGIZE TO CLOSE AIR VENT VALVE (AV)

DIESEL STARTS CRANKING

86-2/2153-24C-2

VOLTAGE RESTRAINT OVERCURRENT RELAY (51V)

REVERSE POWER RELAY (32)

OVERCURRENT RELAY (50/1)

GROUND FAULT RELAY (3ØGS)

52/STA. 3462-24C-2 CLOSED

52/STA. 2153-24C-2 CLOSED

52/STA. 2153-24C-2 CLOSED

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

SYNC. SWITCH ON (LOCAL) (25-NCS)

CONTROL SWITCH IS IN CLOSED POSITION (LOCAL) (CS-5)

LOCAL REMOTE SEL. SW. IS IN REMOTE POSITION (43-MSS)

SYNC. SWITCH ON (REMOTE) (25-NCS)

CONTROL SWITCH IS IN CLOSED POSITION (REMOTE) (CS-2)

LOCAL CONT'L SW IS IN TRIP POS. (CS-5)

REMOTE CONT'L SW. IS IN TRIP POSITION (43-MSS)

A3 BUS UNDERVOLTAGE

D.G. BKR CLOSING CIR. BLOCK RESET (3 RELAY)

LOCAL REMOTE SW. IS IN REMOTE POSITION (43-MSS)

CS-2 IS IN AUTO. POSITION (REMOTE)

LOCAL REMOTE SW. IS IN LOCAL POSITION (43-MSS)

CS-3 IS IN AUTO. POSITION (LOCAL)

FIELD RESET PUSH BUTTON ACTUATED (DZA) (PB-12)

EXCITER FIELD ENERGIZED (FIELD FLASH)

DIESEL GENERATOR OUTPUT VOLTAGE ESTABLISHED

TRIP BREAKER 52-A312

CLOSE BREAKER 52-A312

(B) COIX

(W) COIX

INTERLOCK TO DIESEL SEQUENCER

CONTROL SWITCHES LOCATED AT MAIN CONTROL BOARD C08

SB1	SBM	SBM	SBM	SB1	SB1	SBM	SB1	SB1
MAN. AUTO	LOWER N RAISE	LOWER N RAISE	LOWER N RAISE	LOCAL REMOTE	OFF NORM UNIT PARALLEL	STOP AUTO START	OFF ON	TRIP NORM CLOSE
43-VRMS VOLTAGE REGULATOR MAN-AUTO MAINT.	CS-6S GOVERNOR RAISE-LOWER SPRING RET.	CS-7Ø MAN. VOLTAGE ADJUST RAISE-LOWER SPRING RET.	CS-9Ø AUTO VOLTAGE ADJUSTER RAISE-LOWER SPRING RET.	43-MSS L/R SELECTOR MAINT SEE NOTE #1	43-UPS UNIT PARALLEL SPRING RET	DGH7A START-STOP SPRING RET.	25-CS SYNC. MAINT.	CS-2 DIESEL BKR CONTROL SPRING RET.

ENGINE EMERG SHUTDOWN PB-021

ENGINE EMERG SHUTDOWN PB-022

STOP START CR2940

CS-Ø23A ENGINE PRE-LUBE MAINTAINED CONTACTS

CONTROL SWITCHES LOCATED AT DIESEL CONTROL PANEL C38

SB1	SB1	SBM	SBM	SBM	SB1	SB1	SB1	SB1
NORMAL SLOW START	MAN AUTO	LOWER N RAISE	LOWER N RAISE	LOWER N RAISE	OFF NORM UNIT PARALLEL	OFF ON	NORMAL MAINT.	TRIP NORM CLOSE
43-SLST SLOW-START KEYLOCK	43-VRMS VOLTAGE REGULATOR MAN-AUTO MAINT.	CS-65A GOVERNOR RAISE-LOWER SPRING RET.	CS-7ØA MAN. VOLTAGE ADJUST. RAISE-LOWER SPRING RET.	CS-9ØA AUTO. VOLTAGE ADJUSTER RAISE-LOWER SPRING RET.	43-UPS UNIT PARALLEL SPRING RET.	25-CS SYNC. MAINT	43-MSS3 MAINTENANCE MAINT. SEE NOTE #1	CS-3 DIESEL BKR CONTROL SPRING RET.

EXCITER RESET CR2940 PB-12

ENGINE START CR2940 PB-15

ENGINE STOP CR2940 PB-14

DIESEL ROOM EMERGENCY CR2940 BREAK GLASS PUSHBUTTON PB-100

ENGINE EMERG SHUTDOWN PB-031 CR2940

ENGINE EMERG SHUTDOWN PB-032 CR2940

EXCITER SHUTDOWN CR2940 PB-10

EMERGENCY SHUTDOWN CR2940 PB-13

CONTROL SWITCHES AT ENGINE PANEL TO40

STOP START CR2940 RS-15 ENGINE PRE-LUBE MAINTAINED CONTACTS

ENGINE START CR2940 PB-1

ENGINE STOP CR2940 PB-02

ALARM RESET CR2940 PB-3

LOCAL ALARMS AT C38

AUX. CONT. SW. NOT IN AUTO. POSITION	JACKET COOLANT PRESSURE LOW
GENERATOR NEUTRAL GROUND FAULT	480 V. AUXILIARY POWER LOSS
D.C. AIR COMPRESSOR STARTED	GENERATOR STATOR TEMP. HIGH
CRANKCASE PRESSURE HIGH	GENERATOR OVERVOLTAGE
ENGINE START FAILURE	FUEL OIL PRESSURE LOW
ENGINE OVERSPEED	LUBE OIL PRESSURE LOW
GENERATOR LOCKOUT TRIP (DIFFERENTIAL)	LUBE OIL TEMPERATURE LOW
GENERATOR NON-LOCKOUT TRIP	JACKET COOLANT TEMP. LOW
SERVICE WATER FLOW LOW	D.C. CONTROL POWER FAILURE
LUBE OIL LEVEL LOW	GENERATOR EXCITATION LOSS
JACKET COOLANT LEVEL LOW	GENERATOR UNDERFREQUENCY
4160 V. AUXILIARY POWER LOSS	STARTING AIR PRESSURE LOW
GENERATOR BEARING TEMP. HIGH	LUBE OIL TEMPERATURE HIGH
GENERATOR UNDERVOLTAGE	JACKET COOLANT TEMP. HIGH
FUEL OIL DAY TANK LEVEL LOW	FUEL OIL SUPPLY VLV. SHUT
PRE-LUBE PUMP RUNNING	
KEYLOCK SW IN SLOW START	

REMOTE ALARMS AT C08 AND PLANT COMPUTER INPUTS

- DIESEL GENERATOR 12U BKR. TRIP
- DIESEL GEN. 12U CONT. SW. IN MAINT.
- DIESEL GEN. 12U AUTO VOLT REG SETPOINT NOT AT 4160
- DIESEL GEN. 12U TROUBLE
- DIESEL GEN. 12U EXCITER NOT RESET
- DIESEL GEN. 12U SUPPLY TANK LEV. HI/LO
- DIESEL GEN. 12U VENT FAN FAILURE
- DIESEL GEN. 12U ROOM TEMP. HI/LO
- DIESEL GEN. 12U READY TO LOAD
- DIESEL GEN. 12U AUTO. START
- *DIESEL GEN 12U SIAS START*
- DIESEL GEN. 12U DIFFERENTIAL LOCKOUT
- DIESEL GEN. 12U BKR. CLOSING CKT. BLOCKED
- DIESEL GEN. 12U START AIR PRESS LOW
- DIESEL GEN. 12U DISABLED

(A) (V) (PC)

NOTES:

1. SAME KEY OPERATES 43-MSS (LOCAL-REMOTE) SWITCH ALSO OPERATES 43-MSS3 (MAINT.) SWITCH.
2. IF DIESEL DOES NOT START WITHIN 12 SEC. THE TDDO RELAY OUTPUT WILL ENERGIZE SFR RELAY AND STOP ENGINE
3. REFER TO 25203-28110 SH.12 FOR LOGIC DIAGRAM, FUEL OIL SYSTEM ALARMS AT C08.

Q.A.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
13	6-6-06	INCORP. DCN DM2-00-0027-06	DLW	WBZ	-	ASK
12	7-21-05	INCORP. DCN DM2-00-0287-05	SMC	DLW	-	ASK
11	10-4-04	INCORP. DCN DM2-00-0556-02	DLW	WBZ	-	DED
10	3-11-02	INCORP'D DCNS DM2-00-1736-98 & DM2-00-0480-01	SSP	DVP	RSTG	MAG
9	10-10-01	INCORP'D DCN DM2-00-0865-99	TAM	RAH	MAG	GMO
8	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
7	10-12-98	INCORP. DCN's DM2-00-0256-97, 01-0354-97, 01-0490-97	WBZ	MM	JPP	JPP

Dominion® Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE UNIT 2
LOGIC DIAGRAM
DIESEL GENERATOR (H7A) & DIESEL BKR. (A312) CONTROL
WATERFORD, CT.

BY	JMM	CHKD.	APP.	APP.
DATE	4/15/96	DATE	DATE	DATE
SCALE	N/A	DWG. NO.	25203-30034 SH. 1	
	N/A			

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UNIT PARALLEL SW. TURN TO PARALLEL POSITION

SETS GOVERNOR TO DROOP MODE

SETS VOLTAGE ADJUSTER TO DROOP MODE

SETS GOVERNOR TO ISOCHRONOUS MODE

SETS VOLTAGE REGULATOR TO AUTO MODE

LOSS OF PLANT POWER2

SIAS2

LOSS OF CONTROL DC POWER

SELECTOR SW. IS IN LOCAL POSITION (43-MSS) LOCAL CONT. PUSHBUTTONS START (OFF SHD PNL) (PB-01)

CONT'L RM. EMERG. ACTUATED PB DIESEL ROOM

SELECTOR SW. IS IN REMOTE POSITION (KEY LOCKED) (43-MSS)

REMOTE CONT'L SW. START POSITION (DG-47B)

MAINT. SW. IS NOT IN MAINT. POS. (43-MSS3)

OIL PRESS. ESTABLISHED (OPS)

DIESEL AT 810 R.P.M.

DIESEL AT 200 R.P.M. (LSR)

AND

(HSR)

4 START RELAY

LOCAL START PUSH-BUTTON ACTUATED (HS-8298)

START PUSHBUTTON AT M.C.C. ACTUATED

LOSS OF PLANT POWER

DIESEL GENERATOR SCHEMER ACTUATED

DE-ENERGIZE TO OPEN JACKET COOLANT VALVE

STARTS DIESEL GENERATOR VENT FAN MF-38B

OPS1 LUBE OIL PRESSURE LOW

OPS2 LUBE OIL PRESSURE LOW

OPS3

JACKET COOLANT TEMPERATURE HIGH (CTH)

CRANKCASE PRESSURE HIGH (CCP)

FUEL OIL PRESSURE LOW (FLA)

LUBE OIL TEMPERATURE HIGH (OTHA)

DC CONTROL POWER FAILURE (DR)

ENGINE OVERSPEED (EOS)

JACKET COOLANT PRESSURE LOW (CPLA)

ENGINE SPEED 810 RPM

DIESEL ENGINE SLOW START SW IN NORM POS

ALARM PUSHBUTTON RESET (AT ENGINE PANEL)

ENGINE OVERSPEED RESET (MECHANICAL)

ENGINE OVERSPEED 118% FULL R.P.M (EOR)

86 (86-12A-2 ENERGIZED)

2/3

LUBE OIL PRESS. LOW

SEE NOTE 2

TDDO 12 SEC.

SFR

ALARM PUSHBUTTON RESET (AT ENGINE PANEL)

2-REMOTE EMERGENCY STOP PUSHBUTTONS LOCAL CONT. PUSHBUTTONS

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

2-LOCAL EMERGENCY STOP PUSHBUTTONS (PB-031, 2 PER SET)

LOSS OF PLANT POWER

SIAS 2

LOCAL START PUSH BUTTON (PB-15)

LOCAL/REMOTE SWITCH IS IN LOCAL (43-MSS)

REMOTE START SWITCH TO START (DG-47B)

MAINTANCE SWITCH IS IN MAINTANCE POSITION (43-MSS3)

LOCAL STOP PUSH BUTTON ACTUATED AT ENGINE PANEL (PB-2)

NORMAL/MAINTENANCE SWITCH IS IN MAINTENANCE POSITION (43-MSS3)

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

LOCAL STOP PUSH BUTTON ACTUATED (PB-14)

LOCAL REMOTE SEL. SW. IS IN REMOTE POS. (43-MSS)

REMOTE CONT'L SW. START POS. (DG-47B)

ALR COMMON ALARM RELAY

ESS1

86 (86-12A-2 ENERGIZED)

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION

LOCAL STOP PB. NOT ACTUATED (PB-14)

LOCAL FIELD FLASHING PUSHBUTTONS NOT ACTUATED (PB-13) (PB-10)

SHUTDOWN RELAY (SD)

REMOTE CONT. SW. NOT IN STOP POS. (DG47B)

LOCAL REMOTE SEL. SW. IS IN REMOTE POSITION (43-MSS)

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

ENGINE SPEED 250 R.P.M.

ENGINE SPEED 810 R.P.M.

OIL PRESS. ESTABLISHED (OPS)

LOSS OF DC CONTROL POWER

LOCAL REMOTE SW. IS IN LOCAL POS.

MAINTANANCE SW. IS IN MAINTENANCE. (43-MSS3)

LOCAL START PUSH-BUTTON AT ENGINE PANEL (PB-1)

LOSS OF DC CONTROL POWER

LOSS OF DC CONTROL POWER

TDDO 60 SEC

DE-ENERGIZE TO OPEN AIR SUPPLY VALVE (AS1)

DE-ENERGIZE TO OPEN AIR SUPPLY VALVE (AS2)

DE-ENERGIZE TO CLOSE AIR VENT VALVE (AV)

DIESEL STARTS CRANKING

ENERGIZE GOVERNOR SOL TO STOP ENGINE

B4-2/2133-24D-2

VOLTAGE RESTRAINT OVERCURRENT RELAY (51V)

REVERSE POWER RELAY (32)

OVERCURRENT RELAY (50/1)

GROUND FAULT RELAY (50GS)

52/STA. 34B-24E-2 CLOSED

ST/STA. 2133-24D-2 CLOSED

52/STA. 2133-24D-2 CLOSE

LOCAL REMOTE SEL. SW. IS IN LOCAL POSITION (43-MSS)

SYNC SWITCH ON (LOCAL) (25-NCS)

CONTROL SWITCH IS IN CLOSED POSITION (LOCAL) (CS-3)

LOCAL REMOTE SEL. SWITCH IS IN REMOTE POSITION (43-MSS)

SYNC. SWITCH ON (REMOTE) (25-NCS)

CONTROL SWITCH IS IN CLOSED POSITION (REMOTE) (CS-3)

LOCAL CONT. SW IS IN LOCAL POS. (CS-3)

REMOTE CONTL SW. IS IN TRIP POSITION (CS-2)

A4 BUS UNDERVOLTAGE

D/G BKR CLOSING CIR. BLOCK RESET (52/BLK)

LOCAL REMOTE SW. IS IN LOCAL POSITION (43-MSS)

LOCAL REMOTE SW. IS IN AUTO. POSITION (REMOTE)

LOCAL REMOTE SW. IS IN LOCAL POSITION (43-MSS)

CS-3 IS IN AUTO. POSITION (LOCAL)

FIELD RESET PUSHBUTTON DE-ENERGIZED (PB-12)

EXCITER FIELD ENERGIZED (FIELD FLASH)

DIESEL GENERATOR OUTPUT VOLTAGE ESTABLISHED

TRIP BREAKER 52-A401

CLOSE BREAKER 52-A401

(B) COIX

(W) COIX

INTERLOCK TO DIESEL SEQUENCER

CONTROL SWITCHES LOCATED AT MAIN CONTROL BOARD C08

SB1 43-VRMS VOLTAGE REGULATOR MAN-AUTO MAINT.

SBM CS-6S GOVERNOR RAISE-LOWER SPRING RET.

SBM CS-7O MAN. VOLTAGE ADJUST RAISE-LOWER SPRING RET.

SBM CS-9O AUTO. VOLTAGE ADJUSTER RAISE-LOWER SPRING RET.

SB1 43-MSS L/R SELECTOR MAINT SEE NOTE #1

SB1 43-UPS UNIT PARALLEL SPRING RET

SBM DG-47B START-STOP SPRING RET

SB1 25-CS SYNC MAINT.

SB1 CS-2 DIESEL BKR CONTROL SPRING RET.

ENGINE EMERG. SHUTDOWN PB-021

ENGINE EMERG. SHUTDOWN PB-022

CR2940 CS-023B ENGINE PRE LUBE MAINTAINED CONTACTS

CONTROL SWITCHES LOCATED AT DIESEL CONTROL PANEL C39

SB1 43-SLST SLOW-START KEY LOCK

SB1 43-VRMS VOLTAGE REGULATOR MAN-AUTO MAINT.

SBM CS-6SA GOVERNOR RAISE-LOWER SPRING RET.

SBM CS-7OA MAN. VOLTAGE ADJUST. RAISE-LOWER SPRING RET.

SBM CS-9OA AUTO VOLTAGE ADJUSTER RAISE-LOWER SPRING RET.

SB1 43-UPS UNIT PARALLEL SPRING RET.

SB1 25-CS SYNC. MAINT

SB1 43-MSS3 MAINTENANCE MAINT. SEE NOTE #1

SB1 CS-3 DIESEL BKR CONTROL SPRING RET.

EXCITER RESET CR2940 PB-12

ENGINE START CR2940 PB 15

ENGINE STOP CR2940 PB-14

DIESEL ROOM EMERGENCY CR2940 BREAK GLASS PUSHBUTTON PB-100

ENGINE EMERG SHUTDOWN PB-031 CR2940

ENGINE EMERG SHUTDOWN PB-032 CR2940

EXCITER SHUTDOWN CR2940 PB-10

EMERGENCY SHUTDOWN CR2940 PB-13

CONTROL SWITCHES AT ENGINE PANEL T041

CR2940 RS-6 ENGINE PRE LUBE MAINTAINED CONTACTS

ENGINE START CR2940 PB-1

ENGINE STOP CR2940 PB-2

ALARM RESET CR2940 PB-3

LOCAL ALARMS AT C39

AUX. CONT. SW. NOT IN AUTO POSITION	JACKET COOLANT PRESSURE LOW
GENERATOR NEUTRAL GROUND FAULT	480 V. AUXILIARY POWER LOSS
D. C. AIR COMPRESSOR STARTED	GENERATOR STATOR TEMP. HIGH
CRANKCASE PRESSURE HIGH	GENERATOR OVERVOLTAGE
ENGINE START FAILURE	FUEL OIL PRESSURE LOW
ENGINE OVERSPEED	LUBE OIL PRESSURE LOW
GENERATOR LOCKOUT TRIP (DIFFERENTIAL)	LUBE OIL TEMPERATURE LOW
GENERATOR NON-LOCKOUT TRIP	JACKET COOLANT TEMP. LOW
SERVICE WATER FLOW LOW	D. C. CONTROL POWER FAILURE
LUBE OIL LEVEL LOW	GENERATOR EXCITATION LOSS
JACKET COOLANT LEVEL LOW	GENERATOR UNDERFREQUENCY
4160 V. AUXILIARY POWER LOSS	STARTING AIR PRESSURE LOW
GENERATOR BEARING TEMP. HIGH	LUBE OIL TEMPERATURE HIGH
GENERATOR UNDERVOLTAGE	JACKET COOLANT TEMP. HIGH
FUEL OIL DAY TANK LEVEL LOW	FUEL OIL SUPPLY VLV. SHUT
PRE LUBE PUMP RUNNING	
KEYLOCK SW IN SLOW START	

REMOTE ALARMS AT C08 AND PLANT COMPUTER INPUTS

- DIESEL GENERATOR 13U BKR. TRIP
- DIESEL GEN. 13U CONT. SW. IN MAINT.
- DIESEL GEN 13U AUTO VOLT REG SETPOINT NOT AT 4160
- DIESEL GEN. 13U TROUBLE
- DIESEL GEN. 13U EXCITER NOT RESET
- DIESEL GEN. 13U SUPPLY TANK LEV. HI/LO
- DIESEL GEN. 13U VENT FAN FAILURE
- DIESEL GEN. 13U ROOM TEMP. HI/LO
- DIESEL GEN. 13U READY TO LOAD
- DIESEL GEN. 13U AUTO. START
- DIESEL GEN. 13U SIAS START
- DIESEL GEN. 13U DIFFERENTIAL LOCKOUT
- DIESEL GEN. 13U CLOSING CKT. BLOCKED
- DIESEL GEN. 13U START AIR PRESS LOW
- DIESEL GEN. 13U DISABLED

NOTES:

1. SAME KEY OPERATES 43-MSS (LOCAL-REMOTE) SWITCH ALSO OPERATES 43-MSS3 (MAINT.) SWITCH.
2. IF DIESEL DOES NOT START WITHIN 12 SEC. THE TDDO RELAY OUTPUT WILL ENERGIZE SFR RELAY AND STOP ENGINE
3. REFER TO 25203-28110 SH.12 FOR LOGIC DIAGRAM, FUEL OIL SYSTEM ALARMS AT C08.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
13	6-6-06	INCORP DCN DM2-00-0027-06	DLW	WBZ	-	ASK
12	7-21-05	INCORP DCN DM2-00-0287-05	SMC	DLW	-	ASK
11	3-13-03	INCORP DCN DM2-00-0506-02	SMH	DLW	N/A	DED
10	4-8-02	INCORP DCN DM2-00-0480-01 DM2-00-1736-98	JSJ	DLW	N/A	RJY
9	10-10-01	INCORP'D DCN DM2-00-0865-99	TAM	RAH	MAG	GMO
8	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
7	10-12-98	INCORP. DCN's DM2-00-0356-97, 01-0354-57,01-0490-97	WBZ	MM	JPP	JPP

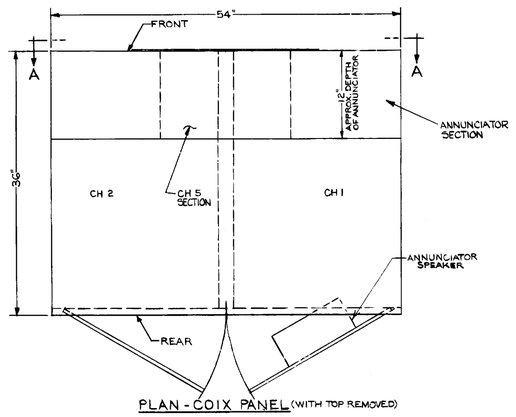
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

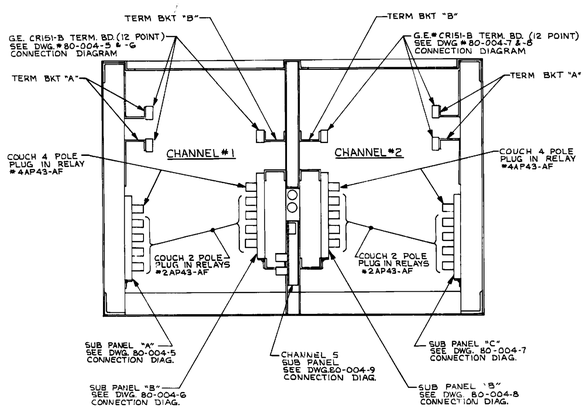
TITLE: MILLSTONE UNIT 2
LOGIC DIAGRAM
DIESEL GENERATOR (H7B) &DIESEL BKR. (A401) CONTROL
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ	—	—	—
DATE 1-31-96	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-30034 SH. 2		
N/A			

25203-30071-SH-0000000



PLAN - COIX PANEL (WITH TOP REMOVED)

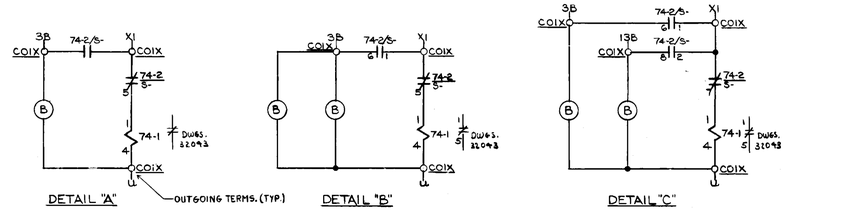


SECTION A-A
(INTERNAL SUB-PANEL LAYOUT)

ITEM/ DETAIL #	DEVICE NO.	DESCRIPTION	SCHEME NO.	SCHEMATIC DWG. NO.
1104	RB-28.1A	CTMT AIR RECIRC COOLER INLET VLV	HV6072	32015 SH. 26
1105	RB-28.1C	CTMT AIR RECIRC COOLER INLET VLV	HV6075	32015 SH. 28
1106	CS-13.1A	RWST STOP MOV	B5124	32009 SH. 27
1107	SI-656	HPSI PP DIS AUX HDR MOV	B5106	32008 SH. 34
1108	SI-306	SDC BYP E/P THROTTLE VLV	SI306	32008 SH. 61
1109	SI-657	SDC FLOW CNTL THROTTLE VLV	SI657	32008 SH. 60
1103	MF-134	VITAL SWGR RM CLG FAN	B5163	32023 SH. 71
1110	HV5276	AUX FEEDWATER VALVE	HV5276	32012 SH. 21
1111	MP9A	AUX FEEDWATER PUMP	A307	32012 SH. 11
1112	H-5A	MAIN FEED PUMP A	NF07	32012 SH. 7
1113	MS-265B	#1 S.G. MAIN STEAM LOW POINT DRAIN	HV4193	32020 SH. 45
1114	FW-5A	#1 S.G. F.W. CK. VLV.	HV5419	32012 SH. 25
1115	FW-51A	#1 S.G. F.W. CNTL VLV	IN25268	28500 SH. 1120
1116	FW-41A	#1 F.W. CNTL BYPASS VLV	N25215	28500 SH. 570
2103	MF-133	VITAL SWGR RM CLG FAN	B5162	32023 SH. 70
2104	RB-28.1B	CAR CLR INLET VLV	HV6073	32015 SH. 27
2105	RB-28.1D	CAR CLR INLET VLV	HV6077	32015 SH. 29
2106	CS-13.1B	RWST STOP MOV	B6125	32009 SH. 28
2107	SI-654	HPSI PP DIS AUX HDR MOV	B6106	32008 SH. 8
2108	MF-142	VITAL SWGR RM EXHAUST FAN	B6243	32023 SH. 80
2110	HV-5279	AUX FEEDWATER VALVE	HV5279	32012 SH. 22
2111	MP9B	AUX FEEDWATER PUMP	A406	32012 SH. 12
2112	H-5B	MAIN FEED PUMP B	NF08	32012 SH. 9
2113	MS-266B	#2 S.G. MAIN STEAM LOW POINT DRAIN	HV4209	32020 SH. 46
2114	FW-5B	#2 S.G. F.W. CK. VLV	HV5420	32012 SH. 26
2115	FW-51B	#2 S.G. F.W. CNTL VLV	2N25269	28500 SH. 1121
2116	FW-41B	#2 F.W. CNTL BYPASS VLV	N25216	28500 SH. 571
2117	HV5247	MAIN FEEDWATER PP 1B DISCH. VLV	2B2113	32012 SH. 20
2118	HV5264	MAIN FEEDWATER TO STM GEN NO. 2	2B2114	32012 SH. 18
1117	HV5245	MAIN FEEDWATER PP 1A DISCH VLV	1B1210	32012 SH. 19
1118	HV5263	MAIN FEEDWATER TO STM GEN NO.1	1B1211	32012 SH. 17

ITEM/ DETAIL #	DEVICE NO.	DESCRIPTION	SCHEME NO.	SCHEMATIC DWG. NO.
501/A	P-5B	SERVICE WTR PUMP B	Z5A502	32013 SH. 6
502/A	P-41B	HPSI PUMP B	Z5A503	32008 SH. 38&9
503/A	P-18B	CHARGE PUMP B	Z5MCV01	32009 SH. 418.48
504/A	P-11B	RBCCW PUMP B	Z5A504	32015 SH. 2
505				
101/A	P-5A	SERVICE WATER PUMP A	Z1A306	32013 SH. 5
102/A	SW-8.1B	RBCCW HX B OUTLET VLV	Z1TV6307	32013 SH. 41
103/A	SW-8.1A	RBCCW HX A OUTLET VLV	Z1TV6308	32013 SH. 40
104/A	SW-3.2B	TBCCW HDR A INLET VLV	Z1HV6438	32013 SH. 19
105/A	E-42A	LPSI PUMP A	Z1A305	32008 SH. 48&4
106/A	P-41A	HPSI PUMP A	Z1A308	32008 SH. 18&9
107/A	SI-618	SI HDR 1A CK VLV LEAK DRN CNTL VLV	Z1SI618	32008 SH. 56
108/A	SI-628	SI HDR 1B CK VLV LEAK DRN CNTL VLV	Z1SI628	32008 SH. 57
109/A	CH-515	LETDN SYS SIAS ISO VLV	Z1CH515	32009 SH. 33
110/A	F-15A	ENG. SFGRD FROM FAN A	Z1B5139	32022 SH. 17
111/A	X-169A	CHILLER SYSTEM A	Z1HV8846	32023 SH. 49
112/A	CH-512	VCT MAKE-UP STOP VLV	Z1CH512	32009 SH. 21
113/A	P-11A	RBCCW PUMP A	Z1A311	32015 SH. 1
114/A	RB-28.1A	ENG. SFGRD COOLER A OUTLET	Z1HV6731	32015 SH. 8
115/A	RB-28.3A	CAR CLR A EMRG OUTLET	Z1HV6080	32015 SH. 31
116/A	RB-28.3C	CAR CLR C EMRG OUTLET	Z1HV6088	32015 SH. 35
117/A	RB-8.1A	CCWB OUT SFPC HX A & BA EVAP	Z1HV6315	32015 SH. 12
118/A	SI-614	SI TK #1 ISOL VLV	Z1B5111	32008 SH. 32
119/A	SI-624	SI TK #2 ISOL VLV	Z1B5112	32008 SH. 38
120/A	SI-615	LPSI HDR 1A INJ VLV	Z1B5118	32008 SH. 21
121/A	SI-625	LPSI HDR 1B INJ VLV	Z1B5119	32008 SH. 22
122/A	SI-617	HPSI HDR A INJ TO LOOP 1A	Z1B5114	32008 SH. 17
123/A	SI-627	HPSI HDR A INJ TO LOOP 1B	Z1B5117	32008 SH. 18
124/A	SI-637	HPSI HDR A INJ TO LOOP 2A	Z1B5115	32008 SH. 19
125/A	SI-647	HPSI HDR A INJ TO LOOP 2B	Z1B5116	32008 SH. 20
126/A	CH-508	B A TK B GRAVITY FEED VLV	Z1B5151	32009 SH. 8
127/A	CH-509	B A TK A GRAVITY FEED VLV	Z1B5149	32009 SH. 3
128/A	CH-501	VCT OUTLET TO CHG PP SUC	Z1B5145	32009 SH. 6
129/A	P-18A	CHARGING PUMP A	Z1B5103	32009 SH. 40&48
130/C	F-14A	CAR FAN A	Z1B0509	32022 SH. 1
131/A	CH-196	VCT MAKE-UP BYPASS VLV	Z1CH196	32009 SH. 54
132/C	F-14C	CAR FAN C	Z1B0511	32022 SH. 2
133	156-12U	DG H7A READY TO LOAD	Z1A312	32041 SH. 1
134				
135				
136				
137				
138				
139				
140				
141/A	P-43A	CTMT. SPRAY PUMP A	Z1A310	32009 SH. 6&69
144/A	CS-4.1A	CS HDR ISOL VLV	Z1B5150	32008 SH. 29
145/A	MS-220A	#1 SG BLDN CNTL VLV	Z1HV4246	32020 SH. 14
146/A	MS-191A	#1 SG SAMPLE ISOL VLV	Z1HV4250	32020 SH. 11
147/A	PWR-43C	PWR CTMT. ISOL VLV	Z1HV7311	32007 SH. 31
148/A	AC-1	CTMT. & EB PURGE FAN DIS	Z1HV8020	32022 SH. 23
149/A	CH-5058/CH-198	RCP CONTROL BLEED OUTBD ISOL	Z1CH505	32009 SH. 30
150/A	LRR-43.2	PDT CTMT. ISOL OUTBD	Z1HV9016	32025 SH. 11
151/A	GR-11.2	WASTE GAS HDR CTMT. ISOL OUTBD	Z1HV9126	32026 SH. 2
152/A	SSP-16.2	CTMT. SUMP PP DISCH ISOL OUTBD	Z1HV9150	32030 SH. 3
153/A	RC-45	RC COMBINED SAMPLE ISOL VLV	Z1HV7690	32027 SH. 3
154				
155/A	AC-3	PURGE SUPPLY TO EB	Z1HV8081	32022 SH. 46
156				
157/A	AC-8	PURGE EXH. FROM EB	Z1HV8127	32022 SH. 48
158/A	AC-12	CTMT. RAD MON ISOL	Z1HV8122	32022 SH. 27
159/A	AC-15	CTMT. RAD MON ISOL	Z1HV8124	32022 SH. 43
160/A	F-39A	CTMT. RAD MON FAN/ISOL VLV'S	Z1B5220	32022 SH.14A-C
161/A	EB-88	H2 MON ISOL (EMERG)	Z1HV8150	32022 SH. 30
162/A	EB-99	HYD PURGE ISOL VLV IN EB	Z1HV8377	32022 SH. 59
163/A	EB-100	H2 PURGE ISOL VLV IN CTMT.	Z1HV8378	32022 SH. 60
164				
165				
166				
167				
168				
169/A	SI-659	SI/CS PP MIN FLOW HDR STOP	Z1SI659	32008 SH. 54
170/A	RB-13.1A	SDC HX A OUTLET VLV	Z1HV6050	32015 SH. 10
171/A	CS-16.1A	CTMT SUMP OUTLET HDR A ISOL	Z1B5122	32008 SH. 25
172/A	P-42A	LPSI PUMP A	Z1A309	32008 SH. 48&4
173				
174	MF32A	CONTROL ROOM FILTER FAN	Z1B5213	32023,SH.15
175/B	EB-51	INLET TO EBFS FILTER A	Z1HV8074	32022, SH. 35
176/A	HV-107	SUPPLY DAMPER TO ESFRM	Z1HV8247	32021. SH. 12
177/A	HV-116	EXH DAMPER FROM ESFRM	Z1HV8249	32021. SH. 14
178/C	EB-50	SPLY FROM EB TO EBSF	Z1HV8153	32022. SH. 37
179/C	EB-60	SPLY FROM FUEL HAND TO EBSF	Z1HV8443	32022. SH. 52
180/B	F-25A	E B FILTRATION FAN A	Z1B5215	32022, SH. 21
181/B	EB-73	SPLY FROM PURGE SYS	Z1HV8851	32022 SH.68
182/B	EB-56	AIR REMOVAL DISCHARGE TO EBSF	Z1HV8695	32022. SH. 66
183/B	EB-52	EBFS FAN A DIS DMPR	Z1HV8079	32022. SH. 34
184/B	C25A	CONTROL ROOM A/C SYS A	Z1VA1005	32023, SH. 3.8
185	MF32A	CONTROL ROOM FILTER FAN	Z1B5213	32023,SH.15
186/B	C25A	CONTROL ROOM A/C SYS A	Z1VA1005	32023, SH. 3.8
187/B	EB-51	INLET TO EBFS FILTER A	Z1HV8074	32022, SH. 35
188/B	F-25A	EBFS FAN A	Z1B5215	32022, SH. 21
189/B	EB-73	SPLY FROM PURGE SYS	Z1HV8851	32022 SH.68
190/B	EB-56	AIR REMOVAL DIS TO EBFS	Z1HV8695	32022. SH. 66
191/A	F-20	FUEL HANDLING AREA FAN	Z1B5234	32021. SH. 6
192/A	HV-171	EXH B/U TO MN EXH	Z1HV8275	32021. SH. 4
193/C	EB-60	SPLY FROM FUEL HAND SYS TO EBFS	Z1HV8145	32022. SH. 52
194/C	EB-50	SPLY FROM EB TO EBFS	Z1HV8153	32022. SH. 37
195/B	EB-52	EBFS FAN A DIS DMPR	Z1HV8079	32022. SH. 34
196/A	F-51	VITAL SWGR RM. CLG FAN	Z1B5160	32023, SH. 42
197/A	MS-65A	#1 SG MSIV BYPASS	Z1B5207	32020. SH. 1
198/A	MS-64A	#1 SG MSIV	Z1HV4217	32020. SH. 18
199/A	MS-64B	#2 SG MSIV	Z1HV4221	32020. SH. 20
1100/A	MF-38A	DG VENT FAN A	Z1B5140	32023,SH.7
1101	156-12U	D.G REMOTE CONTROL	2L D601	32041, SH. 2
1102/A	156-12U-2	D.G. BKR.	2EIA312	32041,SH. 1

ITEM/ DETAIL #	DEVICE NO.	DESCRIPTION	SCHEME NO.	SCHEMATIC DWG. NO.
201/A	P-5C	SERVICE WATER PUMP C	Z2A407	32013, SH. 7
202/A	SW-8.1C	RBCCW HX C OUTLET VLV	Z2TV6306	32013, SH. 42
203/A	SW-8.1B	RBCCW HX B OUTLET VLV	Z2TV6307	32013, SH. 43
204/A	SW-3.2A	TBCCW HDR B INLET VLV	Z2HV6435	32013, SH. 20
205/A	P-42B	LPSI PUMP B	Z2A404	32008, SH. 24&5
206/A	P-41C	HPSI PUMP	Z2A403	32008, SH. 24&5
207/A	SI-638	SI HDR 2A CK VLV LEAK DRN CNTL LVL	Z2SI638	32008, SH. 58
208/A	SI-648	SI HDR 2B CK VLV LEAK DRN CNTL LVL	Z2SI648	32008, SH. 59
209/A	CH-511	BA. TK REGINC TO TK B	Z2CH511	32009, SH. 25
210/A	P-11C	RBCW PUMP C	Z2A310	32015, SH. 3
211/A	RB-68.1B	ENG. SFGRD COOLER B OUTLET	Z2HV6735	32015, SH. 9
212/A	RB-28.3B	CAR CLR B EMRG OUTLET	Z2HV6084	32015, SH. 33
213/A	RB-28.3D	CAR CLR D EMRG OUTLET	Z2HV6092	32015, SH. 37
214/A	RB-210	SAMPLE A DEGAS CLR OUTLET	Z2HV7353	32015, SH. 38
215/A	RB-8.1B	COMB OUTLET SFPC HX B & AUX WST EVAP	Z2B5139	32022 SH. 17
216/A	SI-634	SI TANK #3 ISOL VLV	Z2B5109	32008, SH. 39
217/A	SI-644	SI TANK #4 ISOL VLV	Z2B5110	32008, SH. 40
218/A	SI-635	LPSI HDR 2A INJ VLV	Z2B5116	32008, SH. 23
219/A	SI-645	LPSI HDR 2B INJ VLV	Z2B5111	32008, SH. 24
220/A	SI-616	HPSI HDR B INJ TO LOOP 1A	Z2B5112	32008, SH. 14
221/A	SI-626	HPSI HDR B INJ TO LOOP 1B	Z2B5113	32008, SH. 14
222/A	SI-636	HPSI HDR B INJ TO LOOP 2A	Z2B5113	32008, SH. 15
223/A	SI-646	HPSI HDR B INJ TO LOOP 2B	Z2B5114	32008, SH. 16
224/A	CH-514	BA TANK TO CHG PP SUCT	Z2B5162	32009, SH. 10
225/A	P-18C	CHARGING PUMP C	Z2B5102	32008, SH. 43&48
226/A	P-19A	BORIC ACID PUMP A	Z2B5138	32009, SH. 4
227/A	P-19B	BORIC ACID PUMP B	Z2B5135	32009, SH. 5
228/C	F-14B	CAR FAN B	Z2B0604	32022, SH. 3
229				
230/C	F-14D	CAR FAN D	Z2B0602	32022, SH. 4
231				
232/A	F-15B	ENG. SFGRD ROOM FAN B	Z2B5145	32022, SH. 18
233	156-13U	DG H7B READY TO LOAD	Z2A401	32041 SH 2
234/A	CH-510	BA TK REGINC TO TK A	Z2CH510	32009 SH 24
235/A	X-169B	CHILLER SYSTEM B	Z2 Left	32023,SH.50
236				
237/A	P-43B	CTMT SPRAY PUMP B	Z2A405	32008, SH. 7&8
238				
239				
240/A	CS-4.1B	CS HDR B ISOL VLV	Z2B5124	32008, SH. 30
241/A	MS-220B	#2 SG BLWDN CNTL VLV	Z2HV4248	32020, SH. 15
242/A	MS-191B	#2 SG SAMPLE ISOL VLV	Z2HV4251	32020, SH. 12
243/A	RC-003	PZR STM SAMPLE ISOL	Z2N1002	32007, SH. 28
244/A	RC-002	PZR SURGE SAMPLE ISOL	Z2HV1002	32007, SH. 29
245/A	RC 001	RC HOT LEG ISOL	Z2HV1004	32007, SH. 30
246/A	SI-312	N2 ISOL VLV OUTSIDE CTMT	Z2HV7312	32008, SH. 49
247/A	CH-506	RCP CNTL BLEEDOFF INBD ISOL	Z2CH506	32009, SH. 31
248/C	CH-089	REGEN HX OUTLET OUTSIDE/CTMT ISOL	Z2HV2525	32009, SH. 39
249/A	LRR-43.1	PDT CTMT ISOL INB'D	Z2HV3015	32025, SH. 10
250/A	LRR-61.1	PDT SAMPLE ISOL	Z2HV3230	32025, SH. 17
251/A	GR-11.1	WASTE GAS HDR CTMT ISOL INBD	Z2HV9126	32026, SH. 2
252/A	SSP-16.1	CTMT SUMP PUMP DISCH ISOL INB'D	Z2HV9151	32030, SH. 4
253				
254				
255/C	RB-402	LETDOWN HX RBCW OUTLET VLV RB402	Z2HV2525	32009 SH. 38
256/A	AC-47	CTMT RAD MON ISOL	Z2HV8121	32022, SH. 26
257/A	AC-11	PURGE EXH FLTR DIS DMPR	Z2HV8126	32022, SH. 38
258/A	EB-89	H2 MON SAMPLE ISOL(EMERG)	Z2HV8181	32022, SH. 31
259/A	EB-92	H2 PURGE ISOL VLV IN EB	Z2HV8376	32022, SH. 61
260/A	EB-91	H2 PURGE ISOL VLV IN CTMT	Z2HV8380	32022, SH. 62
261/A	F-23	CTMT PURGE SUPPLY FAN	Z2B6183	32022, SH. 10
262/A	F-39B	CTMT. RAD MON FAN/ISOL VLV'S	Z2B5219	32022 SH.15A-15C
263/B	CH-516	LETDOWN SYS CIAS ISOL	Z2CH516	32009, SH. 34
264/A	AC-20	H2 MON SAMPLE ISOL (NORM)	Z2HV8556	32022, SH.73
265				
266				
267/A	SI-660	SI/CS PP MIN FLOW HDR STP	Z2SI660	32008, SH. 55
268/A	RB-13.1B	SDC HX B OUTLET VLV	Z2HV6055	32015, SH. 11
269/A	CS-16.1B	CTMT SUMP OUTLET HDR B ISOL	Z2B5211	32008, SH. 24
270/A	P-42B	LPSI PUMP B	Z2A404	32008, SH. 5&48
271				
272				
273/B	EB-41	INLET TO EBFS FILTER B	Z2HV8070	32022, SH. 33
274/A	HV-106	SUPPLY DAMPER TO ESFRM	Z2HV8133	32021, SH. 13
275/A	HV-117	EXH DAMPER FROM ESFRM	Z2HV8248	32021, SH. 15
276/C	EB-40	SPLY FROM EB TO EBFS	Z2HV8063	32022, SH. 53
277/C	EB-61	SPLY FROM FUEL HAND TO EBFS	Z2HV8062	32022, SH. 39
278/B	F-25B	EB FILTRATION FAN B	Z2B5230	32022, SH. 22
279/A	EB-72	SPLY FROM PURGE SYS.	Z2HV8850	32022 SH.67
280/B	EB-55	AIR REMOVAL DIS TO EBFS	Z2HV8254	32022, SH. 65
281/B	EB-42	EBFS FAN B DISCH DMPR	Z2HV8073	32022, SH. 49
282/A	C25B	CONTROL ROOM A/C SYS B	Z2VA1006	32023, SH. 3.9
283	MF32B	CONTROL ROOM FILTER FAN	Z2B6244	32023,SH.6
284				
285/C	EB-40	SPLY FROM EB TO EBFS	Z2HV8063	32022, SH. 53
286/A	HV-165	FUEL HAND FAN DIS DMPR	Z2HV8141	32021, SH. 8
287/A	EB-61	INLET TO EBFS FILTER B	Z2HV8071	32022, SH. 33
288/B	F-25B	EBFS FAN B	Z2B6230	32022, SH. 22
289/A	EB-72	SPLY FROM PURGE SYS	Z2HV8850	32022 SH.67
290/B	EB-55	AIR REMOVAL DIS TO EBFS	Z2HV8254	32022, SH. 65
291/A	HV-170	EXH DAMPER TO MN EXH	Z2HV8336	32021, SH. 10
292/B	EB-61	SPLY FROM FUEL HAND SYS TO EBFS	Z2HV8062	32022, SH. 39
293/B	EB-42	EBFS FAN B DISCH DMPR	Z2HV8073	32022, SH. 49
294/A	C25B	CONTROL ROOM A/C SYS B	Z2VA1006	32023, SH. 3.9
295	MF-38B	CONTROL ROOM FILTER FAN	Z2B6244	32023,SH. 6
296/A	F-52	VITAL SWGR RM. CLG FAN	Z2B6154	32023,SH.43
297/A	MS-65B	#2 MSIV BYPASS	Z2B6208	32020, SH. 2
298/A	MS-64A	#1 SG MSIV	Z2HV4217	32020, SH. 18
299/A	MS-64B	#2 SG MSIV	Z2HV4221	32020, SH. 20
		ALARM ANNUNCIATOR		
		ALARM ANN. RESET		
		ALARM ANN. FUNCTIONAL TEST		
		ALARM ANN. ACKNOWLEDGE		
		CHANNEL 5 TEST		
		CHANNEL 1 TEST		
		CHANNEL 2 TEST		
2100/A	MF38B	DG VENT FAN B	Z2B6147	32023 SH. 8
2101	156-13U	D.G REMOTE CONTROL	2L D608	32041 SH.15
2102/A	156-13U-2	D G BKR	2EIA401	32041,SH.2



REFERENCE DWGS.
25203-30084 SH.1 THRU 9 - NAMEPLATE ENGRAVINGS
ENG. SAFETY EQUIP. PNL. (COIX)

74-1
COUCH 2CP RELAY
OR DEUTSCH 2AP X1596

74-2
COUCH TYPE 4C RELAY

NOTES
1. CONTACT NOS. ARE TYPICAL SEE DWGS. 32043 FOR CONTACT WIRING.
2. DESCRIPTIONS GIVEN ARE NOT ACTUAL NAMEPLATE ENGRAVINGS.
3. DIMENSIONS ARE APPROX. SEE F.P. 7604-E46-2
4. REFERENCE DWG B7006, SH.6 FOR WINDOW ENGRAVING.
5. THE SAFETY EQUIPMENT STATUS PANEL MUST BE ASSEMBLED, WIRED & TESTED IN STRICT ACCORDANCE WITH BECHTEL SPEC #7604-E-46.
6. REFER TO EMD DWG. 80-004-1 THRU 4 FOR FABRICATION DETAILS.
7. REFER TO EMD DWG. 80-004-5 THRU 9 FOR ELECTRICAL CONNECTION DIAGRAM.
8. CAUTION LABEL BLK/YEL BORDER

CIRCUIT DETAILS

THIS DWG. SUPERCEDES DWG. 25203-39030 SH.15

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REDRAWN FROM SAME MNECO DWG. NO. REV. 11
CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

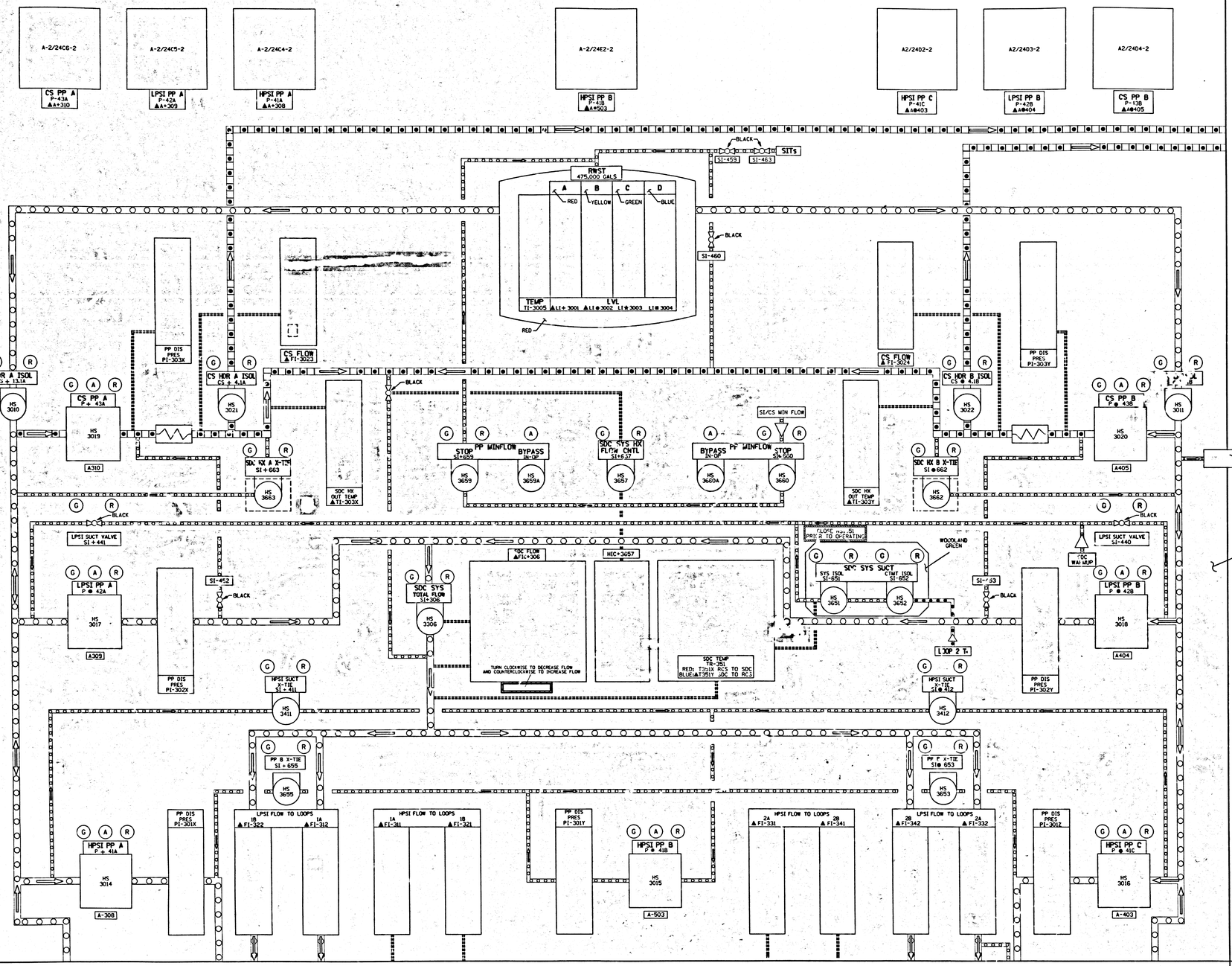
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
		REVISIONS DURING CONSTRUCTION				
17	4-14-05	INCORP. DCN DM2-00-0024-04	JSJ	MEL	—	WBZ
16	12-1-04	INCORP. DCN DM2-00-0025-04	DLW	WBZ	-	DED
15	4-6-02	INCORP. DCN DM2-00-0113-01	BLM	GHH	TP	CAM
14	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
13	5-25-99	INCORP. DCN DM2-04-1005-97, 04-1006-97, 00-0220-98, 01-0382-98 & 00-1695-98	JSJ	WBZ	JPP	JPP
N/A	12	3-31-98 INCORP. DCN DM2-S-0535-96	DSJ	GFD	RWD	RW

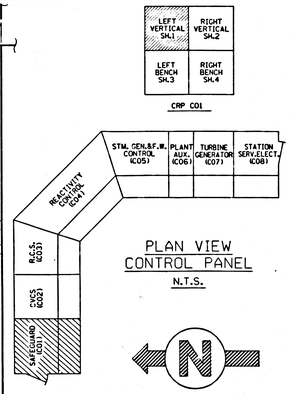
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
ENGINEERED SAFETY EQUIPMENT
STATUS PANEL LAYOUT (C01X)
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JMM	—	—	—
DATE 5-15-96	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-30071		

ANNUNCIATOR
70 ½"





PLAN VIEW
CONTROL PANEL
N.T.S.

MIMIC LEGEND

BLUE	
ORANGE	
RED	
BLACK/WHITE	
PURPLE	
BLACK/YELLOW	

NAMEPLATE LINE SPACING

SEE DWG	
1/8"	SPACE
	LINE 1
1/8"	SPACE
	LINE 2
1/8"	SPACE
	LINE 3
1/8"	SPACE

AS REQ'D

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .
3. [ ] - PROTECTIVE COVERING

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

- ▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
- + = INDICATES AN ½" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
- ● = INDICATES AN ½" YELLOW DOT (Z2 POWERED EQUIPMENT)
- ★ = INDICATES AN ½" GREEN DOT (Z3 POWERED EQUIPMENT)
- ■ = INDICATES AN ½" BLUE DOT (Z4 POWERED EQUIPMENT)
- ✳ = INDICATES AN ½" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28002
25203-29097,SH.45
25203-29097,SH.50A TO 50K

CRP. C01 (LEFT VERTICAL)

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

NO	DATE	REVISIONS	BY	CHK	APP	APP
8	2-11-04	INCORP DCN DM2-00-0150-02	SMH	A-3	-	DJH
7	7-25-01	INCORP. DCN DM2-00-0531-00	JKY	JSJ	DJH	RJY
6	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
5	7-21-98	INCORP. DCN's DM2-00-0532-98, DM2-04-0532-98	WBZ	JSJ	JPP	JPP

REVISIONS DURING CONSTRUCTION | P.A. NO.

Dominion Nuclear Connecticut Inc.
FOR Millstone Power Station

TITLE: MILLSTONE UNIT 2
C01, CRP. FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION
WATERFORD, CONNECTICUT

BY	T.A.SWIFT	CHKD	MBR	APP	MCH	APP	LJC
DATE	7/9/87	DATE	11-24-87	DATE	11-24-87	DATE	11-24-87
SCALE	N.T.S.	DWG. NO	25203-30208 SH.1				

86-230

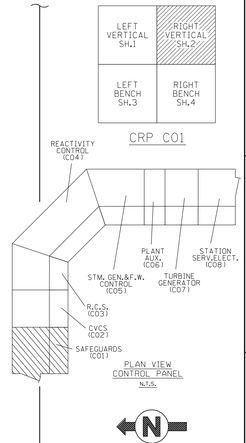
ANNUNCIATOR

A/MF-14A | A/MF-14C | A/MF-24A | A/MF-18A | A/MF-18B | A/MF-24B | A/MF-14B | A/MF-14D

F+14A B0508 CAR F+14C B0510 | CTMT AUX CIRC F-24A B3IB03 | F+18A B5228 PIR F●18B B6214 | CTMT AUX CIRC F-24B B-41B03 | F●14B B0605 CAR F●14D B0603

CONTAINMENT

WOODLAND GREEN



H2 RECOMB A: KW J1+8723 | TEMP IND SW TIS-8721 | POT JC-8723 | TEMP SEL HJS-8721

H2 RECOMB B: TEMP IND SW TIS-8722 | KW J1●8724 | TEMP SEL HJS-8722 | POT JC-8724

SIT 1: PRES ▲PI+311 | LVL ▲LI+311 | N2 HDR | CTMT VENT SI●613 HS 3613 | CHK VLV LKG PRES PI-319 | HDR-1A CK VLV LKG DRN STOP HIC-618 HIC 3618 | FILL & DRN SI-611 HS 3611 | SI OVERRIDE SEL SW+ HS 3614-1 | ISOL ▲SI+614 HS 3614 | SI HS 3618 PUSH TO RESET SI-618 AFTER ESAS OR LCP | LOOP 1A

SIT RECIRC/FILL/DRAIN: RWST | PDT | DRAIN SI-661 HS 3661 | SIT RECIRC HDR FLOW FI-305 | RECIRC HDR PRES PI-305

SIT 3: PRES ▲PI 331 | LVL ▲LI●331 | N2 HDR | CTMT VENT SI+633 HS 3633 | CHK VLV LKG PRES PI-339 | HDR-2A CK VLV LKG DRN STOP HIC-638 HIC 3638 | FILL & DRN SI-631 HS 3631 | SI OVERRIDE SEL SW● HS 3634-1 | ISOL ▲SI● 634 HS 3634 | SI HS 3638 PUSH TO RESET SI-638 AFTER ESAS OR LCP | LOOP 2A

EB△P PDI-8060 | EB△P SEL SW HJS-8060

N2 SPLY HDR STOP ▲SI●312 HS 7312 | N2

CAR COOLING SYS A C | RBCCW | CAR COOLING SYS B D | RBCCW

LT GREEN | EBFS TRAIN A | MILLSTONE STACK | FAN A DIS DMPR EB-52 | FAN A F + 25A | HS 8079 | B5215 | FAN A SUCT X-TIE DMPR EB + 76 | HS 8078 | FLTR A | FLTR B | TEMP ▲TI-6082 | TEMP ▲TI-6090 | FLOW ▲FI-6081 | FLOW ▲FI-6089 | NORM OUTLET RB-28.2C HS 6087 | EMERG. OUTLET RB+28.3C HS 6088 | TEMP TI-8076 | △P PDI-8075

SIT 1: N2 SPLY SI-612 HS 3612 | N2 SPLY SI-622 HS 3622 | RED | SIT 2 | CTMT VENT SI●623 HS 3623 | CHK VLV LKG PRES PI-329 | HDR 1B CK VLV LKG DRN STOP HIC-628 HIC 3628 | FILL & DRN SI-621 HS 3621 | SI OVERRIDE SEL SW+ HS 3624-1 | ISOL ▲SI+624 HS 3624 | SI HS 3628 PUSH TO RESET SI-628 AFTER ESAS OR LCP | LOOP 1B

PRES-LR PI-8117 | TEMP ▲TI-8096 | TEMP SEL SW HS 8096 | SEE DETAIL "A"

A	B	C	D
RED	YELLOW	GREEN	BLUE

PRES-WR ▲PI+8113 ▲PI●8114 PI★8115 PI■8116

SIT 3: N2 SPLY SI-632 HS 3632 | N2 SPLY SI-642 HS 3642 | RED | SIT 4 | CTMT VENT SI-643 HS 3643 | CHK VLV LKG PRES PI-349 | HDR 2B CK VLV LKG DRN STOP HIC-648 HIC 3648 | FILL & DRN SI-641 HS 3641 | SI OVERRIDE SEL SW● HS 3644-1 | ISOL ▲SI● 644 HS 3644 | SI HS 3648 PUSH TO RESET SI-648 AFTER ESAS OR LCP | LOOP 2B

TEMP ▲TI-6086 | TEMP ▲TI-6093 | FLOW ▲FI-6085 | FLOW ▲FI-6094 | NORM OUTLET RB-28.2B HS 6083 | EMERG OUTLET RB ● 28.3B HS 6084

LT GREEN | EBFS TRAIN B | MILLSTONE STACK | FAN B DIS DMPR EB-42 | FAN B F ● 25B | HS 8073 | B6230 | FAN B SUCT X-TIE DMPR EB ● 77 | HS 8254 | FLTR A | FLTR B | △P PDI-8071 | TEMP TI-8072

MIMIC LEGEND

GRAY
BLACK/YELLOW
RED
BLACK/WHITE
LT-BLUE
BLACK
PURPLE
AQUA

NOTES

CRP. C01 (RIGHT VERTICAL)

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

- ▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
- + = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
- ● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
- ★ = INDICATES AN 1/8" GREEN DOT (Z3 POWERED EQUIPMENT)
- ■ = INDICATES AN 1/8" BLUE DOT (Z4 POWERED EQUIPMENT)

NAMEPLATE LINE SPACING — SEE DWG

1/8"	SPACE
	LINE 1
1/8"	SPACE
	LINE 2
1/8"	SPACE
	LINE 3
1/8"	SPACE

AS REQ'D

TEMP. POSITIONS
#1 RX CAVITY N.E.
#2 RX CAVITY S.W.
#3 S/G NO.1 CAVITY
#4 S/G NO.2 CAVITY
#5 38' ELEV. S.E. WALL
#6 38' ELEV. S.W. WALL
#7 95' ELEV. CTMT. DOME
#8 RX CAVITY N.W.

DETAIL "A"

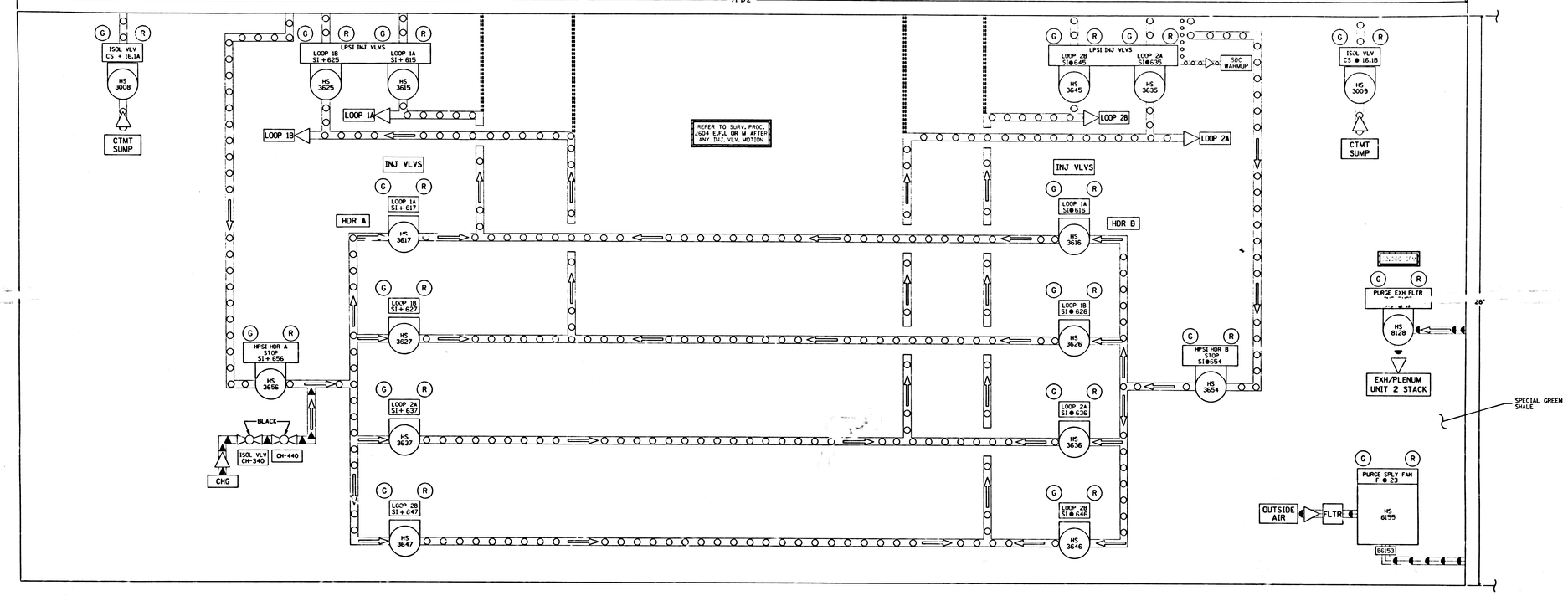
THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28002
25203-29097,SH.45
25203-29097,SH.50A TO 50K

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

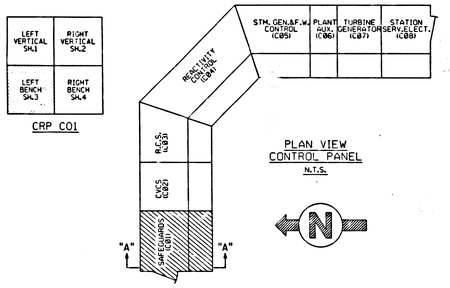
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION

REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER MSDUR 11-800102 & DUR 11-800343
CAD NO: DGNSPEC FOR FILE VERIFICATION
DRAWING NO: 25203-30208 REV 8
DSGN DLW | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 2

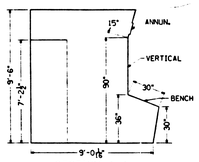


CRP. C01 (LEFT BENCH)



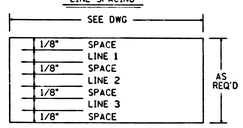
CRP C01

PLAN VIEW
CONTROL PANEL
N.T.S.



SECTION A-A
N.T.S.

NAMEPLATE
LINE SPACING



MIMIC LEGEND

RED
BLACK/WHITE
BLACK/YELLOW
LT-BLUE

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT (TRAIN-A), EXCEPT FOR CAUTION LABELS
   • - INDICATES AN 1/8" YELLOW DOT (TRAIN-B)
   • - INDICATES AN 1/8" BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE.

NON Q.A.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28002
25203-29097,SH.45
25203-29097,SH.50A TO 50K

	5	12-12-03	INCORP. DCN DM2-00-0204-03	JCV			
	4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
	3	11-9-96	INCORP. DCN DM2-S-0403-95 PER DCR-M2-S-0105-96	JMM	MM	JPP	JPP
86-230	2	1-11-91	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
86-230	1	7/26/88	AS BUILT PER DSR-M2-S-302-88	TAS	DJK		LJC
REV	NO	DATE	REVISIONS	D	CHK	APP	APP

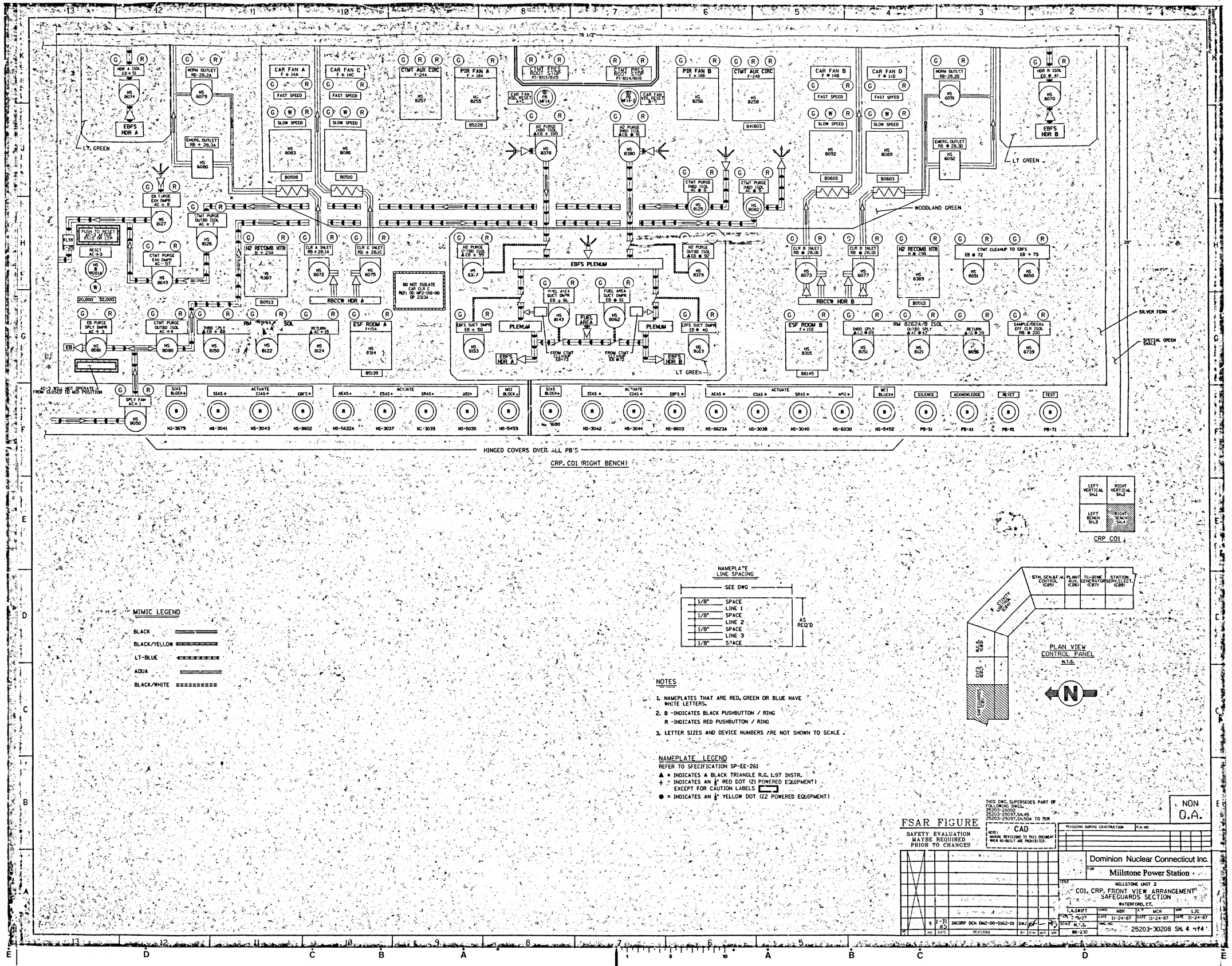
REVISIONS DURING CONSTRUCTION | P.A. NO.

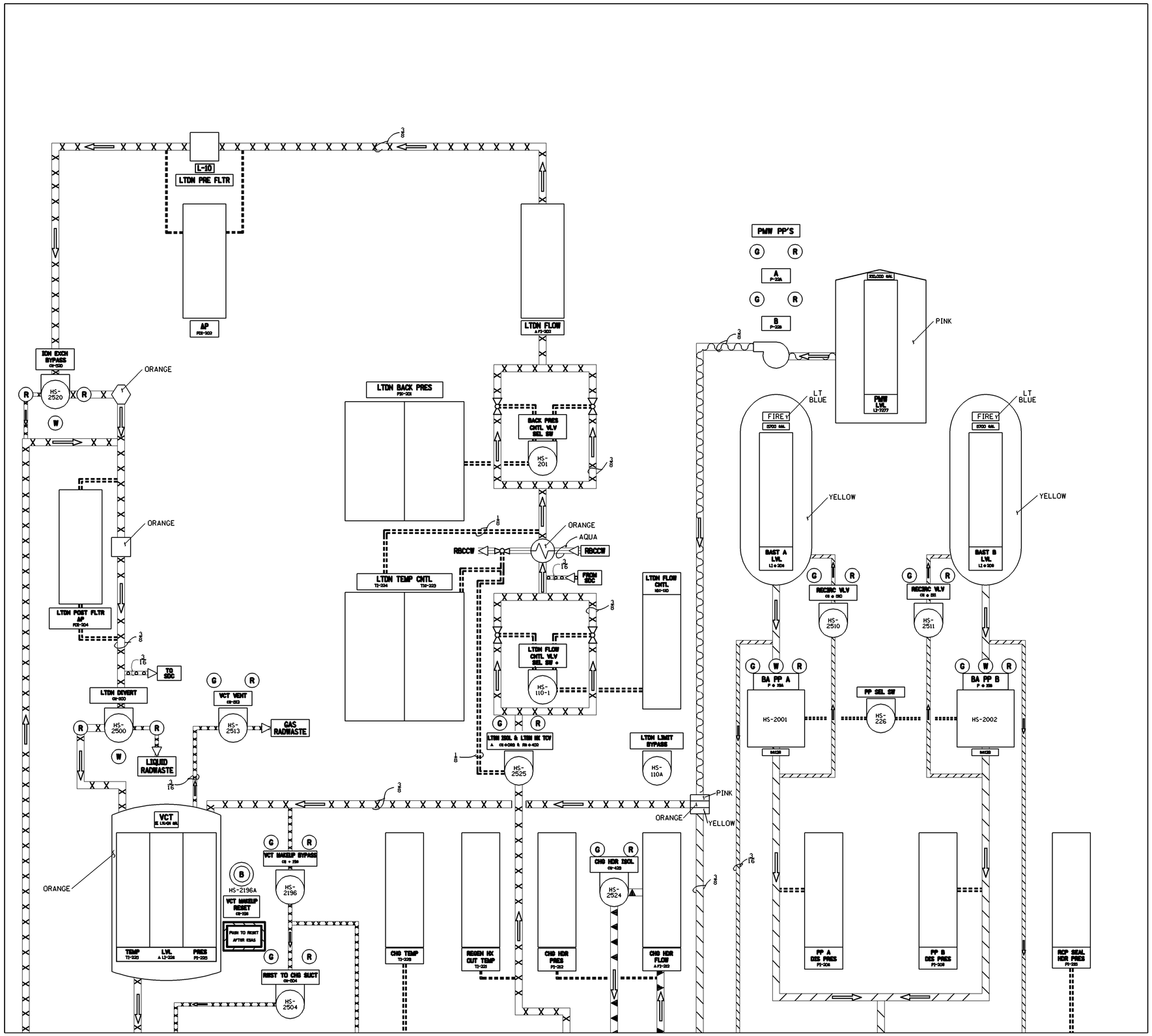
Dominion Nuclear Connecticut Inc.

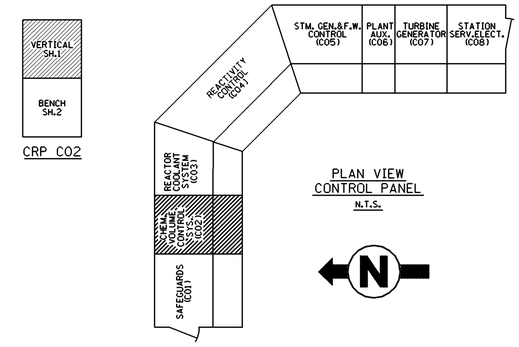
FOR Millstone Power Station

TITLE: MILLSTONE UNIT 2
C01, CRP, FRONT VIEW ARRANGEMENT
SAFEGUARDS SECTION
WATERFORD, CONNECTICUT

T.A.SWIFT	MBR	MCH	LJC
DATE 7/9/37	DATE 11-24-87	DATE 11-24-87	DATE 11-24-87
SCALE N.T.S.	DWG NO. 25203-30208,SH.3 of4		
86-230			


76 1/2"
28"
HDR A ISOL EB ● 51
EBFS HDR A
LT. GREEN
NORM OUTLET RB ● 28.2A
EMERG. OUTLET RB ● 28.3A
CAR FAN A F ● 14A
CAR FAN C F ● 14C
FAST SPEED
SLOW SPEED
B0508
B0510
CTMT AUX CIRC F-24A
PIR FAN A F ● 18A
B5228
CTMT PRESS ROOT STOP PT-8113/8115
CAR FAN VIB. RESET A-1
CTMT PRESS ROOT STOP PT-8114/8116
CAR FAN VIB. RESET B-1
H2 PURGE INBD ISOL AEB ● 100
H2 PURGE INBD ISOL AEB ● 91
PIR FAN B F ● 18B
CTMT AUX CIRC F-24B
B41803
CAR FAN B F ● 14B
CAR FAN D F ● 14D
B0605
B0603
NORM OUTLET RB ● 28.2D
EMERG. OUTLET RB ● 28.3D
HDR B ISOL EB ● 41
EBFS HDR B
LT GREEN
WOODLAND GREEN
EB PURGE EXH DMPR AC ● 8
CTMT PURGE OUTBD ISOL AC ● 7
CTMT PURGE EXH DMPR AC ● 57
PUSH TO RESET AC-3 AFTER FLTR OR CTR
RESET
FLTR
EB
20,000 32,000
H2 RECOMB HTR H ● 29A
B0513
CLR A INLET RB ● 28.1A
CLR C INLET RB ● 28.1C
RBCCW HDR A
DO NOT ISOLATE CAR CLR C REFS: DD MP2-016-00 DP 2303A
H2 PURGE OUTBD ISOL AEB ● 99
H2 PURGE OUTBD ISOL AEB ● 92
EBFS PLENUM
EBFS SUCT DMPR EB ● 50
PLENUM
FUEL AREA SUCT DMPR EB ● 64
FUEL AREA SUCT DMPR EB ● 61
FUEL AREA
EBFS SUCT DMPR EB ● 40
FROM CTMT CLEANUP EB ● 73
FROM CTMT PURGE EB ● 72
EBFS HDR A
EBFS HDR B
LT GREEN
CLR B INLET RB ● 28.1B
CLR D INLET RB ● 28.1D
RBCCW HDR B
CTMT CLEANUP TO EBFS EB ● 72
EB ● 73
ESF ROOM B F ● 15B
B6145
RM 8282A/B ISOL
INBD SPLY
OUTBD SPLY
RETURN
SAMPLE/DECAD EFF. CLR ISOL RB ● 210
SILVER FERN
SPECIAL GREEN DABLE
AC-3 WILL NOT OPERATE FROM CLOSED TO MID POSITION
SPLY FAN AC ● 1
EB PURGE SPLY DMPR AC ● 3
CTMT PURGE OUTBD ISOL AC ● 4
RM SOL
ESF ROOM A F ● 15A
B5139
SIAS BLOCK ●
ACTUATE CIAS ●
EBFS ●
AEAS ●
CSAS ●
SRAS ●
MSI ●
HS-3679
HS-3041
HS-3043
HS-8602
HS-5622A
HS-3037
HC-3039
HS-5030
HS-5453
HS-3680
HS-3042
HS-3044
HS-8603
HS-8623A
HS-3038
HS-3040
HS-6030
HS-5452
SILENCE
ACKNOWLEDGE
RESET
TEST
PB-S1
PB-A1
PB-R1
PB-T1
HINGED COVERS OVER ALL PB'S
CRP. C01 (RIGHT BENCH)
LEFT VERTICAL SHL1
RIGHT VERTICAL SHL2
LEFT BENCH SHL3
RIGHT BENCH SHL4
CRP C01
STM. GEN. & F.W. CONTROL (C05)
PLANT AUX. (C06)
TURBINE GENERATOR (C07)
STATION SERV. & ELECT. (C08)
ENTRY DOOR
R.C.S. (C03)
CVCS (C02)
SAFETY INJECTION (C04)
PLAN VIEW CONTROL PANEL
N.T.S.
N
MIMIC LEGEND
BLACK
BLACK/YELLOW
LT-BLUE
AQUA
BLACK/WHITE
NAMEPLATE LINE SPACING
SEE DWG
1/8" SPACE
LINE 1
1/8" SPACE
LINE 2
1/8" SPACE
LINE 3
1/8" SPACE
AS REQ'D
NOTES
1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. B -INDICATES BLACK PUSHBUTTON / RING
R -INDICATES RED PUSHBUTTON / RING
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE.
NAMEPLATE LEGEND
REFER TO SPECIFICATION SP-EE-261
▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES
THIS DWG. SUPERSEDES PART OF FOLLOWING DWGS.
25203-28002
25203-28097,SH.45
25203-28097,SH.50A TO 50R
CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.
NON Q.A.
REVISIONS DURING CONSTRUCTION
P.A. NO.
9 1-31-89 INCORP DCN DM2-00-0162-01 DWJ
Dominion Nuclear Connecticut Inc.
FOR Millstone Power Station
MILLSTONE UNIT 2
C01, CRP. FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION
WATERFORD, CT.
L.A.SWIFT
CHKD. MBR
APP. MCH
APP. LJC
DATE 7-23-87
DATE 11-24-87
DATE 11-24-87
DATE 11-24-87
SCALE N.T.S.
DWG. NO. 86-230
25203-30208 SH. 4 of 4





MIMIC LEGEND

- GREEN
- AQUA
- ORANGE
- YELLOW
- PINK
- BLACK/WHITE
- BLACK/YELLOW
- RED

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)

NAMEPLATE ENGRAVING

AS REQ'D

1/8"	SPACE
	LINE 1
1/8"	SPACE
	LINE 2
1/8"	SPACE
	LINE 3
1/8"	SPACE

AS REQ'D

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

THIS DWG SUPERCEDES 25203-28005, 29097 SH.36, 25203-29097 SHTS.41A,41B,41C, 41K,41L

REVISIONS DURING CONSTRUCTION

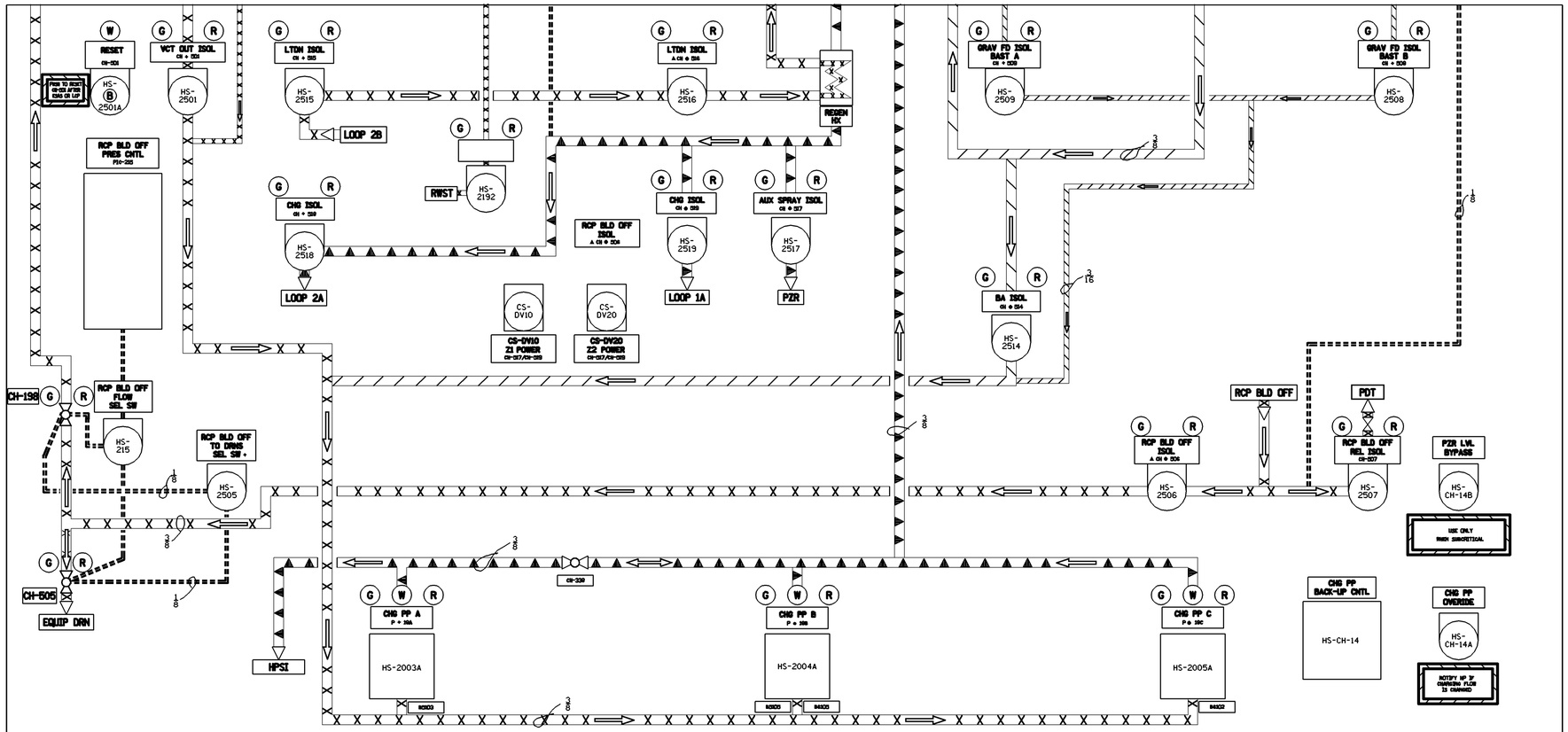
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
9	6-2-05	INCORP. DCN DM2-00-0355-04	JSJ	MEL	—	WBZ

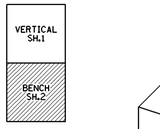
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

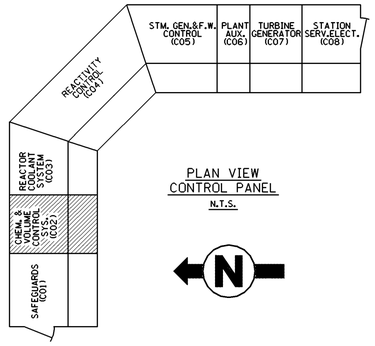
TITLE: MILLSTONE UNIT 2
CO2, CRP. FRONT VIEW ARRANGEMENT
CVCS
WATERFORD, CONNECTICUT

BY	CHKD.	APP.	APP.
T.A.SWIFT	MBR		LJC
DATE 3/25/88	DATE 5-12-88	DATE	DATE 5-19-88
SCALE NTS	DWG. NO. 86-230	25203-30209 SH.1 of 2	





CRP C02



PLAN VIEW
CONTROL PANEL
N.T.S.

NAMPLATES
ENGRAVINGS

AS REQ'D		
1/8"	SPACE	AS REQ
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

MIMIC LEGEND

GREEN

ORANGE

YELLOW

BLACK/WHITE

BLACK/YELLOW

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.

+ = INDICATES AN $\frac{1}{8}$" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS

● = INDICATES AN $\frac{1}{8}$" YELLOW DOT (Z2 POWERED EQUIPMENT)

✳ = INDICATES AN $\frac{1}{8}$" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

THIS DWG. SUPERCEDES 25203-28005,29097 SH.36 25203-29097 SHTS.41A,41B,41C, 41K,41L.

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

	NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
	5	6-17-05	INCORP. DCN DM2-00-0138-04, DM2-03-0138-04	DLW	WBZ	-	ASK
	4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
	3	7-21-99	INCORP. DCN's DM2-00-0532-98, DM2-00-1630-98	WBZ	JSJ	JPP	JPP
86-230	2	1-11-91	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
86-230	1	6/16 89	AS BUILT PER PDCR-2-27-88 & DSR-M2-S-492-89	WBZ	MEL	JTK	LJC

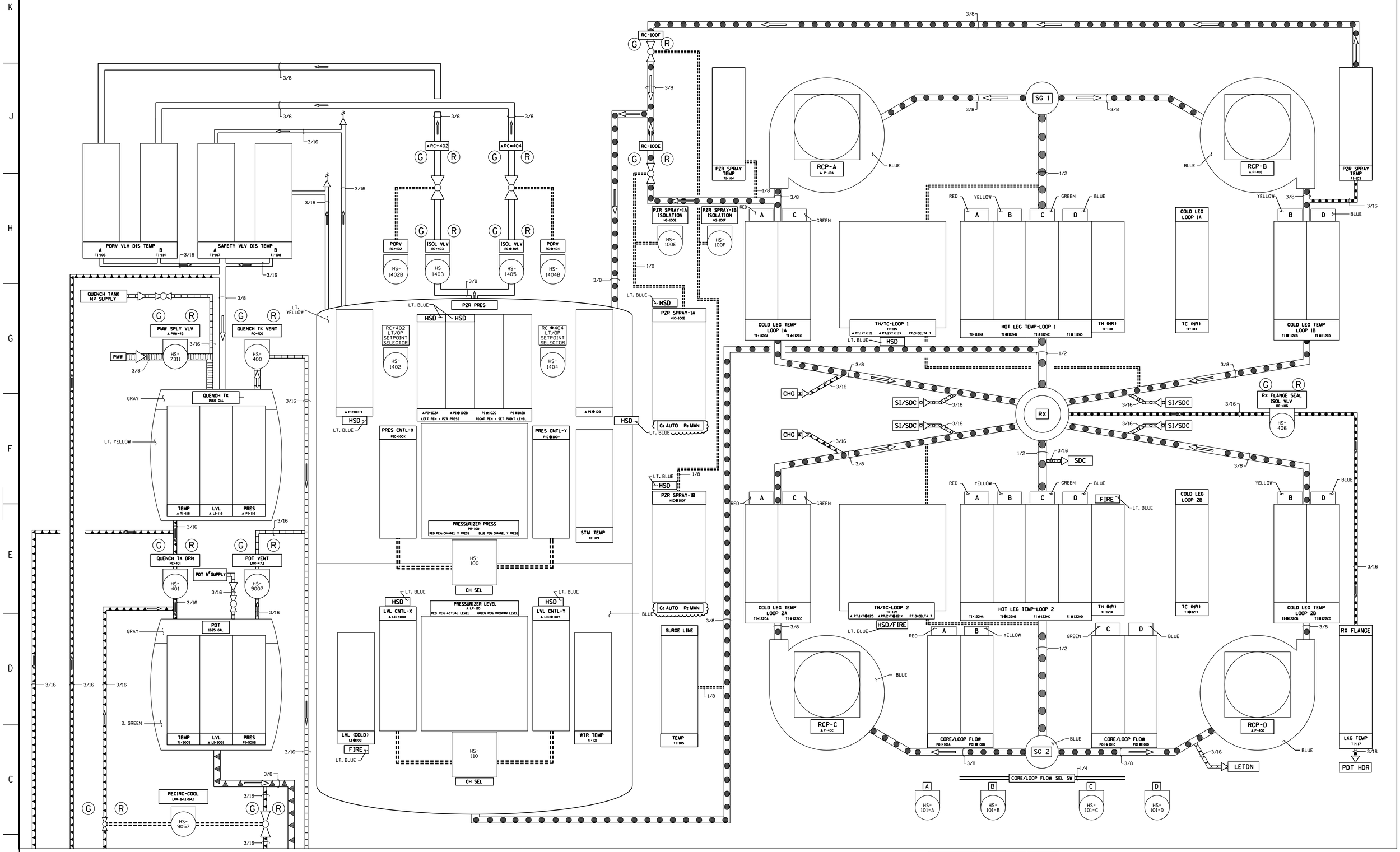
Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
C02, CRP. FRONT VIEW ARRANGEMENT
CVCS
WATERFORD, CONNECTICUT

BY	CHKD.	APP.	APP.
T.A.SWIFT	MBR		LJC
DATE 3/25/88	DATE 5-12-88	DATE	DATE 5-19-88

SCALE NTS
DWG. NO. 25203-30209,SH.2of2
86-230

CPS P679
6/02/89



MIMIC LEGEND

BLUE	
GREEN	
IVORY	
GRAY	
BLACK	
ORANGE	
RED	
BLACK/WHITE	
PINK	

NAMEPLATES ENGRAVING

AS REQ'D		
1/8"	SPACE	AS REQ'D
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

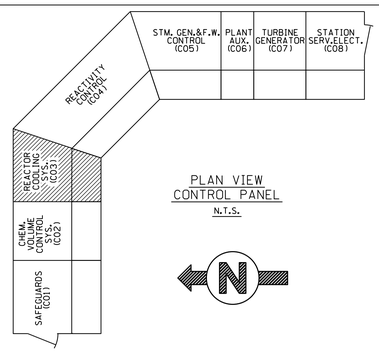
REFER TO SPECIFICATION SP-EE-261

- ▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
- + = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
- ● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
- ★ = INDICATES AN 1/8" GREEN DOT (VA30 SUPPLIED)
- ■ = INDICATES AN 1/8" BLUE DOT (VA40 SUPPLIED)
- ✳ = INDICATES AN 1/8" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

VERTICAL SH.1

BENCH SH.2

CRP C03



PLAN VIEW CONTROL PANEL

N.T.S.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE POWER STATION UNIT 2
C03, CRP. FRONT VIEW ARRANGEMENT R.C.S

REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER MSDUR 23-000033 & DUR 23-800106

CAD NO: C302101.DGN

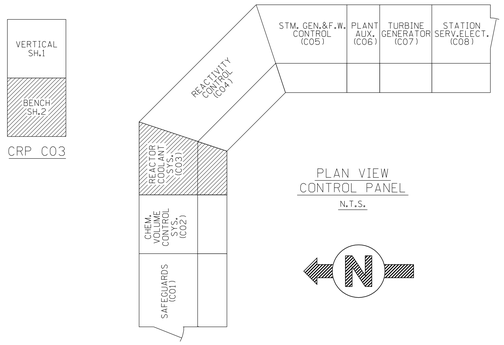
DRAWING NO: 25203-30210

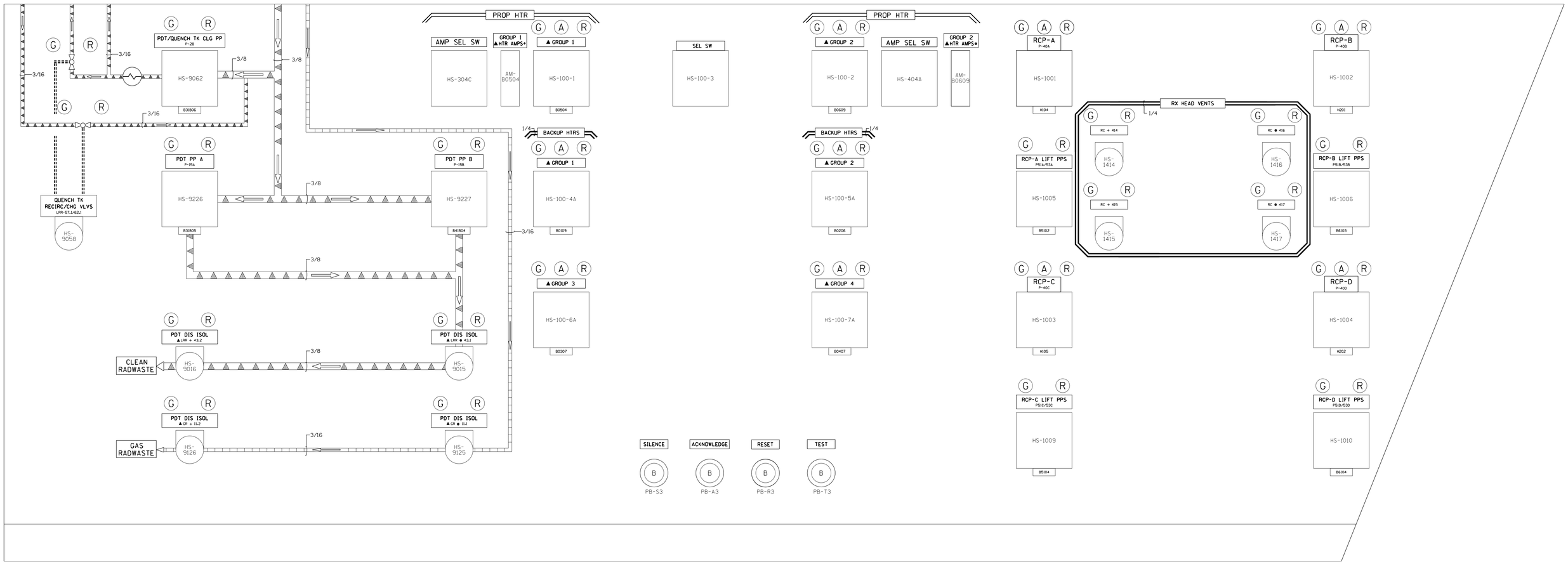
REV 13

DSGN WJP | SCALE: N/A | UNLESS OTHERWISE NOTED | SH 1

THIS DWG. SUPERCEDES 25203-28005, 29097 SH. 36 25203-29097 SHTS. 41C, 41D, 41E, 41F, 41K, 41L

Authenticated By: MBULOGIN w:i10765
30-APR-2023 07:21
C302101.DGN





MIMIC LEGEND

GREEN	▲▲▲▲▲▲▲▲▲
BLACK/WHITE	==========
GRAY	□□□□□□□□□□□□
BLACK	————

NAMEPLATE ENGRAVING

AS REQ'D

1/8"	SPACE	AS REQ'D
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
\+ = INDICATES AN $\frac{1}{8}$" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN $\frac{1}{8}$" YELLOW DOT (Z2 POWERED EQUIPMENT)

THIS DWG. SUPERCEDES
25203-28005,29097 SH.36
25203-29097 SHTS.41C,41D,41E,41F, 41K,41L

NON Q.A.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
6	4-27-10	INCORP. DCN DM2-00-0019-09	EAL	JCV	-	DLW
5	12-19-06	INCORP. DCN DM2-00-0237-05	DLW	MEL	-	ASK
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
3	7-21-99	INCORP. DCN DM2-00-0532-98	WBZ	JSJ	JPP	JPP
86-230	2	1-10-91 AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
86-230	1	6/16/89 AS BUILT PER PDCR-2-27-88 PDCR-2-13-88 & DSR-M2-S-492-89	WBZ	MEL	JTK	LJC

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE POWER STATION-UNIT 2
C03, CRP. FRONT VIEW ARRANGEMENT R.C.S
WATERFORD, CONNECTICUT

BY T.A.SWIFT	CHKD. MBR	APP.	APP. LJC
DATE 3/29/88	DATE 5-12-88	DATE	DATE 5-19-88
SCALE NTS	DWG. NO. 25203-30210 SH.2		
	86-230		

FOR UPPER PANEL SEE DET.1 30211 SH.2

(COMPUTER DISPLAY -ALARM- PC71)

MONITOR 1

MAG/JACK MG A
22E3 +

(CEA TOP/BOTTOM LIMIT LIGHT DISPLAY)

RED -UPPER ELECTRICAL UNIT
BLUE -SHUTDOWN EXERCISE LIMIT
WHITE -BETWEEN ELECTRICAL LIMITS
GREEN -LOWER ELECTRICAL LIMIT
AMBER -ROD DROP LIGHT

MEGAWATTS ELECTRIC

CONT. ON 30211 SH.2

CEAPDS MONITOR
(CEA-TS1)

MON 3
(PPC MONITOR-CEAPDS BACK-UP)

BA MAKEUP FLOW CONTROLLER
FC-210Y

BA FLOW-RED
MAKEUP WTR FLOW-GREEN
FR-210

PRI MAKEUP WTR FLOW CONTROLLER
FC-210X

RED YELLOW GREEN BLUE

A B C D

J1-1011 J1-1012 J1-1013 J1-1014 J1-1015 J1-1016 J1-1017 J1-1018 J1-1019 J1-1020 J1-1021 J1-1022

LOCAL POWER DENSITY
J1-1011 THRU J1-1022

AXIAL PWR RATIO
JR-012

COMPUTER TREND
UR-243

UTILITY TREND
UR-203

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

+ = INDICATES AN $\frac{1}{8}$" RED DOT (Z1 POWERED EQUIPMENT), EXCEPT FOR CAUTION LABELS

● = INDICATES AN $\frac{1}{8}$" YELLOW DOT (Z2 POWERED EQUIPMENT)

★ = INDICATES AN $\frac{1}{8}$" GREEN DOT (Z3 POWERED EQUIPMENT)

■ = INDICATES AN $\frac{1}{8}$" BLUE DOT (Z4 POWERED EQUIPMENT)

NAMEPLATE ENGRAVING

AS REQ

1/8"	SPACE
	LINE 1
1/8"	SPACE
	LINE 2
1/8"	SPACE
	LINE 3
1/8"	SPACE

AS REQ

COLOR CODE LABEL

GREEN- LOWER ELECTRICAL LIMIT
RED- UPPER ELECTRICAL LIMIT
WHITE- BETWEEN ELECTRICAL LIMITS
BLUE- BELOW EXERCIZE LIMITS
AMBER- DROPPED ROD.

THIS DWG SUPERCEDES 25203-28001, 25203-29097 SH.30A, 30B & 32

FSAR FIGURE

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
C04 CRP, FRONT VIEW ARRANGEMENT
REACTIVITY CONTROL SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C30211.DGN	
REVISED PER MSDUR 22-000020 AND DUR 22-800052	DRAWING NO: 25203-30211	REV 13
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

MSULOGIN\jome694
03-MAR-2022 10:35
C30211.DGN

FOR UPPER PANEL SEE DETAIL 1 THIS SH.

CONT. ON 30211 SH.1

MAG/JACK MG B
ZPF6 ●

(COMPUTER DISPLAY -TREND- PC73)

MONITOR 2

Tavg

RPS

RED A YELLOW B C GREEN D BLUE

POWER RANGE
J1+005 J1●006 J1●007 J1■008

LEFT PEN = PR RANGE RPS CHA-D RIGHT PEN = SET POINT LEVEL

START UP RATE
J41+001 J41●002 J41●003 J41■004

WIDE RANGE POWER
▲ J1 + 001 J1 ● 002 J1 ● 003 ▲ J1 ■ 004

LT BLUE HSD HSD/FIRE LT BLUE

WIDE RANGE POWER
JR-011

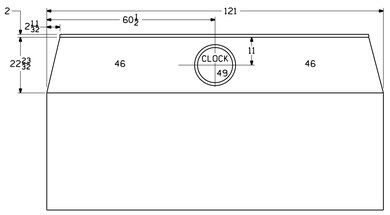
POWER RANGE
JR-009
RED PEN CHANNEL X GREEN PEN CHANNEL Y

COMPUTER-TREND
UR-7660

Tavg/Tref
TR-121

CONTROL CHANNEL
X Y

POWER RANGE
J1-009 J1-010



DETAIL 1

NOTES

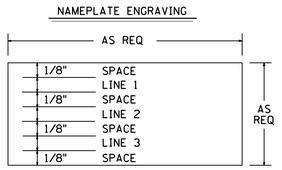
1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
★ = INDICATES AN 1/8" GREEN DOT (Z3 POWERED EQUIPMENT)
■ = INDICATES AN 1/8" BLUE DOT (Z4 POWERED EQUIPMENT)

NAMEPLATE ENGRAVING



THIS DWG SUPERCEDES 25203-28001, 25203-29097 SH.30A, 30B & 32

FSAR FIGURE

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
C04 CRP, FRONT VIEW ARRANGEMENT
REACTIVITY CONTROL SYSTEM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302112.DGN	
REVISED PER MSDUR 22-000020 AND DUR 22-800052	DRAWING NO: 25203-30211	REV 11
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

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03-MAR-2022 18:35
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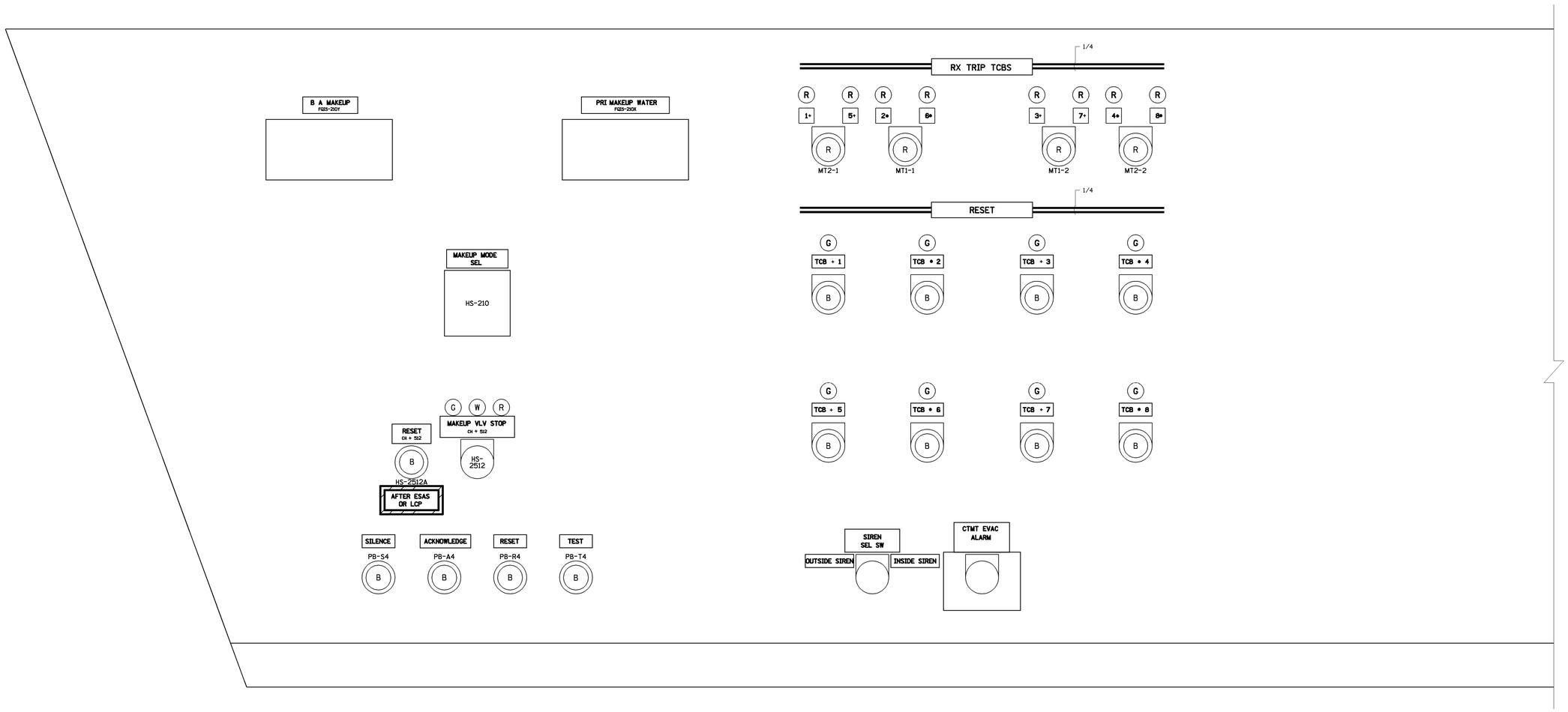


NAMEPLATE ENGRAVING

AS REQ		
	1/8"	SPACE
		LINE 1
	1/8"	SPACE
		LINE 2
	1/8"	SPACE
		LINE 3
AS REQ	1/8"	SPACE

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8'' RED DOT., EXCEPT FOR CAUTION LABELS
   - • -INDICATES AN 1/8'' YELLOW DOT
   - * - INDICATES AN 1/8'' BLACK DOT.

THIS DWG SUPERCEDES 25203-28001, 25203-29097 SH.30A, 308 & 32

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
PDR 0203 1927	4	4-6-02	INCORP. DCN DM2-00-0113-01	BLM	GHH	TP	CAM
X	3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
86-230	2	12-6-90	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
86-230	1	6/16 89	AS BUILT PER PDCR-2-27-88 & DCR-M2-S-492-89	WBZ	MEL	JTK	LJC

REVISIONS DURING CONSTRUCTION

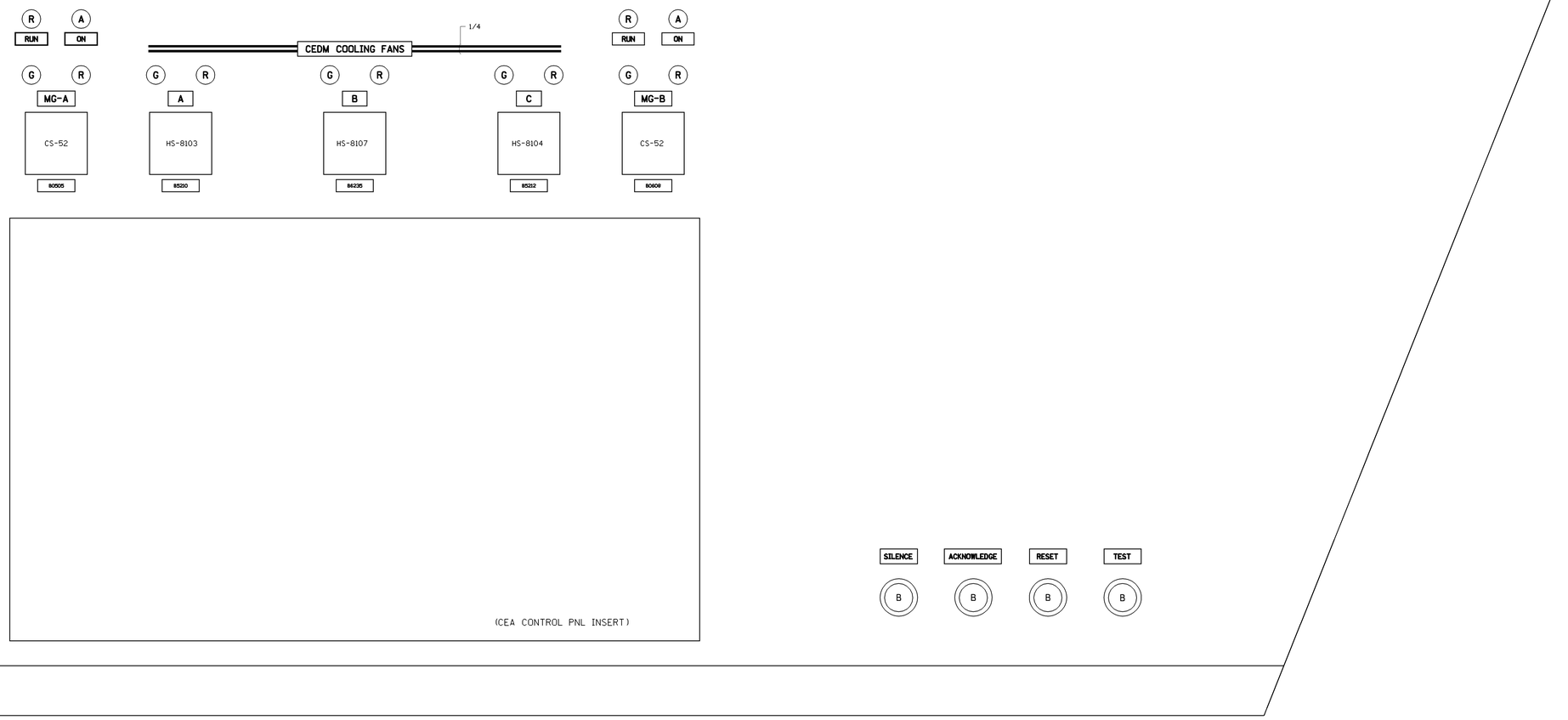
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

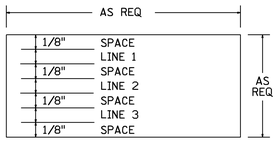
TITLE
MILLSTONE UNIT 2
C04 CRP, FRONT VIEW ARRANGEMENT
REACTIVITY CONTROL SYSTEM
WATERFORD, CONN.

BY	CHKD.	APP.	APP.
NAP	MBR		LJC
DATE 8-18-88	DATE 11-3-88	DATE	DATE 11-3-88

SCALE: NTS
DWG. NO.: 86-230
25203-30211 SH.3



NAMEPLATE ENGRAVING



NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT., EXCEPT FOR CAUTION LABELS
   - • -INDICATES AN 1/8" YELLOW DOT
   - * - INDICATES AN 1/8" BLACK DOT.

THIS DWG SUPERCEDES 25203-28001, 25203-29097 SH.30A, 308 & 32

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
5	12-15-06	INCORP. DCN DM2-00-0243-05	JSJ	MEL	—	DLW
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
3	11-9-96	INCORP. DCN DM2-S-0403-95 PER DCR M2-S-0105-96	JMM	MM	JPP	JPP
2	12-6-90	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
1	6/16 89	AS BUILT PER PDCR-2-27-88 & DSR-M2-S-492-89	WBZ	MEL	JTK	LJC

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

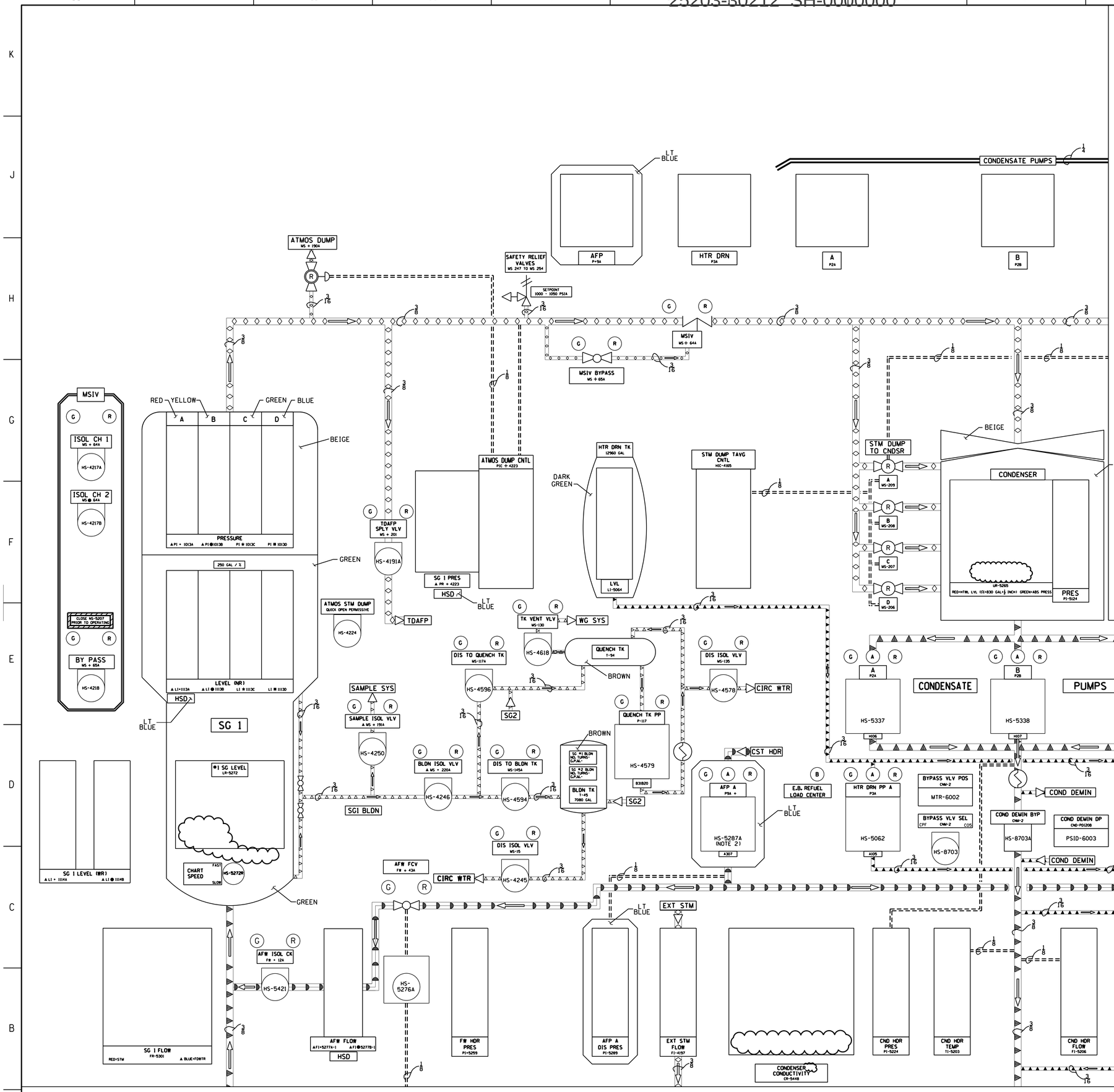
TITLE
MILLSTONE UNIT 2
C04 CRP, FRONT VIEW ARRANGEMENT
REACTIVITY CONTROL SYSTEM
WATERFORD, CONN.

BY	CHKD.	APP.
NAP	MBR	LJC
DATE 8-18-88	DATE 11-3-8	DATE 11-3-8

SCALE NTS

DWG. NO. 86-230

25203-30211 SH.4



NAMEPLATE ENGRAVING

AS REQ

1/8"	SPACE	AS REQ
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. RED HANDLE INSTALLED ON CONTROL SWITCH.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED /EQUIPMENT)
★ = INDICATES AN 1/8" GREEN DOT (Z3 POWERED EQUIPMENT)
■ = INDICATES AN 1/8" BLUE DOT (Z4 POWERED EQUIPMENT)
✳ = INDICATES AN 1/8" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

MIMIC LEGEND

LT BLUE
GREEN
BLACK/WHITE
BEIGE
BLACK
BROWN

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

THIS DWG SUPERCEDES 25203-28003, 25203-29097 SH. 21, 26A-E, 26J & 26K

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
C05 CRP, FRONT VIEW ARRANGEMENT
STEAM GENERATOR & FEEDWATER CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302121.DGN	
REVISED PER DC MP2-17-00171 & DUR 18-800261	DRAWING NO: 25203-30212	REV 16
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 1

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28-OCT-2018 13:17
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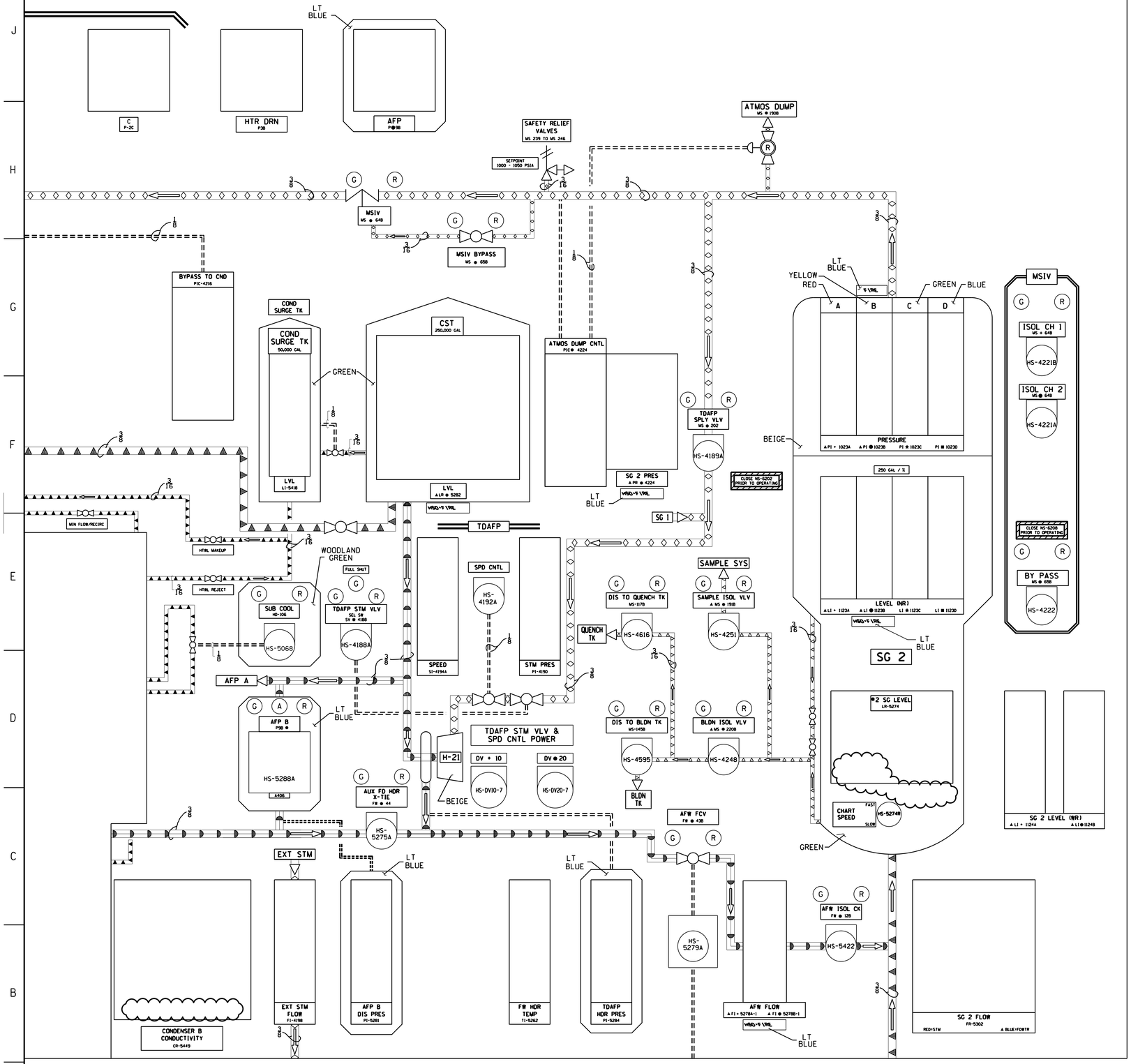


NAMEPLATE ENGRAVING

AS REQ

1/8"	SPACE	AS REQ
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. RED HANDLE INSTALLED ON CONTROL SWITCH.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

- ▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
- + = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
- ● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
- ★ = INDICATES AN 1/8" GREEN DOT (Z3 POWERED EQUIPMENT)
- ■ = INDICATES AN 1/8" BLUE DOT (Z4 POWERED EQUIPMENT)
- ✳ = INDICATES AN 1/8" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

MIMIC LEGEND

- LT BLUE
- GREEN
- BEIGE
- BLACK/WHITE
- BLACK
- BROWN

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

THIS DWG SUPERCEDES 25203-28003, 25203-29097 SH.21, 26A-E, 26J & 26K

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE

C05 CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302122.DGN	
REVISED PER DC MP2-17-00171 & OUR 18-800261	DRAWING NO: 25203-30212	REV 14
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

MBULOGIN\jame694
28-OCT-2018 13:19
C302122.DGN

NAMEPLATE ENGRAVING

AS REQ		
	1/8"	SPACE
		LINE 1
	1/8"	SPACE
		LINE 2
	1/8"	SPACE
		LINE 3
	1/8"	SPACE
AS REQ		

NOTES

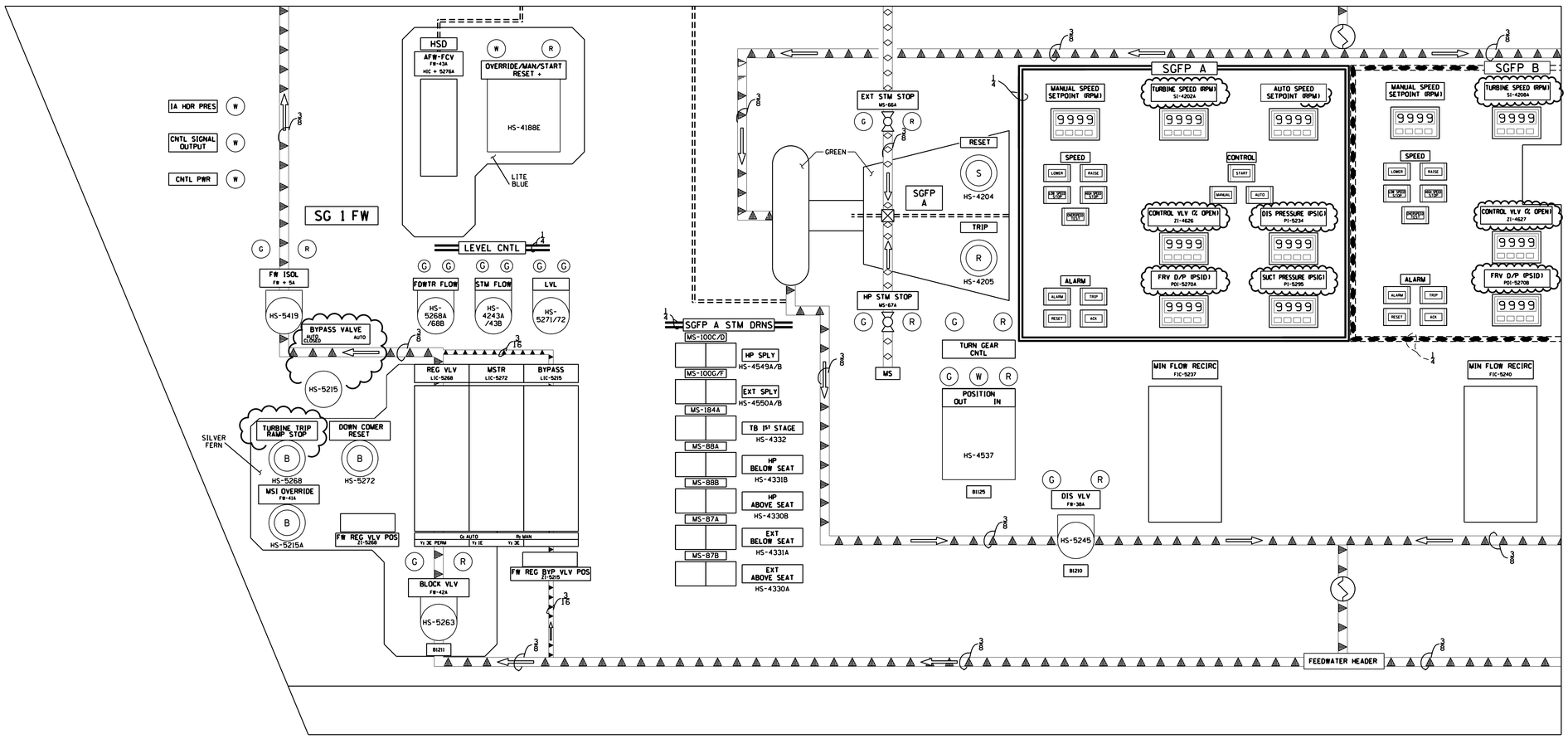
1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT), EXCEPT FOR CAUTION LABELS

● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)



MIMIC LEGEND

GREEN	
BEIGE	
BLACK/WHITE	
BLACK	
WHITE	

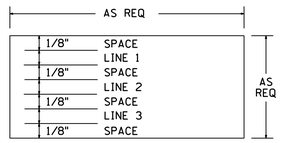
FSAR FIGURE

Dominion Energy®	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE UNIT 2 C05 CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302123.DGN
REVISED PER MSDUR 19-000029 AND DUR 19-800080	DRAWING NO: 25203-30212 REV 10
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED SH 3

THIS DWG SUPERCEDES 25203-28003, 25203-29097 SH.21, 26A-E, 26J & 26K

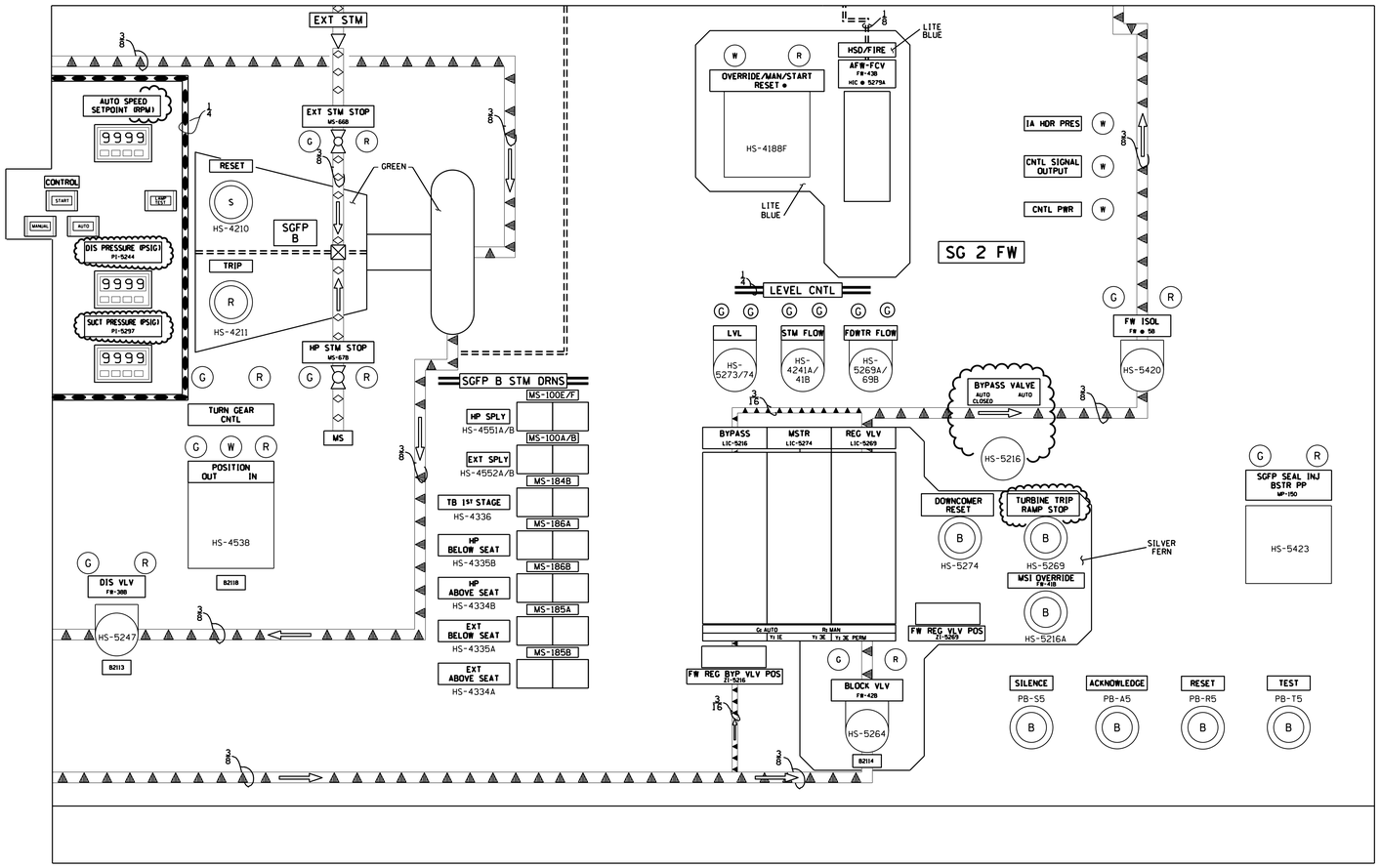
MBULOGIN\dlw+49
14-APR-2019 17:58
C302123.DGN

NAMEPLATE ENGRAVING



NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT., EXCEPT FOR CAUTION LABELS ▭
   - ● -INDICATES AN 1/8" YELLOW DOT
   - ✱ - INDICATES AN 1/8" BLACK DOT.



MIMIC LEGEND

GREEN
BEIGE
BLACK/WHITE
BLACK
WHITE

FSAR FIGURE

THIS DWG SUPERCEDES 25203-28003, 25203-29097 SH.21, 26A-E, 26J & 26K

Dominion Energy®	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE UNIT 2 C05 CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302124.DGN
REVISED PER MSDUR 19-000029 AND DUR 19-800080	DRAWING NO: 25203-30212 REV 9
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED SH 4

MBULOGIN\diane48
14-APR-2019 17:54
C302124.DGN

ANNUNICIATOR

54 1/4"

CIRC WATER PUMPS

READY TO RUN (W) | VFD MODE (B) | BYP MODE (A)

P-6A	P-6B	P-6C	P-6D
A2/24A4-2	A2/24B1-2	A2/24A5-2	A2/24B2-2
A106	A201	A107	A202

BACKWASH PERMISSIVE

(B) P-6A | (B) P-6B | (B) P-6C | (B) P-6D

TBCCW PUMPS

P-7A	P-7B	P-7C
A2/22A5-2	A2/22B4-2	A2/22D3-2
B0107	B0204	B0403

RBCCW PUMPS

DK GN

P-11A	P-11B	P-11C
A2/24C7-2	A2/24E3-2	A2/24D1-2
A311 +	A504 ●	A402 ●

SERVICE WATER PUMPS

AQUA

P-5A	P-5B	P-5C
A2/24C2-2	A2/24E1-2	A2/24D6-2
A306 +	A502 ●	A407 ●

PLAN VIEW
CONTROL PANEL
N.T.S.

STM. GEN.&F.W. CONTROL (C05) | PLANT AUX. (C06) | TURBINE GENERATOR (C07) | STATION SERV.ELECT. (C08)
REACTIVITY CONTROL (C04)
R.C.S. (C03)
CVCS (C02)
SAFEGUARDS (C01)

VERTICAL SH.1
BENCH SH.2
CRP C06

SPECIAL GREEN SHALE

DIESEL OIL STOR TANK 25,000 GAL — LVL LI-7004A

RCP TB/OC TEMP — 1A TI-6098 | 1B TI-6099 | 2A TI-6101 | 2B TI-6103

RBCCW — PP DIS PRESS — HX OUT TEMP
PP-11A PI-6008 | PP-11B PI-6009 | PP-11C PI-6010 | HX-A ▲TI-6031 | HX-B ▲TI-6032 | HX-C ▲TI-6033

SERV WTR HDR FLOW — A FI-6471 | B FI-6472

RBCCW — HDR DIS FLOW — SURGE TK HDR LVL (4,480 GAL)
A ▲FI-6035 | B ▲FI-6034 | A LI-6001 | B LI-6730

TRAV SCREENS Δ LVL — 2A LI-6499A | 2B LI-6499B | 2C LI-6499C | 2D LI-6499D

CTMT SUMP LVL 3.35 GAL / ½ — ▲LI-9155

STACK RADIATION/FLOW RR-8132 — PT.1-R-8132A ▲PT.2-R-8132B ▲PT.3-F-8412
E-PLAN-CATEGORY A

SDC HX A RBCCW OUT — TEMP TI-6051 | FLOW FI-6043

ESG ROOM HX A OUT — TEMP TI-6734 | FLOW FI-6732

AIR PRESSURE — SA PI-7099 | IA PI-7079

RBCCW HX SERV WTR FLOW — A FI-6433 | B FI-6434 | C FI-6435

AUX STM — PRES PI-8367

AQUA

RBCCW HX SERV WTR OUTLET VLVS
SW-8.1A RESET + SW-8.1B — HS-6307A/6308
SW-8.1B RESET ● SW-8.1C — HS-6306/6307B
PUSH TO RESET AFTER ESAS OR LCP

TBCCW HX OUT — PRES PI-6273 | TEMP TI-6274

TBCCW HX SERV WTR FLOW — A FI-6467 | B FI-6468 | C FI-6469

SDC HX B RBCCW OUT — TEMP TI-6056 | FLOW FI-6042

ESG ROOM HX B OUT — TEMP TI-6737 | FLOW FI-6736

DK GN — DKGN

RBCCW OUTLET VALVES
SDC HX A RB + 13.1-A (HS 6050) | SFPC HX A RB + 81-A (HS 6315) | ESG RM CLR A RB + 68.1-A (HS 6731)

CTMT ISOL IA IA + 27.1 (HS 7083) | CTMT SA/IA X-TIE SA-23.1 (HS 7808)

RBCCW HX SERV WTR INLET VLVS
SW + 3.1A (HS 6399) | SW ● 3.1B (HS 6400)

RBCCW SURGE TK MAKEUP RB-215 (HS 6000)

TBCCW HX SERV WTR INLET VALVES
SW + 3.2B (HS 6439) | SW ● 3.2A (HS 6438)

RBCCW OUTLET VALVES
SDC HX B RB ● 13.1B (HS 6055) | SFPC HX B RB ● 8.1B (HS 6316) | ESG RM CLR B RB ● 68.1B (HS 6735)

TBCCW PUMPS

P-7A	P-7B	P-7C
HS-6282A	HS-6282B	HS-6282C
B0107	B0204	B0403

FUEL HANDLE AREA SPLY FAN DIS DMPR HV ● 165 (HS 8141B)

RBCCW CTMT ISOL HDR A — SPLY RB + 30.1A (HS 6096) | RTN RB + 37.2A (HS 6100) — DK GN

CTMT SUMP ISOL OUTBOARD ▲SSP-16.2 (HS 9150)

DG A HX SERV WTR — SPLY SW + 89A (HS 6389) | BYP SW + 23/A (HS 6341) — AQUA

DG B HX SERV WTR — SPLY SW ● 89B (HS 6397) | BYP SW ● 23.1B (HS 6342) — AQUA

CTMT SUMP ISOL INBOARD ▲SSP ● 16.1 (HS 9151)

RBCCW CTMT ISOL HDR B — SPLY RB ● 30.1B (HS 6095) | RTN RB ● 37.2B (HS 6106) — DK GN

OUTSIDE AIR MAKEUP DMPR AC-59 (HS 8715) — 12,000/20,000 CFM

RADWASTE ISOL VALVES
EB-SOUTH HV-113 (HS 8967) | EB-EAST HV-144 (HS 8969)

VENTILATION

ESG ROOM INBD ISOL DMPR
SPLY HV + 107 (HS 8247) | EXH HV + 116 (HS 8249)

20,000 CFM

FUEL HANDLE AREA EXH MOD DIS DMPR HV-173 (HS 8144) | EXH ISOL DMPR HV + 171 (HS 8275)

L-27

SA HDR ISOL SA-11.1

SA/IA X-TIE SA-1.01 (HS 7064)

CTMT SUMP PP A P-33A — HS-9154 — B31B04

STACK MONITOR EXHAUST (RM 8132) MUST BE TURNED OFF WHEN NO MAIN EXHAUST FANS ARE RUNNING

MAIN EXH FANS

F-34A	F-34B	F-34C
HS-8129	HS-8130	HS-8131
B3208	B41B14	B41B07

CTMT SUMP PP B P-33B — HS-9157 — B41B08

RADWASTE ISOL VALVES
EB-SOUTH HV-109 (HS 8966) | EB-EAST HV-143 (HS 8968)

VENTILATION

44,000 CFM

FUEL HANDLE AREA EXH ISOL DMPR HV ● 170 (HS 8326)

RADWASTE EXH DIS DMPR HV-118 (HS 8138)

L-26

ESG ROOM OUTBD ISOL DMPR
SPLY HV ● 106 (HS 8133) | EXH HV ● 117 (HS 8248)

69 1/8"

CRP C06 (VERTICAL)

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN 1/8" RED DOT (Z1 POWERED EQUIPMENT) EXCEPT FOR CAUTION LABELS
● = INDICATES AN 1/8" YELLOW DOT (Z2 POWERED EQUIPMENT)
✳ = INDICATES AN 1/8" BLACK DOT (Z5 POWERED SWING EQUIPMENT)

NAMEPLATE LINE SPACING

SEE DWG	
1/8"	SPACE
	LINE 1
1/8"	SPACE
	LINE 2
1/8"	SPACE
	LINE 3
1/8"	SPACE

AS REQ

TYPICAL ALL DWGS

MIMIC LEGEND

BLACK
BLACK/YELLOW

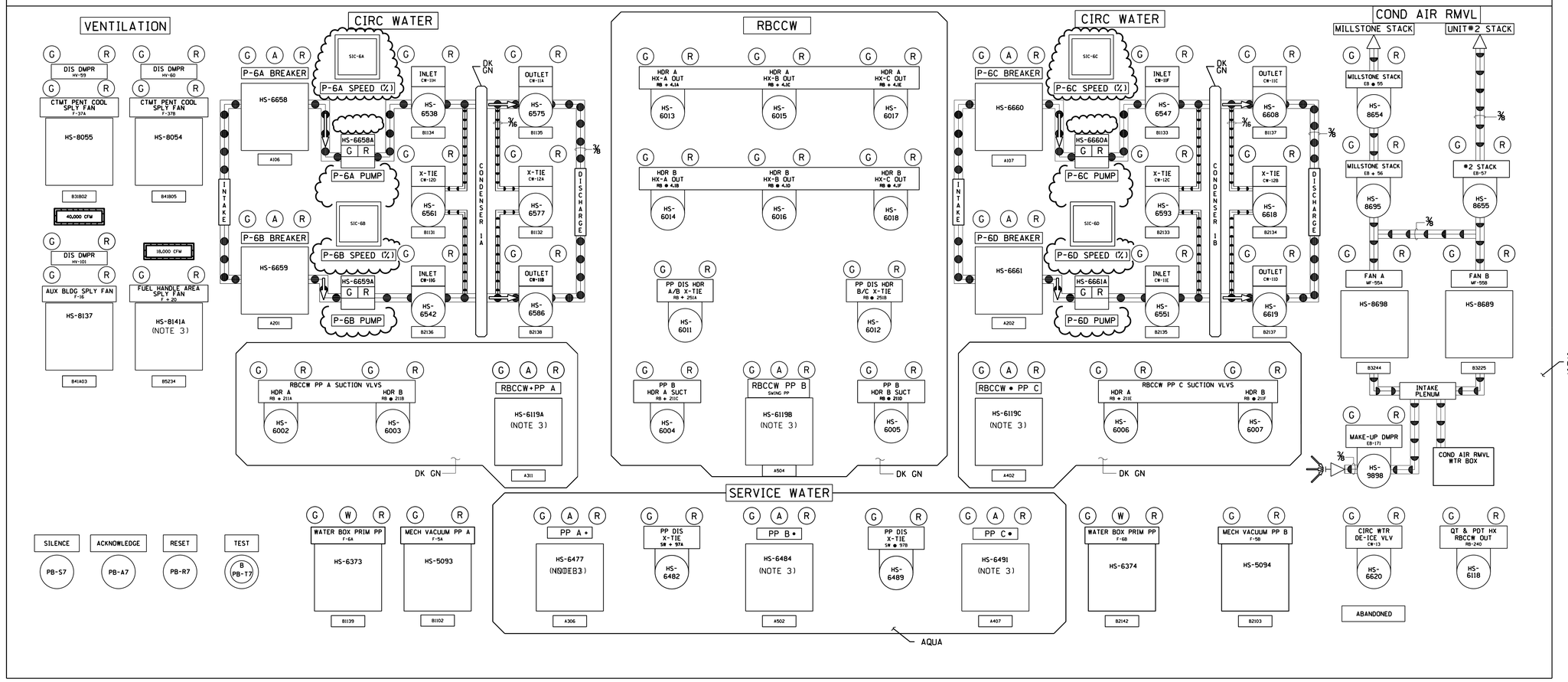
FSAR FIGURE

Dominion Energy® — Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
C06, CRP. FRONT VIEW ARRANGEMENT
PLANT AUXILIARY

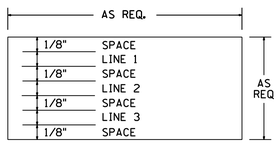
REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER MSDUR 19-000029 & DUR 19-800080
CAD NO: C30213L.DGN
DRAWING NO: 25203-30213
REV: 9
DSGN: DLW | SCALE: NONE | UNLESS OTHERWISE NOTED | SH: 1

MBLOGIN\dlw 14-APR-2019 18:34 C30213L.DGN



CRP. C06 (BENCH)

NAMEPLATE ENGRAVING



MIMIC LEGEND

BLUE
BLACK
BLACK/YELLOW
LT BLUE

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT., EXCEPT FOR CAUTION LABELS
   - ● -INDICATES AN 1/8" YELLOW DOT
   - ✱ - INDICATES AN 1/8" BLACK DOT.
3. RED HANDLES INSTALLED ON THE CONTROL SWITCHES.

FSAR FIGURE

THIS DWG. SUPERSEDES THE FOLLOWING DWGS.
25203-28007
25203-29097,SH.8
25203-29097,SH.14D, 14E, 14F, 14H, 14K 14R, 14S, 14T

Dominion Energy	Dominion Nuclear Connecticut, Inc. Millstone Power Station, Waterford, CT.
TITLE	MILLSTONE POWER STATION UNIT No. 2 C06, CRP. FRONT VIEW ARRANGEMENT PLANT AUXILIARY
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302132.DGN
REVISED PER MSDUR 19-000029 AND DUR 19-800080	DRAWING NO: 25203-30213 REV 11
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED SH 2

MBULOGIN\diane48
14-APR-2019 18:37
C302132.DGN

ANNUNCIATOR

45 1/4"

54"

ELECTROHYDRAULIC CONTROL (EHC) SYSTEM CRM1 MONITOR

32" TOUCH SCREEN DISPLAY

SPECIAL GREEN SHALE

MASTER TRIP / EMERGENCY TRIP

MASTER TRIP / EMERGENCY TRIP
PUSH BOTH BUTTONS AT THE SAME TIME TO TRIP TURBINE

LOAD/SPEED CONTROL

LOWER

RAISE

ELECTROHYDRAULIC CONTROL PANEL (EHC)

TURB BEARING OIL PRESSURE

NORMAL PI-7210

EMERG. PI-7193

SPENT FUEL POOL TEMP
RED: POOL TEMP GRN: INLET TEMP
TR-7030

STEAM SEAL

PACK EXH PRES PI-4340

PRES PI-4345

TEMP TI-4344

GEN HYDROGEN

PRES PI-7836

PURITY QI-7835

2010

AM DB3

VM DB3

AM DO3

BATT AMPS VOLTS

BUS AMPS

CRP. C07 (LEFT VERTICAL)

LEFT VERTICAL SH.1

RIGHT VERTICAL SH.2

LEFT BENCH SH.3

RIGHT BENCH SH.4

CRP C07

REACTIVITY CONTROL (C04)

STM. GEN.&F.W. CONTROL (C05)

PLANT AUX. (C06)

TURBINE GENERATOR (C07)

STATION SERV.ELECT. (C08)

R.C.S. (C03)

CVCS (C02)

SAFEGUARDS (C01)

PLAN VIEW
CONTROL PANEL
N.T.S.

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8'' RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS
   • -INDICATES AN 1/8'' YELLOW DOT (TRAIN B)
   * - INDICATES AN 1/8'' BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

NAMEPLATE LINE SPACING

SEE DWG

1/8"	SPACE	
	LINE 1	
1/8"	SPACE	
	LINE 2	AS REQ'D
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

TYPICAL ALL DWGS

MIMIC LEGEND

BLACK

REFERENCE DWGS

25203-29224 SHEETS 5A AND 5B EHC PANEL INSERT

THIS DWG. SUPERCEDES THE FOLLOWING DWGS.
25203-29097,SH.14F,14G,14H,14J,14K,14T
25203-28007
25203-29097 SH.8

FSAR FIGURE

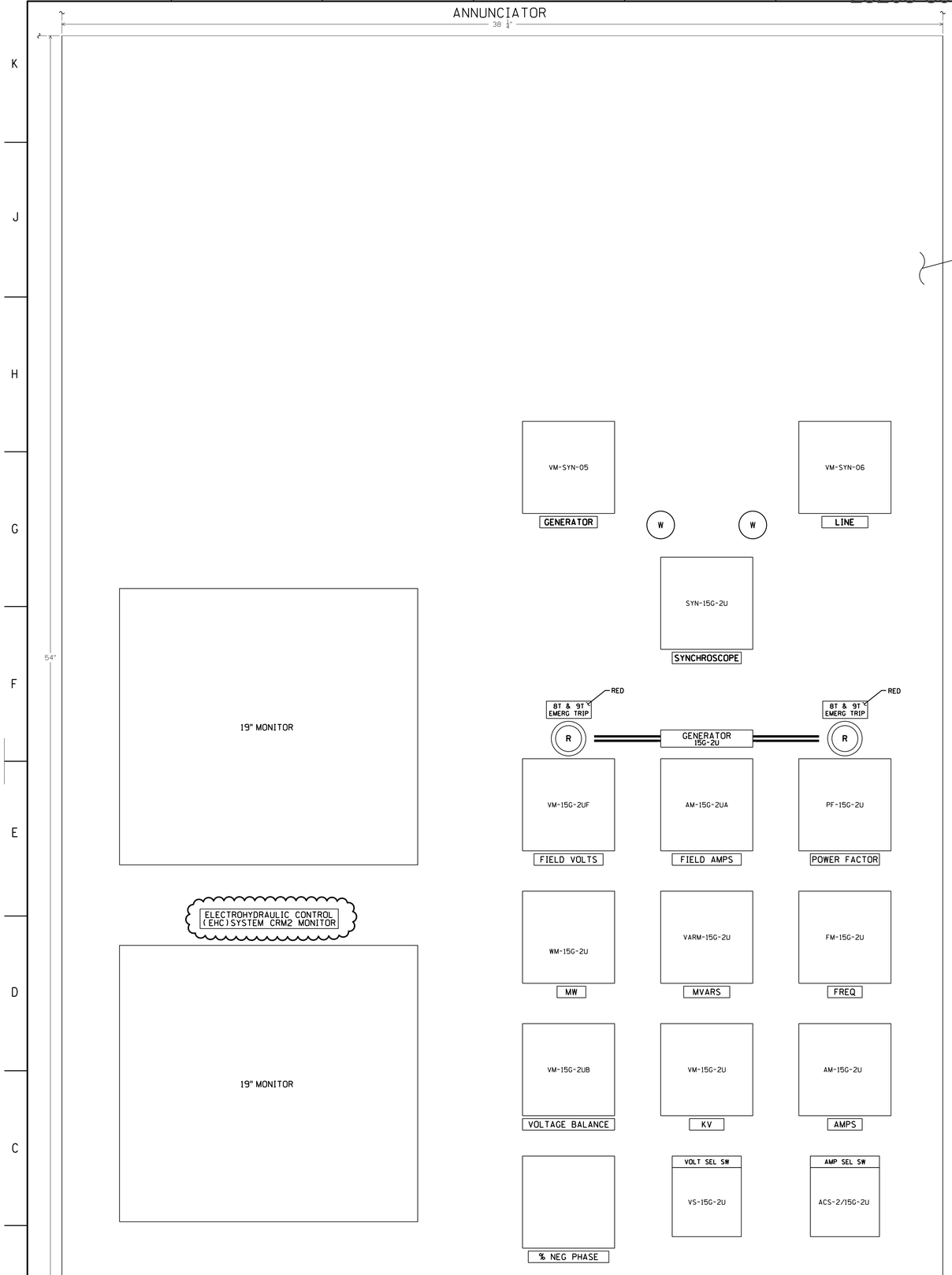
Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

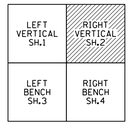
TITLE
MILLSTONE UNIT 2
C07, CRP. FRONT VIEW ARRANGEMENT TURBINE - GENERATOR

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C30214.DGN	
REVISED PER MSDUR 19-000029 AND DUR 19-800080	DRAWING NO: 25203-30214	REV 8
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

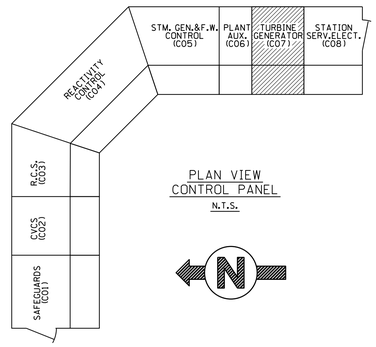
MBULOGIN\diane48
14-APR-2019 19:08
C30241.DGN



CRP. C07 (RIGHT VERTICAL)



CRP C07

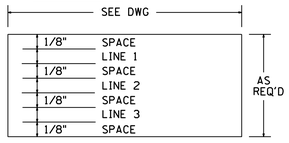


PLAN VIEW
CONTROL PANEL
N.T.S.

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS
   • -INDICATES AN 1/8" YELLOW DOT (TRAIN B)
   * - INDICATES AN 1/8" BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE.

NAMEPLATE LINE SPACING



TYPICAL ALL DWGS

MIMIC LEGEND

BLACK

BLACK/YELLOW

FSAR FIGURE

THIS DWG. SUPERCEDES THE FOLLOWING DWGS.
25203-29097,SH.14F,14G,14H,14J,14K,14T
25203-28007
25203-29097 SH.8

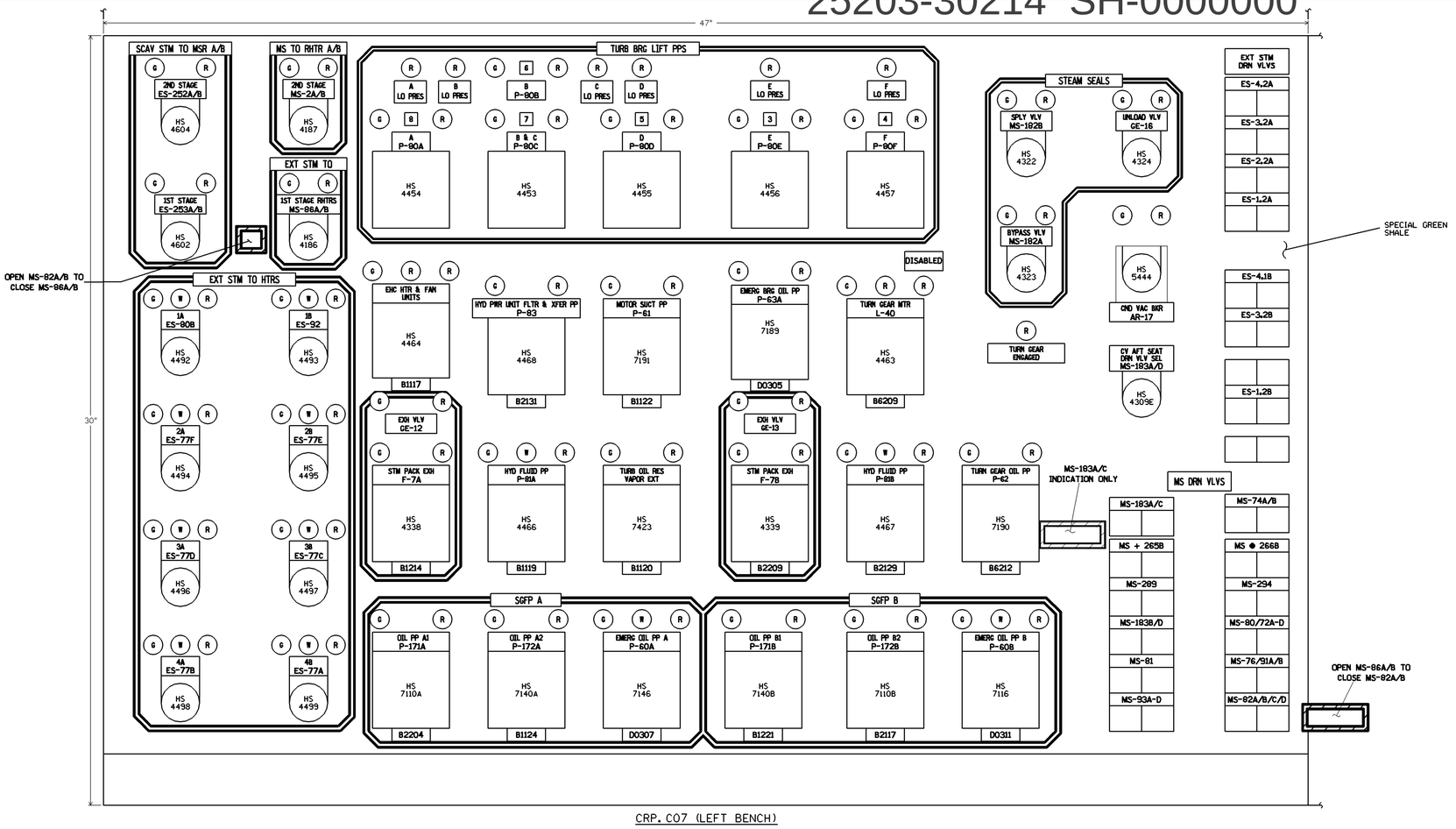
Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
C07, CRP. FRONT VIEW ARRANGEMENT TURBINE - GENERATOR

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302142.DGN	
REVISED PER MSDUR 19-000029 & DUR 19-800080	DRAWING NO: 25203-30214	REV 14
DSGN DLW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

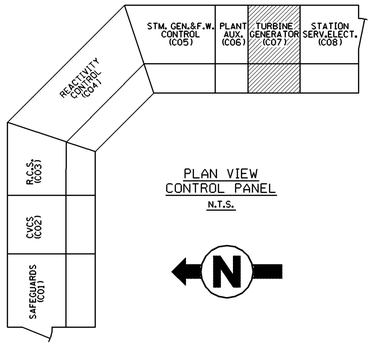
MSULOGIN\diane48
14-APR-2019 19:17
C302142.DGN



CRP. C07 (LEFT BENCH)

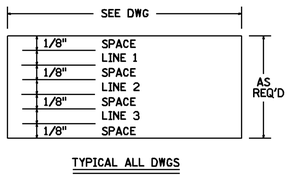


CRP C07



PLAN VIEW
CONTROL PANEL
N.T.S.

NAMEPLATE
LINE SPACING



TYPICAL ALL DWGS

MIMIC LEGEND

BLACK

BLACK/YELLOW

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS
   • - INDICATES AN 1/8" YELLOW DOT (TRAIN B)
   * - INDICATES AN 1/8" BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

THIS DWG. SUPERSEDES THE FOLLOWING DWGS.
25203-29097,SH.14F,14G,14H,14J,14K,14T
25203-28007
25203-29097 SH.8

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

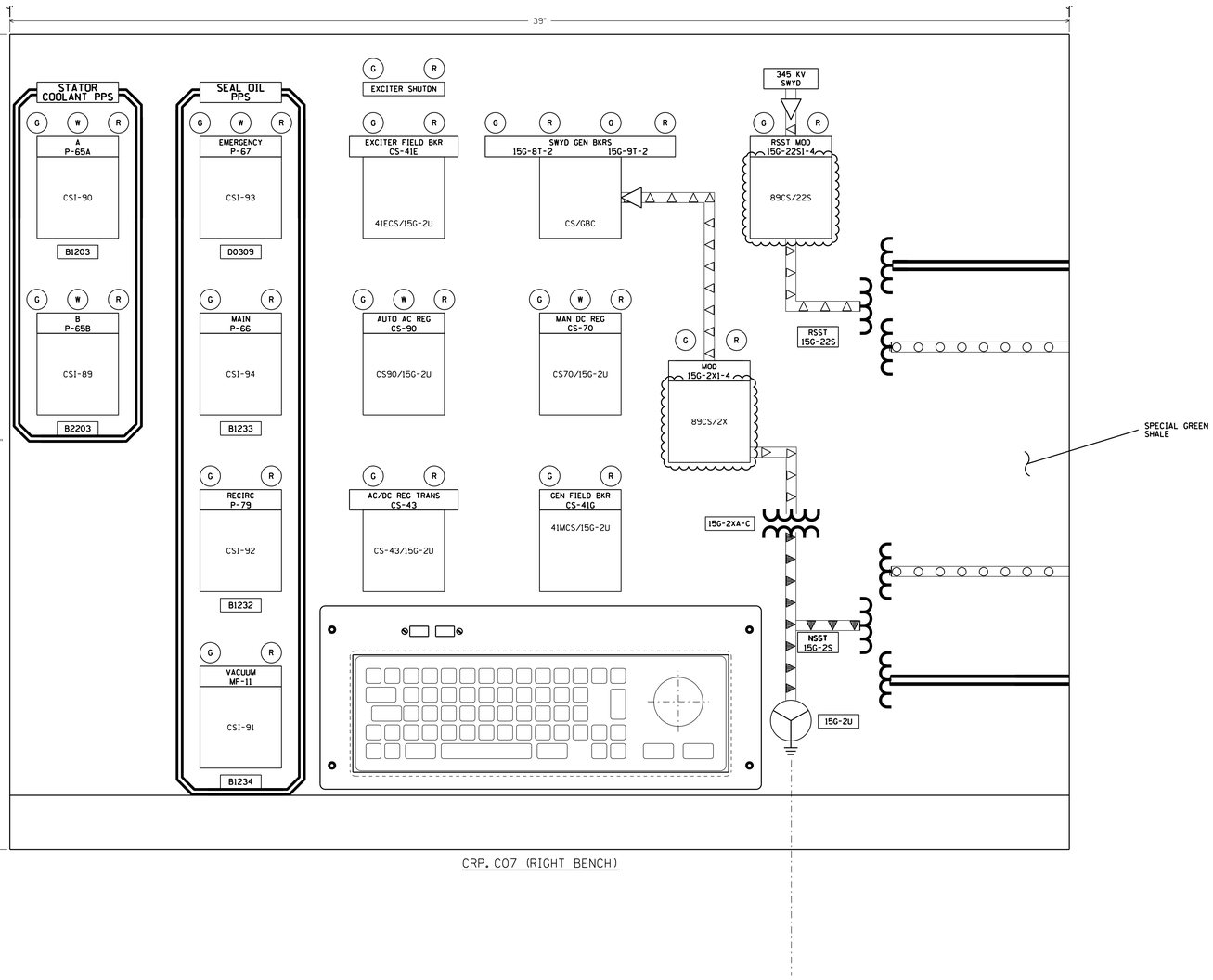
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
5	6-10-05	INCORP. DCN DM2-00-0217-05	DLW	WBZ	-	ASK
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
N/A	10-14-97	INCORP. DCN DM2-S-0216-95	DSJ	GFD	RWD	RW
86-230	5/10/91	AS BUILT PER DCR-M2-S-1218-89	WBZ	MEL PJK	JTK	LJC
86-230	7/27/88	AS BUILT PER DSR-M2-S-302-88	TAS	MEL		LJC

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

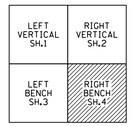
TITLE
MILLSTONE UNIT 2
C07, CRP. FRONT VIEW ARRANGEMENT
TURBINE - GENERATOR
WATERFORD, CT.

BY	CHKD.	APP.	APP.
T.A.SWIFT	AHS		LJC
DATE 7/9/87	DATE 8-19-87	DATE	DATE 8-21-87

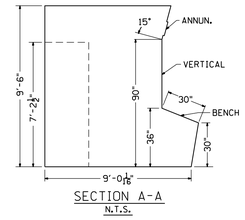
SCALE N.T.S.
DWG. NO. 25203-30214 SH. 3 of 4
86-230



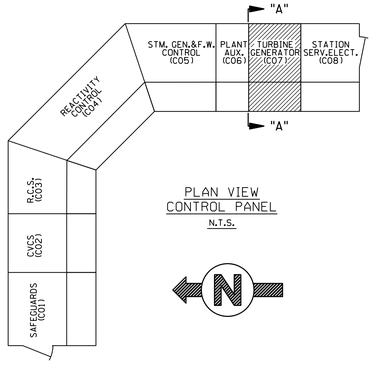
CRP. CO7 (RIGHT BENCH)



CRP CO7

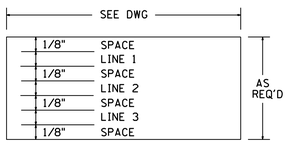


SECTION A-A
N.T.S.



PLAN VIEW
CONTROL PANEL
N.T.S.

NAMEPLATE
LINE SPACING



TYPICAL ALL DWGS

MIMIC LEGEND

BLACK
GREEN
BROWN
RED

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS
   • -INDICATES AN 1/8" YELLOW DOT (TRAIN B)
   * - INDICATES AN 1/8" BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

THIS DWG. SUPERCEDES THE FOLLOWING DWGS.
25203-29097,SH.14F,14G,14H,14J,14K,14T
25203-28007
25203-29097 SH.8

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

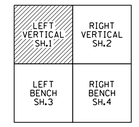
TITLE
MILLSTONE UNIT 2
CO7, CRP. FRONT VIEW ARRANGEMENT TURBINE - GENERATOR
WATERFORD, CONNECTICUT

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302144.DGN	
REVISED PER DC MP2-21-01048 & DUR 23-80141	DRAWING NO: 25203-30214	REV 7
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 4

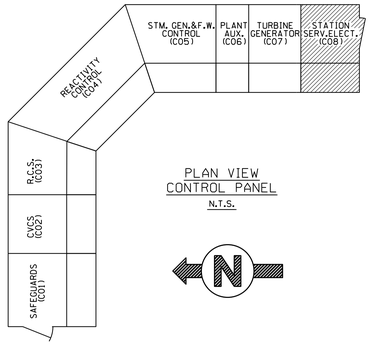
Authenticated By: MBULOGIN\jame694
12-JUN-2023 09:05
C302144.DGN

ANNUNCIATOR
61 ¾"

54"



CRP C08



PLAN VIEW
CONTROL PANEL
N.T.S.

6.9 KV

NSST

WM 15G-252
MW
A-2 2S2-25A-2
25A SPLY AMPS 2S2-25A
A-2 2S2-25B-2
25B SPLY AMPS 2S2-25B
V-2 15G-252
NSST VOLTS
25A SPLY AMP SEL SW
ACS-2 2S2-25A-2
25B SPLY AMP SEL SW
ACS-2 2S2-25B-2
NSST VOLT SEL SW
VCS-2 15G-252

RSST

WM 15G-22S2
MW
A-2 22S2-25A-2
25A SPLY AMPS 22S2-25A
A-2 22S2-25B-2
25B SPLY AMPS 22S2-25B
V-2 15G-22S2
RSST VOLTS
25A SPLY AMP SEL SW
ACS-2 22S2-25A-2
25B SPLY AMP SEL SW
ACS-2 22S2-25B-2
RSST VOLT SEL SW
VCS-2 15G-22S2

SYNCHRONIZING
25VI-1
INCOMING
25VR-1
RUNNING

W W

25SS-1
SYNCHROSCOPE

6.9 KV BUS VOLTS
VM 25A-1
VM 25B-1
25A
25B

201A
AM-DB1 | VM-DB1 | AM-D01
▲ BATT + AMPS | VOLTS | ▲ BUS + AMPS

4.16 KV

NSST

WM 15G-2S3
MW
A-2 2S3-24A-2
24A SPLY AMPS 2S3-24A
A-2 2S3-24B-2
24B SPLY AMPS 2S3-24B
V-2 15G-2S3
NSST VOLTS
24A SPLY AMP SEL SW
ACS-2 2S3-24A-2
24B SPLY AMP SEL SW
ACS-2 2S3-24B-2
NSST VOLT SEL SW
VCS-2 15G-2S3

RSST

WM 15G-22S3
MW
A-2 24C-1T-2
24A/24C AMPS
A-2 24D-1T-2
24B/24D AMPS
V-2 24C
▲24C BUS VOLTS+
RED
24A/24C AMP SEL SW
ACS-2 24C-1T-2
24B/24D AMP SEL SW
ACS-2 24D-1T-2
24C BUS VOLT SEL SW +
VCS-2 24C

A-2 22S3-24C-2
24C SPLY AMPS 22S3-24C+
A-2 22S3-24D-2
24D SPLY AMPS 22S3-24D ●
V-2 15G-22S3
RSST VOLTS
BUS 24C SPLY AMP SEL SW
ACS-2 22S3-24C-2
BUS 24D SPLY AMP SEL SW
ACS-2 22S3-24D-2
RSST VOLT SEL SW
VCS-2 15G-22S3

A-2 24C-2T-2
24C/24E AMPS
A-2 24D-2T-2
24D/24E AMPS
V-2 24D
▲24D BUS VOLTS●
YELLOW
24C/24E AMP SEL SW
ACS-2 24C-2T-2
24D/24E AMP SEL SW
ACS-2 24D-2T-2
24D BUS VOLT SEL SW ●
VCS-2 24D

4.16 KV BUS VOLTS
VM 24A-1
VM 24B-1
24A
24B

201B
AM-DB2 | VM-DB2 | AM-D02
▲ BATT ● AMPS | VOLTS | ▲ BUS ● AMPS

CRP - C08 (LEFT VERTICAL)

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
+ = INDICATES AN ⅛" RED DOT (Z1 POWERED EQUIPMENT), EXCEPT FOR CAUTION LABELS
● = INDICATES AN ⅛" YELLOW DOT (Z2 POWERED EQUIPMENT)

NAMEPLATE LINE SPACING

SEE DWG

1/8"	SPACE	AS REQ'D
	LINE 1	
1/8"	SPACE	
	LINE 2	
1/8"	SPACE	
	LINE 3	
1/8"	SPACE	

TYPICAL ALL DWGS

MIMIC LEGEND

BLACK
BLACK/WHITE
BLACK/YELLOW

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28008
25203-29097,SH.1
25203-29097,SH.16A TO 16G

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
C08 CRP. FRONT VIEW ARRANGEMENT
STATION SERVICE ELECTRIC
WATERFORD, CONNECTICUT

REVISION DESCRIPTION (SIGNATURES ON FILE): REVISED PER DC MP2-19-01094 & DUR 21-800297
CAD NO: C302151.DGN
DRAWING NO: 25203-30215
REV 7
DSGN JCV | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 1

\BBLOGIN\jenne594
12-JAN-2022 09:59
C302151.DGN



CRP - C08 (RIGHT VERTICAL)

CRP C08

PLAN VIEW
CONTROL PANEL
N.T.S.

NOTES

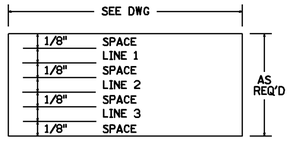
1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE.

NAMEPLATE LEGEND

REFER TO SPECIFICATION SP-EE-261

▲ = INDICATES A BLACK TRIANGLE R.G. 1.97 INSTR.
✦ = INDICATES AN $\frac{1}{8}$" RED DOT (Z1 POWERED EQUIPMENT), EXCEPT FOR CAUTION LABELS
● = INDICATES AN $\frac{1}{8}$" YELLOW DOT (Z2 POWERED EQUIPMENT)

NAMEPLATE LINE SPACING



TYPICAL ALL DWGS

MIMIC LEGEND

BLACK

BLACK/DASHED

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28008
25203-29097,SH.1
25203-29097,SH.16A TO 16G

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

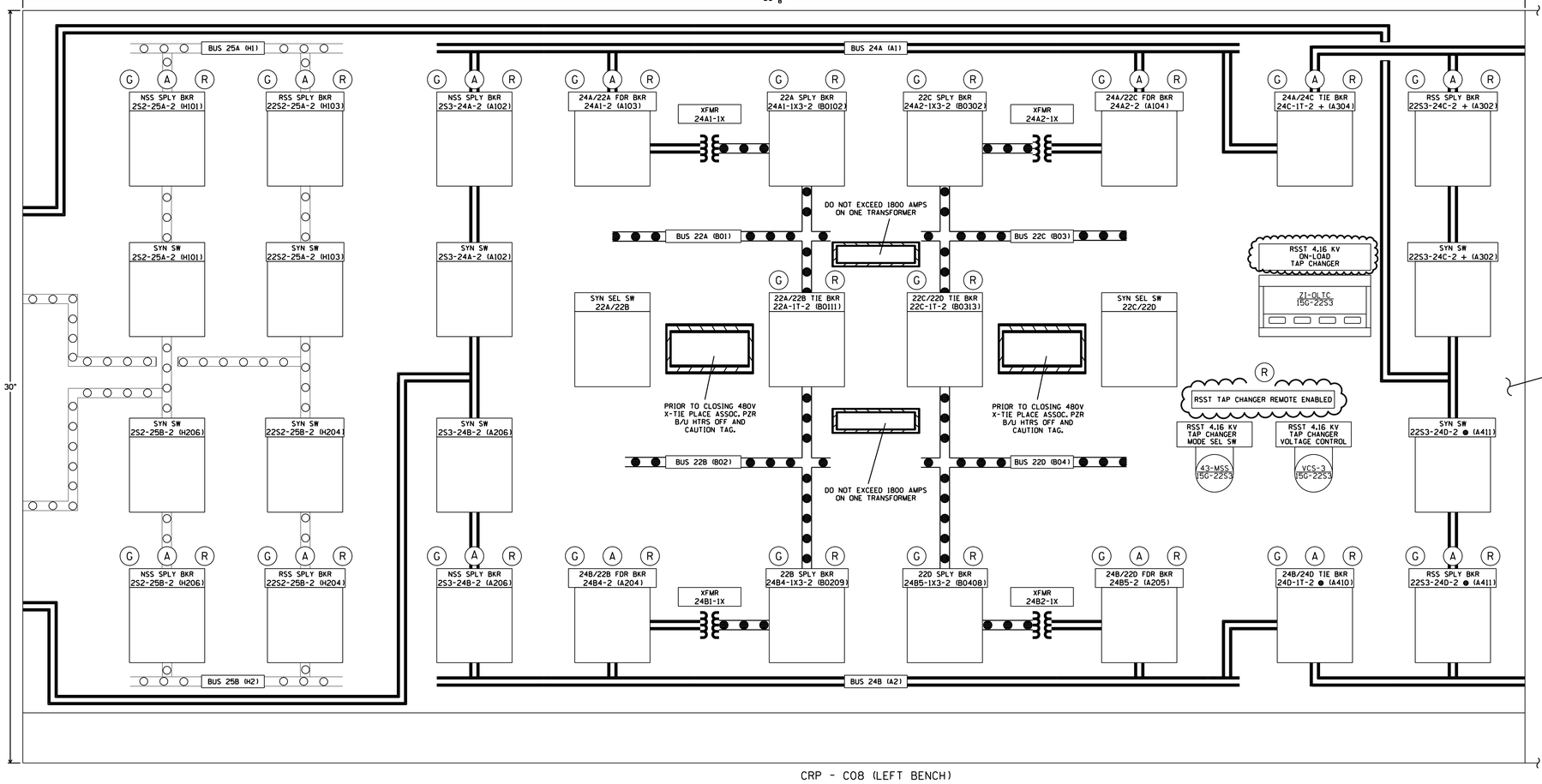
NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
7	9-17-01	INCORP'D DCN DM2-00-0865-99	VP	KBB	MAG	GMO
6	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
5	9/25/99	INCORP. DCN(S)DM2-00-0532-98, DM2-02-0532-98	MEL	WBZ	JPP	JPP
4	11-9-96	INCORP. DCN DM2-S-0134-94 PER DCR M2-S-0105-96	JMM	MM	JPP	JPP
N/A	3	10-31-95	INCORP. DCN DM2-S-1054-94 PER DCR M2-S-0390-94	JMM	RCH RTM	JPP	JPP
86-230	2	12-7-90	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC
86-230	1	6/3/88	AS BUILT PER DSR-M2-S-302-88	TAS	DJK		LJC

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

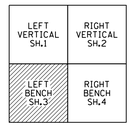
TITLE
MILLSTONE UNIT 2
C08, CRP. FRONT VIEW ARRANGEMENT
STATION SERVICE ELECTRIC
WATERFORD, CONNECTICUT

BY	CHKD.	APP.	APP.
T.A.SWIFT	AHS	———	LJC
DATE 7/9/87	DATE 8/19/87	DATE ———	DATE 8/21/87

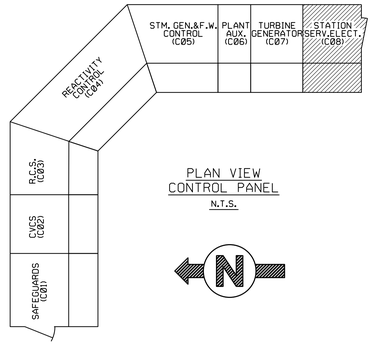
SCALE N.T.S.
DWG. NO. 25203-30215, SH. 2 of 4
86-230



CRP - C08 (LEFT BENCH)

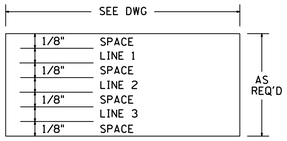


CRP C08



PLAN VIEW
CONTROL PANEL
N.T.S.

NAMEPLATE
LINE SPACING



TYPICAL ALL DWGS

MIMIC LEGEND

BLUE
BLACK
RED
BLACK/YELLOW

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8'' RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS
   ● -INDICATES AN 1/8'' YELLOW DOT (TRAIN B)
   * - INDICATES AN 1/8'' BLACK DOT.
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

FSAR FIGURE

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28008
25203-29097,SH.1
25203-29097,SH.16A TO 16G

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
C08, CRP. FRONT VIEW ARRANGEMENT
STATION SERVICE ELECTRIC

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C302153.DGN	
REVISED PER MSDUR 19-000029 AND DUR 19-800080	DRAWING NO: 25203-30215	REV 5
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH 3

MBULOGIN\diane48
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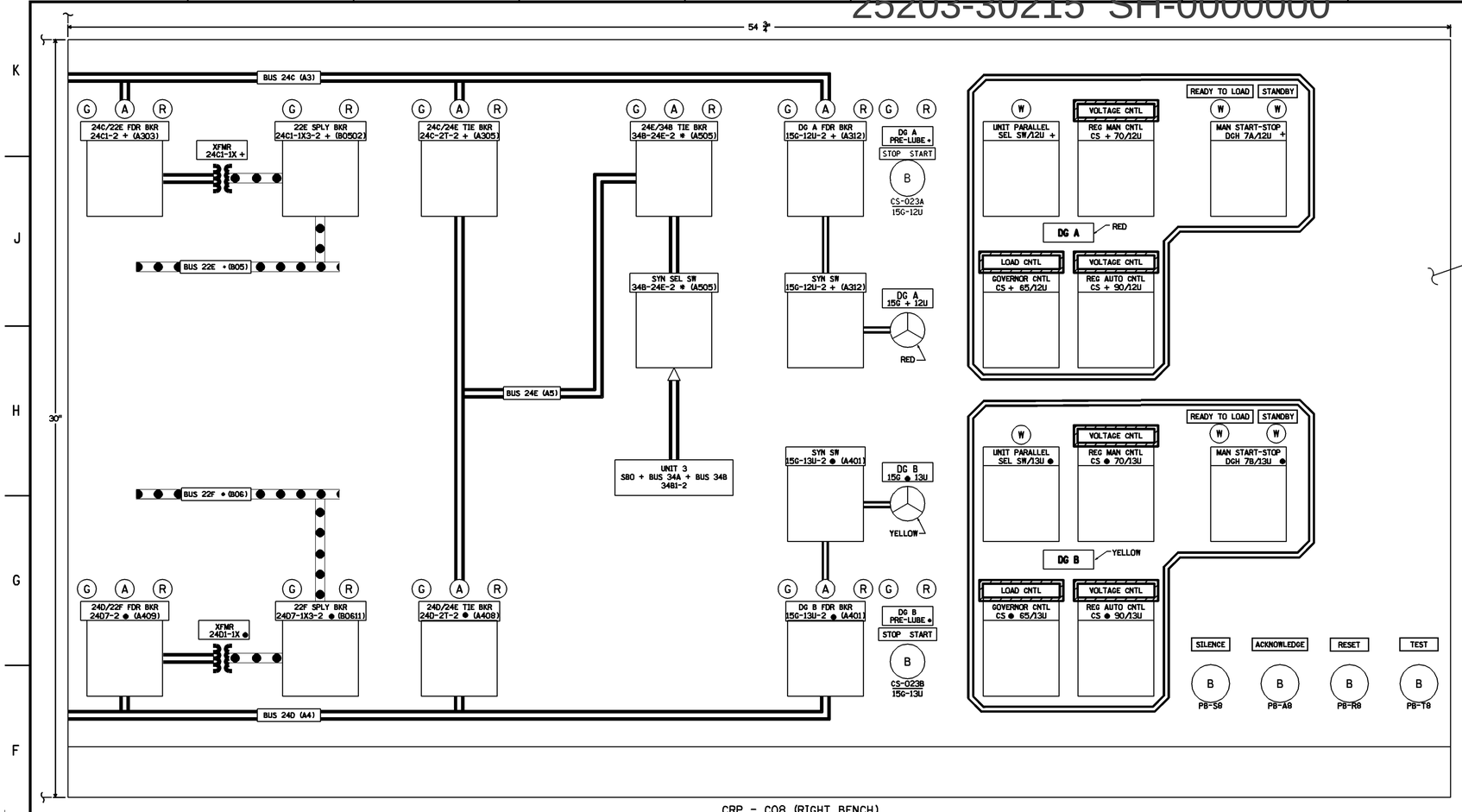
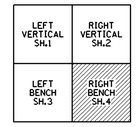
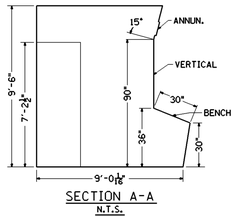
C302153.DGN



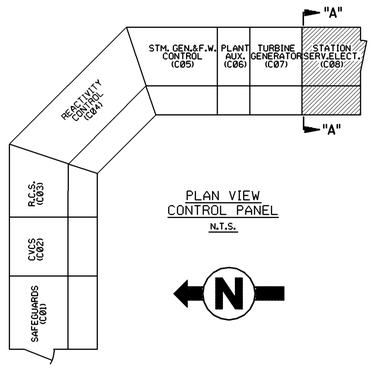
CRP - C08 (RIGHT BENCH)



CRP C08

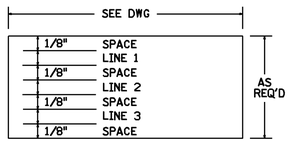


SECTION A-A
N.T.S.



PLAN VIEW
CONTROL PANEL
N.T.S.

NAMEPLATE LINE SPACING



TYPICAL ALL DWGS

NOTES

1. NAMEPLATES THAT ARE RED, GREEN OR BLUE HAVE WHITE LETTERS.
2. + - INDICATES AN 1/8" RED DOT (TRAIN A), EXCEPT FOR CAUTION LABELS ▭
   ● -INDICATES AN 1/8" YELLOW DOT (TRAIN B)
   * -INDICATES AN 1/8 " BLACK DOT. (Z5 POWERED SWING EQUIPMENT)
3. LETTER SIZES AND DEVICE NUMBERS ARE NOT SHOWN TO SCALE .

MIMIC LEGEND

BLUE
BLACK
BLACK/YELLOW

THIS DWG. SUPERCEDES PART OF FOLLOWING DWGS.
25203-28008
25203-29097,SH.1
25203-29097,SH.16A TO 16G

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK.	APP.	APP.
8	9-17-01	INCORP'D DCNS DM1-00-0118-99, DM2-00,03-0865-93	VP	DVP	MAG	GMO
7	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
6	9/25/98	INCORP. DCN DM2-04-0356-97	WBZ	JSJ	JPP	JPP
N/A	5	4-7-98	INCORP. DCN DM2-03-0256-97	AP	GFD	RWD	RW
N/A	4	11-9-96	INCORP. DCN DM2-S-0403-95 PER DCR M2-S-0105-96	JMM	MM	JPP	JPP
91-056	3	5-13-93	AS BUILT PER DM2-S-1377-92, DM2-S-1993-92, DCR M2-S-1265-92	JSJ	MBR DME		RCD
86-230	2	12-7-90	AS BUILT PER DCR-M2-S-1381-90	WBZ	MEL MBR	JTK	LJC

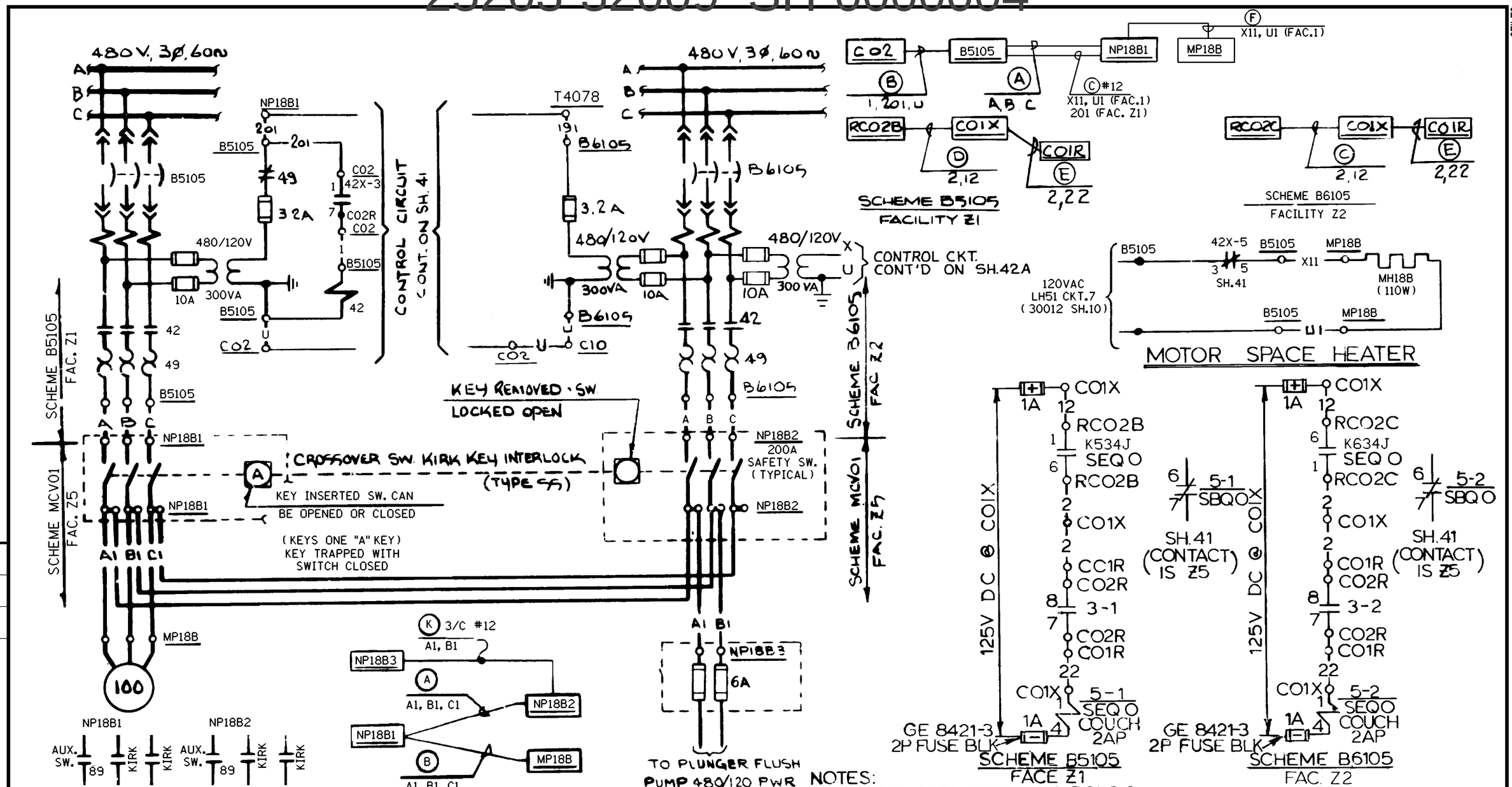
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
C08, CRP, FRONT VIEW ARRANGEMENT
STATION SERVICE ELECTRIC
WATERFORD, CT.

BY	CHKD.	APP.	APP.
T.A.SWIFT	AHS	———	LJC
DATE 7/9/87	DATE 8-19-87	DATE ———	DATE 8-21-87

SCALE N.T.S. | DWG. NO. 86-230 | 25203-30215 SH. 4 of 4

25203-32009 SH-0000004



NOTES:
1- FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
13	3-29-00	INCORP. DCN DM2-00-0823-98	DM	WBZ	JPP	JPP
12	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
11	9/9/99	INCORP. DCN's DM2-00-1989-98, DM2-00-0375-99	WBZ	JSJ	JPP	JPP

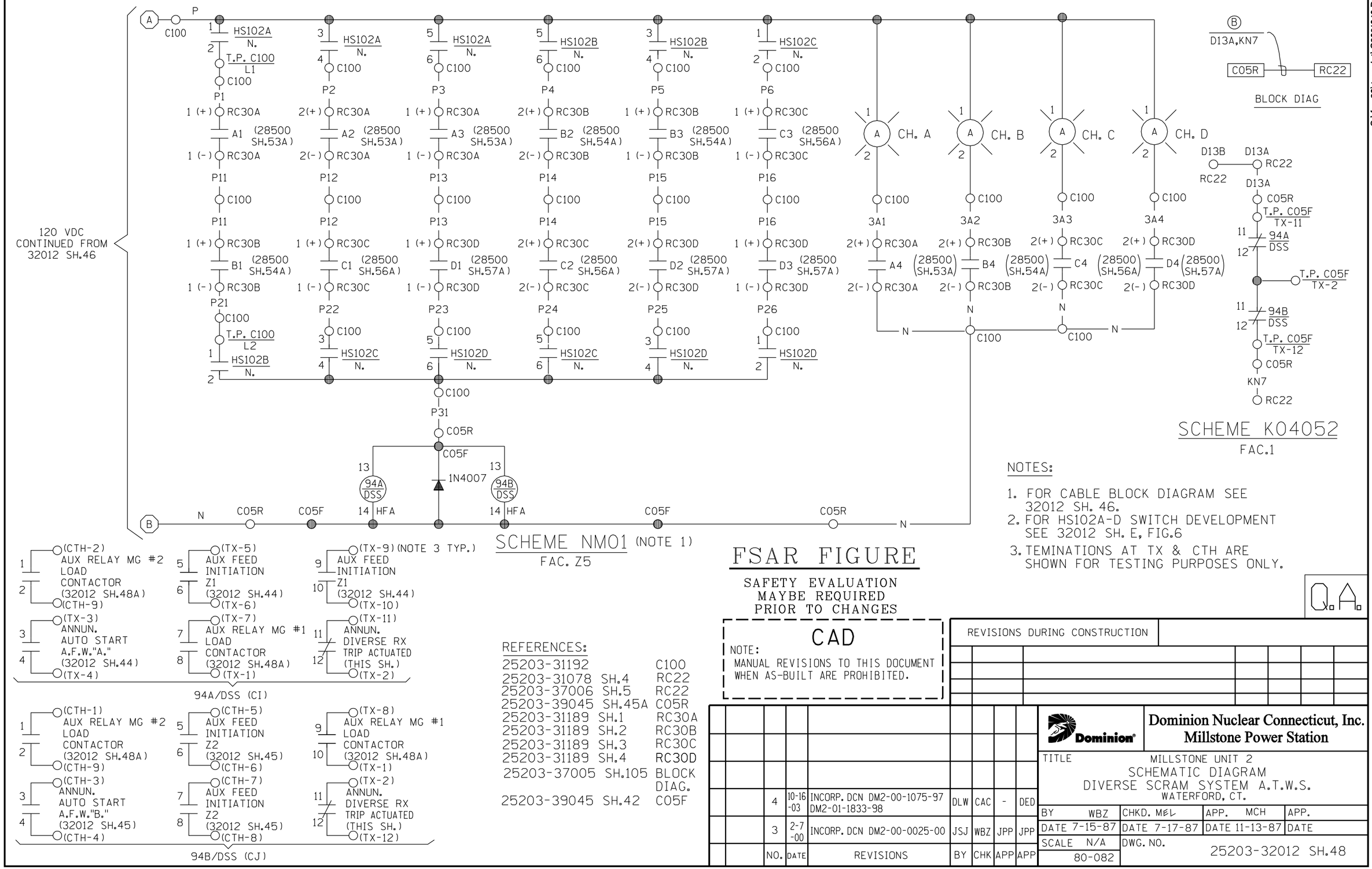
Q.A. CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE MILLSTONE UNIT 2
CHARGING PUMP
PWR. SUPPLY CROSSOVER P18B
WATERFORD, CT.

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JSJ	CHKD.	APP. —	APP. —
DATE 2-13-97	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-32009 SH. 42		
N/A			

25203-32012 SH-0000004



SCHEME NM01 (NOTE 1)
FAC. Z5

SCHEME K04052
FAC.1

NOTES:

1. FOR CABLE BLOCK DIAGRAM SEE 32012 SH. 46.
2. FOR HS102A-D SWITCH DEVELOPMENT SEE 32012 SH. E, FIG.6
3. TEMINATIONS AT TX & CTH ARE SHOWN FOR TESTING PURPOSES ONLY.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

REFERENCES:

25203-31192	C100
25203-31078 SH.4	RC22
25203-37006 SH.5	RC22
25203-39045 SH.45A	C05R
25203-31189 SH.1	RC30A
25203-31189 SH.2	RC30B
25203-31189 SH.3	RC30C
25203-31189 SH.4	RC30D
25203-37005 SH.105	BLOCK DIAG.
25203-39045 SH.42	C05F

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
4	10-16-03	INCORP. DCN DM2-00-1075-97 DM2-01-1833-98	DLW	CAC	-	DED
3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

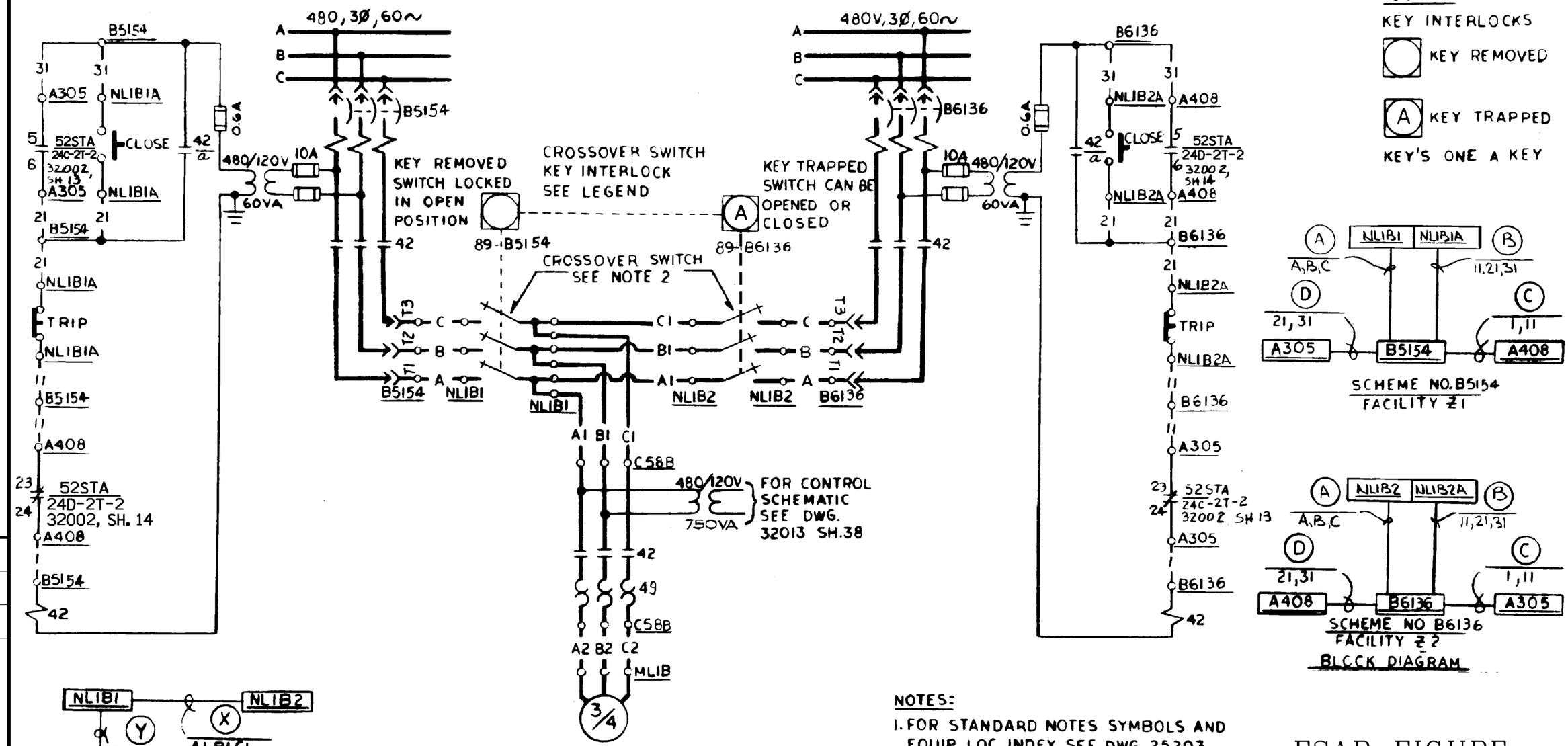
Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
SCHEMATIC DIAGRAM
DIVERSE SCRAM SYSTEM A.T.W.S.
WATERFORD, CT.

BY WBZ	CHKD. MEL	APP. MCH	APP.
DATE 7-15-87	DATE 7-17-87	DATE 11-13-87	DATE
SCALE N/A	DWG. NO. 25203-32012 SH.48		
80-082			

C:\fms32\tmpplot\C3201248.DGN 8/17/2011



NOTES:

1. FOR STANDARD NOTES SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2. CROSSOVER SW. 89, 30A, 600 VOLT, 3 POLE S.T. NEMA 1 NON-FUSIBLE DISCONNECT SW. SEE DWG. 7604-E-39-ML, ITEM 6.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
5	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
4	9/25/99	INCORP. DCN DM2-01-0153-97	MEL	JSJ	JPP	JPP

CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE: MILLSTONE UNIT 2
SERVICE WATER STRAINER ML1B POWER SUPPLY CROSSOVER SCHEME
WATERFORD, CT.

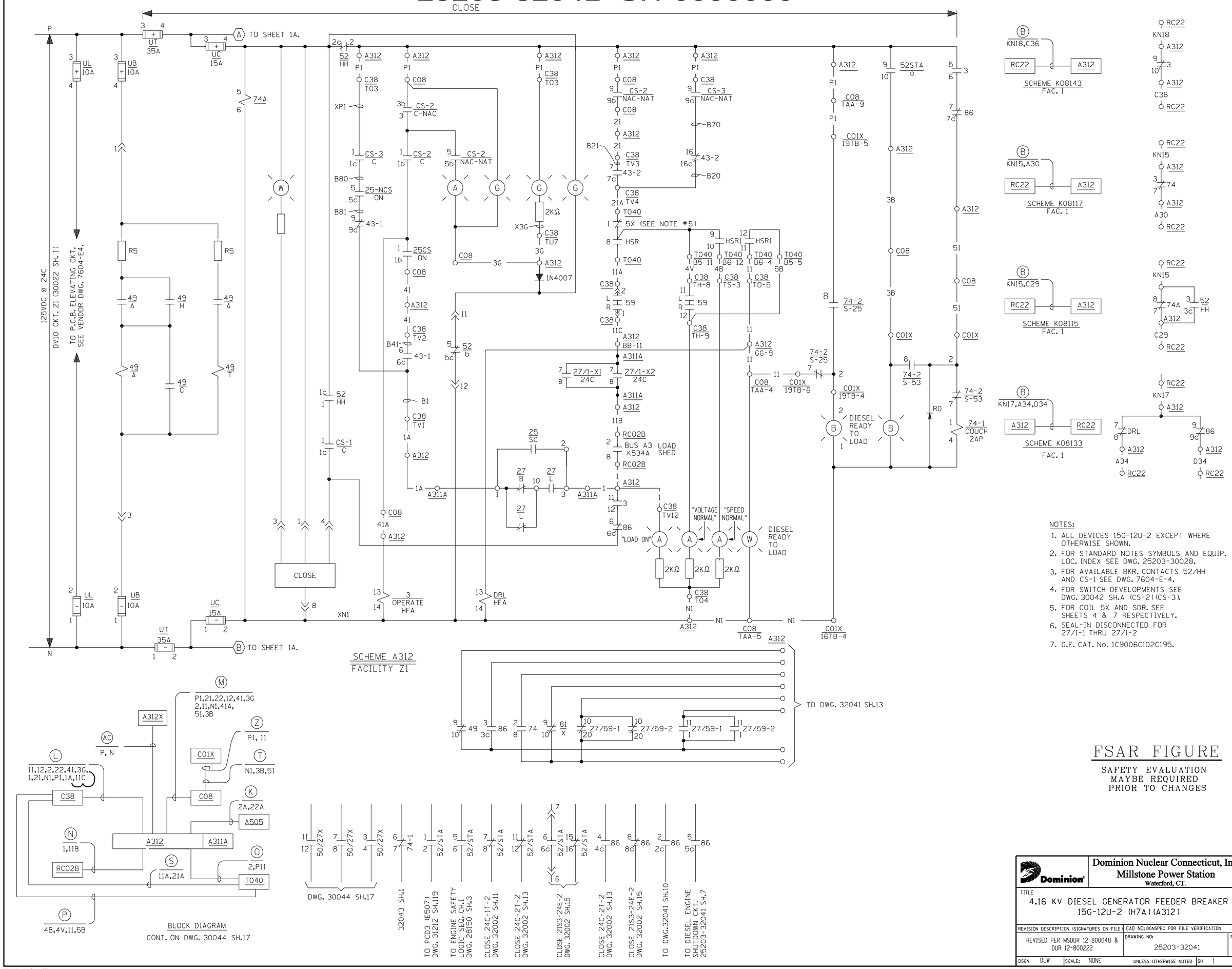
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JMM/JSJ	CHKD.	APP.	APP.
DATE 10/22/97	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-32013 SH. 39	
N/A			

$FILE$
$DATE$

25203-32041 SH-0000000



NOTES:

1. ALL DEVICES 15G-12U-2 EXCEPT WHERE OTHERWISE SHOWN.
2. FOR STANDARD NOTES SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-30028.
3. FOR AVAILABLE BKR. CONTACTS 52/HH AND CS-1 SEE DWG. 7604-E-4.
4. FOR SWITCH DEVELOPMENTS SEE DWG. 30042 SH.A (CS-2)(CS-3).
5. FOR COIL 5X AND SDR. SEE SHEETS 4 & 7 RESPECTIVELY.
6. SEAL-IN DISCONNECTED FOR 27/1-1 THRU 27/1-2
7. G.E. CAT. No. 1C9006C102C195.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
4.16 KV DIESEL GENERATOR FEEDER BREAKER 15G-12U-2 (H7A)(A312)

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: DGNSPEC FOR FILE VERIFICATION	
REVISED PER MSDUR 12-800048 & DUR 12-800222	DRAWING NO: 25203-32041	REV 15
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

25203-32041 SH-0000000

CLOSE

P

3 UT 4 — 35A — TO SHEET 2A.

3 UC 4 — 15A

3 UL 4 — 10A

3 UA 4 — 10A

125VDC @ 24D
(DV20 CKT. 21)(30022 SH. 3)

TO P.C.B. ELEVATING CKT.
SEE VENDOR DWG. 7604-E4.

74A

R5 R5

49/A 49/H 49/A 49/C 49/A 49/T

W

2c 52/HH

A401 P1 C39/T03 XP1

1 CS-3/C 1c B80 5 25-NCS/ON 5c B81 9 43-1 9c

A401 P1 C08

3b CS-2/C-NAC 3

1 CS-2/C 1b

1 25CS/ON 1b C08 41 A401 41 B41 C39/TV2 6 43-1 6c B1 C39 1A A401/HH5 1A A401A

5 CS-2/NAC-NAT 5b A G

C08 3G A401 1N4007

A401 P1 C39/T03

G G (SEE NOTE 5)

2KΩ X3G C39/TU7 3G

11 5 52/b 5c 12

1c 52/HH 1 1c CS-1/C

3 1 4

CLOSE

8

C08 41A A401

13 3 OPERATE 14

13 DRL/HFA 14

XN1

2 UL 1 — 10A

2 UA 1 — 10A

1 UC 2 — 15A

1 UT 2 — 35A — TO SHEET 2A.

N

A401 P1 C08/TN-9 P1 C01X/11TB-11

A401 P1 C08/9 CS-2/NAC-NAT 9b C08 21 A401 21 B21 C39/TV3 7 43-2 7c C39/TV4 21A T041 1 5X (SEE NOTE #5) 8 HSR T041 11A C39/TV-10 2 L 59 R 1 C39/TV-11 11C A401/BB-11

A401 P1 C39 9 CS-3/NAC-NAT 9c B70 16 43-2 16c B20

9 HSR1 10 T041/B5-11 4V C39/TR-1 11 L 59 R 12 C39/TR-2

12 HSR1 11 T041/B6-12 4B C39/TS-3

T041/B6-4 11 C39/TD-5

T041/B6-5 5B

A401A 7 27/1-X1/24D 8 7 27/1-X2/24D 8 A401A 11B A401 RC02C A4 LOAD SHED 8 K634A 2 RC02C 1 A401 11 3 12 6 86 6c

25/SC 2 27/B 10 27/L 1 3 A401A 1 A401A

27/L

C39/TV12 1 L10 A "LOAD ON" 2KΩ L12 A "VOLTAGE NORMAL" 2KΩ L11 A "SPEED NORMAL" 2KΩ W DIESEL READY TO LOAD 2KΩ

C38/T04 N1

A401 N1 C08/TN-5 N1 C01X/16TB-4

A401 12 74-2/S-44 11 C01X/11TB-12 11 C08/TN-4 11 A401/HH-1

A401 P1 C08/TN-9 P1 C01X/11TB-11 4 2 DIESEL READY TO LOAD 1 B

9 52STA/a 10 A401 3B C08 3B C01X 8 74-2/S-54 2 B RD

5 3 6 7 86 7c A401 51 C08 51 C01X 2 74-2/S-54 7 1 74-1/COUCH 2AP 4

B KN18,D36 — RC22 — A401 — BLOCK DIAGRAM

RC22 KN18 A401 9 3 10 A401 D36 RC22 — SCHEME KO8144 FAC. 2

B KN16,A31 — RC22 — A401 — BLOCK DIAGRAM

RC22 KN16 A401 3 74 7 A401 A31 RC22 — SCHEME KO8121 FAC. 2

B KN15,D29 — RC22 — A401 — BLOCK DIAGRAM

RC22 KN15 A401 3 74A 7 A401 D29 RC22 — 3 52/HH 3c — SCHEME KO8145 FAC. 2

B KN18,A35,D35 — A401 — RC22 — BLOCK DIAGRAM

RC22 KN18 A401 7 DRL 8 A401 A35 RC22 — 9 86 9c A401 D35 RC22 — SCHEME KO8137 FAC.2

NOTES

1. ALL DEVICES 15G-13U-2 EXCEPT WHERE OTHERWISE SHOWN.
2. FOR STANDARD NOTES SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-30028.
3. FOR AVAILABLE BKR. CONTACTS 52/HH AND CS-1 SEE DWG. 7604-E-4.
4. FOR SWITCH DEVELOPMENTS SEE DWG. 30042 SH.A (CS-2)(CS-3).
5. FOR COIL 5X AND SDR. SEE SHEETS 16 & 19 RESPECTIVELY.
6. SEAL-IN DISCONNECTED FOR 27/1-1 THRU 27/1-2
7. G.E. CAT. No. 1C9006C102C195.

SCHEME A401
FACILITY Z2

M — P1,21,22,12,41,3G 2,11,N1,41A, 51,3B

A401X

Z — P1, 11

AC — P, N

L — 11,12,2,22,41,3G, 1,21,N1,P1,1A,11C

C01X

T — N1,3B,51

C39

C08

A505

N — 1,11B

A401 | A401A

K — 2,P11

RC02C

T041

S — 11A,21A

AA — 4B,4V,11,5B

BLOCK DIAGRAM
CONT. ON DWG. 30044 SH.19

A401 — TO DWG. 32041 SH.25

9 10 49 — 3 3c 86 — 8 2 74 — 10 9 81/X — 10 20 27/59-1 — 10 20 27/59-2 — 11 1 27/59-1 — 11 1 27/59-2

11 12 50/27X — 7 8 50/27X — 3 4 50/27X — 6 7 74-1

1 2 52/STA — TO PCD3 (E512) DWG. 31212 SH.119

5 6 52/STA — TO ENGINE SAFETY LOGIC SEQ. CH.II DWG. 28150 SH.3

7 8 52/STA — CLOSE 24D-1T-2 DWG. 32002 SH.12

11 12 52/STA — CLOSE 24D-2T-2 DWG. 32002 SH.14

7 6c 52/STA 15 16 52/STA 6 — CLOSE 21S3-24E-2 DWG. 32002 SH.15

4 4c 86 — CLOSE 24D-2T-2 DWG. 32002 SH.14

8 8c 86 — CLOSE 21S3-24E-2 DWG. 32002 SH.15

2 2c 86 — TO DWG.32041 SH.22

5 5c 86 — TO DIESEL ENGINE SHUTDOWN CKT. 25203-32041 SH.19

7 7c 86 — THIS DWG. (C01X)

9 9c 86 — THIS DWG. (ANNUN.)

DWG. 30044 SH.19

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
4.16 KV DIESEL GENERATOR FEEDER BREAKER
15G-13U-2 (H7B)(A401)

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: DGNSPEC FOR FILE VERIFICATION	
REVISED PER MSDUR 11-800048 & DUR 11-800160	DRAWING NO: 25203-32041	REV 15
DSGN DLW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 2

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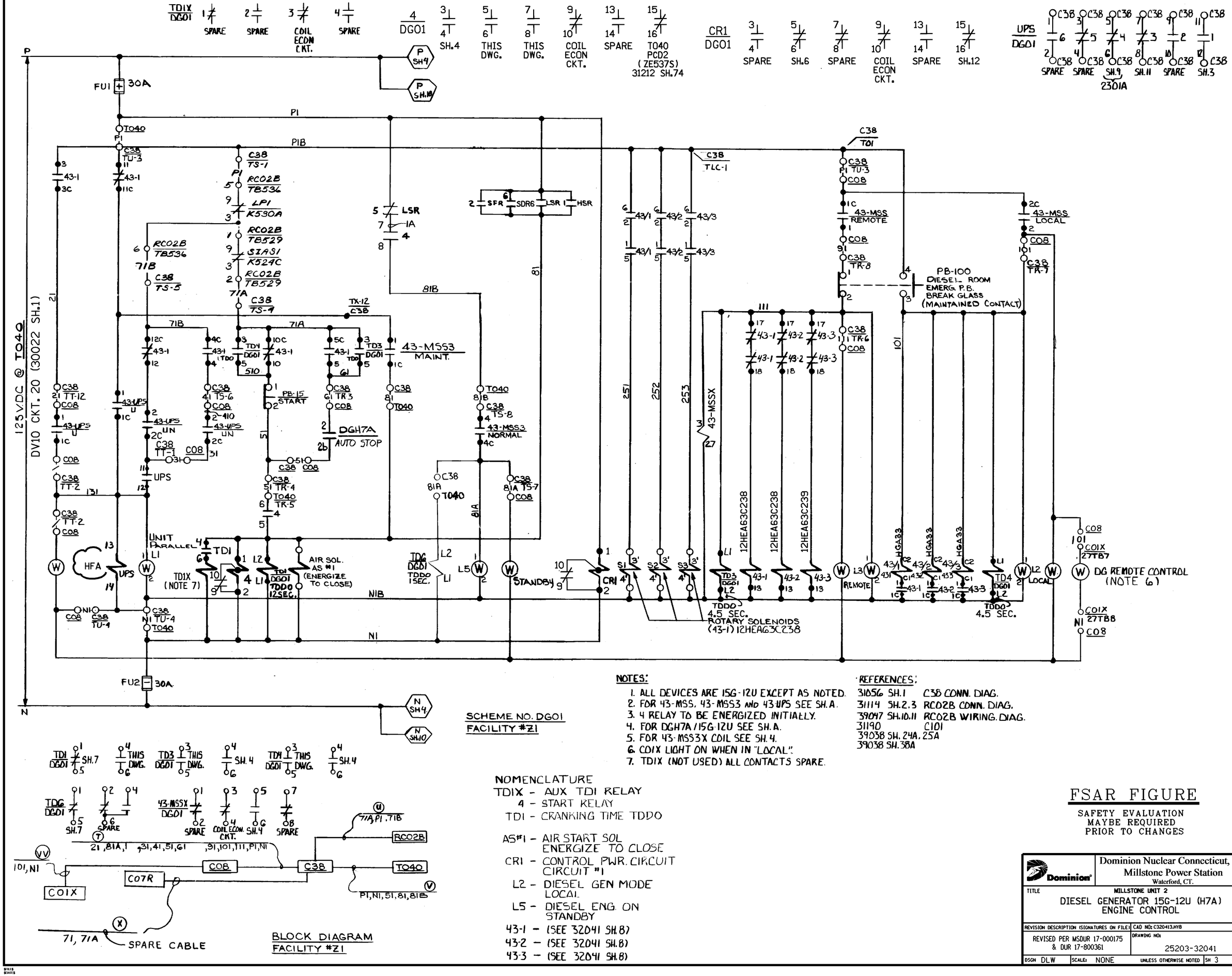
5/6/2011



BLOCK DIAGRAM
FACILITY #Z1

SCHEME NO. DG01
FACILITY #Z1

NOTES:

1. ALL DEVICES ARE 15G-12U EXCEPT AS NOTED.
2. FOR 43-MSS, 43-MSS3 AND 43-UPS SEE SH. A.
3. 4 RELAY TO BE ENERGIZED INITIALLY.
4. FOR DG17A/15G-12U SEE SH. A.
5. FOR 43-MSS3X COIL SEE SH. 4.
6. C01X LIGHT ON WHEN IN "LOCAL".
7. TD1X (NOT USED) ALL CONTACTS SPARE.

REFERENCES:

31056 SH.1	C38 CONN. DIAG.
31114 SH.2.3	RC02B CONN. DIAG.
39047 SH.10.11	RC02B WIRING DIAG.
31190	C101
39038 SH. 24A, 25A	
39038 SH. 38A	

NOMENCLATURE

TD1X - AUX TD1 RELAY
4 - START RELAY
TD1 - CRANKING TIME TDDO
AS#1 - AIR START SOL ENERGIZE TO CLOSE
CR1 - CONTROL PWR. CIRCUIT CIRCUIT #1
L2 - DIESEL GEN MODE LOCAL
L5 - DIESEL ENG. ON STANDBY
43-1 - (SEE 32041 SH.8)
43-2 - (SEE 32041 SH.8)
43-3 - (SEE 32041 SH.8)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A) ENGINE CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320413.HYB	
REVISED PER MSDUR 17-000175 & DUR 17-800361	DRAWING NO: 25203-32041	REV 15
DSGN DLW	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

CR2 DG02: 3/4 SPARE; 5/6 SH.6; 7/8 SPARE; 9/10 COIL ECON CKT.; 13/14 SPARE; 15/16 SH.12

5X DG02: 1 SH.1; 2 THIS DWG.; 3 COIL ECON CKT.; 4 SH.1A; 6 SH.13

5A DG02: 3/4 SPARE; 5/6 SH.10; 7/8 SH.10; 9/10 COIL ECON CKT.; 13/14 SPARE; 15/16 SPARE

LSR: 1 SH.6; 2 SH.10; 3 COIL ECON CKT.; 4 39038 SH.14; 5 SH.3; 6 SH.3; 7 THIS DWG.; 8 32023 SH.7; 9 39038 SH.14; 10 39038 SH.14; 11 SPARE; 12 ANNUN SH.6

P SH3 — P SH5

FU3 30A.

125 VDC

P11 — T040 — C38 TU8

TD0 TD3 DGØ1 (4/6) — 5 43-MSSX G — 14C 43-1 14 — TD0 TD4 DGØ1 (4/6)

43-MSS3 NORMAL 2C 2 — 12

C38 2 TS10 T040 — C38 12 T39 T040 — 3 43-MSS3 MAINT. 3C

PB1 START 1F

2 — 8 SDR — 7 LSR — 2 HSR — 3/4

IC

AIR SOL. AIR VENT AV — ENERGIZE TO CLOSE — ENERGIZE TO OPEN

1 — 10/9 CR2 AS#2 — 2

N2 FU4 30A.

C38 22 TS11 T040 — C38 22 C08 — 3 43-MSS3X DGØ2 — 27 — 1B PB2 STOP

W L4 W MAINTENANCE 2

C08 C38 N2O — C38 N2 TY12 T040

6 ESS — T040 33 C38 TS2

13 43-1 13C — 2 43-1 2C B42 C38 42 TV7 C08

321

1 — 2 PB-14 STOP

32

1 DGH7A STOP 1B

C08 32 — C38 32 TRIO T040

5 DGØ2 TDDO 60SEC

SDR11 — P11 — T4035 TB2-11 — A2 SST-4 810 RPM (NOTE 4) A1 — TB035 TB2-12 — 1J

T040 B9 — L1 — HSR1 8 7 — L2

5 DGØ2 6 TDO — 4 5X

2 5X — 7 SDR — 1E

T4040 B22-1

T4037 TB3-4

T4037 TB3-6 — T4038 TB1-5

M — GOV. EGB-13P — N

5 ESS

T4038 TB1-6

T4037 TB3-7

T4037 TB3-5 — T040 B22-2 — N2

1 — 5A 10/9 — 2

1E — 1F — 1H

T4035 TB2-5 — A2 SST-1 250 RPM (NOTE 4) A1 — T4035 TB2-6 — T040 B22-3

OPS PS-8789 — T040 B29-2

T4035 TB2-7 — A2 SST-2 810 RPM (NOTE 4) A1 — T4035 TB2-8 — T040 B22-4

2 TD2 G DGØ2 — HSR — TD2X

TD2 DGØ2 TDPU 20 SEC.

TD3 DGØ2 TDPU 50 SEC. @ T040

HSR: 1 SH.3; 2 THIS DWG.; 3 COIL ECON. CKT.; 4 SH.10; 5 32013 SH.21; 6 SPARE; 7 39038 SH.14; 8 SH.1; 9 SPARE; 10 SPARE; 11 SPARE; 12 32013 SH.21

HSR1: 1/2 SH.7; 3/4 SP.; 5/6 SP.; 9/10 SH.1; 11/12 SH.1; 13/14 SP.; 15/16 SP.; 20/21 SP.; 22/23 SP.; 7/8 COIL ECONOMIZER

TD2X: 1 SPARE; 2 SPARE; 3 COIL ECON CKT.; 4 SH.13

NOMENCLATURE:

CR2 - CONTROL POWER CIRCUIT #2
AS2 - AIR START SOL. ENERGIZE TO CLOSE
AV - AIR VENT SOL. ENERGIZE TO OPEN
"5" - STOPPING RELAY T.D.D.O. 60 SEC
5X - AUX 5 RELAY (NOT A TIMING RELAY)
GOV - GOV. SHUTDOWN SOL. ENERGIZE STOP
5A - STOPPING RELAY
LSR - LOW SPEED RELAY
HSR, HSR1 - ENGINE AT NORMAL SPEED.
TD2X - AUX TD2 RELAY (NOT A TIMING RELAY)
TD2 - ALARM DELAY T.D.P.U 20 SEC
TD3 - T.D.P.U. 50 SEC
OPS - LUBE OIL PRESSURE (PS-8789)
SSW - SPEED SWITCH
SST - DYNALCO SST SPEED SW AUX RELAY

GOV: GOV SHUTDOWN SOLENOID ENERGIZE TO STOP

N SH3 — N SH5

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES 25203-39038 SH. 2,9

TD3 DGØ2 (5/1) T040 T040 — 6/4 T040 T040 — TO SH.6 — TO DWG. 32023 SH.7

5 DGØ2 (10/5) — 4 G THIS DWG. — SH.7

C08 — C38 — (B) 22,32,42,N2 — (D) P11,2,12,22,32,N2,33 — (G) P11,1F,1H

EGB-13P — (K1) 1E,N2 — T4038 — (K) 1E,N2 — T4037 — (J) 1E,N2 — T040 — (L) 1J — T4035

TD2 DGØ2 (10/5) TDC — 20/6 SH.7 THIS DWG.

43MSS3X DGØ2 (10/2) SPARE — 30/4 COIL ECON. CKT. — 50/6 SH.11 — 70/8 SPARE

BLOCK DIAGRAM FACILITY #Z1

SCHEME NO. DG02
FACILITY #Z1

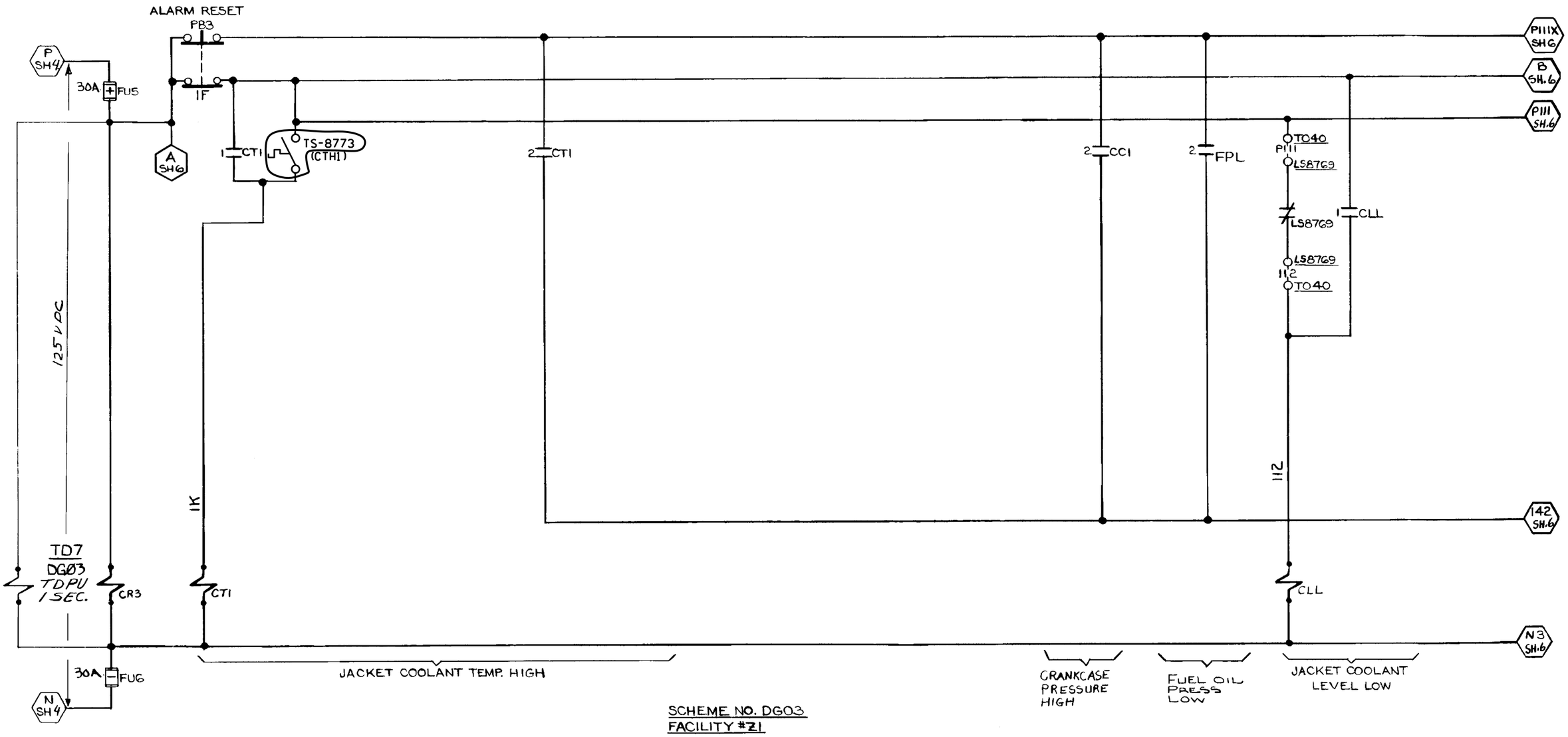
REF. DWGS.
25203-39038 SH. 24A, 25A

NOTES:
1. ALL DEVICES ARE 15G-12U EXCEPT AS NOTED.
2. FOR 43-MSS3 SEE SH A.
3. FOR DGH7A/15G-12U, SEE SH. A.
4. SST-1, SST-2, SST-4 REFER DWG 28500 SH. 11A
5. GOV. EGB-13P TAG: H7AD-GOVERNOR
6. PS-8789 REF. DWG. 28500 SH.883

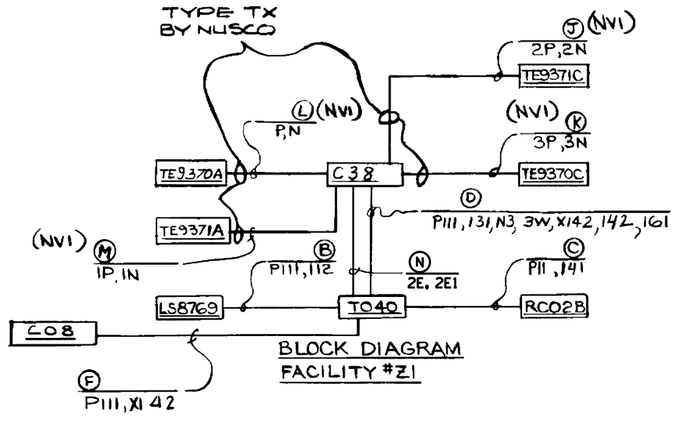
Dominion — Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
ENGINE CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C320414.HYB
REVISED PER MSDUR 13-000114 & DUR 13-800524 | DRAWING NO: 25203-32041 | REV 12
DSGN DLW | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 4



SCHEME NO. DG03
FACILITY #21



BLOCK DIAGRAM
FACILITY #21

NOMENCLATURE
CR3 - CONTROL POWER CIRCUIT No 3
CT1 - JACKET COOLANT TEMP HIGH
CLL - JACKET COOLANT LEVEL LOW
TD7 - TDPU 1 SEC LUBE OIL PRES. LOW COMMON ALARM

REF DWGS.
25203-39038 SH. 24A, 25A

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES
25203-39038 SH.2

REDRAWN FROM SAME
NNECO DWG. No. REV.4

TD7 DG03: 1P 5 SH.7; 2P 6 SH.7

CR3: 1 SP.; 2 SP.; 3 COIL ECON. CKT.; 4 SP.; 6 SH.12

CLL: 1 THIS DWG.; 2 SP.; 3 COIL ECON. CKT.; 4 SH.12

CT1: 1 THIS DWG.; 2 THIS DWG.; 3 COIL ECON. CKT.; 4 SP.; 6 SP.; 7 SH.12

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
ENGINE CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C320415.HYB
REVISED PER DC MP2-19-01207 & DUR 23-800119 | DRAWING NO: 25203-32041 | REV 7
DSGN JSJ | SCALE: NONE UNLESS OTHERWISE NOTED | SH 5

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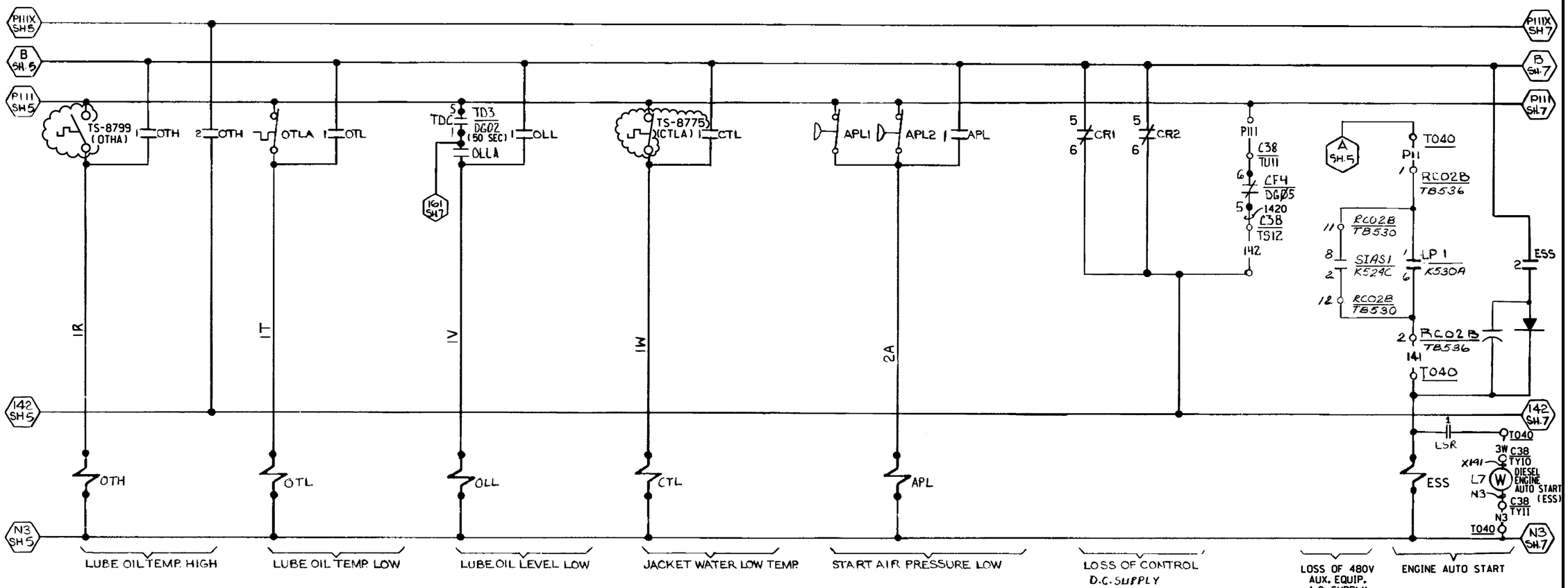


OTH: 1 THIS DWG. · 2 THIS DWG. · 3 COIL ECON CKT. · 4 SPARE · 6 SPARE · 7 SH.12

OTL: 1 THIS DWG. · 2 SPARE · 3 COIL ECON CKT. · 4 SPARE · 6 SPARE · 7 SH.12

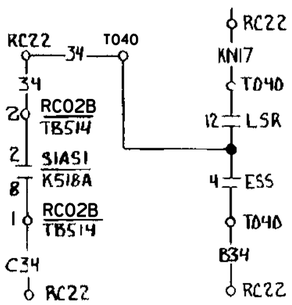
ESS: 1 TO SH.7 · 2 THIS SH. · 3 COIL ECON. CKT. · 4 TO SH.14 · 5 TO SH.4 · 6 TO SH.4 · 7 TO SH.9 · 8 SPARE

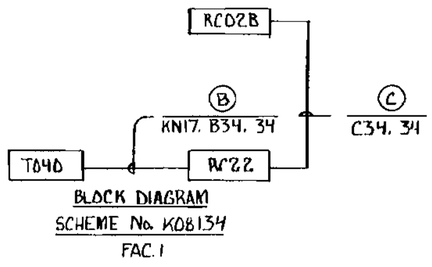
OLL: 1 THIS DWG. · 2 SPARE · 3 COIL ECON CKT. · 4 SPARE · 6 SPARE · 7 SH.12

APL: 1 THIS DWG. · 2 SH.11 · 3 COIL ECON CKT. · 4 PCD3 (P8314) 31212 SH.12D · 6 SPARE · 7 SH.12

CTL: 1 THIS DWG. · 2 SPARE · 3 COIL ECON CKT. · 4 SPARE · 6 SPARE · 7 SH.12

SCHEME NO. DG03
FACILITY #Z1





BLOCK DIAGRAM
SCHEME No. K08134
FAC. 1

REF. DWGS.
25203-39038 SH.24A, 25A

REFERENCES
31114 SH.2.3 RC02B CONN. DIAG.
39047 SH.10 RC02B WIRING DIAG.

SUPERCEDES PART OF
25203-39038 SH.2,9,3

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT No.2
DIESEL GENERATOR 15G-12U (H7A)
ENGINE CONTROLS

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C320416.HYB
REVISED PER DC MP2-19-01207 & DUR 21-800167 | DRAWING NO: 25203-32041 | REV 10
DSGN JCV | SCALE: NONE UNLESS OTHERWISE NOTED | SH 6

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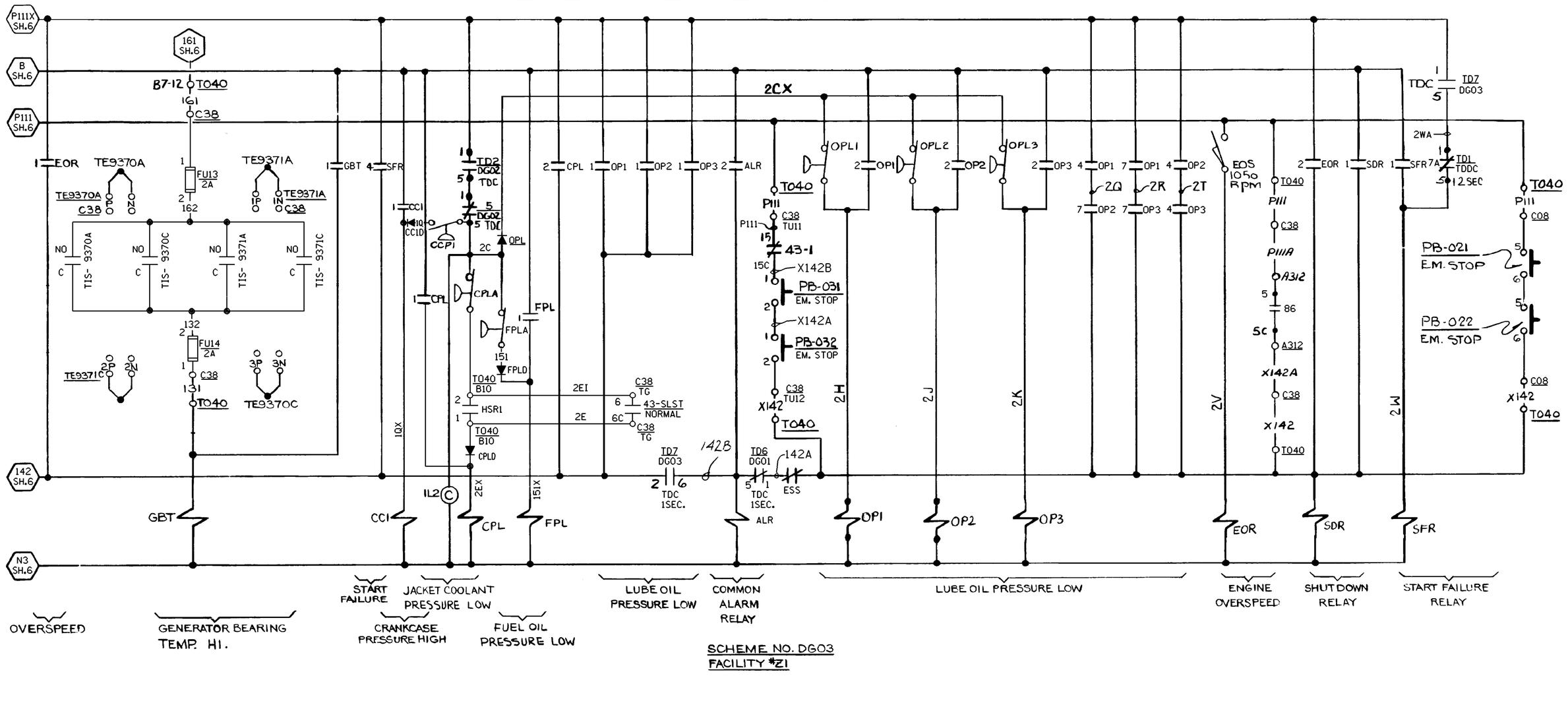


SCHEME NO. DG03
FACILITY #21

GBT
1 THIS DWG | 2 SP. | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH. 12

CC1
1 THIS DWG | 2 SH.5 | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH. 12

CPL
1 THIS DWG | 2 THIS DWG | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH. 12

FPL
1 THIS DWG | 2 SH.5 | 3 COIL ECON. CKT. | 4 SH.13 | 6 SP. | 7 SP.

ALR
1 SP. | 2 THIS DWG | 3 COIL ECON. CKT. | 4 SP.

OP1
1 THIS DWG | 2 THIS DWG | 3 COIL ECON. CKT. | 4 THIS DWG | 5 SP. | 6 SP. | 7 THIS DWG | 8 SH. 12

OP2
1 THIS DWG | 2 THIS DWG | 3 COIL ECON. CKT. | 4 THIS DWG | 5 SP. | 6 SP. | 7 THIS DWG | 8 SH. 12

OP3
1 THIS DWG | 2 THIS DWG | 3 COIL ECON. CKT. | 4 THIS DWG | 5 SP. | 6 SP. | 7 THIS DWG | 8 SH. 12

EOR
1 THIS DWG | 2 THIS DWG | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH. 12

SFR
1 THIS DWG | 2 SH.3 | 3 COIL ECON. CKT. | 4 THIS DWG | 5 SP. | 6 SH. 12 | 7 SP.

SDR
1 THIS DWG | 2 SH.10 | 3 COIL ECON. CKT. | 4 SH 1A | 5 SH.11 | 6 SH.3 | 7 SH.4 | 8 SH.4 | 10 SP. | 11 SH.4

SUPERSEDES
25203-39038 SH.3,9,13

REF. DWGS.
25203-39038 SH. 24A, 25A

NOTES:
1. FOR BLOCK DIAGRAM SEE SH 5.

FSAR FIGURE

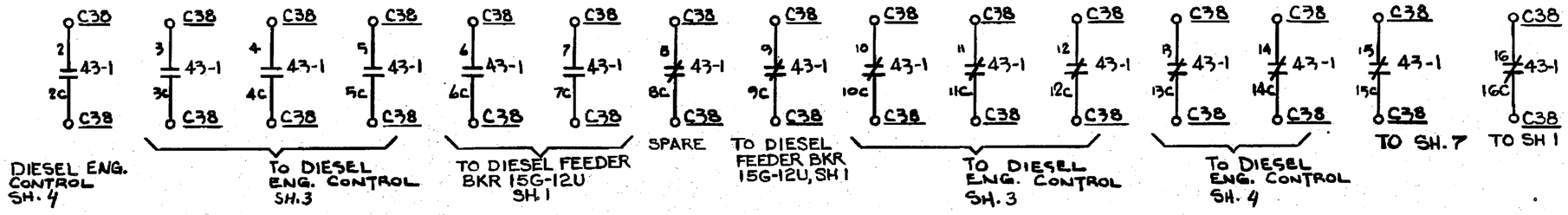
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

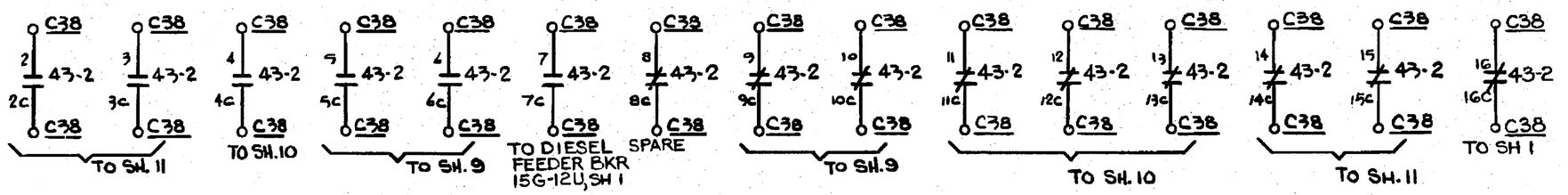
Dominion

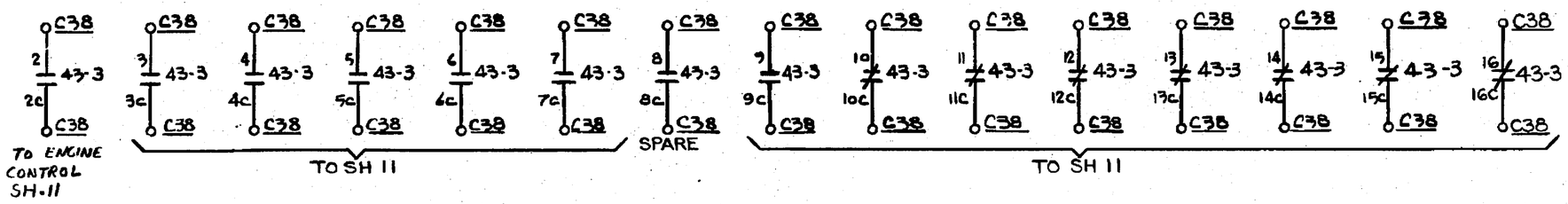
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
ENGINE CONTROLS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320417.HYB	
REVISED PER MSDUR 13-000114 & DUR 13-800524	DRAWING NO: 25203-32041	REV 9
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH 7







NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2. ALL RELAY CONTACTS SHOWN IN RESET POSITION (LOCAL)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REDRAWN FROM SAME NNECO DWG. No. REV.1

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

REVISIONS DURING CONSTRUCTION	P.A.*

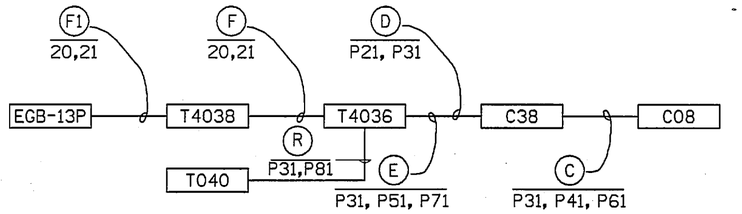
MF	P.A.*	NO.	DATE	REVISIONS	BY	CHK	APP	APP
		2	2/11/00	INCORP. DCN DM2-00-0025-00	JSJ		JPP	JPP

Northeast Utilities System

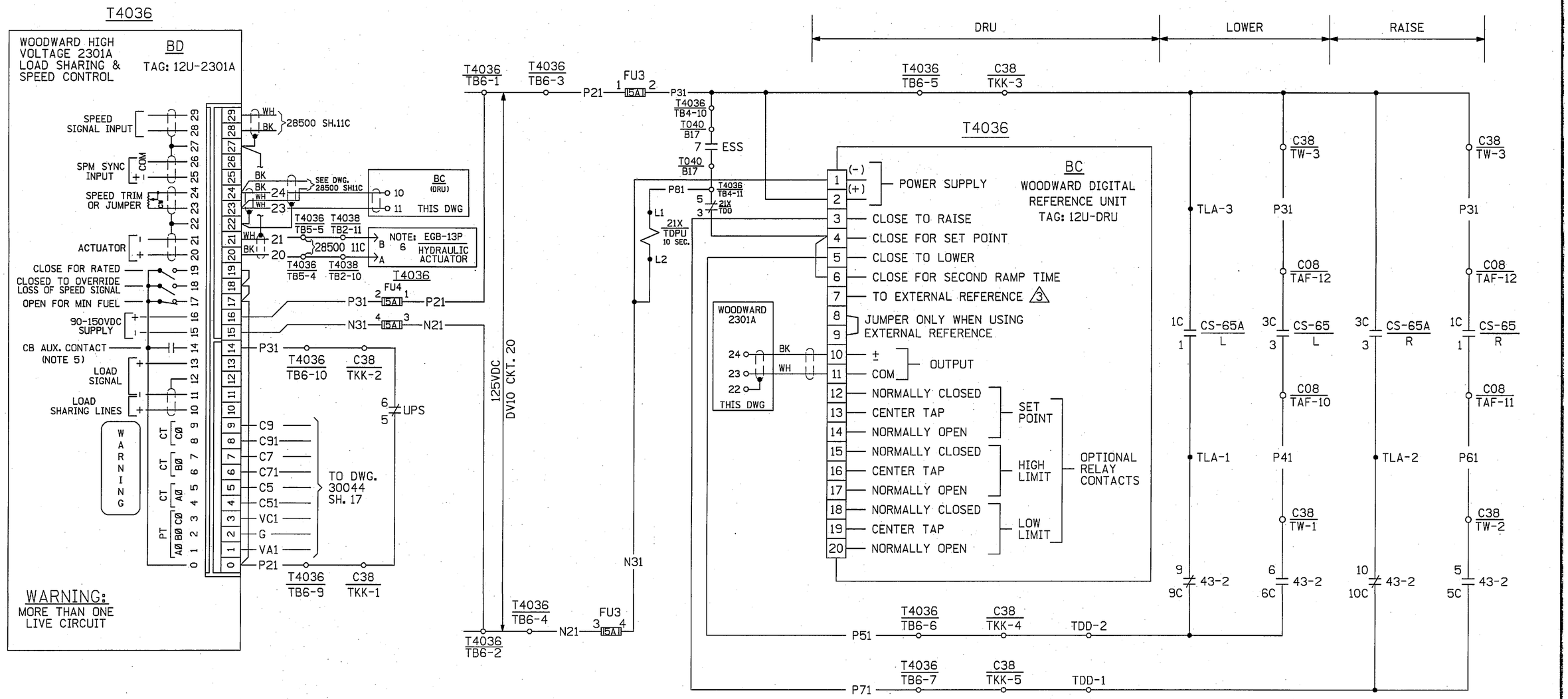
FOR NORTHEAST NUCLEAR ENERGY CO.

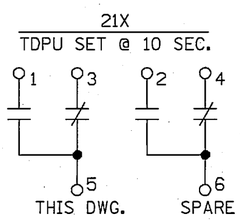
TITLE: MILLSTONE UNIT No.2
DIESEL GENERATOR 15G-12U (H7A)
ENGINE CONTROL RELAYS
WATERFORD, CT.

BY JSJ/WSB	CHKD.	APP. JPP	APP. JPP
DATE 1-27-99	DATE	DATE 2-7-00	DATE 2-7-00
SCALE N/A	DWG. NO.	25203-32041 SH.8	
P.A.* N/A			



BLOCK DIAGRAM
SCHEME: DG04
FAC: Z1





SCHEME. NO. DG04
FACILITY #Z1

NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX, SEE DWG. 25203-30028.
2. FOR CONTROL SWITCH DEVELOPEMENT, SEE DWG. 32041 SH. A.
3. FOR RELAYS UPS & 43-2, SEE DWG. 32041 SHEETS 3 & 8.
4. 43-2 CONTACTS SHOWN IN THE D/G POSITION (LOCAL CONTROL).
5. OPEN FOR DROOP CLOSE FOR ISOCHRONOUS
6. GOV. EGB-13P TAG: H7AD-GOVERNOR

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NOTE: C.A.D.
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED. REFER TO GED 3.01 FOR MANUAL REVISIONS DURING CONSTRUCTION.

THIS DWG. SUPERCEDES PART OF 25203-39038 SH. 13

Q.A.

REVISIONS DURING CONSTRUCTION | P.A.•

MF	P.A.•	NO.	DATE	REVISIONS	BY	CHK	APP	APP
		6	9/12	INCORP. DCN DM2-00-0025-00	JSJ	MBR	JFP	JFP
	N/A	5	3-25-96	INCORP. DCN DM2-S-1171-95 PER DCR M2-S-0025-96	JMM	MBR	JFP	JFP
	91-056	4	5/94	AS BUILT PER DCN-DM2-S-0141-93 & DCR-M2-S-1452-93	DK	MBR DME	JFP	JFP
	91-056	3	5-13-93	AS BUILT PER DM2-S-0637-92, DM2-S-0847-92, DM2-S-1437-92, DM2-S-1791-92, DM2-S-1993-92, DCR M2-S-0779-92	JSJ	MBR DME		RCD

NORTHEAST UTILITIES SERVICE CO.
FOR NORTHEAST NUCLEAR ENERGY CO.

TITLE: MILLSTONE UNIT NO. 2
DIESEL GENERATOR 15G-12U (H7A)
GOVENOR CONTROL
WATERFORD, CT.

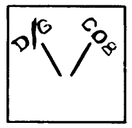
BY DMM	CHKD. RJH	APP. KWF	APP. KWF
DATE 5/13/92	DATE 7-7-92	DATE 7-6-92	DATE 7-6-92

SCALE N/A
P.A.• 91-056
DWG. NO. 25203-32041 SH. 9

MR

43-MSS 15G-12U 43-MSS 15G-13U

CONTACT HANDLE END	CONT. Nº	POSITION COS	POSITION D/G
1 1C 2C 2	1	X	
	2		X
3 3C 4C 4	3	X	
	4		X
5 5C 6C 6	5	X	
	6		X
7 7C 8C 8	7	X	
	8		X

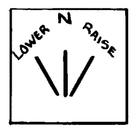


MAINTAINED CONTACTS SB1 SEL. SW.
KEY OPERATED, KEY REMOVABLE IN LOCAL POSITION ONLY.
(TWO DIFFERENT KEYS ARE REQ'D FOR DIESELS H7A & H7B)

FIGURE 1

CS-65 15G-12U CS-65 15G-13U
CS-70 15G-12U CS-70 15G-13U CS-90 15G-12U CS-90 15G-13U

CONTACT HANDLE END	CONT. Nº	POSITION LOWER	POSITION N	POSITION RAISE
1 1C 2C 2	1			X
	2		X	
3 3C 4C 4	3	X		
	4			X
5 5C 6C 6	5		X	
	6	X		
7 7C 8C 8	7			X
	8		X	
9 9C 10C 10	9	X		
	10			X
11 11C 12C 12	11		X	
	12	X		

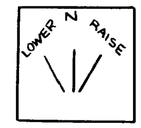


SBM. SPRING RETURN TO NORMAL CONTR. SW.

FIGURE 2

CS-65A 15G-12U CS-65A 15G-13U
CS-70A 15G-12U CS-70A 15G-13U CS-90A 15G-12U CS-90A 15G-13U

CONTACT HANDLE END	CONT. Nº	POSITION LOWER	POSITION N	POSITION RAISE
1 1C 2C 2	1	X		
	2		X	
3 3C 4C 4	3			X
	4	X		
5 5C 6C 6	5		X	
	6			X
7 7C 8C 8	7	X		
	8		X	
9 9C 10C 10	9			X
	10	X		
11 11C 12C 12	11		X	
	12			X



SBM SPRING RETURN TO NORMAL CONTR. SW.

FIGURE 2A

43-MSS3 15G-12U 43-MSS3 15G-13U (G.E. 16SB1BB226SAM2Y)

CONTACT HANDLE END	CONT. Nº	POSITION MAINT.	POSITION NORMAL
1 1C 2C 2	1	X	
	2		X
3 3C 4C 4	3	X	
	4		X

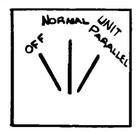


MAINTAINED CONTACTS SB1 SEL. SW.
KEY OPERATED (SAME KEY FOR FIG.1)
KEY REMOVABLE IN NORMAL POSITION ONLY. - SEE NOTE # 2

FIGURE 5

43-UPS 15G-12U 43-UPS 15G-13U (UPS = UNIT PARALLEL SW.)

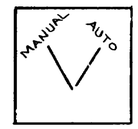
CONTACT HANDLE END	CONT. Nº	POSITION OFF	POSITION NORMAL	POSITION UNIT PARALLEL
1 1C 2C 2	1			X
	2		X	X
3 3C 4C 4	3			X
	4		X	X



SB1 SPRING RETURN TO NORMAL CONTROL SW.

FIGURE 4

CONTACT HANDLE END	CONT. Nº	POSITION AUTO	POSITION MANUAL
1 1C 2C 2	1	X	
	2		X
3 3C 4C 4	3	X	
	4		X
5 5C 6C 6	5	X	
	6		X
7 7C 8C 8	7	X	
	8		X



G.E. 16SB1DB203SSM3
MAINTAINED CONTACT SELECTOR SWITCH

FIGURE 3

DGH7A 15G-12U , DGH7B 15G-13U

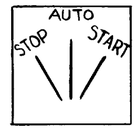
CONTACT HANDLE END	CONT. Nº	POSITION START	POSITION AUTO	POSITION STOP
1 1b 2b 2	1			X
	2		X	X
3 3b 4b 4	3			X
	4		X	X



16SBMB3D41S1S2
SPRING RETURN TO AUTO CONTROL SWITCH

FIGURE 6

CONTACT HANDLE END	CONT. Nº	POSITION START	POSITION AUTO	POSITION STOP
1 1C 2C 2	1	X		
	2	X	X	
3 3C 4C 4	3	X		
	4	X	X	

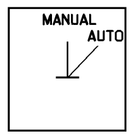


SB1 SPRING RETURN TO AUTO CONTROL SW.

FIGURE 7

43-VRMS 15G-12U 43-VRMS 15G-13U

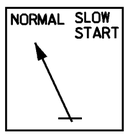
CONTACT HANDLE END	CONT NO	POSITION AUTO	POSITION MANUAL
1 1C 2C 2	1		X
	2		X
3 3C 4C 4	3		X
	4		X
5 5C 6C 6	5	X	
	6	X	
7 7C 8C 8	7	X	
	8	X	



G.E. 16SB1DB204SSM2P
MAINTAINED CONTACT SELECTOR SWITCH

FIGURE 3A

CONTACT HANDLE END	CONT NO	POSITION SLOW START	POSITION NORMAL
1 1C 2C 2	1	X	
	2		X
3 3C 4C 4	3	X	
	4		X
5 5C 6C 6	5	X	
	6		X



MAINTAINED CONTACT SB1 SEL. SW.
KEY OPERATED
KEY REMOVABLE IN NORMAL POSITION

FIGURE 8

NOTES:

1. FOR STANDARD NOTES SYMBOLS & EQUIP. LOCATION; INDEX SEE DWG # 25203-30028.
2. SWITCHES DESIGNATED 15G-12U ARE FACILITY Z1 & 15G-13U ARE FACILITY Z2.

FSAR FIGURE

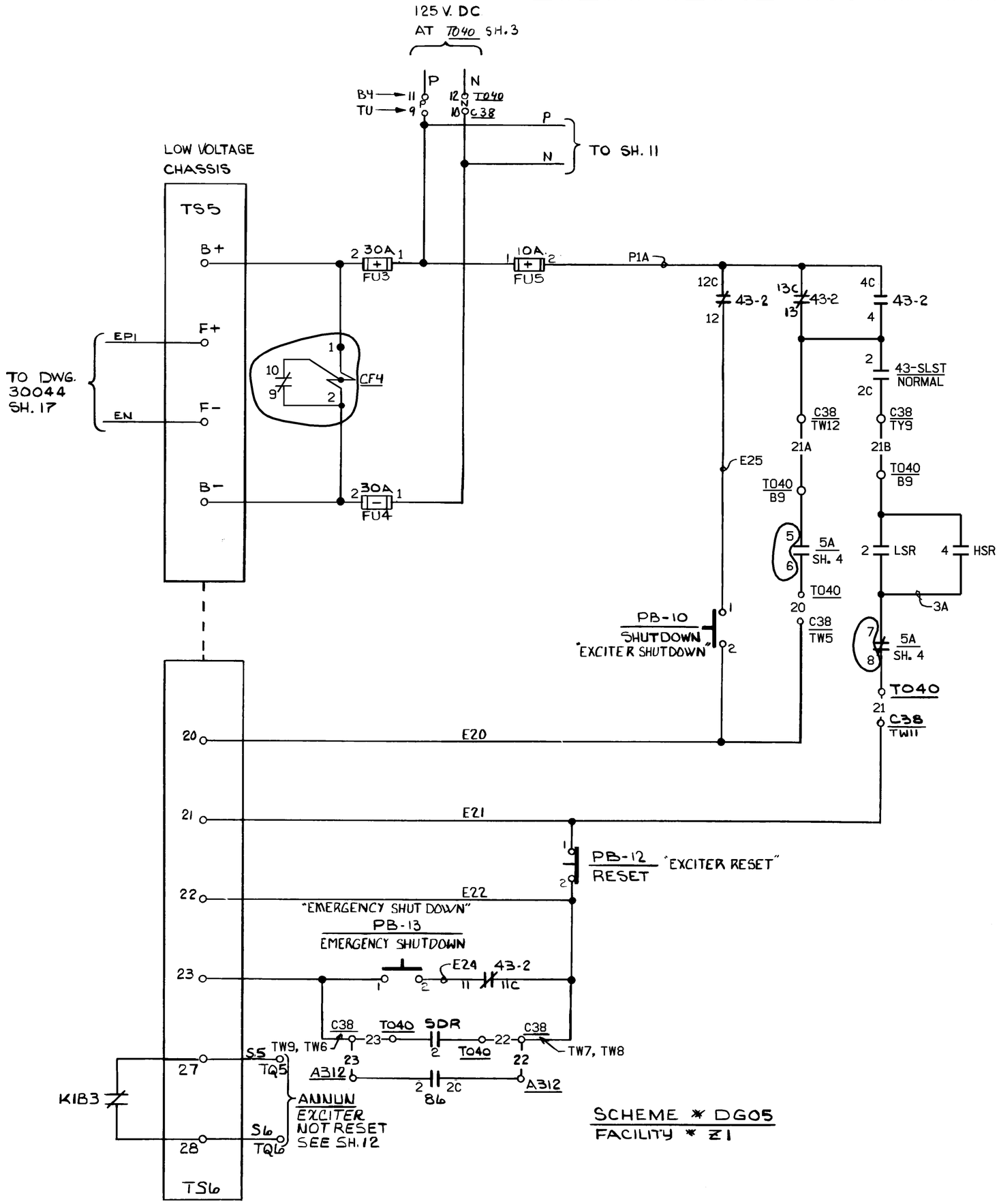
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion

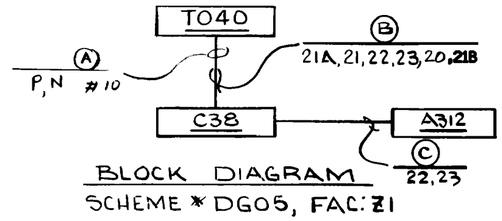
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
DIESEL GENERATOR (H7A) & (H7B) SWITCH DEVELOPMENT

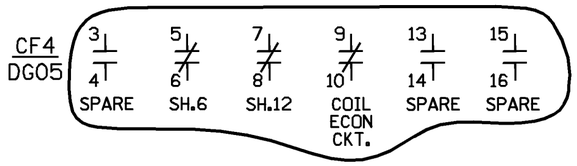
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C32041A.HYB	
REVISED PER MSDUR 13-000091 & DUR 13-800496	DRAWING NO: 25203-32041	REV 8
DSGN DLW	SCALE: NONE UNLESS OTHERWISE NOTED	SH A



SCHEME * DG05
FACILITY * Z1



BLOCK DIAGRAM
SCHEME * DG05, FAC: Z1



NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG. 25203-30028.
2. BASLER INTERNAL SCHEMATIC & INTERNAL CONNECTION DIAGRAM ON 39038 SH.11.

REF. DWG.

25203-39038 SH. 38A

FSAR FIGURE

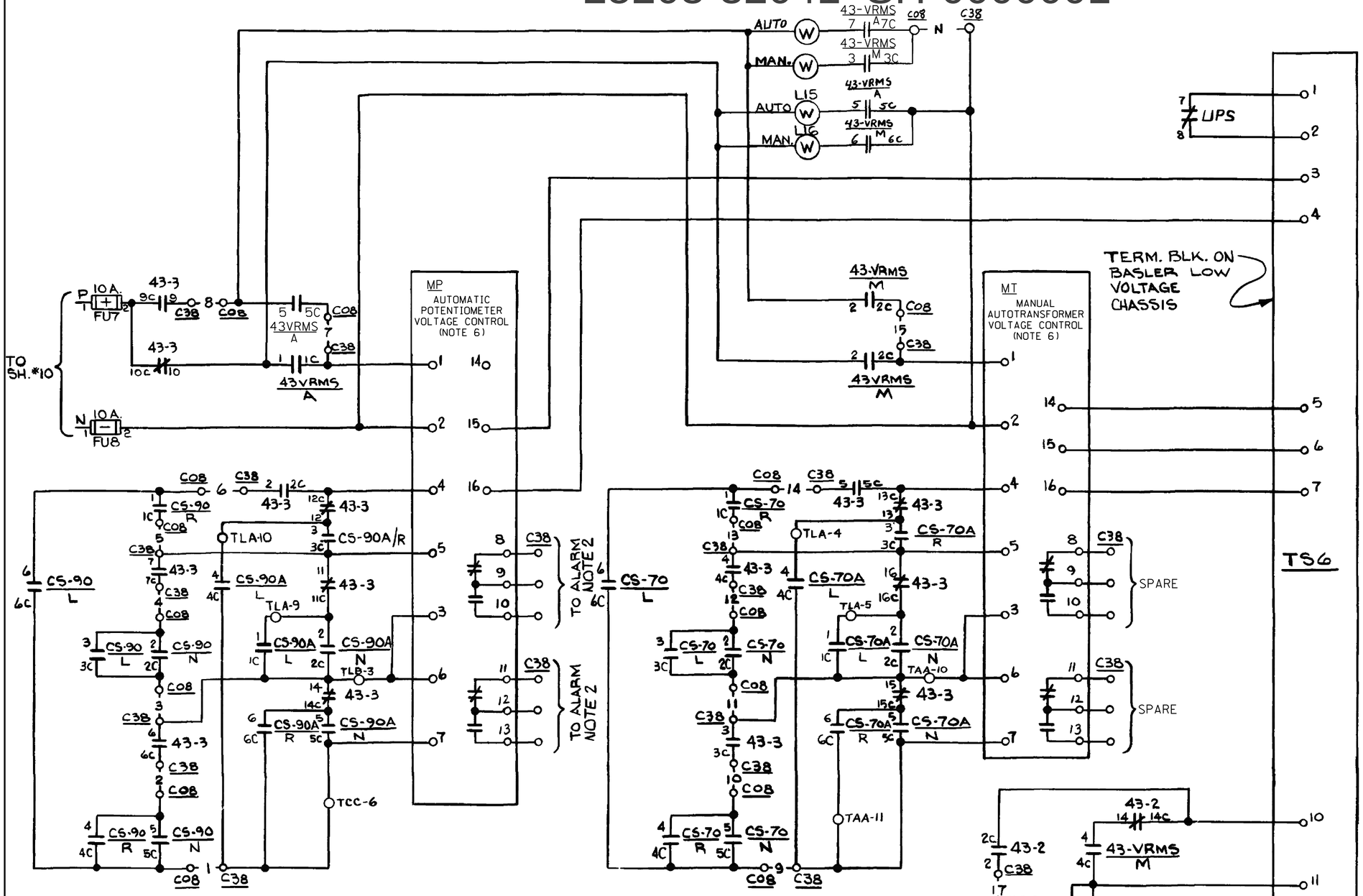
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

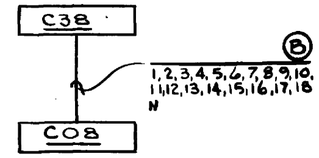
SUPERCEDES
25203-39038
SH. 7, 13

Dominion®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15-12U (H7A)
EXCITER CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204110.HYB	
REVISED PER DC MP2-12-01126 & DUR 13-800199	DRAWING NO: 25203-32041	REV 8
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 10



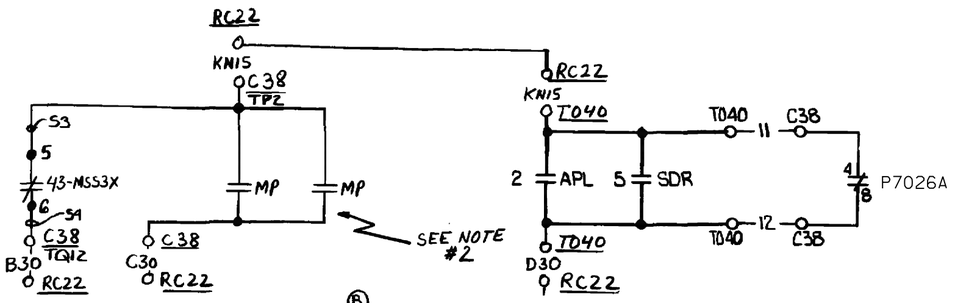


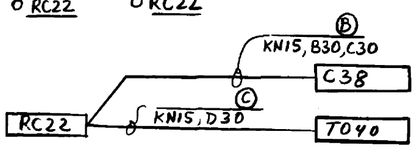
BLOCK DIAGRAM
SCHEME *DG06, FAC. Z1

NOTES:

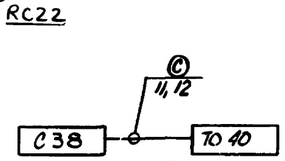
1. FOR STANDARD NOTES, SYMBOLS & EQUIP LOCATION INDEX SEE DWG. 25203-30028.
2. FIELD TO SET LIMIT SWITCH FOR MP SO THAT ANY SETTING ABOVE OR BELOW THE HIGH AND LOW SETPOINTS (HIGH SETPOINT 4310 $^{-30}_{+0}$ VOLTS; LOW SETPOINT 4050 $^{-0}_{+30}$ VOLTS) ON BUS #A3 SHALL ALARM.
3. FOR CONTROL SWITCH DEVELOPMENT SEE DWG. 32041 SH.A
4. 43-3 CONTACTS SHOWN IN THE D/G POSITION (LOCAL CONTROL)
5. REFERENCE LOOP DWG. 28500 SH. 759 AND SCHEMATIC DWG. 32041 SH. 13 FOR P7026A & PS7026A.
6. THE AUTOMATIC POTENTIOMETER (MP) AND MANUAL AUTOTRANSFORMER (MT) HAVE BEEN MODIFIED TO REMOVE THE INPUT DIODE BRIDGE.

SCHEME DG06
FACILITY * Z1





SCHEME NO. K08118
FAC. 1



SCHEME NO. DG078
FAC. 1 (NOTE 5)

THIS DWG. SUPERSEDES 25203-39038 SH. 8, 9

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
10	1-31-08	INCORP. DCN DM2-00-0008-07	JCV	DLW	-	ASK
9	10-4-04	INCORP. DCN DM2-00-0480-01, DM2-00-0556-02	DLW	WBZ	-	DED
8	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
7	8/4/99	INCORP. DCN DM2-00-2028-98	MEL	JSJ	JPP	JPP
6	8/8/98	INCORP. DCN DM2-01-0354-97	WBZ	MM	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
AUTO-MANUAL VOLTAGE CONTROL
WATERFORD, CT.

BY	CHKD.	APP.	APP.
WBZ/JSJ	—	—	—
DATE 7-2-98	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-32041 SH.11	
N/A			

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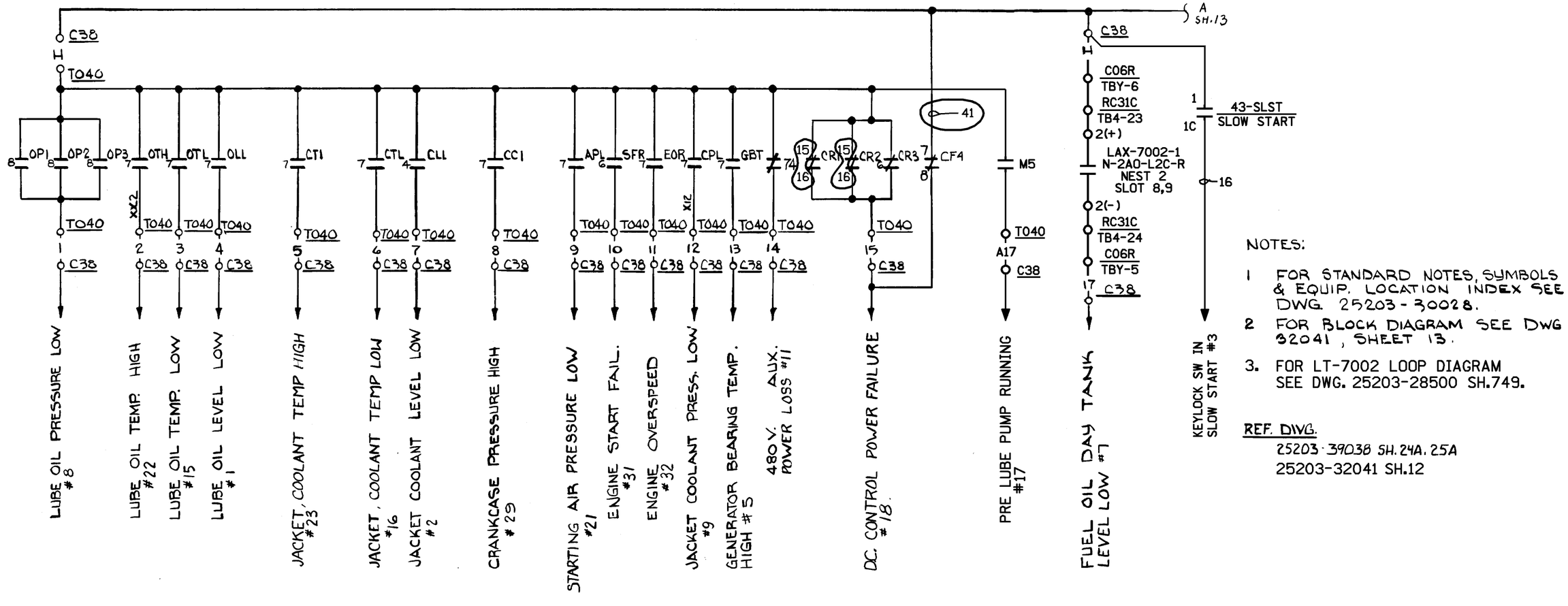
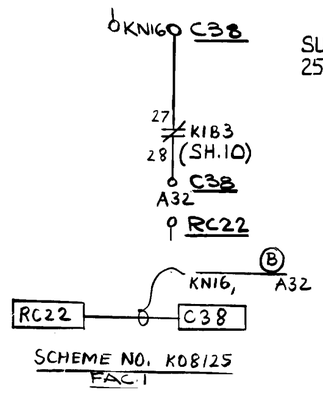


NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG. 25203-30028.
2. FOR BLOCK DIAGRAM SEE DWG 32041, SHEET 13.
3. FOR LT-7002 LOOP DIAGRAM SEE DWG. 25203-28500 SH.749.

REF. DWG.
25203-39038 SH. 24A, 25A
25203-32041 SH.12

SCHEME No DG07
FACILITY *1



SCHEME NO. KD8125
FAC.1

SUPERCEDES
25203-39038 SH.12

FSAR FIGURE

SAFETY EVALUATION
MAYBE REQUIRED
PRIOR TO CHANGES

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT No.2
DIESEL GENERATOR 15G-12U (H7A)
LOCAL ALARMS

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204112.HYB	
REVISED PER DC MP2-12-01126 & DUR 13-800199	DRAWING NO: 25203-32041	REV 7
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 12

SFILES
SCATES

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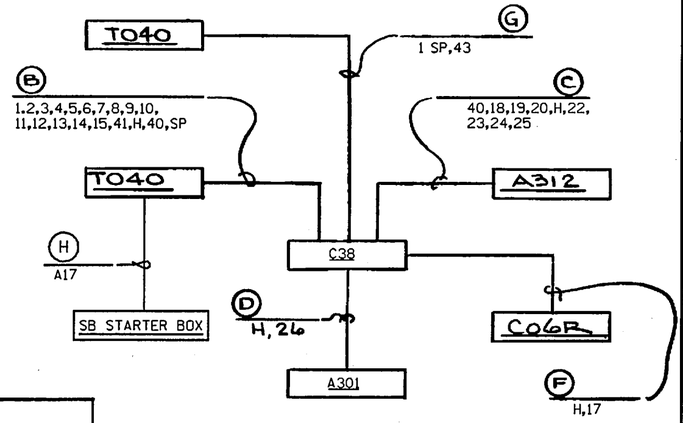
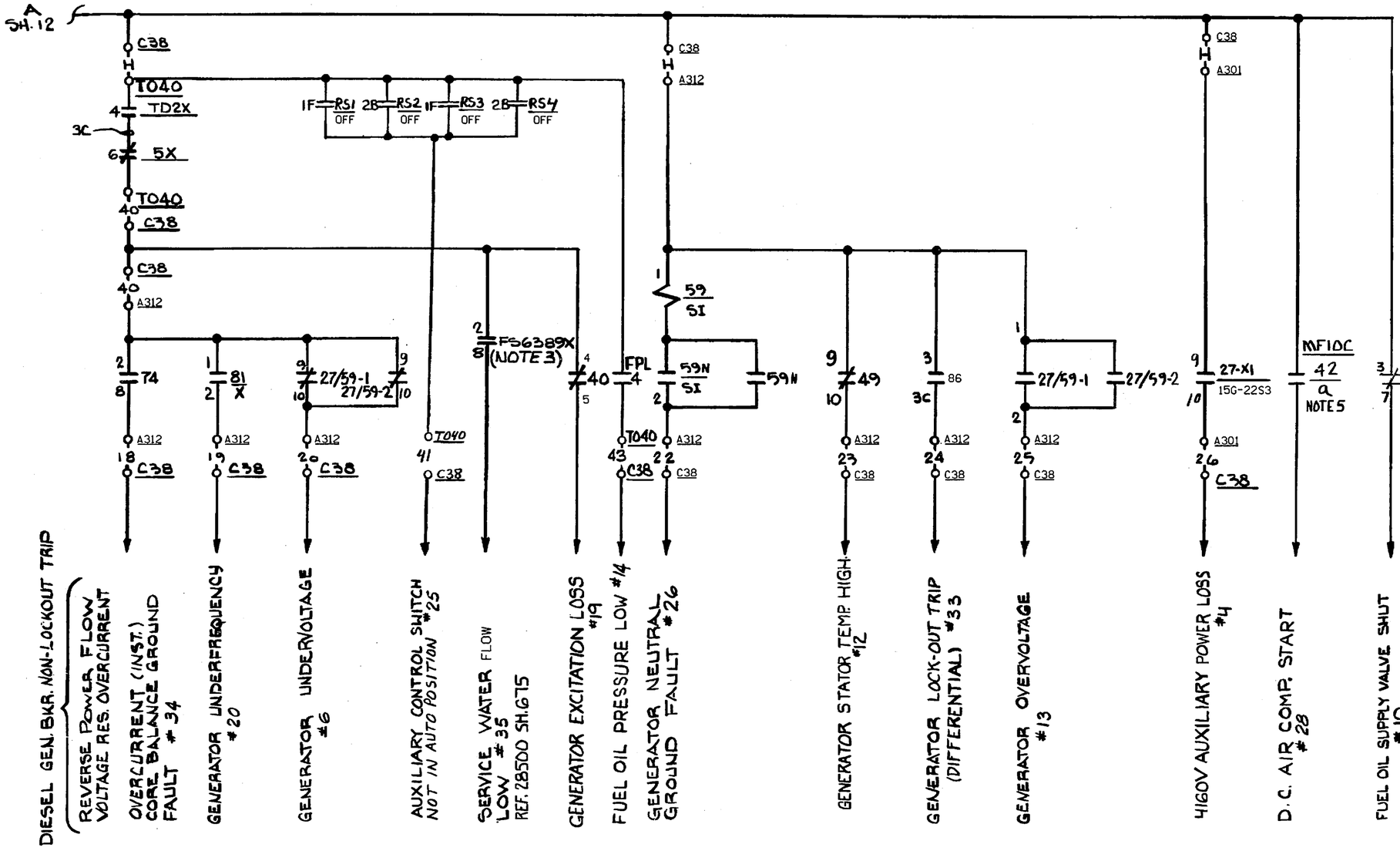


BLOCK DIAGRAM
SCHEME # DG07, FAC: 1

ENGINE	GAUGE BOARD	SWITCHES
RS1	COOLANT HEATER	AUTO / OFF
RS2	COOLANT PUMP	AUTO / OFF / MAN.
RS3	LUBE OIL HEATER	AUTO / OFF
RS4	LUBE OIL PUMP	AUTO / OFF / MAN.

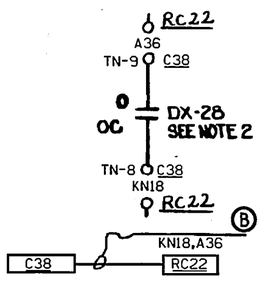


SCHEME Nº DG07
FACILITY Nº 1

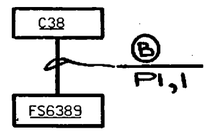
NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2. CONTACT LOCATED AT C38 REMOTE ALARM HORN, CLOSES ON ANY LOCAL ANNUNCIATION. SEE DWG. 39038 SH. 21A
3. REFERENCE DWG 28500 SH. 675 FOR FS6389X & FS6389 LOOP DIAGRAM.
4. REFERENCE LOOP DWG. 28500 SH. 759 AND SCHEMATIC DWG. 32041 SH. 11 FOR P7026A & PS7026A.
5. MF1DC 42/a CONTACT ON SCHEME Z1DG15 ON SH. 27

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES



BLOCK DIAGRAM
SCHEME KO8141
FAC. 1



SCHEME NO. DG07A
FAC. 1

SUPERSEDES
25203-39038 SH 12

NON Q.A.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

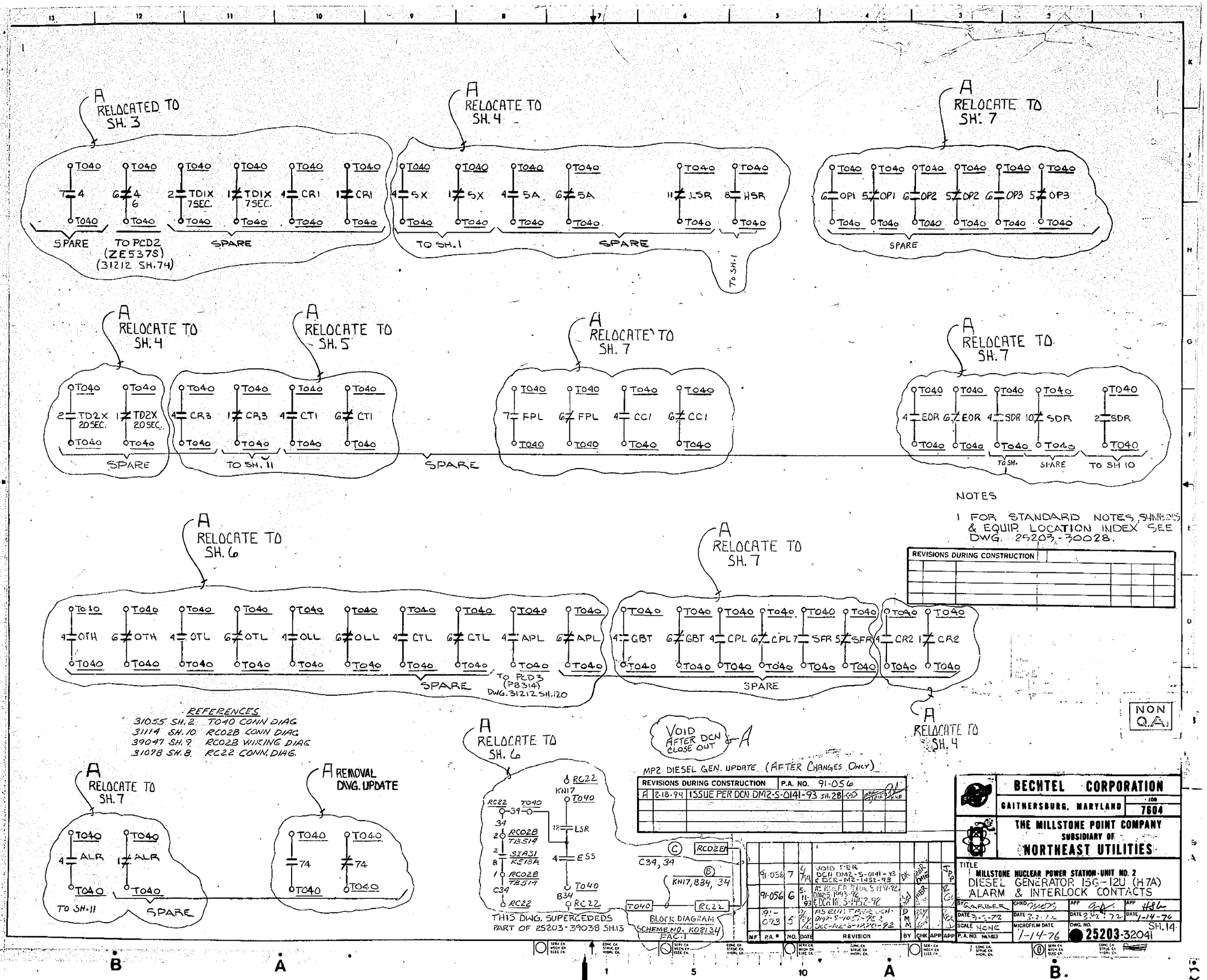
REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
10	5/15/00	INCORP. DCN DM2-00-0150-06	DLW	[illegible]	—	MH
9	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
8	8/4/99	INCORP. DCN DM2-00-2028-98	MEL	JSJ	JPP	JPP
7	8/8/98	INCORP. DCN DM2-00-0256-97	WBZ	MM	JPP	JPP
6	12-3-97	INCORP. DCN DM2-01-0490-97	AP	GFD	RWD	JB

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
LOCAL ALARMS
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ	—	—	—
DATE 4-10-96	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-32041 SH. 13	
N/A			



NOTES

1 FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG. 25203-30028.

REFERENCES

31055 SH. 2 TO40 CONN DIAG
31114 SH. 10 RC02B CONN DIAG
39047 SH. 9 RC02B WIRING DIAG
31078 SH. 8 RC22 CONN. DIAG.

REVISIONS DURING CONSTRUCTION						

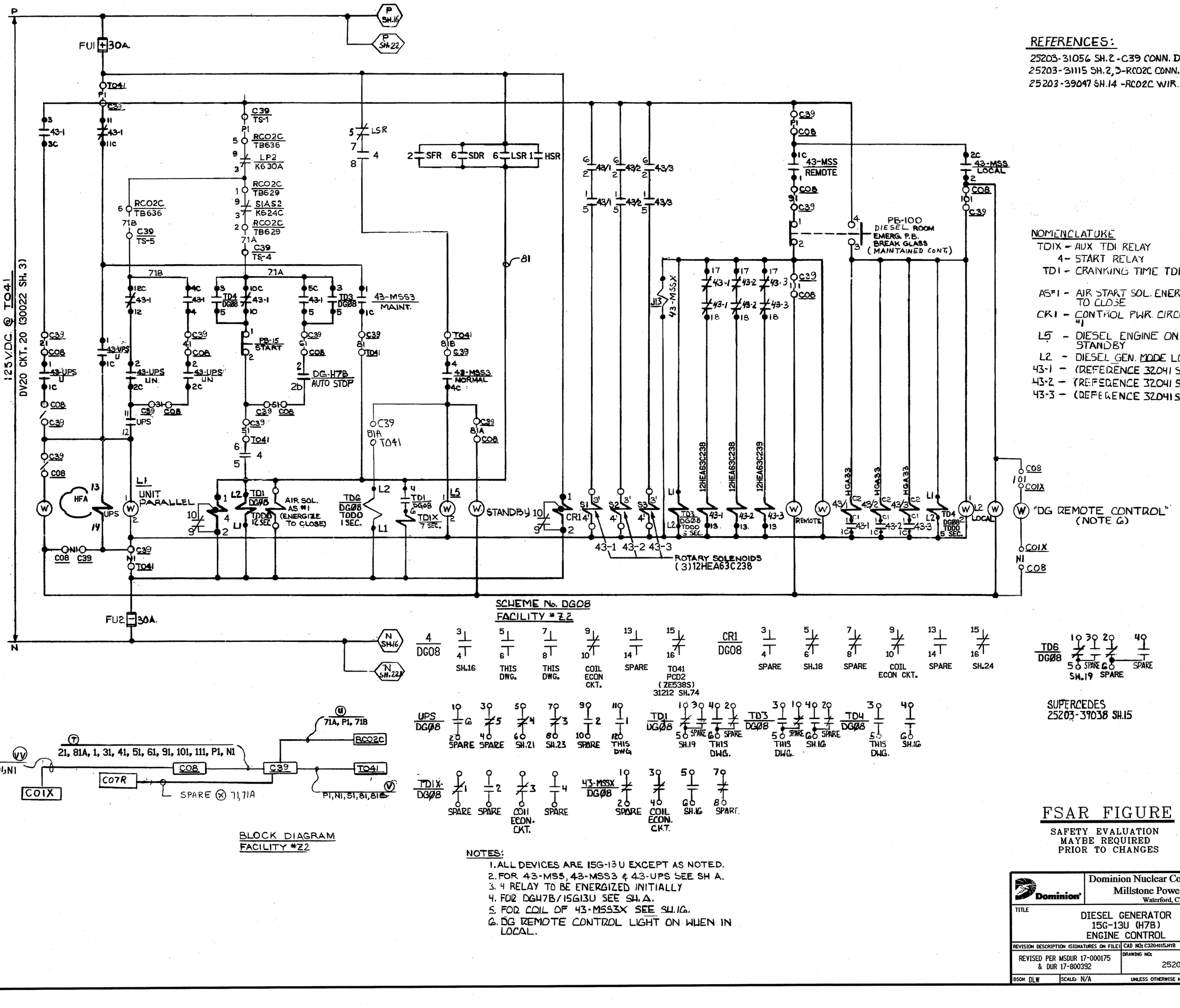
REVISIONS DURING CONSTRUCTION	P.A. NO. 91-056					
A	2-18-94	ISSUE PER DCN DM2-S-0141-93 SH.28				

MF	P.A. #	NO.	DATE	REVISION	BY	CHK	APP	APP
	91-056	7	4/94	VOID PER DCN DM2-S-0141-93 & DCR-M2-1452-93	DK	MBR		
	91-056	6	5-11-93	AS BUILT PER DCN S-1137-92, DM2-S-1093-92 & DCR M2-S-1452-92		MBR		
	91-073	5	3/24/92	AS BUILT PER DCN DM2-S-1051-92 & DCR-M2-S-1220-92	DMM			

BECHTEL CORPORATION
GAITHERSBURG, MARYLAND
JOB 7604
THE MILLSTONE POINT COMPANY
SUBSIDIARY OF
NORTHEAST UTILITIES

TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2
DIESEL GENERATOR 15G-12U (H7A)
ALARM & INTERLOCK CONTACTS

BY	CHKD	APP	APP
GARBER			
DATE 3-2-72	DATE 3-2-72	DATE 3-2-72	DATE 1-14-76
SCALE NONE	MICROFILM DATE 1-14-76	DWG. NO. 25203-32041	SH. 14
P.A. NO. NA1423			



REFERENCES:

25203-31056 SH.2 -C39 CONN. DIAG.
25203-31115 SH.2,3-RCO2C CONN. DIAG.
25203-39047 SH.14 -RCO2C WIR. DIAG.

NOMENCLATURE

TDIX – AUX TDI RELAY
4 – START RELAY
TDI – CRANKING TIME TDDO
AS#1 – AIR START SOL. ENERGIZE TO CLOSE
CR1 – CONTROL PWR. CIRCUIT #1
L5 – DIESEL ENGINE ON STANDBY
L2 – DIESEL GEN. MODE LOCAL
43-1 – (REFERENCE 32041 SH.20)
43-2 – (REFERENCE 32041 SH.20)
43-3 – (REFERENCE 32041 SH.20)

SCHEME No. DG08
FACILITY #Z2

4 DG08: 3 SH.16; 5 THIS DWG.; 7 THIS DWG.; 9 COIL ECON CKT.; 13 SPARE; 15 T041 PCD2 (ZE538S) 31212 SH.74

CR1 DG08: 3 SPARE; 5 SH.18; 7 SPARE; 9 COIL ECON CKT.; 13 SPARE; 15 SH.24

TD6 DG08: 10 30 20 40; 50 SPARE G0 SPARE; SH.19 SPARE

UPS DG08: 10 G 20 SPARE; 30 5 40 SPARE; 50 4 60 SH.21; 70 3 80 SH.23; 90 2 100 SPARE; 110 1 120 THIS DWG

TD1 DG08: 10 30; 50 SPARE SH.19; 40 20; G0 SPARE THIS DWG.

TD3 DG08: 30 10; 50 SPARE THIS DWG.; 40 20; G0 SPARE SH.16

TD4 DG08: 30; 50 THIS DWG.; 40; G0 SH.16

TDIX DG08: 1 SPARE; 2 SPARE; 3 COIL ECON. CKT.; 4 SPARE

43-MSSX DG08: 10 20 SPARE; 30 40 COIL ECON. CKT.; 50 G0 SH.16; 70 80 SPARE

SUPERCEDES
25203-39038 SH.15

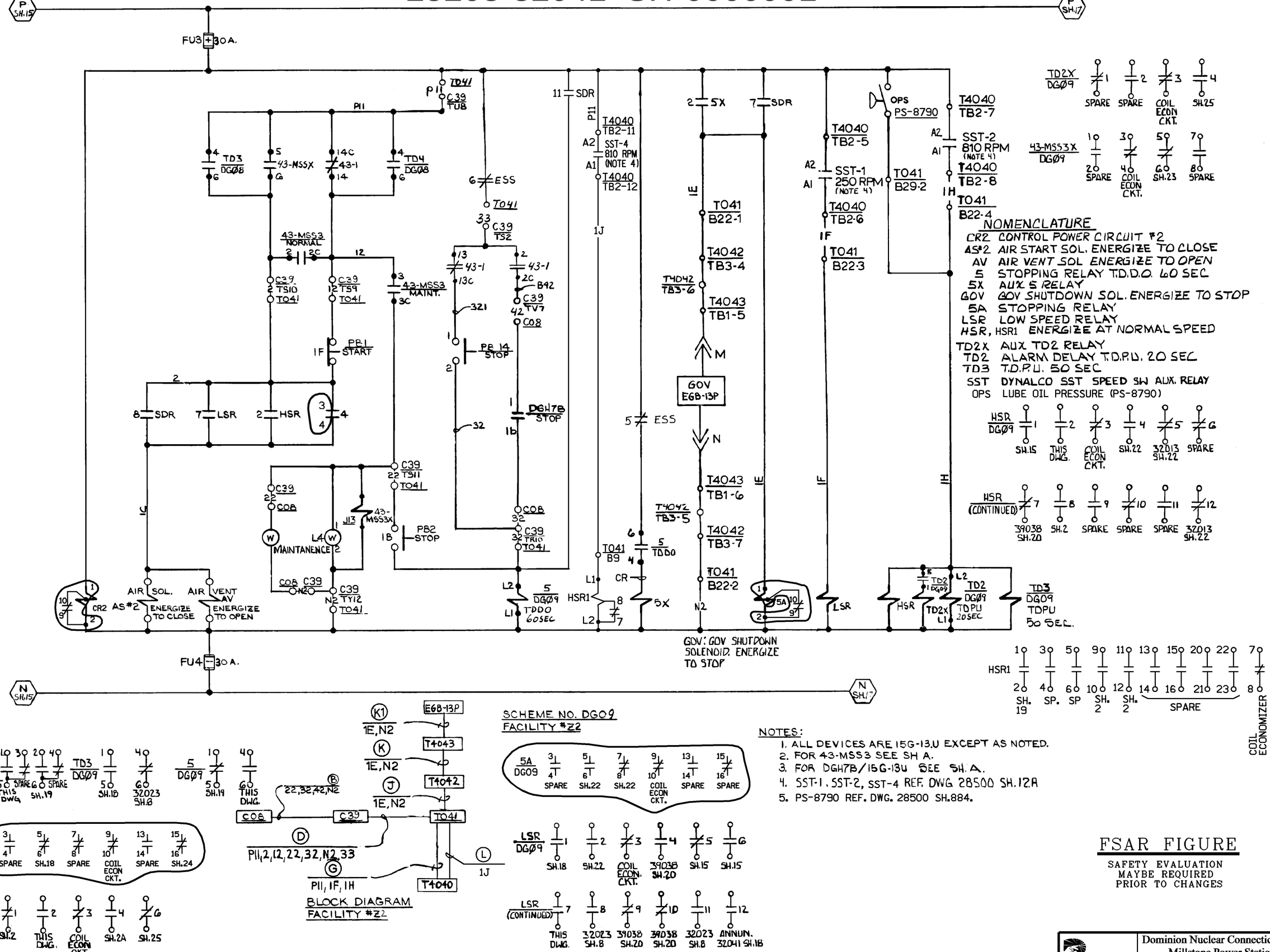
FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NOTES:

1. ALL DEVICES ARE 15G-13U EXCEPT AS NOTED.
2. FOR 43-MSS, 43-MSS3 & 43-UPS SEE SH A.
3. 4 RELAY TO BE ENERGIZED INITIALLY
4. FOR DG#7B/15G13U SEE SH. A.
5. FOR COIL OF 43-MSS3X SEE SH.16.
6. DG REMOTE CONTROL LIGHT ON WHEN IN LOCAL.

Dominion	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	DIESEL GENERATOR 15G-13U (H7B) ENGINE CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204115.HYB
REVISED PER MSDUR 17-000175 & DUR 17-800392	DRAWING NO: 25203-32041 REV 16
DSGN DLW	SCALE: N/A UNLESS OTHERWISE NOTED SH 15



NOTES:

1. ALL DEVICES ARE 15G-13.U EXCEPT AS NOTED.
2. FOR 43-MSS3 SEE SH A.
3. FOR DGH7B/15G-13U SEE SH. A.
4. SST-1, SST-2, SST-4 REF. DWG 28500 SH.12A
5. PS-8790 REF. DWG. 28500 SH.884.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES 25203-39033 SH.16,9

Dominion	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	DIESEL GENERATOR 15G-13U (H7B) ENGINE CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204116.HYB
REVISED PER DC MP2-13-01049 & DUR 14-800224	DRAWING NO: 25203-32041 REV 12
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED SH 16

ALARM RESET PB3

P SH.16

30A FU5

C SH.18

1 CT1

TS-8774 (CTH1)

2 CT1

2 CC1

2 FPL

P111X SH.18

B SH.18

P111 SH.18

P111 TO41 LS8770 LS8770 LS8770 112 TO41

1 CLL

1K

112

142 SH.18

CLL

N3 SH.18

TD7 DG10 TDPU 1 SEC.

CR3

CT1

30A FU6

N SH.15

JACKET COOLANT TEMP. HIGH

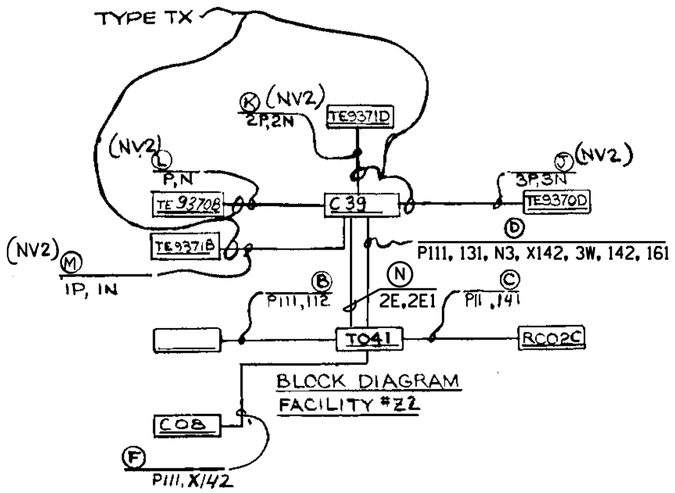
CRANKCASE PRESSURE HIGH

FUEL OIL PRESS. LOW

JACKET COOLANT LEVEL LOW

SCHEME NO. DG10
FACILITY #22

TYPE TX



BLOCK DIAGRAM
FACILITY #22

NOMENCLATURE

CR3 CONTROL POWER CIRCUIT #3
TD7 TDPU 1 SEC LUBE OIL PRES. LOW COMMON ALARM
CT1 JACKET COOLANT TEMP HIGH
CLL JACKET COOLANT LEVEL LOW

TD7 DG10: 1 (5 SH.19), 2 (6 SH.19)

CR3 DG10: 1 SP., 2 SP., 3 COIL ECON. CKT., 4 SP., 6 SH.24

CLL DG10: 1 THIS DWG., 2 SP., 3 COIL ECON. CKT., 4 SH.24

CT1 DG10: 1 THIS DWG., 2 THIS DWG., 3 COIL ECON. CKT., 4 SP., 6 SP., 7 SH.24

REDRAWN FROM SAME NNECO DWG. No. REV.3

SUPERCEDES PART OF 25203-39038 SH.16

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
ENGINE CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204117.HYB	
REVISED PER DC MP2-19-01208 & DUR 21-800137	DRAWING NO: 25203-32041	REV 7
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 17

MBULOGIN\jame694
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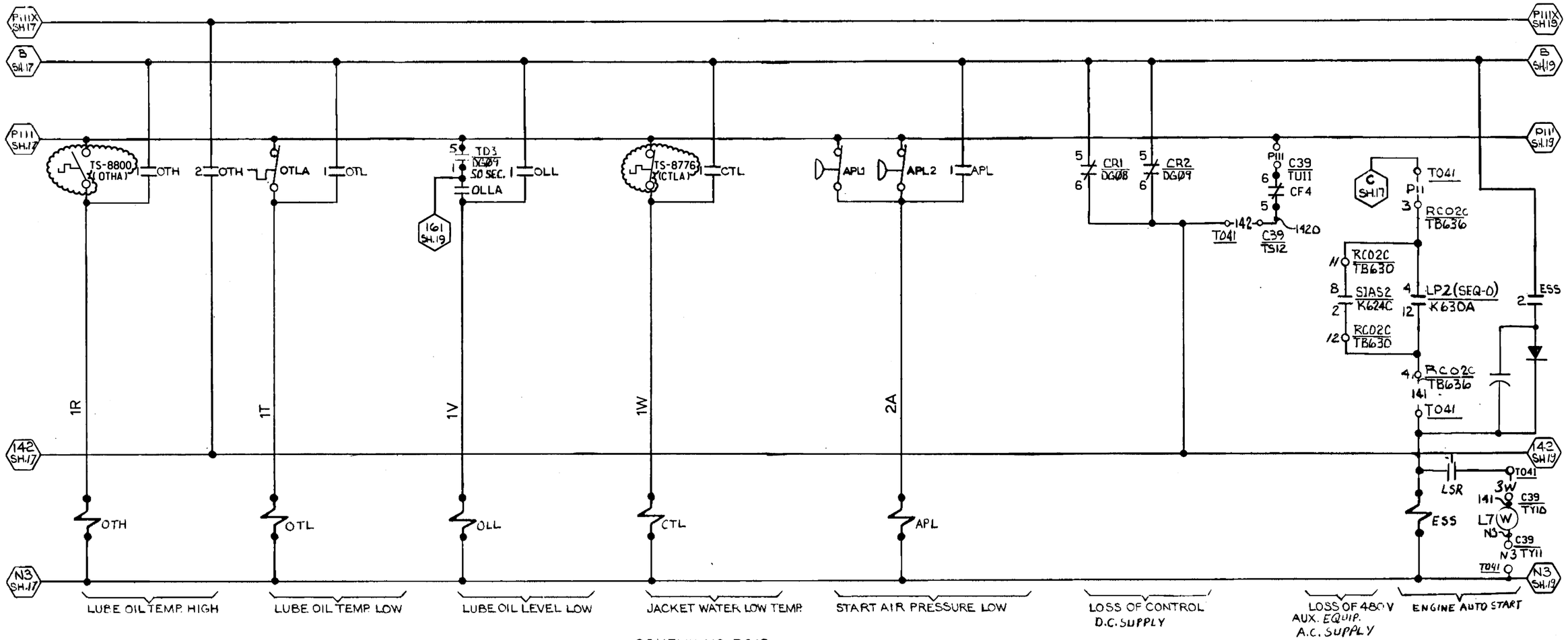
C3204117.HYB



SCHEME NO. DG10
FACILITY #Z2

OTH: 1 THIS DWG. | 2 THIS DWG. | 3 COIL ECON CKT. | 4 SPARE | 6 SPARE | 7 SH.24

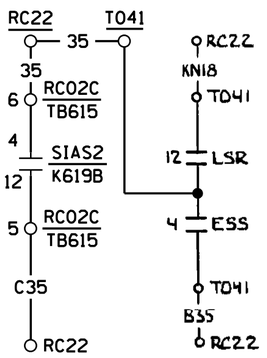
OTL: 1 THIS DWG. | 2 SPARE | 3 COIL ECON CKT. | 4 SPARE | 6 SPARE | 7 SH.24

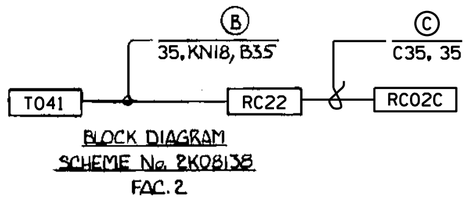
OLL: 1 THIS DWG. | 2 SPARE | 3 COIL ECON CKT. | 4 SPARE | 6 SPARE | 7 SH.24

CTL: 1 THIS DWG. | 2 SPARE | 3 COIL ECON CKT. | 4 SPARE | 6 SPARE | 7 SH.24

APL: 1 THIS DWG. | 2 SH.23 | 3 COIL ECON CKT. | 4 PCD3 (PB818) 31212 SH.120 | 6 SPARE | 7 SH.24

ESS: 1 SH.19 | 2 THIS DWG. | 3 COIL ECON CKT. | 4 ANNUN. THIS DWG. | 5 SH.16 | 6 SH.16 | 7 SH.21 | 8 SPARE





BLOCK DIAGRAM
SCHEME No. 2K08138
FAC. 2

T041 1 ESS T041 TO SH.19
T041 2 ESS T041 THIS SH.
T041 4 ESS T041 TO SH.26
T041 5 ESS T041 TO SH.16
T041 6 ESS T041 TO SH.16
T041 7 ESS T041 TO SH.21
T041 8 ESS T041 SPARE

REFERENCES

31115 SH 2,3	RC02C CONN DIAG
39047 SH 13	RC02C WIRING DIAG
31055 SH. 2	T041
31078 SH. 8	RC22

NOTE:
1. FOR BLOCK DIAGRAM SEE SH. 17.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES PART OF 25203-39038 SH.16, 17.

REDRAWN FROM SAME NNECO DWG. No. REV.4

Dominion Energy®
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
ENGINE CONTROL

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204118.HYB	
REVISED PER DC MP2-19-01208 & DUR 21-800137	DRAWING NO: 25203-32041	REV 10
DSGN JCV	SCALE: NONE UNLESS OTHERWISE NOTED	SH 18

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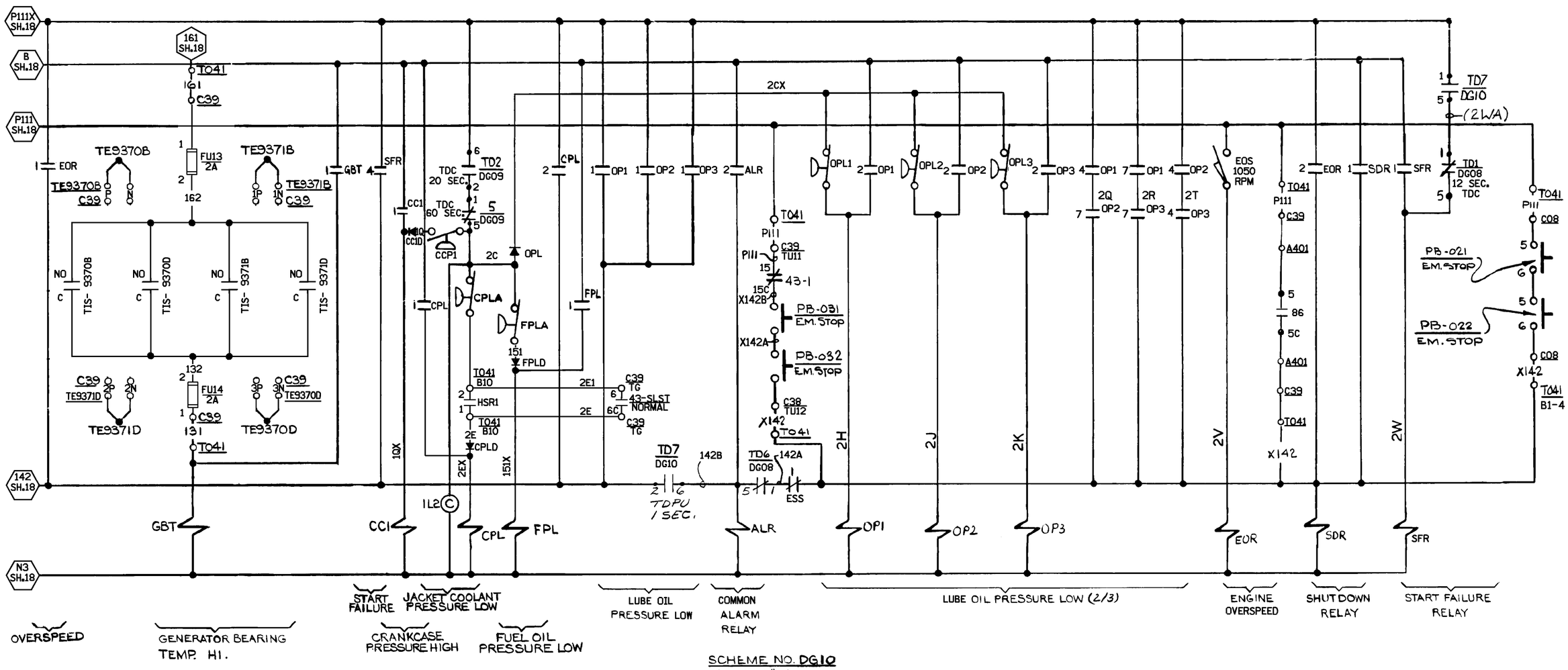
C3204118.HYB



SCHEME NO. DG10
FACILITY #Z2

GBT
1 THIS DWG. | 2 SP. | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH.24

CC1
1 THIS DWG. | 2 SH.17 | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH.24

CPL
1 THIS DWG. | 2 THIS DWG. | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH.24

FPL
1 THIS DWG. | 2 SH.17 | 3 COIL ECON. CKT. | 4 SH.25 | 6 SP. | 7 SP.

ALR
1 SP. | 2 THIS DWG. | 3 COIL ECON. CKT. | 4 SP.

OP1
1 THIS DWG. | 2 THIS DWG. | 3 COIL ECON. CKT. | 4 THIS DWG. | 5 SP. | 6 SP. | 7 THIS DWG. | 8 SH.24

OP2
1 THIS DWG. | 2 THIS DWG. | 3 COIL ECON. CKT. | 4 THIS DWG. | 5 SP. | 6 SP. | 7 THIS DWG. | 8 SH.24

OP3
1 THIS DWG. | 2 THIS DWG. | 3 COIL ECON. CKT. | 4 THIS DWG. | 5 SP. | 6 SP. | 7 THIS DWG. | 8 SH.24

EOR
1 THIS DWG. | 2 THIS DWG | 3 COIL ECON. CKT. | 4 SP. | 6 SP. | 7 SH.24

SDR
1 THIS DWG. | 2 SH.22 | 3 COIL ECON. CKT. | 4 SH.2A | 5 SH.23 | 6 SH.15 | 7 SH.16 | 8 SH.16 | 10 SP. | 11 SH.16

SFR
1 THIS DWG. | 2 SH.15 | 3 COIL ECON. CKT. | 4 THIS DWG. | 5 SP. | 6 SH.24 | 7 SP.

NOTES:
1. FOR BLOCK DIAGRAM SEE SH. 17

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
10	4-13-04	INCORP DCN DM2-00-0479-99	DLW	WBZ	-	DED
9	2-22-02	INCORP'D DCNS DM2-01-1827-98, DM2-01,02-0085-01	MJS	REE	RSTG	MAG
8	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
7	8/8/98	INCORP. DCN DM2-00-0120-97	WBZ	MM	JPP	JPP
6	12-3-97	INCORP. DCN(S) DM2-00-0356-97 & DM2-01-0356-97	AP	GFD	RWD	JB

SUPERSEDES 25203-39038 SH.17,19,9

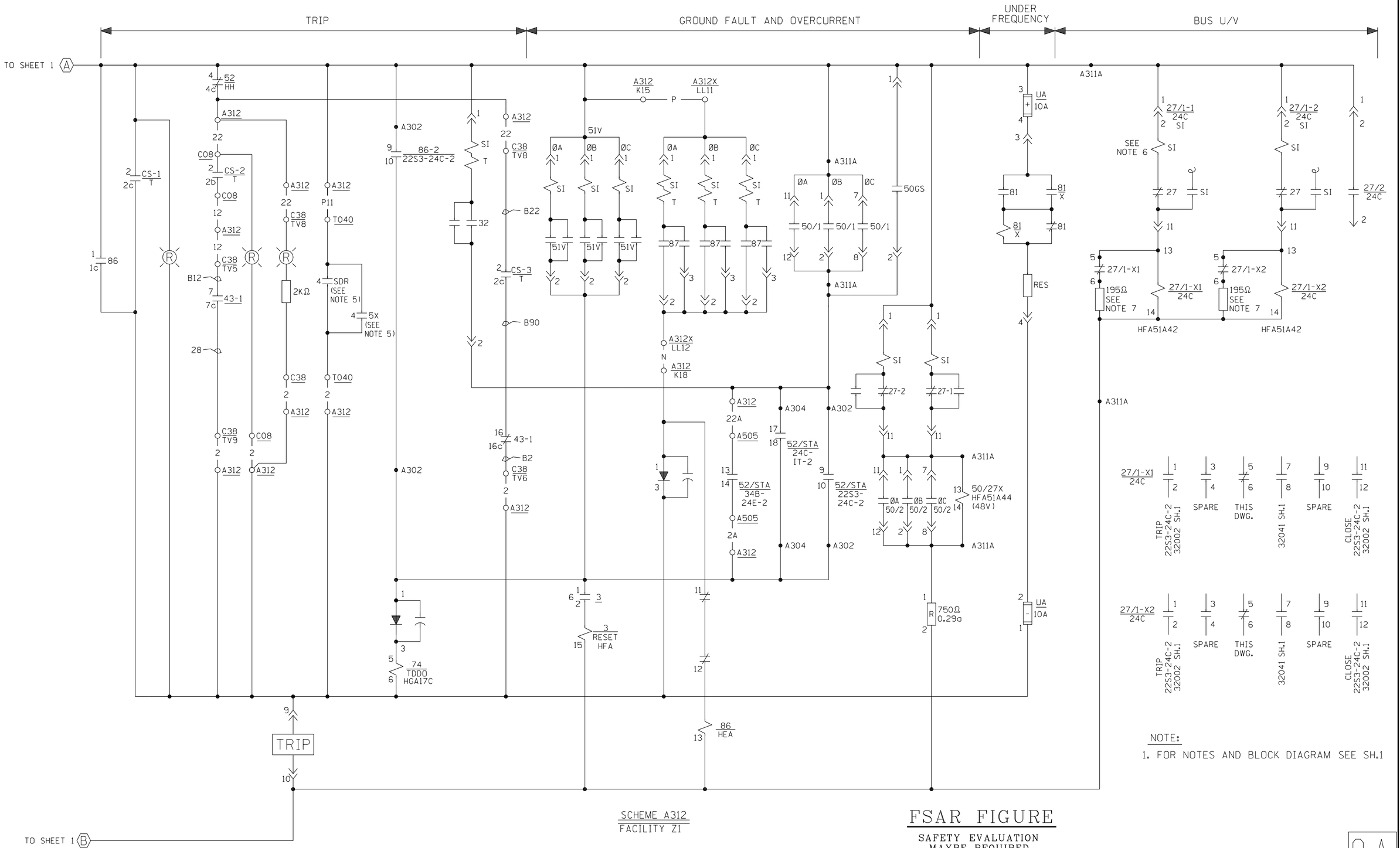
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE: MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
ENGINE CONTROLS
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ	—	—	—
DATE 8-21-95	DATE —	DATE —	DATE —
SCALE N/A	DWG. NO. 25203-32041 SH. 19		
N/A			

$FILE$
$DATE$

25203-32041 SH-0000001



NOTE:

1. FOR NOTES AND BLOCK DIAGRAM SEE SH.1

REFERENCE	
25203-30034 SH.1	LOGIC DIAGRAM
25203-30071	ARRGT. C01X
25203-31063 SH. 1	CONN. DIAG. C08
25203-31073 SH. 1	CONN. DIAG. C01X
25203-32043 SH. 2	SCHEM. DIAG. C01X
25203-39030 SHTS. 7A, 7B	WIRING DIAG. C01X
25203-39045 SH. 8	WIRING DIAG. C08

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
8	9-29-08	INCORP. DCN DM2-00-0235-08	JCV	MEL	-	DLW
7	10-22-01	INCORP'D DCN DM2-00-0865-99	TAM	RNB	MAG	GMO
6	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT 2
4.16 KV DIESEL GEN. FEEDER BKR. 15G-12U-2 (H7A)(A312)
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ	MEL	RJH	GJF
DATE 4/9/92	DATE 10/8/92	DATE 10/8/92	DATE 10/8/92
SCALE N/A	DWG. NO. 25203-32041 SH. 1A		
91-073			

\\10.108.35.47\DATA\MILLSTONE\4\DATA1\MPCAD\MPDesign\USERS\michi599\DGN\MP\25203\PROJECT\30000\P320411A.DGN 09/29/2008

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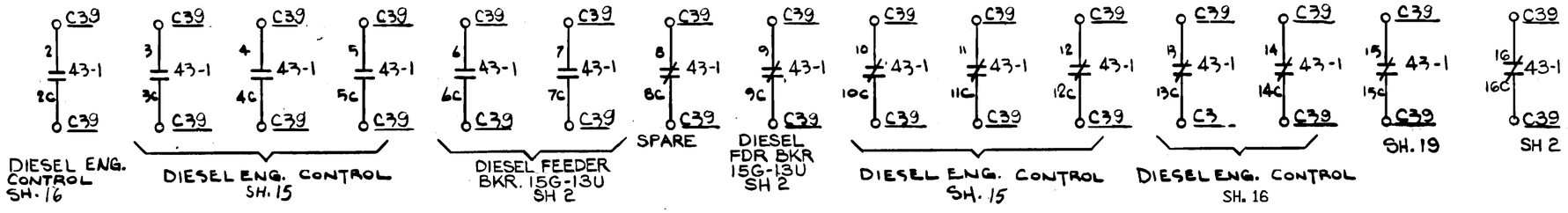
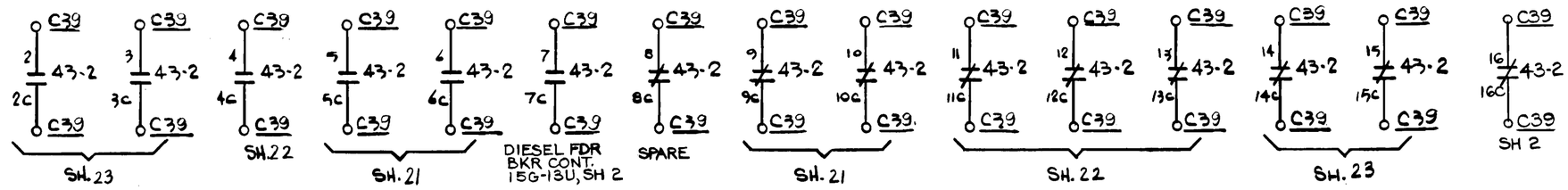
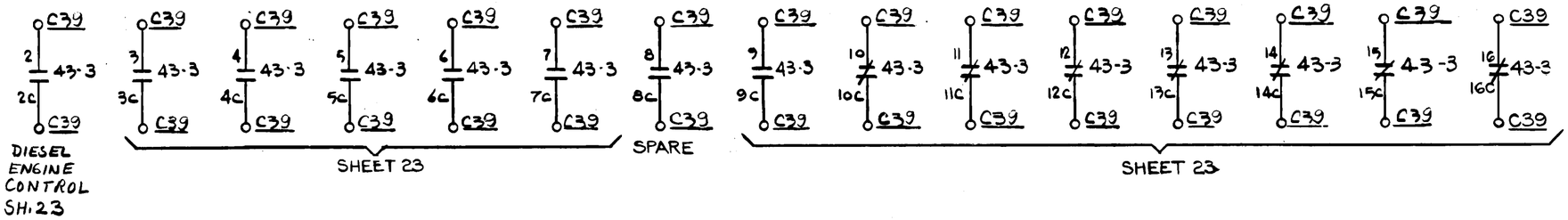




NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30020.
2. ALL RELAY CONTACTS SHOWN IN RESET POSITION (LOCAL)



FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

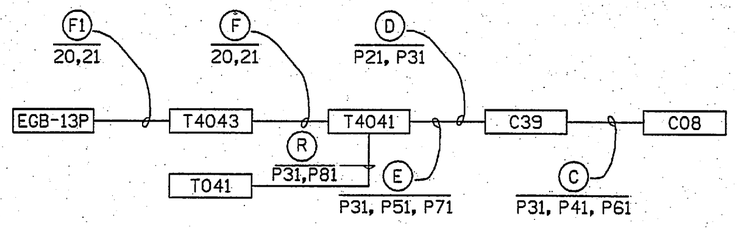
	NO.	DATE	REVISIONS	BY	CHK	APP	APP
	3	4-13-04	INCORP. DCN DM2-00-0479-99	DLW	WBZ	-	DED
	2	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
91-056	1	1-31-96	INCORP. DCN DM2-S-1994-92, 0501-93, DCR M2-S-0422-95	JSJ	MBR	JPP	JPP

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

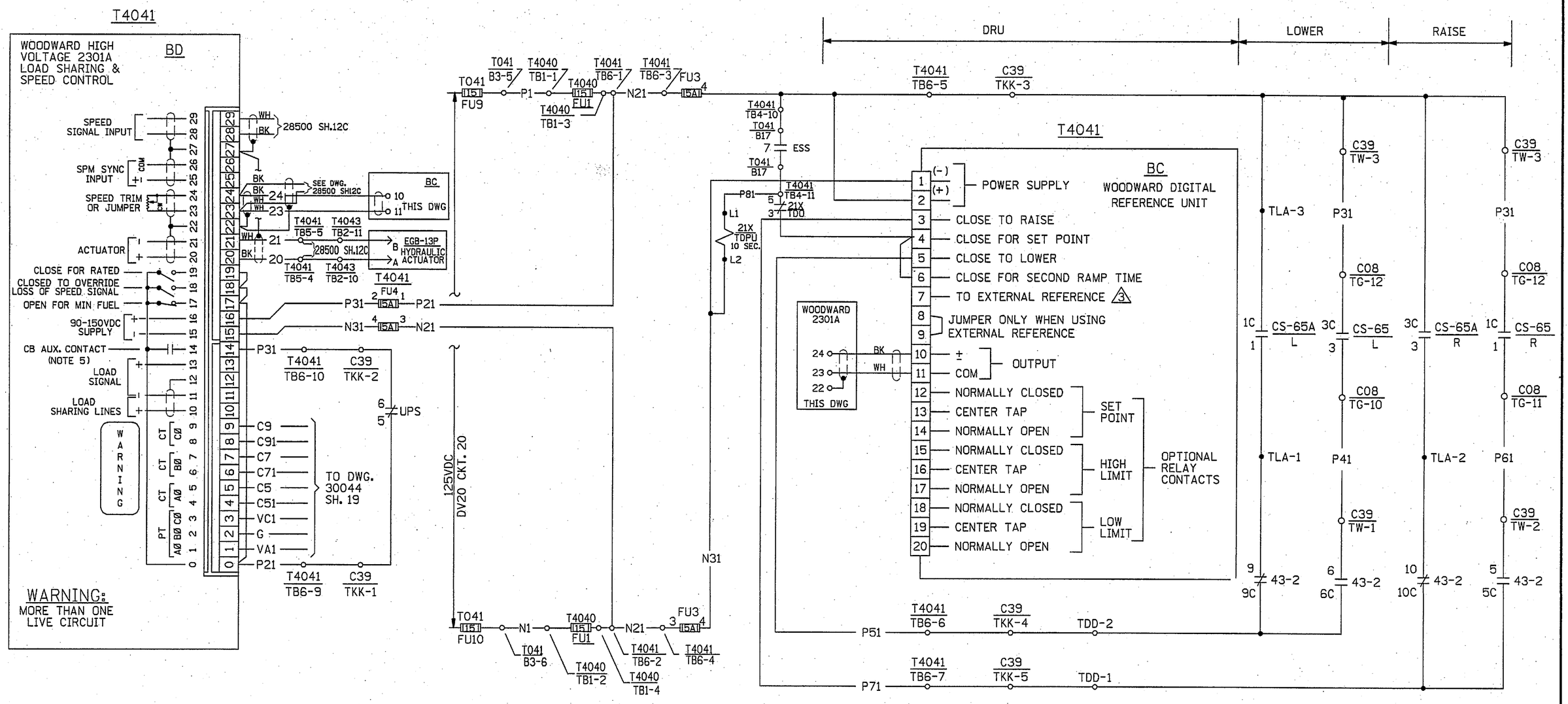
TITLE: MILLSTONE UNIT No.2
DIESEL GENERATOR 15G-13U (H7B)
ENGINE CONTROL RELAYS
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ			
DATE 8-21-95	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-32041 SH.20	
N/A			



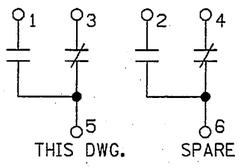
BLOCK DIAGRAM

SCHEME: DG11
FAC: Z2



WARNING:
MORE THAN ONE LIVE CIRCUIT

21X
TDPU SET @ 10 SEC.



SCHEME. NO. DG11
FACILITY #Z2

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

THIS DWG. SUPERCEDES PART OF 25203-39038 SH. 19

REDRAWN FROM SAME NNECO DWG. NO. REV. 2

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

Q.A.

NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX, SEE DWG. 25203-30028.
2. FOR CONTROL SWITCH DEVELOPEMENT, SEE DWG. 32041 SH. A.
3. FOR RELAY UPS DWG. 32041 SH.15.
4. 43-2 CONTACTS SHOWN IN THE D/G POSITION (LOCAL CONTROL).
5. OPEN FOR DROOP CLOSE FOR ISOCHRONOUS
6. FOR RELAY 43-2 SEE DWG. 32041 SH.15.

REVISIONS DURING CONSTRUCTION	P.A.#

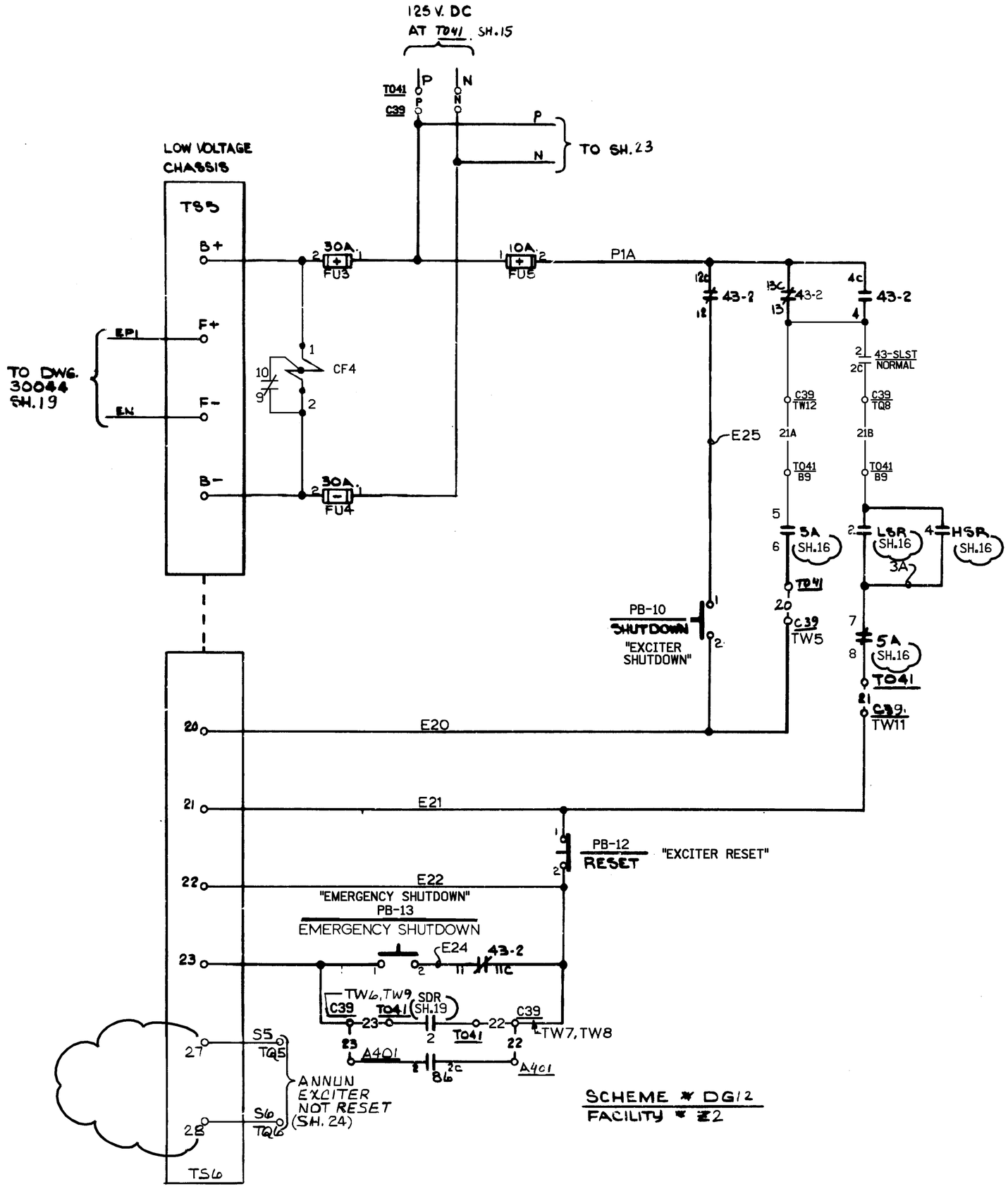
MF	P.A.#	NO.	DATE	REVISIONS	BY	CHK	APP	APP
		4	3/10	INCORP. DCN DM2-00-0025-00	JSJ			
	91-056	3	2-20-96	INCORP. DCN(S) DM2-S-0501-93, DM2-S-1994-92 PER DCR M2-S-0792-92	JMM	MBR	JPP	JPP

Northeast Utilities System
FOR NORTHEAST NUCLEAR ENERGY CO.

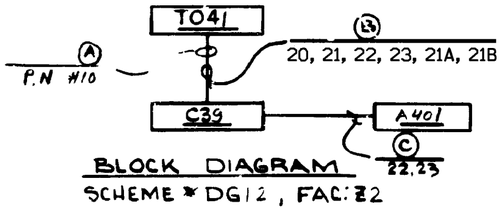
TITLE
MILLSTONE UNIT NO. 2
DIESEL GENERATOR 15G-13U (H7B)
GOVENOR CONTROL
WATERFORD, CT.

BY	CHKD.	APP.	APP.
DMM	RJH	KWF	KWF
DATE 5/18/92	DATE 7-7-92	DATE 7-6-92	DATE 7-6-92

SCALE N/A
P.A.# 91-056
DWG. NO. 25203-32041 SH. 21



SCHEME * DG12
FACILITY * 2



BLOCK DIAGRAM
SCHEME * DG12, FAC: 2

CF4 / DG12

3	5	7	9	13	15
4 SPARE	6 SH.18	8 SH.24	10 COIL ECON CKT.	14 SPARE	16 SPARE

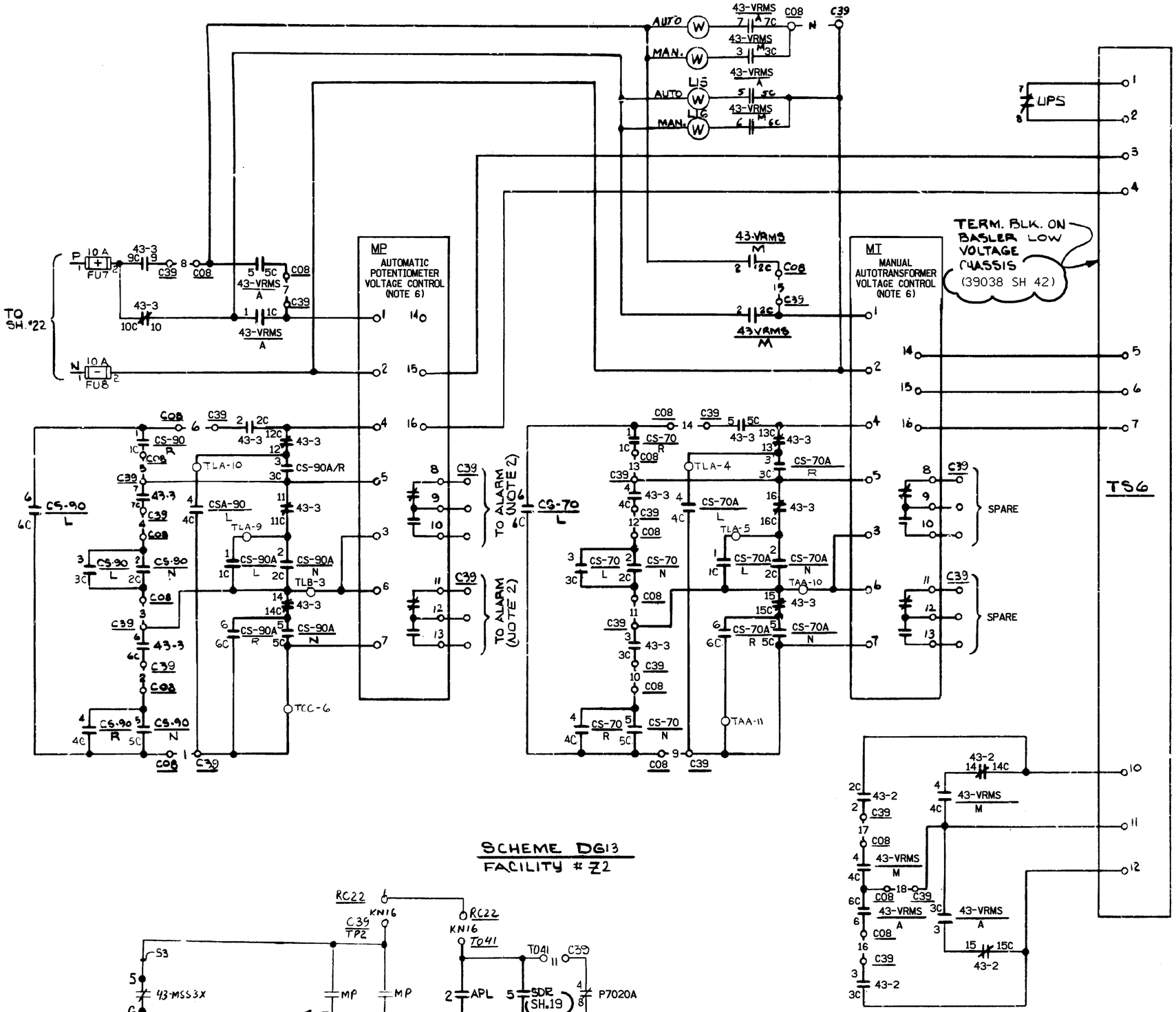
NOTES:

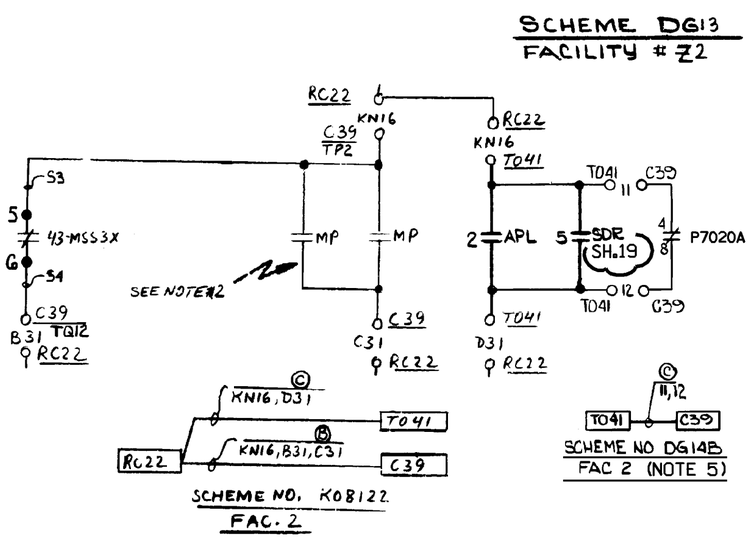
1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG 25203-30028.
2. FOR SCHEMATIC AND INTERCONNECTION DIAGRAM FOR SERIES BOOST EXCITER VOLTAGE REGULATOR SEE DWG. 25203-39038 SH 42.

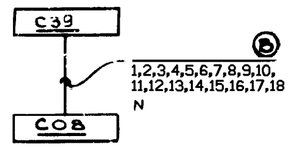
FSAR FIGURE

SUPERCEDES
25203-39038 SH.7,19

Dominion	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE UNIT 2
	DIESEL GENERATOR 15G-13U (H7B) EXCITER CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204122.HYB
REVISED PER MSDUR 17-000093 & DUR 17-800200	DRAWING NO: 25203-32041 REV 10
DSGN DLW	SCALE: N/A UNLESS OTHERWISE NOTED SH 22







BLOCK DIAGRAM
SCHEME #DG13, FAC. Z2

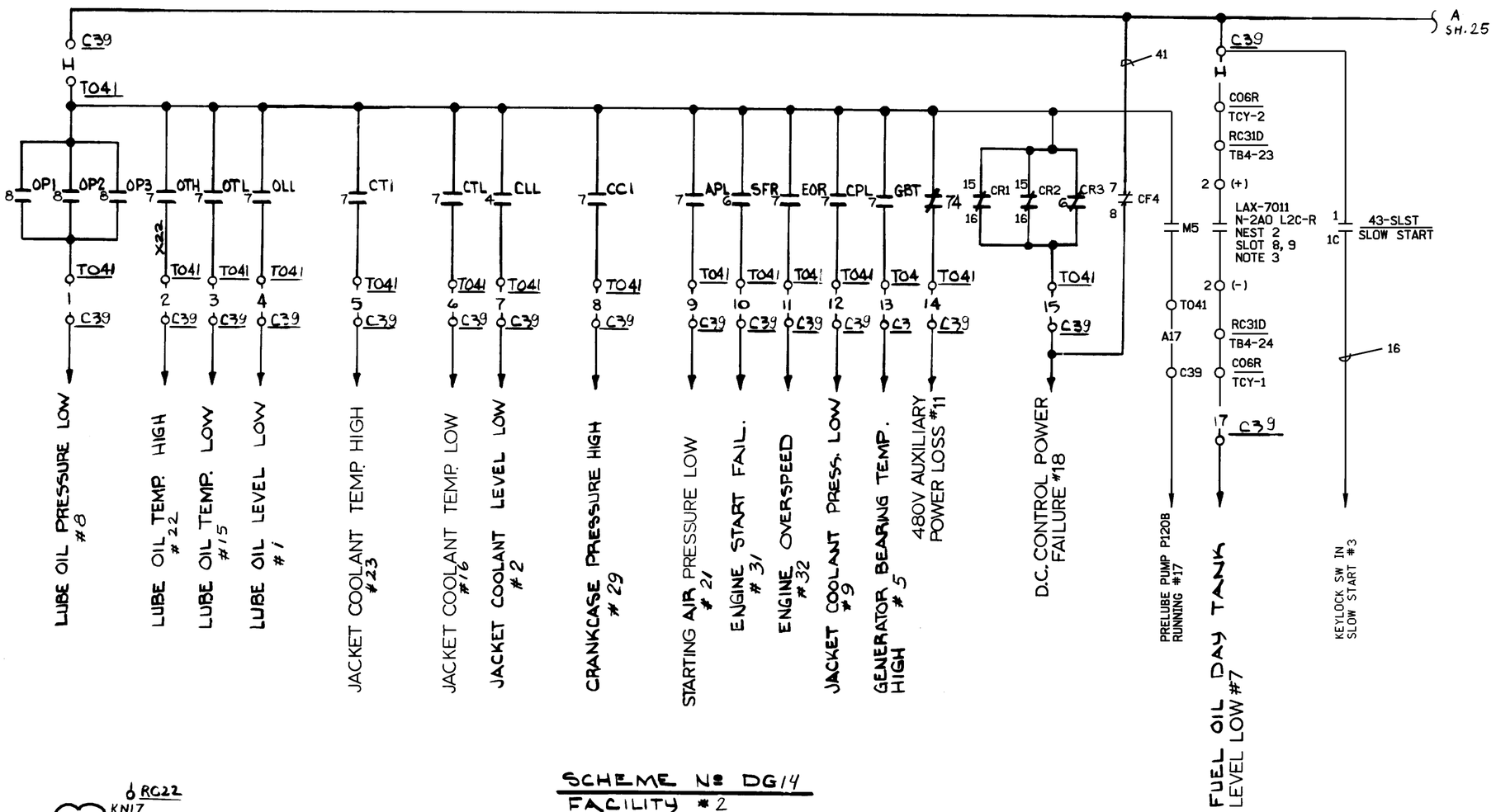
NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP LOCATION INDEX SEE DWG. 25203-30028.
2. FIELD TO SET LIMIT SWITCH FOR MP SO THAT ANY SETTING ABOVE OR BELOW THE HIGH AND LOW SETPOINTS (HIGH SETPOINT 4310 $^{+30}_{+0}$ VOLTS; LOW SETPOINT 4050 $^{+0}_{+30}$ VOLTS) ON BUS #A4 SHALL ALARM.
3. FOR CONTROL SWITCH DEVELOPMENT SEE DWG. 32041 SH. A
4. 43-3 CONTACTS SHOWN IN THE D/G POSITION. (LOCAL CONTROL)
5. REFERENCE LOOP DWG. 28500 SH. 757 AND SCHEMATIC DWG. 32041 SH. 25 FOR P7020A AND PS7020A.
6. THE AUTOMATIC POTENTIOMETER (MP) AND MANUAL AUTOTRANSFORMER (MT) HAVE BEEN MODIFIED TO REMOVE THE INPUT DIODE BRIDGE.
7. FOR SCHEMATIC AND INTERCONNECTION DIAGRAM FOR SERIES BOOST EXCITER VOLTAGE REGULATOR SEE DWG. 25203-39038 SH 42.

FSAR FIGURE

THIS DWG. SUPERCEDES 25203-39038 SH. 8, 9

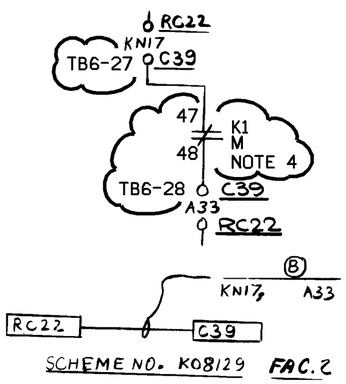
Dominion	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	DIESEL GENERATOR 15G-13U (H7B) AUTO-MANUAL VOLTAGE CONTROL
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204123.HYB
REVISED PER MSDUR 17-000093 & DUR 17-800200	DRAWING NO: 25203-32041 REV 12
DSGN DLW	SCALE: N/A UNLESS OTHERWISE NOTED SH 23



NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG. 25203-30028.
2. FOR BLOCK DIAGRAM SEE DWG 32041, SHEET 25
3. REFERENCE DWG 28500 SH. 754 #755 FOR LT-7011 LOOP DIAGRAM.
4. REFERENCE DWG. 25203-39038 SH 42.

SCHEME No DG14
FACILITY #2



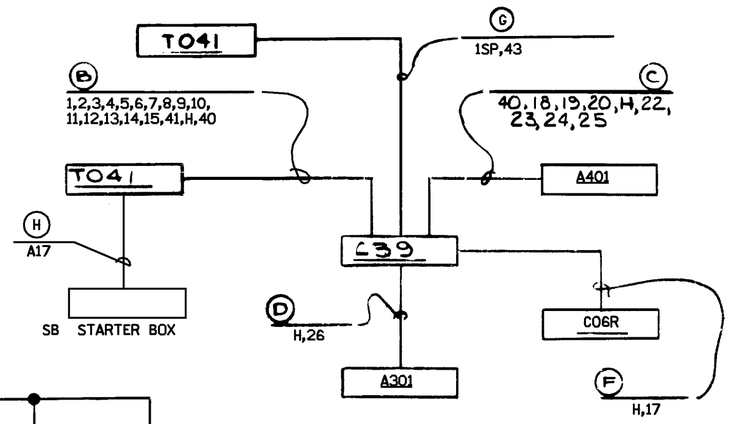
SCHEME NO. K08129 FAC. 2

FSAR FIGURE

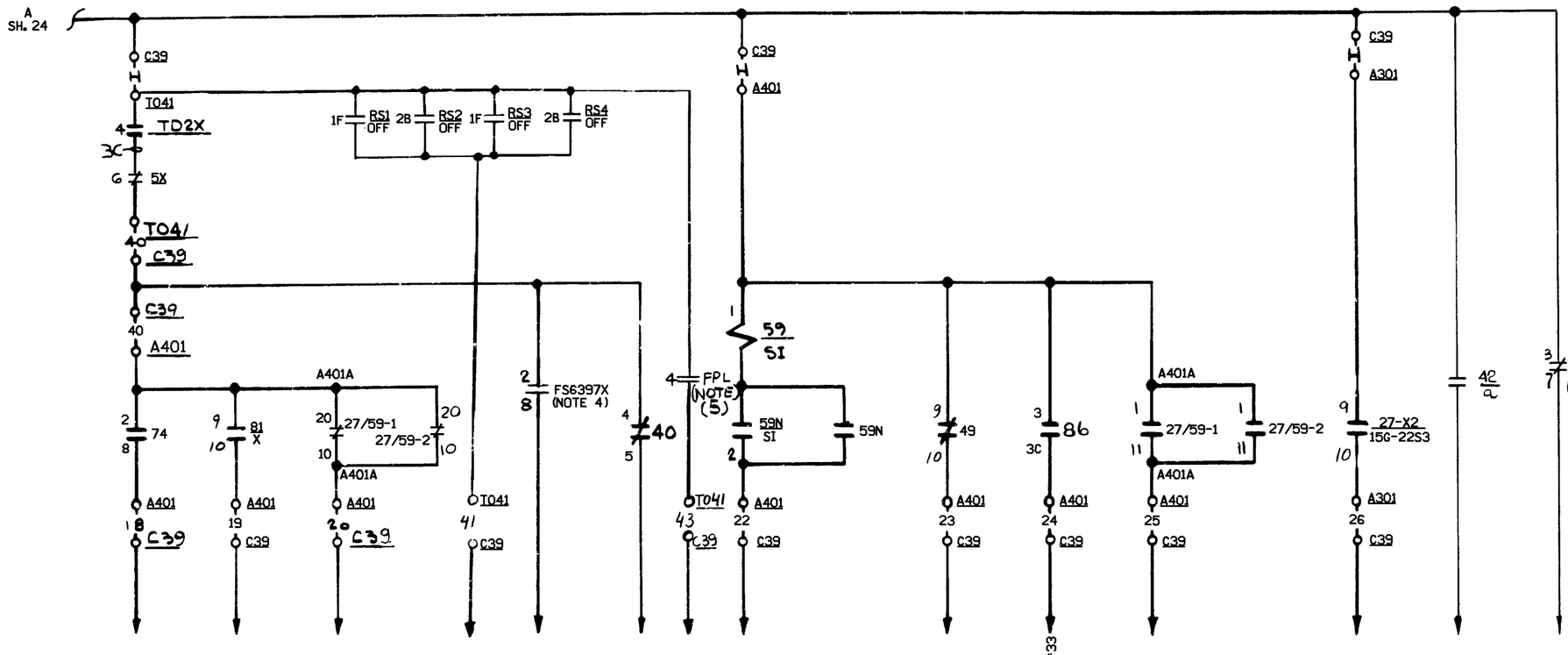
SUPERCEDES PART OF 25203-39038 SH.18

Dominion	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.	
TITLE	MILLSTONE UNIT 2 DIESEL GENERATOR 15G-13U (H7B) LOCAL ALARMS	
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C3204124.HYB	
REVISED PER MSDUR 17-000093 & DUR 17-800200	DRAWING NO: 25203-32041	REV 7
DSGN DLW	SCALE: N/A UNLESS OTHERWISE NOTED	SH 24

$FILE$
$DATE$



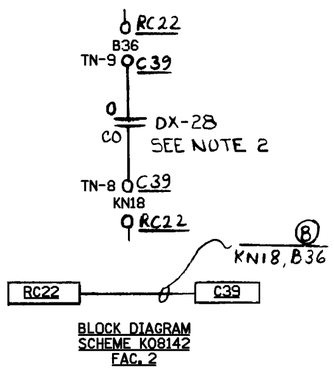
BLOCK DIAGRAM
SCHEME# DG14, FAC.2



ENGINE	GAUGE BOARD	SWITCHES
RS1	COOLANT HEATER	AUTO/OFF
RS2	COOLANT PUMP	AUTO/OFF/MANUAL
RS3	LUBE OIL HEATER	AUTO/OFF
RS4	LUBE OIL PUMP	AUTO/OFF/MANUAL

DIESEL GEN. BKR. NON-LOCKOUT TRIP
REVERSE POWER FLOW
VOLTAGE RES. OVERCURRENT
OVERCURRENT (INST.)
CORE BALANCE GROUND
FAULT #34
GENERATOR UNDERFREQUENCY #20
GENERATOR UNDERVOLTAGE #6
AUXILIARY CONTROL SWITCH NOT IN AUTO POSITION #25
SERVICE WATER FLOW LOW #35
GENERATOR EXCITATION LOSS #19
FUEL OIL PRESS. LOW #14
GENERATOR NEUTRAL GROUND FAULT #26
GENERATOR STATOR TEMP HIGH #12
GENERATOR LOCKOUT TRIP (DIFFERENTIAL)#33
GENERATOR OVERVOLTAGE #13
4160V AUXILIARY POWER LOSS #4
D.C. AIR COMP. START #28
FUEL OIL SUPPLY VALVE SHUT #10

SCHEME NO. DG14
FACILITY NO. 2



BLOCK DIAGRAM
SCHEME KQ8142
FAC. 2

NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2. CONTACT LOCATED AT C39 REMOTE ALARM HORN CLOSES ON ANY LOCAL ANNUNCIATION, SEE DWG. 39038 SH. 21B.
3. REFERENCE LOOP DWG. 28500 SH. 757 AND SCHEMATIC DWG. 32041 SH. 23 FOR P7020A AND PS7020A.
4. REFERENCE DWG. 28500 SH. 677 FOR FS-6397 SERVICE WTR FLOW LOOP DIAGRAM.
5. REFERENCE DWG. 28500 SH. 756 FOR FPL (FUEL OIL PRESSURE LOW) LOOP DIAGRAM.

FSAR FIGURE
SAFETY EVALUATION
MAYBE REQUIRED
PRIOR TO CHANGES

SUPERSEDES PART OF
25203-39038 SH. 18

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

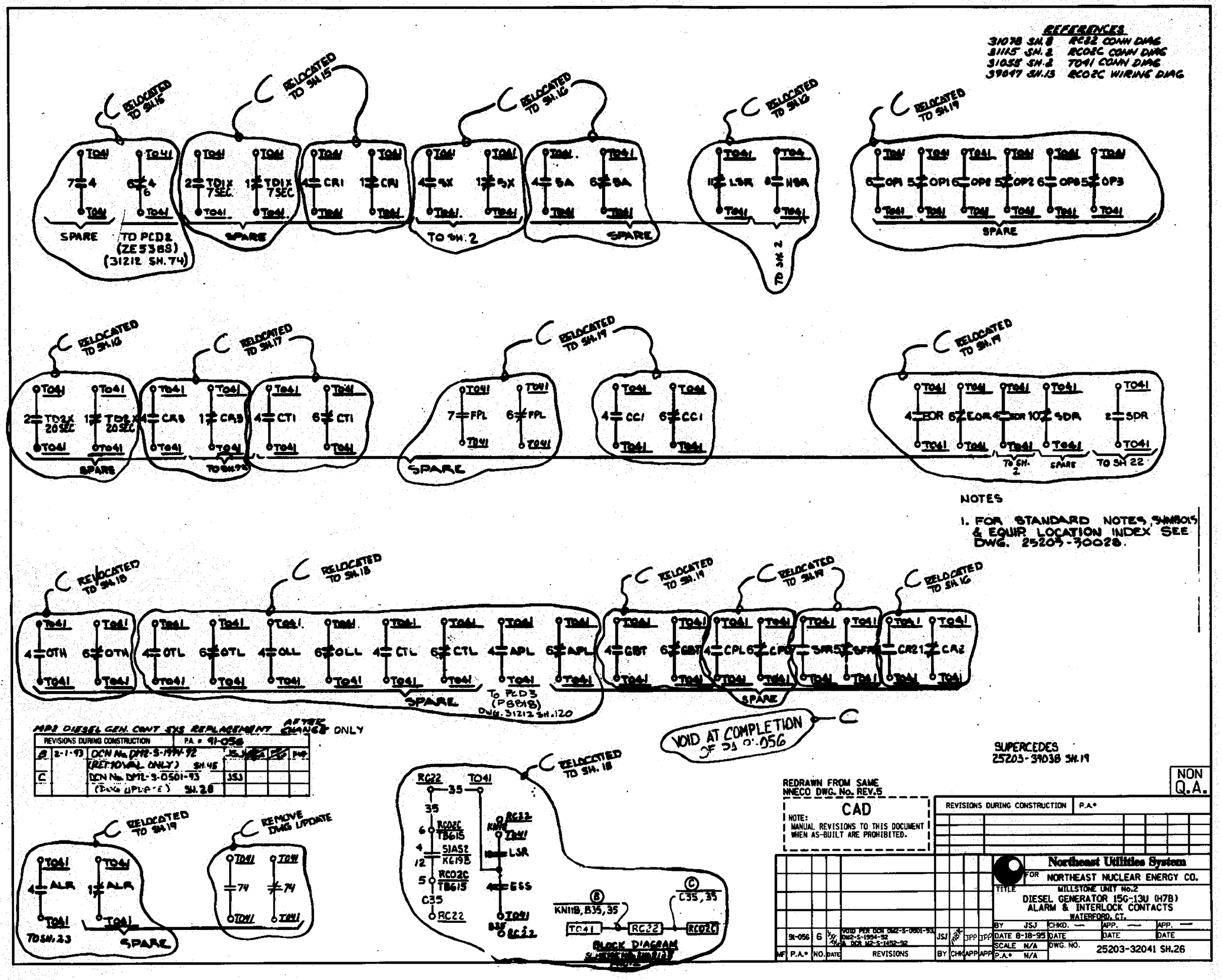
REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
7	5-25-06	INCORP. DCN DM2-00-0150-06	DLW	WBZ	-	ASK
6	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
5	8/4/99	INCORP. DCN DM2-00-2028-98	MEL	JSJ	JPP	JPP

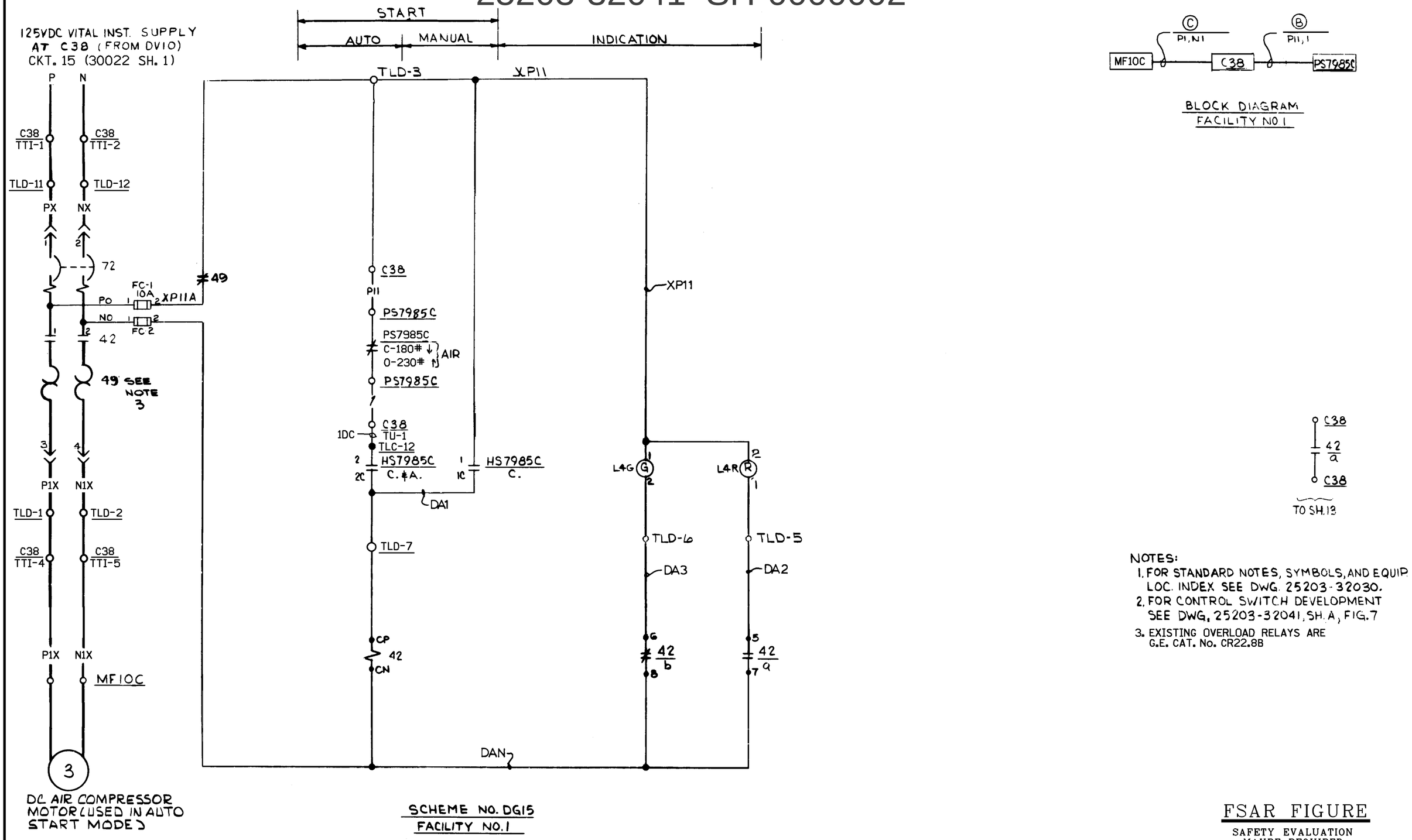
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
LOCAL ALARMS
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JSJ	——	——	
DATE 8-18-95	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-32041 SH. 25	
N/A			


REFERENCES
31078 SH.8 RC22 CONN DIAG
31115 SH.2 RC02C CONN DIAG
31055 SH.2 TO41 CONN DIAG
39047 SH.13 RC02C WIRING DIAG
RELOCATED TO SH.15
RELOCATED TO SH.16
RELOCATED TO SH.17
RELOCATED TO SH.18
RELOCATED TO SH.19
SPARE
TO PCD2 (ZE5388) (31212 SH.74)
TO SH. 2
TO SH. 22
TO PCD3 (PB818) DWG. 31212 SH.120
VOID AT COMPLETION OF PA 91-056
REMOVE DWG UPDATE
BLOCK DIAGRAM
NOTES
1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOCATION INDEX SEE DWG. 25203-30028.
SUPERCEDES 25203-39038 SH.19
NON Q.A.
REDRAWN FROM SAME NNECO DWG. No. REV.5
CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.
REVISIONS DURING CONSTRUCTION
Northeast Utilities System
FOR NORTHEAST NUCLEAR ENERGY CO.
TITLE MILLSTONE UNIT No.2
DIESEL GENERATOR 15G-13U (H7B)
ALARM & INTERLOCK CONTACTS
WATERFORD, CT.
DWG. NO. 25203-32041 SH.26

$FILE$
$DATE$



SCHEME NO. DG15
FACILITY NO. 1

NOTES:
1. FOR STANDARD NOTES, SYMBOLS, AND EQUIP LOC. INDEX SEE DWG. 25203-32030.
2. FOR CONTROL SWITCH DEVELOPMENT SEE DWG. 25203-32041, SH. A, FIG. 7
3. EXISTING OVERLOAD RELAYS ARE G.E. CAT. No. CR22.8B

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES PART OF
25203-39038 SH.41

REDRAWN FROM SAME
NNECO DWG. NO. REV. 1

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

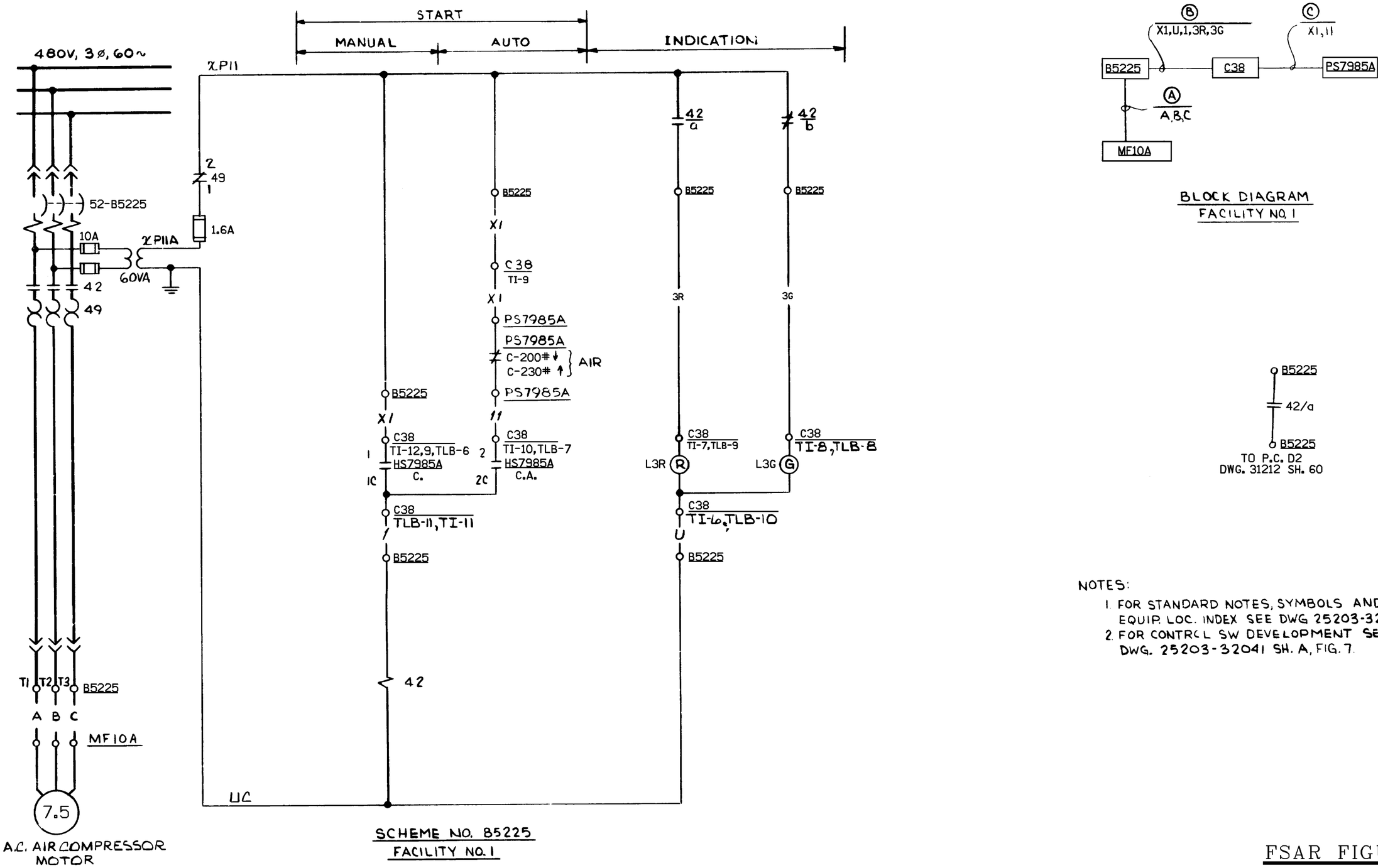
REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
6	2-11-02	INCORP'D DCN DM2-01-1828-98	MJS	REE	RSTG	MAG
5	6-7-01	INCORP. DCN's DM2-00-0084-00 DM2-01-0084-00	DJH	JSJ	DLW	RJY
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
3	8/4/99	INCORP. DCN's DM2-S-0108-94, DM2-00-0010-99	WBZ	JSJ	JPP	JPP
2	3-13-97	INCORP. DCN DM2-S-0806-95 PER DCR M2-S-0121-96	JMM	RNB	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U(H7A)
STARTING AIR COMPRESSOR (DC)
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JMM			
DATE 6/1/94	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-32041 SH. 27		



BLOCK DIAGRAM
FACILITY NO. 1

TO P.C. D2
DWG. 31212 SH. 60

NOTES:
1. FOR STANDARD NOTES, SYMBOLS AND EQUIP. LOC. INDEX SEE DWG 25203-32030
2. FOR CONTROL SW DEVELOPMENT SEE DWG. 25203-32041 SH. A, FIG. 7.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERSEDES
25203-39038 SH. 41

REDRAWN FROM SAME
NNECO DWG. NO. REV. 2

NON Q.A.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
FO* 0203 1927	7	3-4-02	INCORP. DCN DM2-00-0084-00, DM2-01-0084-00	ADS	GHH	TP	CAM
	6	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
	5	8/4/99	INCORP. DCN DM2-S-0108-94	WBZ	JSJ	JPP	JPP
	4	8/8/98	INCORP. DCN DM2-00-1308-97	WBZ	MM	JPP	JPP
N/A	3	12-3-97	INCORP. DCN DM2-S-1199-95	DSJ	GFD	RWD	JB

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

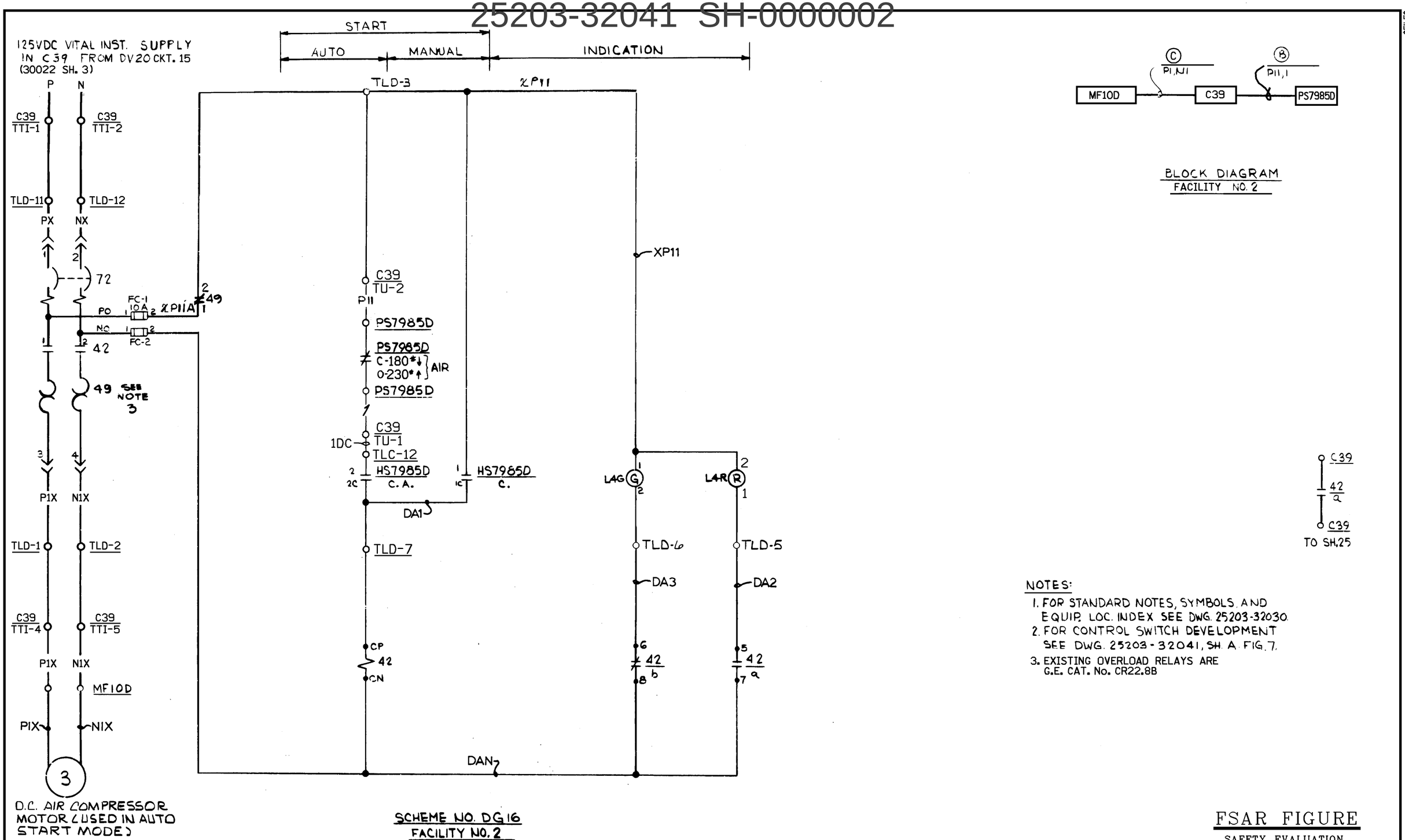
TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-12U (H7A)
STARTING AIR COMPRESSOR (AC)
WATERFORD, CT.

BY	CHKD.	APP.	APP.
DSJ	GFD	RWD	JB
DATE 11/20/97	DATE 12-3-97	DATE 12-3-97	DATE 12-3-97

SCALE N/A
N/A

DWG. NO. 25203-32041 SH. 28

$FILE$
$DATE$



SCHEME NO. DG16
FACILITY NO. 2

NOTES:

1. FOR STANDARD NOTES, SYMBOLS, AND EQUIP. LOC. INDEX SEE DWG. 25203-32030.
2. FOR CONTROL SWITCH DEVELOPMENT SEE DWG. 25203-32041, SH. A FIG. 7.
3. EXISTING OVERLOAD RELAYS ARE G.E. CAT. No. CR22.8B

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

SUPERCEDES PART OF 25203-39038 SH. 41

REDRAWN FROM SAME NNECO DWG. NO. REV. 1

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

P.A. #	NO.	DATE	REVISIONS	BY	CHK	APP	APP
PO# 0203 1927	5	3-22-02	INC, DM2-00-0086-00, DM2-01-0086-00	RJH	GHH	TP	CAM
	4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
	3	8/4/99	INCORP. DCN's DM2-S-0108-94, DM2-00-0010-99	WBZ	JSJ	JPP	JPP
	2	3-6-97	INCORP. DCN DM2-S-1213-95 PER DCR M2-S-0123-96	JMM	RNB	JPP	JPP

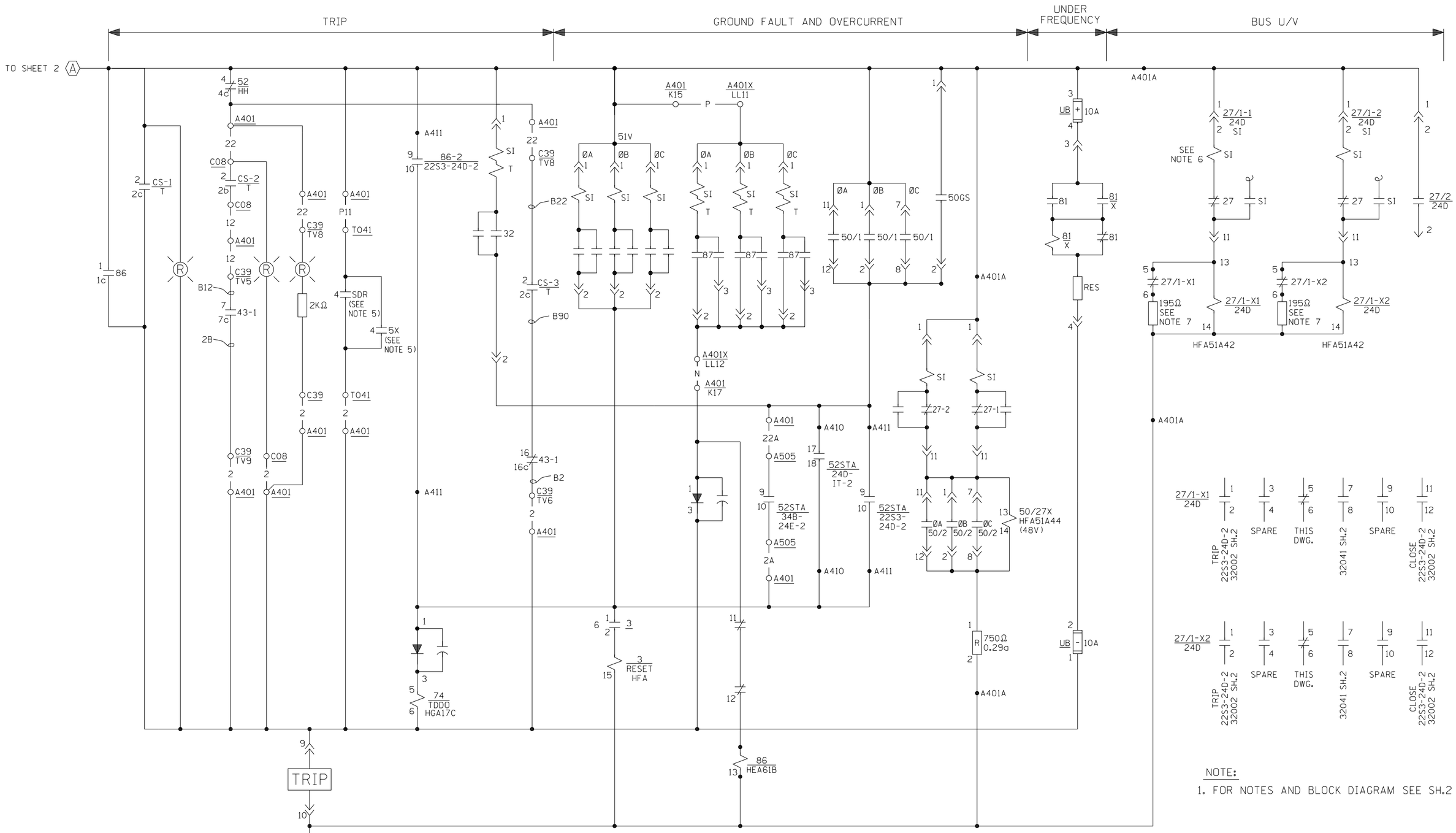
Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
STARTING AIR COMPRESSOR (DC)
WATERFORD, CT.

BY JMM	CHKD. ——	APP. ——	APP. ——
DATE 6/2/95	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-32041 SH. 29		

25203-32041 SH-0000002



NOTE:

1. FOR NOTES AND BLOCK DIAGRAM SEE SH.2

SCHEME A401
FACILITY Z2

FSAR FIGURE

SAFETY EVALUATION
MAYBE REQUIRED
PRIOR TO CHANGES

Q.A.

REFERENCE	
25203-30034 SH. 2	LOGIC DIAG.
25203-30071	ARRGT. C01X
25203-31063 SH. 1	CONN. DIAG. C08
25203-31073 SH. 2	CONN. DIAG. C01X
25203-32043 SH. 4	SCHEM. DIAG. C01X
25203-39030 SH. 11	WIRING DIAG. C01X
25203-39045 SH. 8	WIRING DIAG. C08

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
11	9-29-08	INCORP. DCN DM2-00-0235-08	JCV	MEL	-	DLW
10	10-22-01	INCORP'D DCN DM2-00-0865-99	TAM	RNB	MAG	GMO

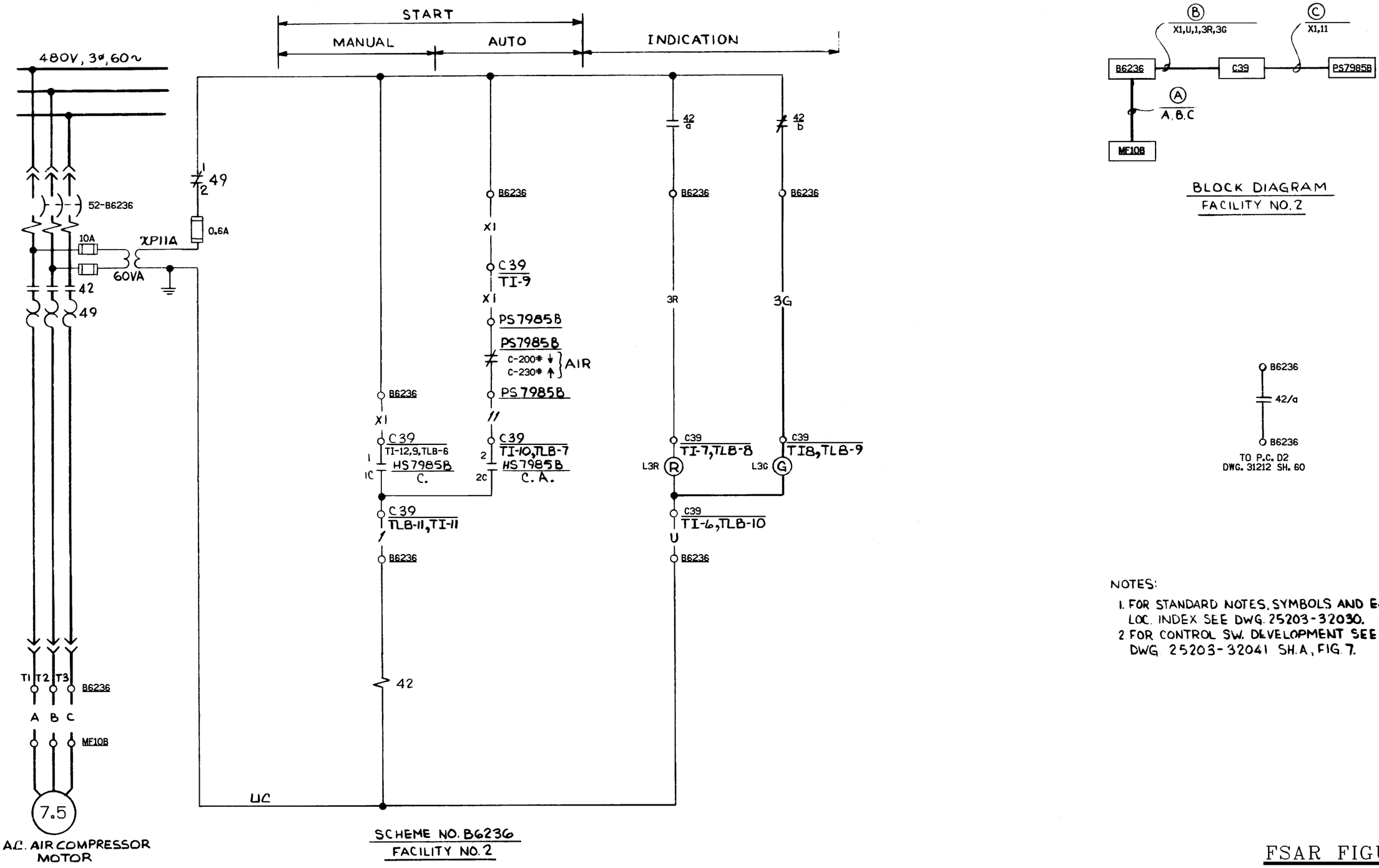
Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT 2
4.16 KV DIESEL GEN. FEEDER BKR.
15G-13U-2 (H7B)(A401)
WATERFORD, CT.

BY JSJ	CHKD. MEL	APP. RJH	APP. GJF
DATE 4/9/92	DATE 10/8/92	DATE 10/8/92	DATE 10/8/92
SCALE N/A	DWG. NO. 25203-32041 SH. 2A		
91-073			

\\10.108.35.47\DATA\MILLSTONE\4\DATA1\MPCAD\MPDesign\USERS\mich599\DGN\MP\25203\PROJECT\30000\P320412A.DGN
09292008

$FILE$
$DATE$



SCHEME NO. B6236
FACILITY NO. 2

BLOCK DIAGRAM
FACILITY NO. 2

TO P.C. D2
DWG. 31212 SH. 60

NOTES:

1. FOR STANDARD NOTES, SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-32030.
2. FOR CONTROL SW. DEVELOPMENT SEE DWG 25203-32041 SH.A, FIG. 7.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NON Q.A.

REDRAWN FROM SAME
NNECO DWG. NO. REV. 2

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

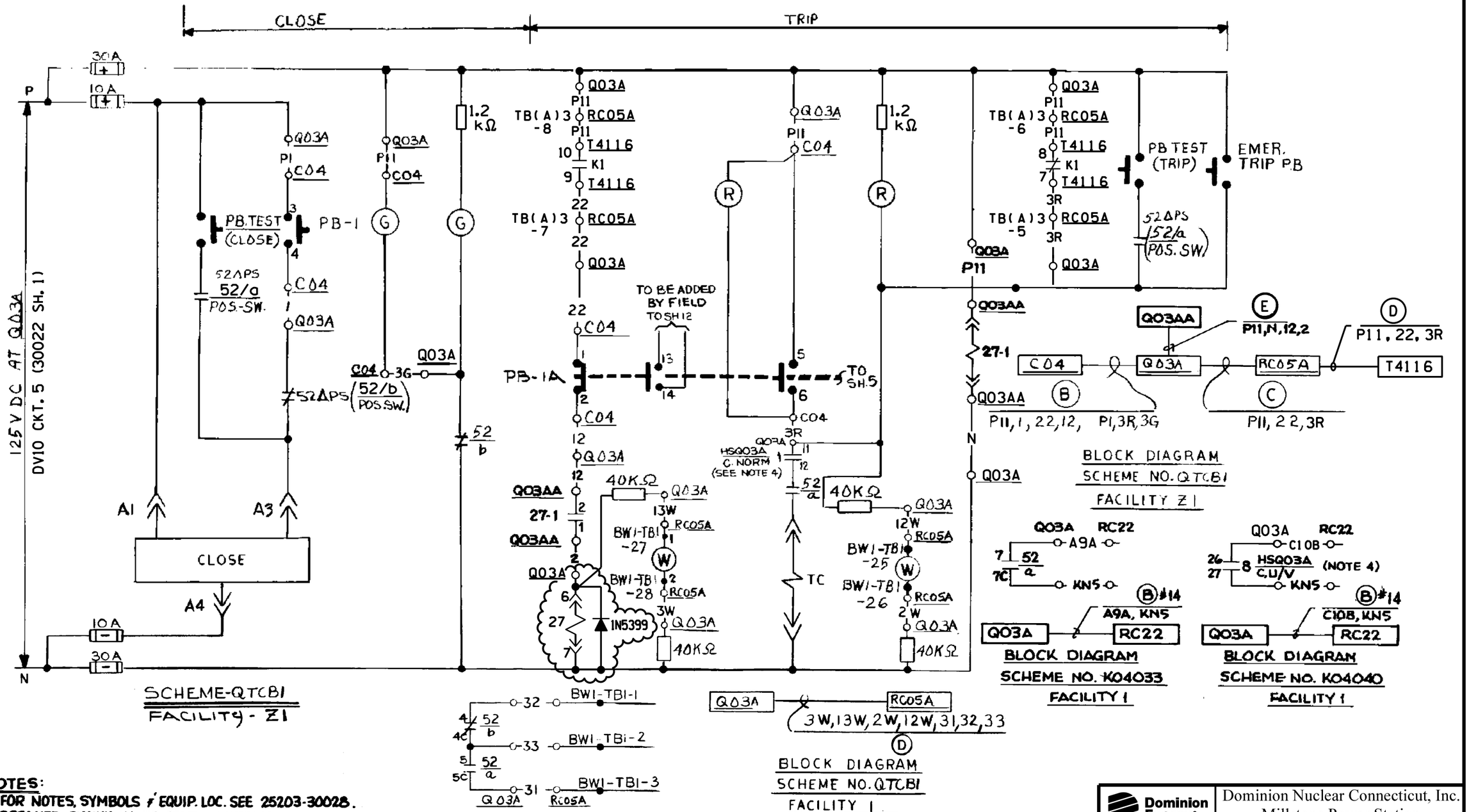
	NO.	DATE	REVISIONS	BY	CHK	APP	APP
PO# 0203 1927	6	3-4 -02	INCORP. DCN DM2-00-0086-00, DM2-01-0086-00	ADS	GHH	TP	CAM
X	5	2-7 -00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
X	4	8/4 /99	INCORP. DCN DM2-S-0108-94	WBZ	JSJ	JPP	JPP
N/A	3	12-3 -97	INCORP. DCN DM2-S-1199-95	DSJ	GFD	RWD	JB

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
DIESEL GENERATOR 15G-13U (H7B)
STARTING AIR COMPRESSOR (AC)
WATERFORD, CT.

BY DSJ	CHKD. GFD	APP. RWD	APP. JB
DATE 11/20/97	DATE 12-3-97	DATE 12-3-97	DATE 12-3-97
SCALE N/A	DWG. NO. 25203-32041 SH. 30		
N/A			

SUPERSEDES
25203-39038 SH. 41



NOTES:

1. FOR NOTES, SYMBOLS & EQUIP. LOC. SEE 25203-30028.
2. BREAKER SHOWN IN TRIPPED POSITION.
3. FOR "K" RELAY LOC. SEE 7604-M806-525 (C.E. DWG. E-18767-411-013 sh. 4.) (39069 SH. 15 & 31175 SH. 393)
4. HAND SWITCH LOCATED ON Q03A SW. GEAR. FOR H.S. DEVELOPMENT SEE -32045 sh.A, FIG. 1.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

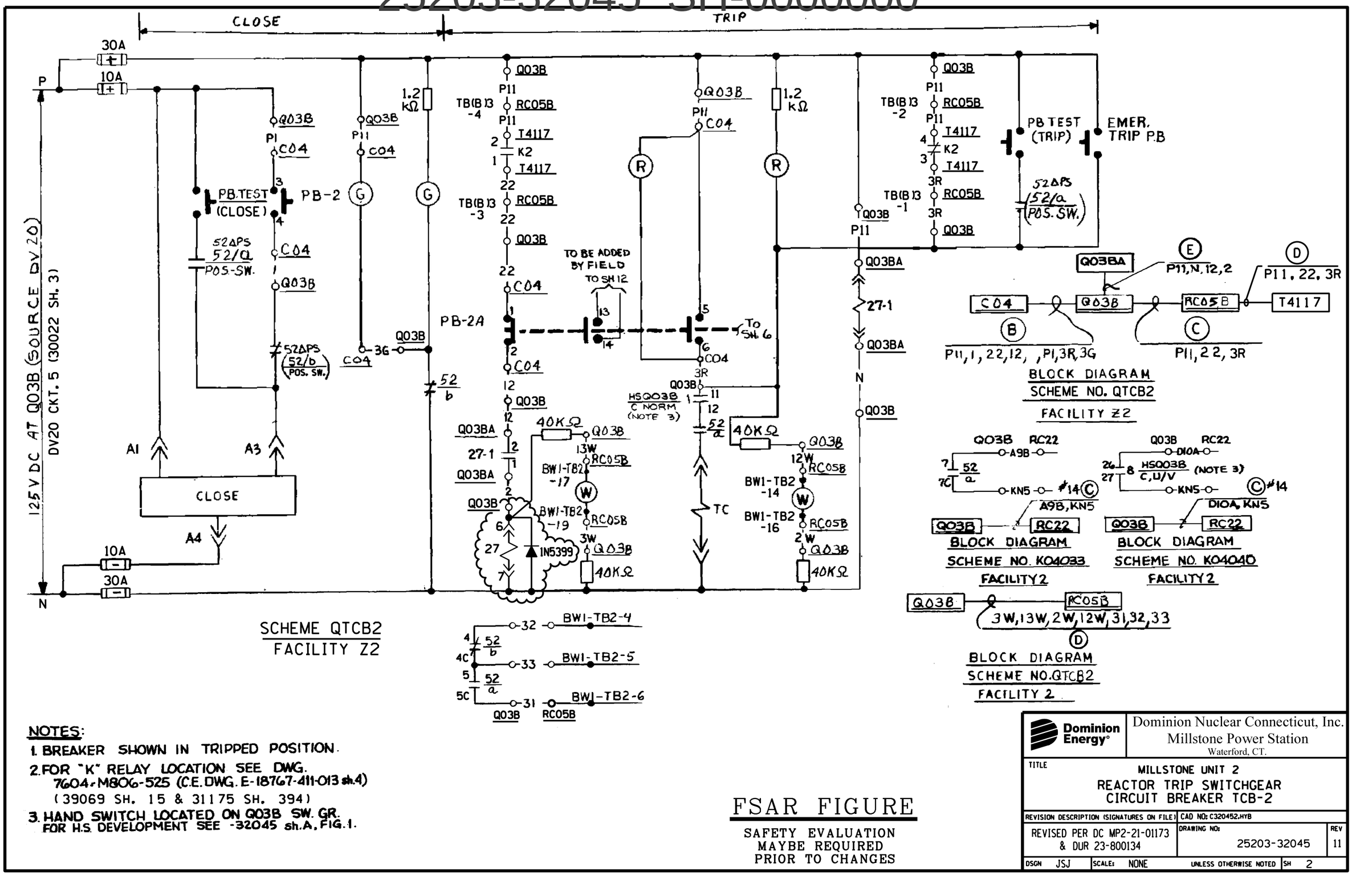
Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-1

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320451.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 1

Authenticated By: MBULOGIN\scott84
22-MAY-2023 22:24
C320451.HYB



SCHEME QTCB2
FACILITY Z2

NOTES:

1. BREAKER SHOWN IN TRIPPED POSITION.
2. FOR "K" RELAY LOCATION SEE DWG. 7604-M806-525 (C.E. DWG. E-18767-411-013 sh.4) (39069 SH. 15 & 31175 SH. 394)
3. HAND SWITCH LOCATED ON Q03B SW. GR. FOR H.S. DEVELOPMENT SEE -32045 sh.A, FIG.1.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy°	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE UNIT 2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER TCB-2
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320452.HYB
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045 REV 11
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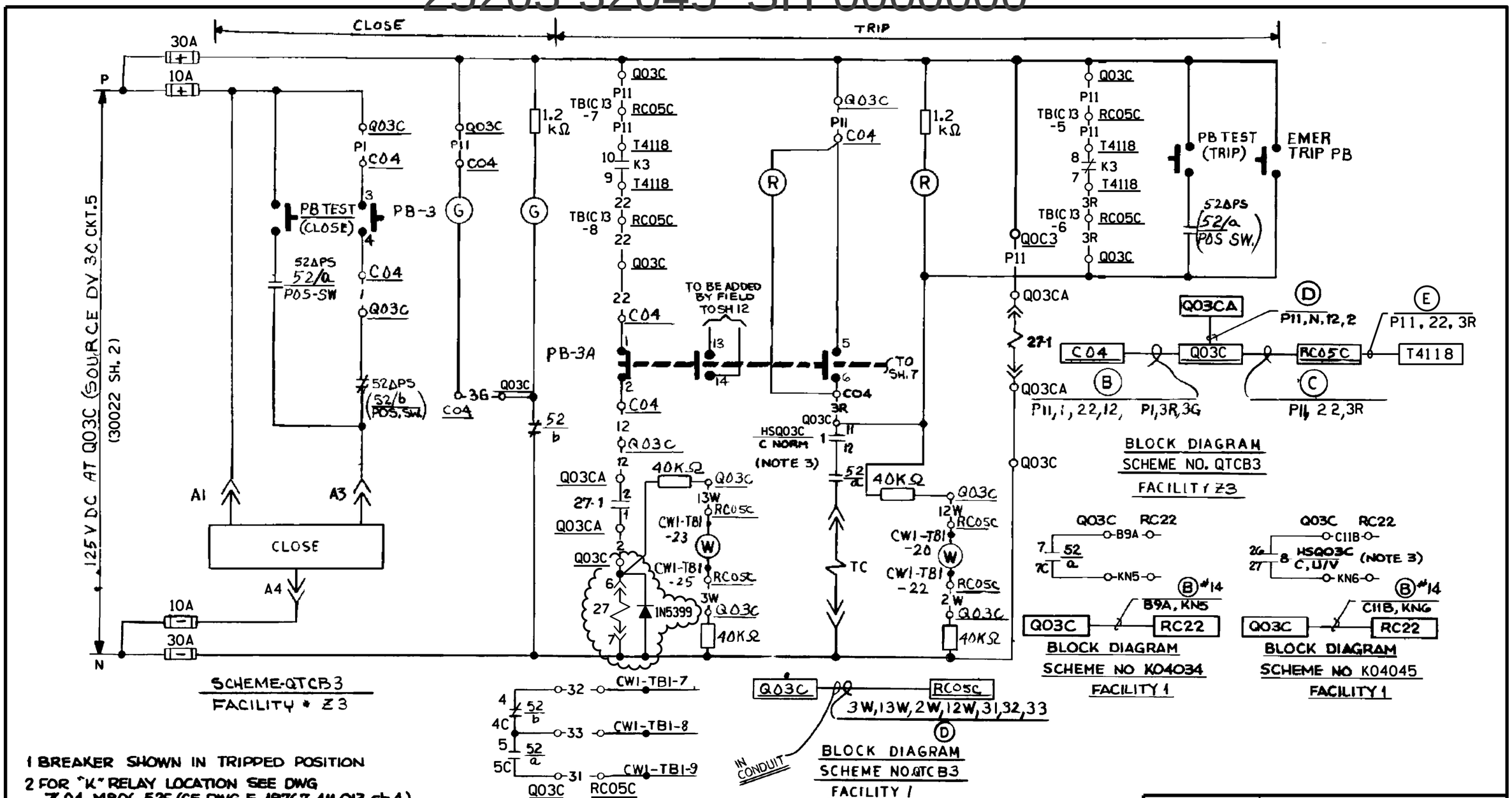

1 BREAKER SHOWN IN TRIPPED POSITION
2 FOR "K" RELAY LOCATION SEE DWG 7604-M806-525 (CE DWG E-18767-411-013 sh.4.) (39069 SH. 15 & 31175 SH. 395)
3 HANDSWITCH LOCATED ON Q03C SWITCHGEAR FOR HS DEVELOPMENT (SEE -32045 sh A, FIG 1)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-3

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320453.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 3

Authenticated By: MBULOGIN\scott84
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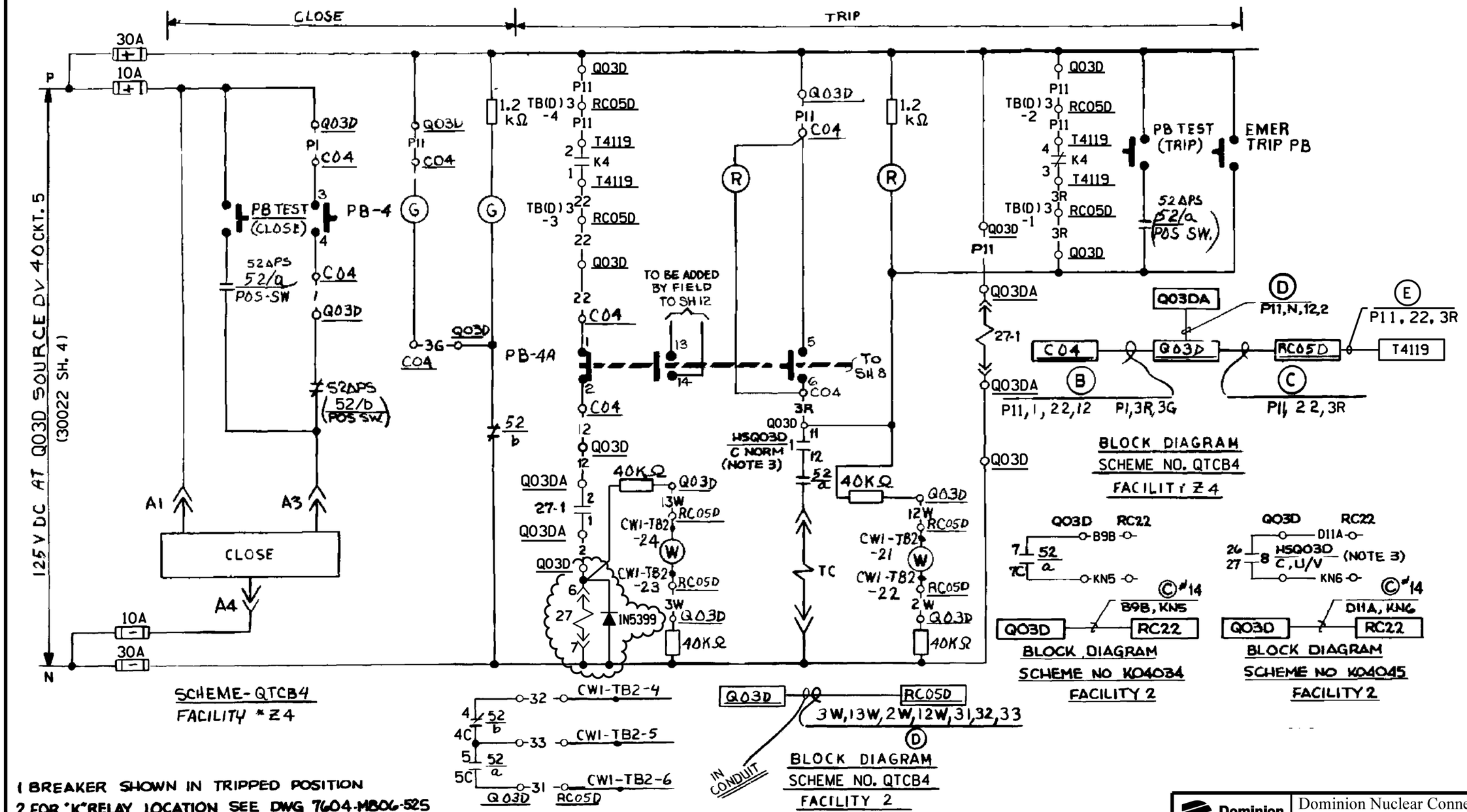

1 BREAKER SHOWN IN TRIPPED POSITION

2 FOR "K" RELAY LOCATION SEE DWG 7604-M806-525
(CE DWG E-18767-411-013 sh 4)
(39069 SH. 15 & 31175 SH. 396)

3 HAND SWITCH LOCATED ON Q03D SWITCHGEAR
FOR HANDSWITCH DEVELOPMENT SEE -32045 sh A FIG 1.

FSAR FIGURE

SAFETY EVALUATION
MAYBE REQUIRED
PRIOR TO CHANGES

Dominion Energy°	Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.
TITLE	MILLSTONE UNIT 2 REACTOR TRIP SWITCHGEAR CIRCUIT BREAKER TCB-4
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320454.HYB
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045 REV 11
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED SH 4

Authenticated By: MBULOGIN\scott84
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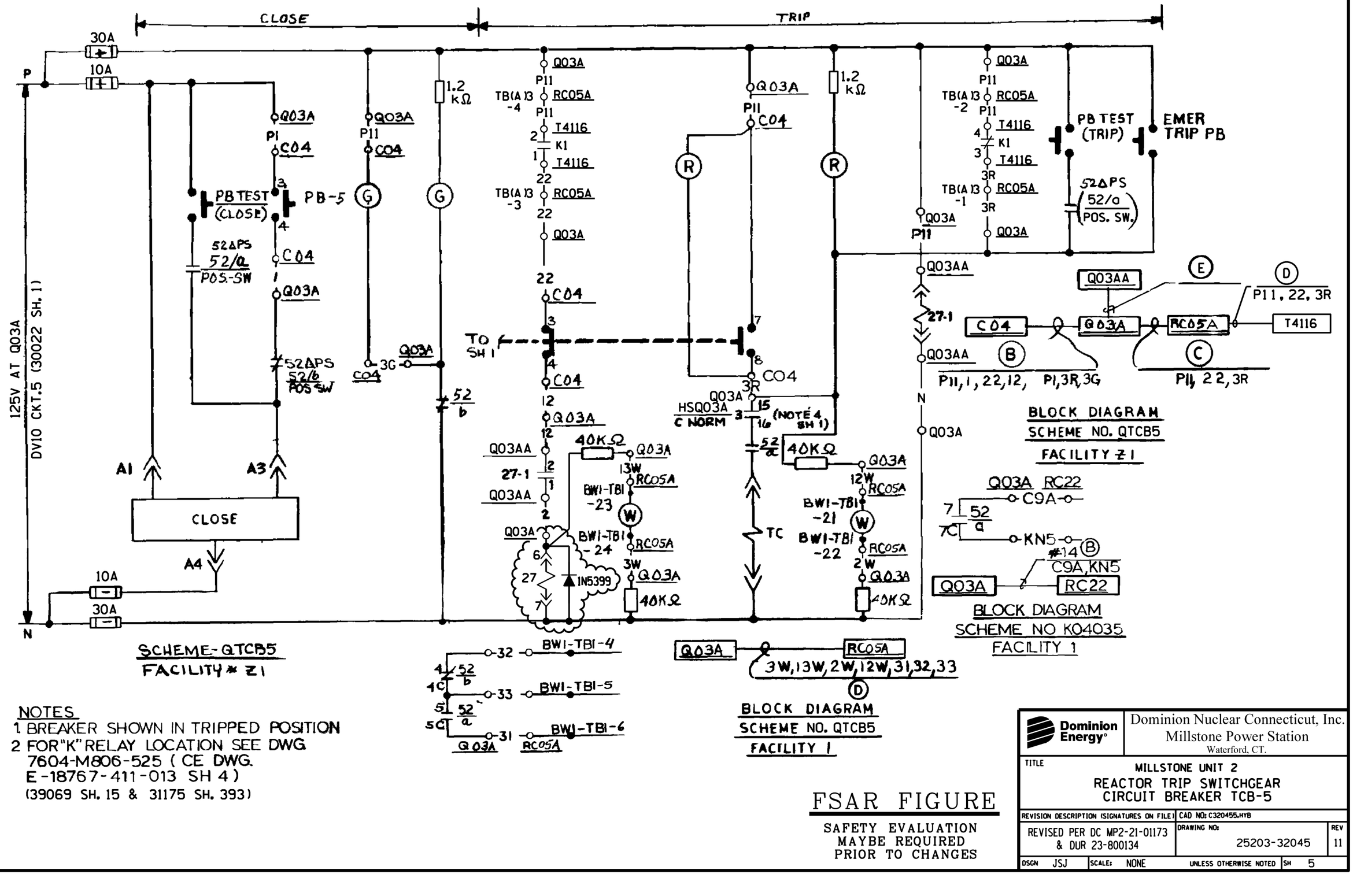
C320454.HYB



NOTES
1. BREAKER SHOWN IN TRIPPED POSITION
2. FOR "K" RELAY LOCATION SEE DWG. 7604-M806-525 (CE DWG. E-18767-411-013 SH 4) (39069 SH. 15 & 31175 SH. 393)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-5

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320455.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE UNLESS OTHERWISE NOTED	SH 5

Authenticated By: MBULOCIN\scottB4
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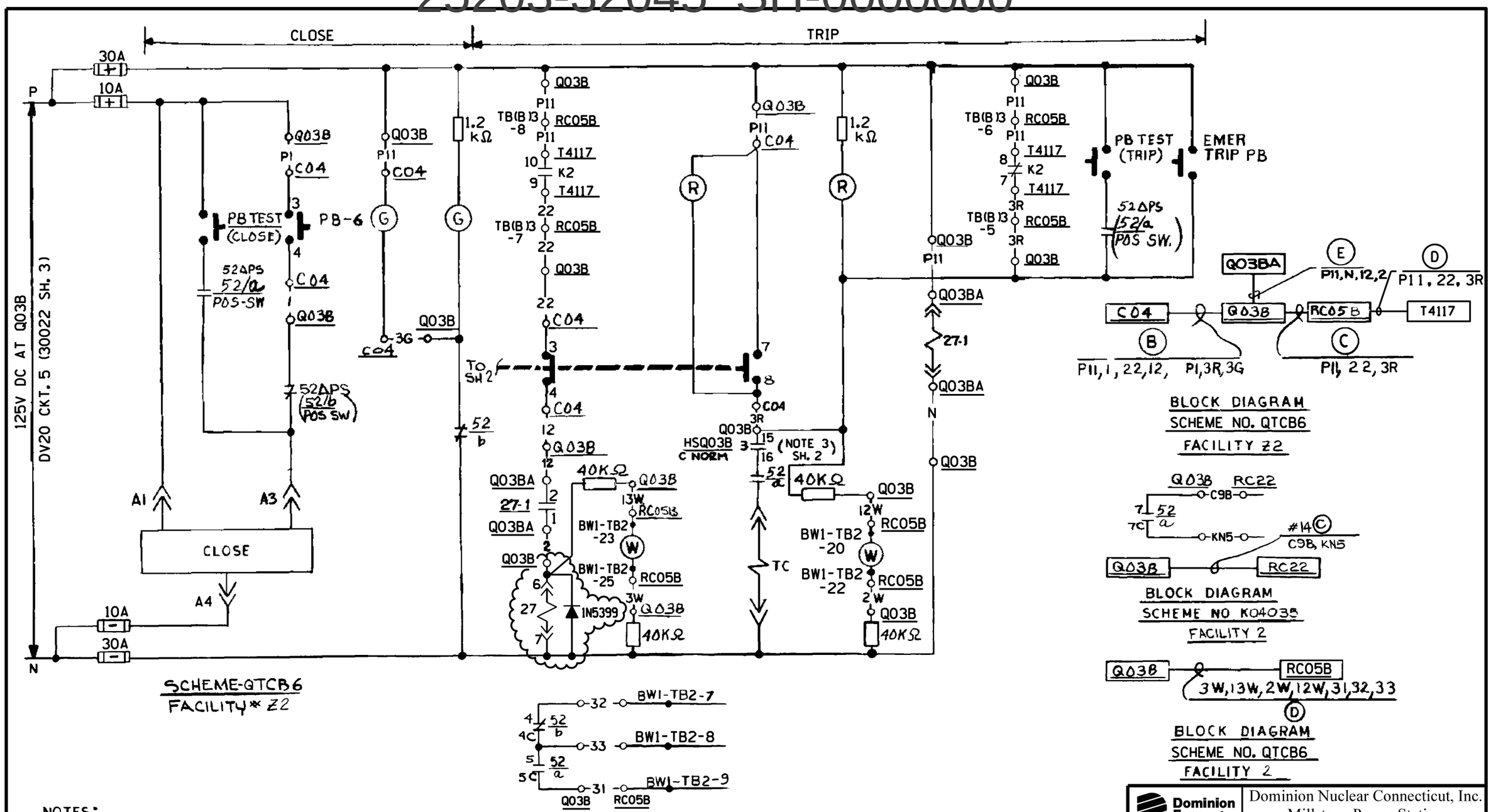

SCHEME-QTCB6
FACILITY* Z2

BLOCK DIAGRAM
SCHEME NO. QTCB6
FACILITY Z2

BLOCK DIAGRAM
SCHEME NO KO4035
FACILITY 2

BLOCK DIAGRAM
SCHEME NO. QTCB6
FACILITY 2

NOTES:

1. BREAKER SHOWN IN TRIPPED POSITION.
2. FOR K" RELAY LOCATION SEE DWG 7604-M806-525(CE DWG E-18767--411-013 SH4) (39069 SH. 15 & 31175 SH. 394)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-6

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320456.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 6

Authenticated By: MBULOGIN\scott84
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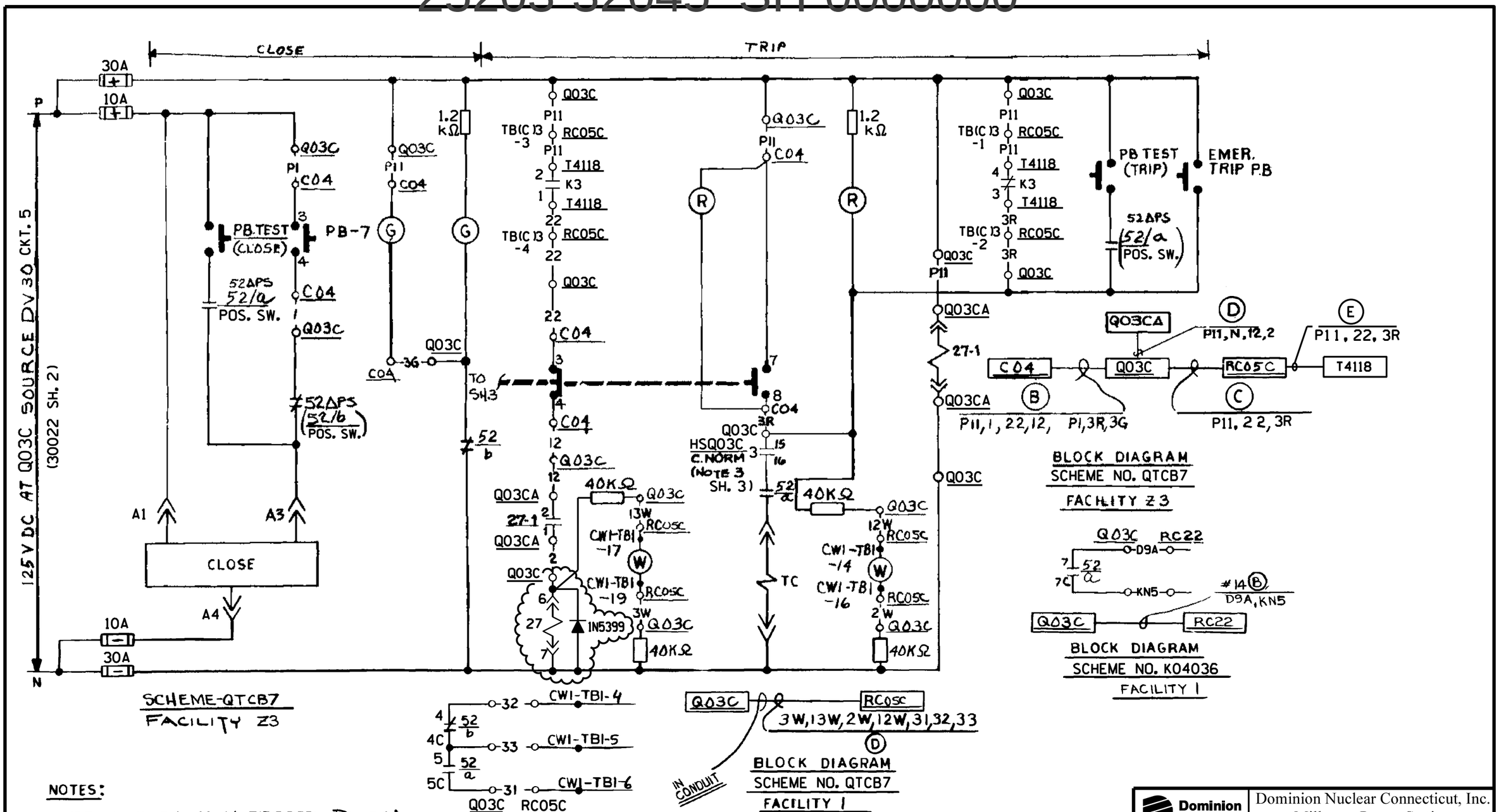


NOTES:

1. BREAKER SHOWN IN TRIPPED POSITION.
2. FOR "K" RELAY LOCATION SEE DWG. 7604-M806-525 (CE DWG. E-18767-411-013 SH4) (39069 SH. 15 & 31175 SH. 395)

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion Energy

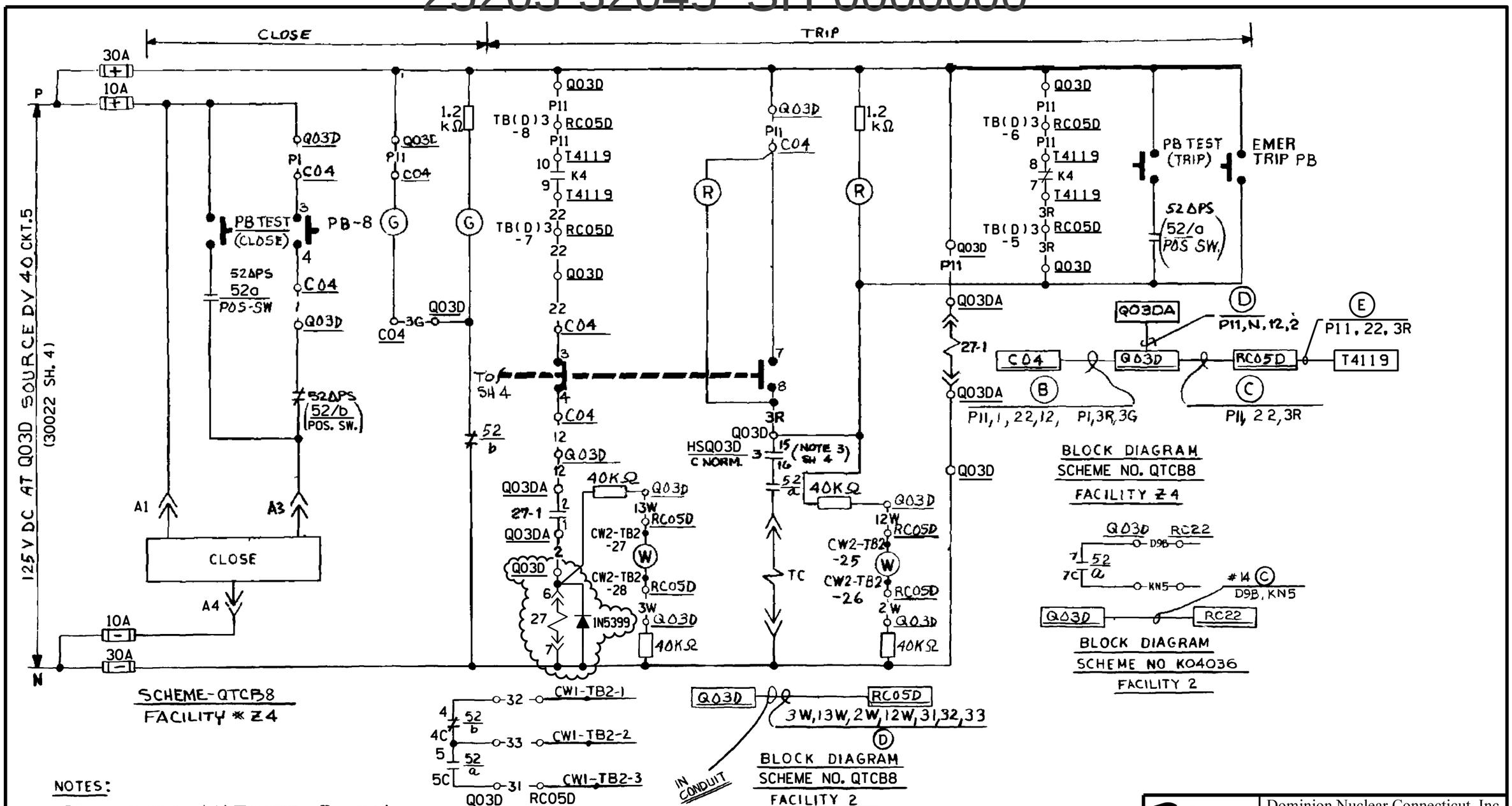
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-7

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320457.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 7

Authenticated By: MBULOGIN\scott84
22-MAY-2023 22:25
C320457.HYB

25203-32045 SH-0000000



NOTES:

1 BREAKER SHOWN IN TRIPPED POSITION.
2 FOR "K" RELAY LOCATION SEE DWG 7604-M806-525 (CE DWG E-18767-411-013 SH4) (39069 SH. 15 & 31175 SH. 396)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy

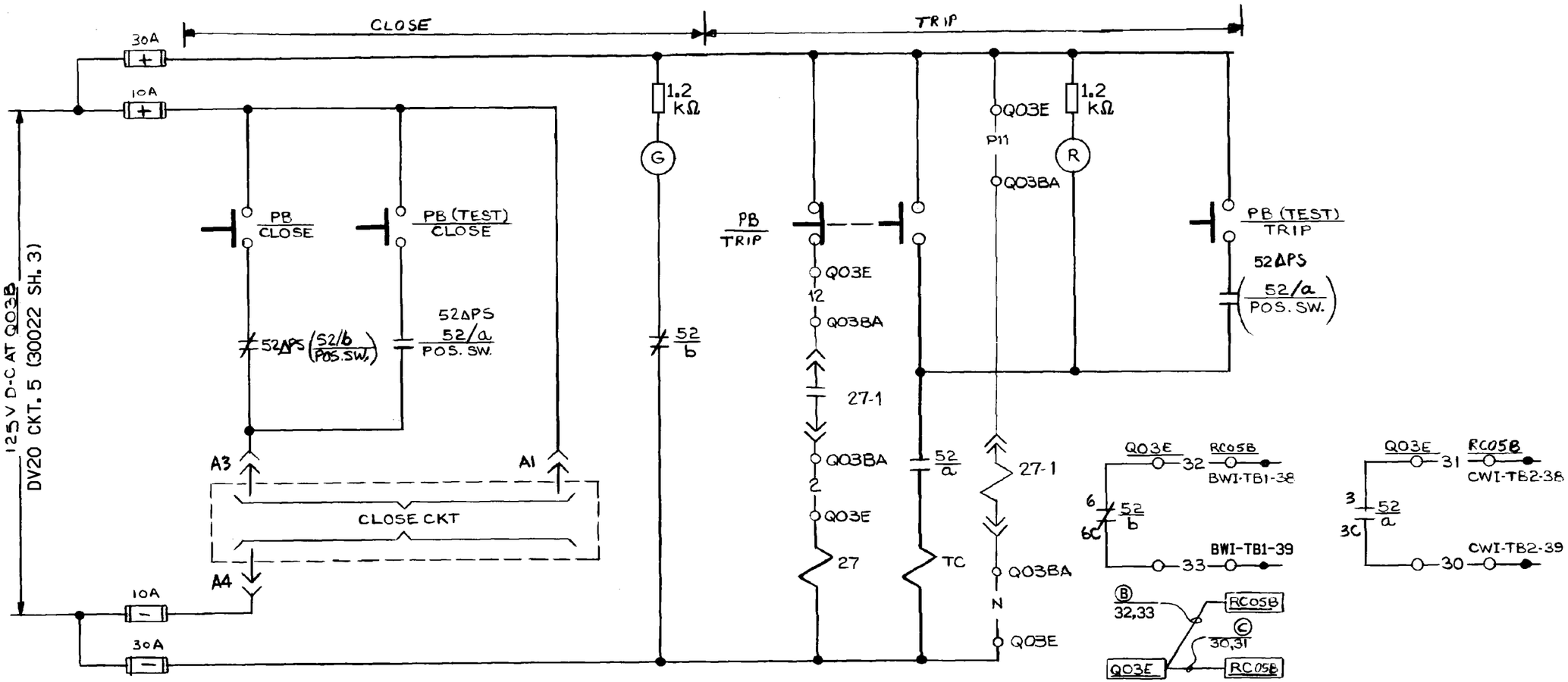
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE: MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
CIRCUIT BREAKER TCB-8

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320458.HYB	
REVISED PER DC MP2-21-01173 & DUR 23-800134	DRAWING NO: 25203-32045	REV 11
DSGN JSJ	SCALE: NONE	UNLESS OTHERWISE NOTED SH 8

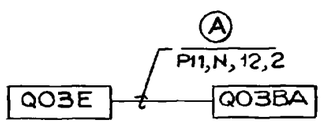
Authenticated By: MBULOGIN\scott84
22-MAY-2023 22:25
C320458.HYB

$FILE$
$DATE$

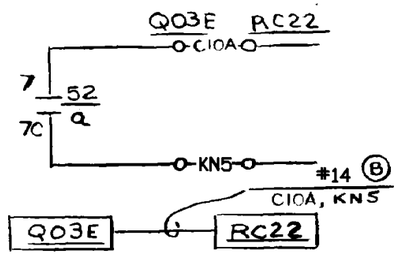


SCHEME - QTCB9
FACILITY 22

BLOCK DIAGRAM
SCHEME NO. QTCB9
FAC: 2



BLOCK DIAGRAM
SCHEME QTCB9
FACILITY 22



BLOCK DIAGRAM
SCHEME KO4039
FACILITY 2

NOTE:
1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028
2. BREAKER SHOWN IN TRIPPED POSITION.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
9	4-26-04	INCORP. DCN DM2-00-1001-99	KBB	DLW	-	DED
8	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
7	3-6-97	INCORP. DCN DM2-S-1213-95 PER DCR M2-S-0123-96	JMM	RNB	JPP	JPP

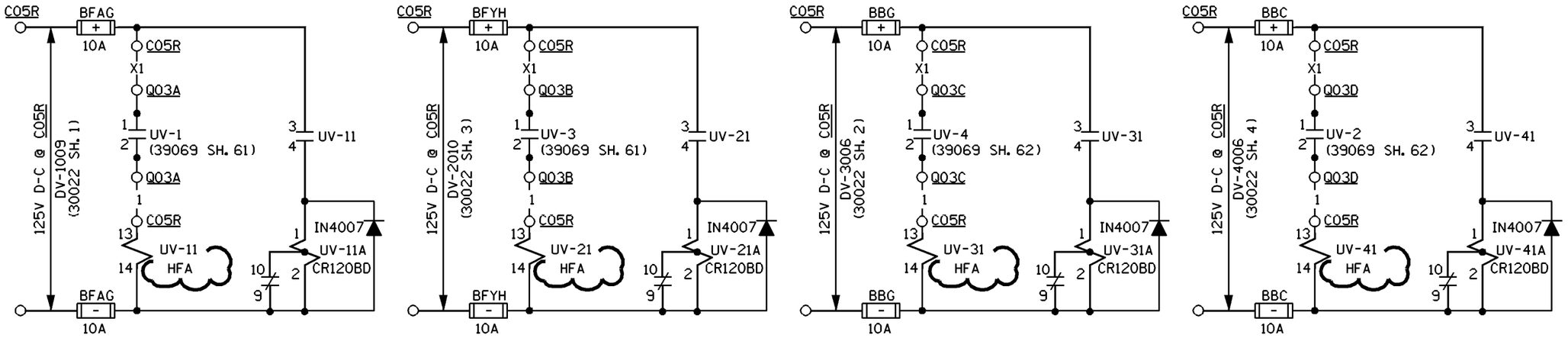
CAD
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

Dominion

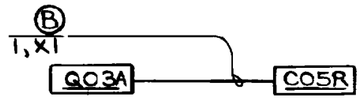
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
REACTOR TRIP SWITCHGEAR
BUS TIE BREAKER TCB-9
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JMM	—	—	—
DATE 8/28/96	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-32045 SH. 9		



SCHEME NO. Q0271
FACILITY NO. Z1



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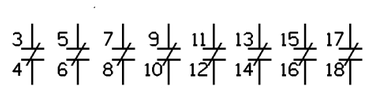
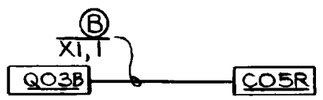


UV-11A
FACILITY 1
(EXCEPT 9/10 WHICH
IS FACILITY Z1)

SCHEME NO. Q0272
FACILITY NO. Z2



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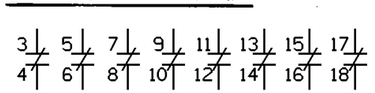
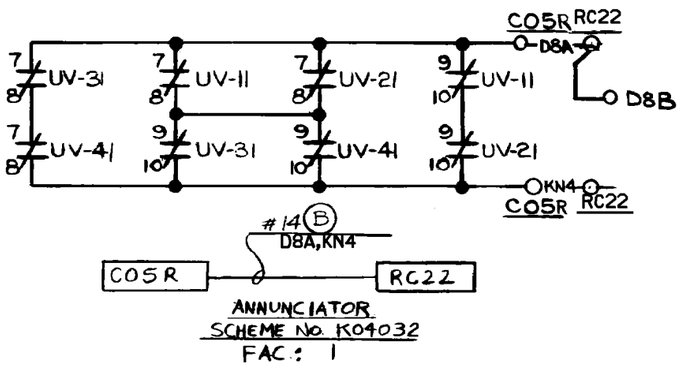
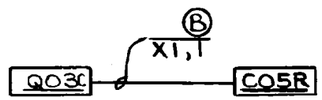


UV-21A
FACILITY 1
(EXCEPT 9/10 WHICH
IS FACILITY Z2)



ANNUNCIATOR
SCHEME NO K04032
FAC: 1

SCHEME NO. Q0273
FACILITY NO. Z3



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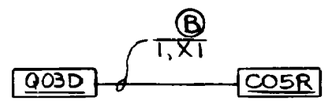

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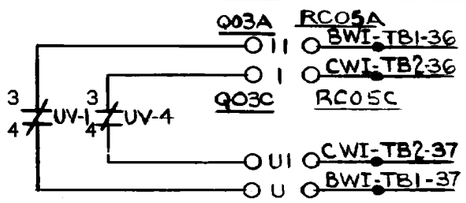
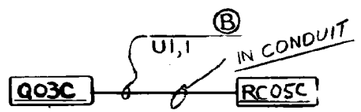
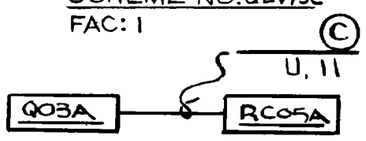
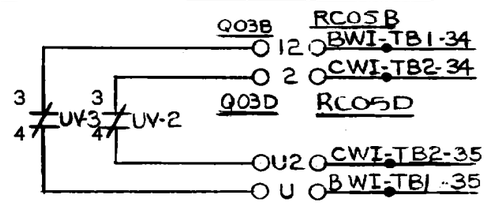


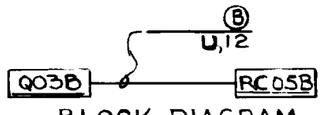


BLOCK DIAGRAM
SCHEME NO. QUV13C
FAC: 1

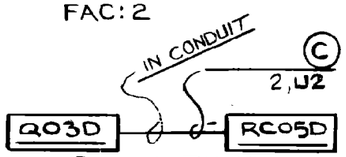


BLOCK DIAGRAM
SCHEME Nº QUV13A
FAC: 1

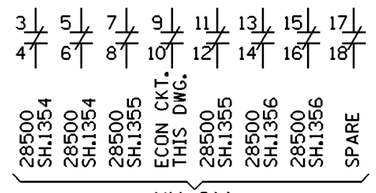




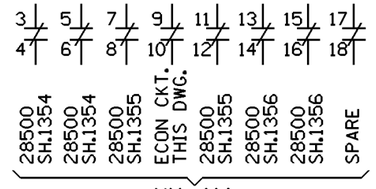
BLOCK DIAGRAM
SCHEME NO. QUV24B
FAC: 2



BLOCK DIAGRAM
SCHEME Nº QUV24D
FAC: 2



UV-31A
FACILITY 1
(EXCEPT 9/10 WHICH
IS FACILITY Z3)



UV-41A
FACILITY 1
(EXCEPT 9/10 WHICH
IS FACILITY Z4)

SCHEME NO. Q0274
FACILITY NO. Z4

NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2. FOR LOCATION IN SWITCHGEAR ASSY REF. DWG E-18767-413-130.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion®

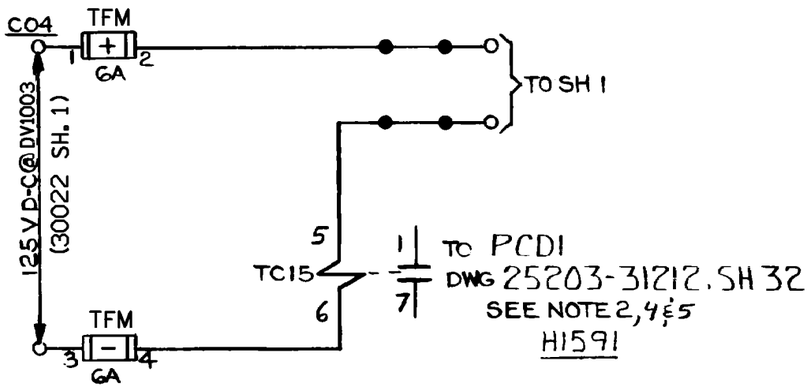
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

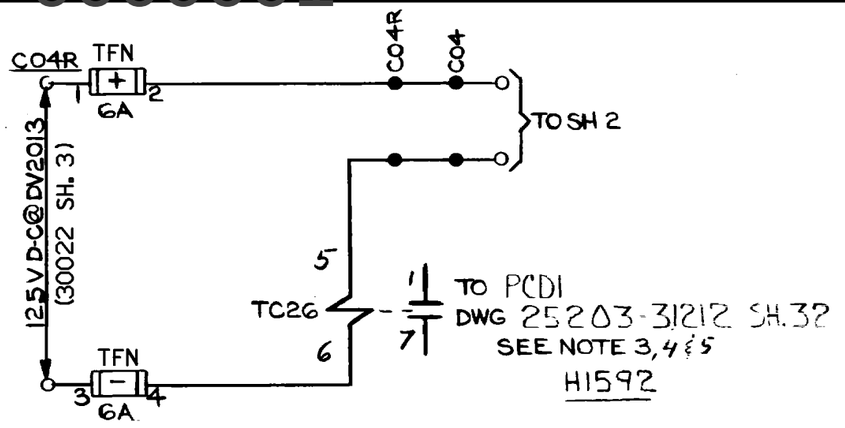
TITLE

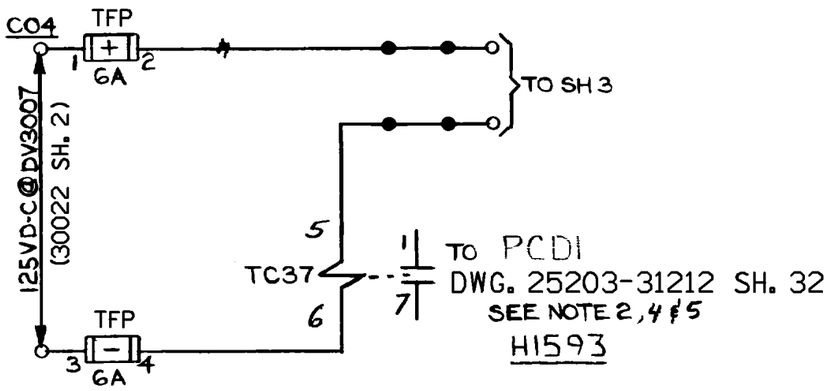
UNDERVOLTAGE TRIP DEVICE & INTERLOCKS

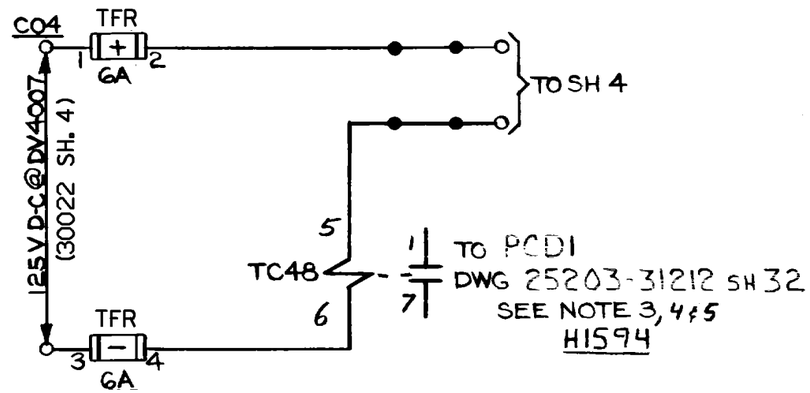
REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C320451LHYB	
REVISED PER MSDUR 17-000175 & DUR 17-800361	DRAWING NO: 25203-32045	REV 8
DSGN DLW	SCALE: N/A	UNLESS OTHERWISE NOTED SH 11

$FILE$
$DATE$









NOTES:

1. FOR STANDARD NOTES, SYMBOLS & EQUIP. LOC. INDEX SEE DWG. 25203-30028
2. CONTACTS WIRED NV1.
3. CONTACTS WIRED NV2.
4. ALL TC RELAYS ARE 12HGA11J52
5. REF. TO FIELD REQUISITION 7604-E-045, SH 14, REV. 0 .

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

REDRAWN FROM SAME NNECO DWG. NO. REV. 2

CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

Dominion

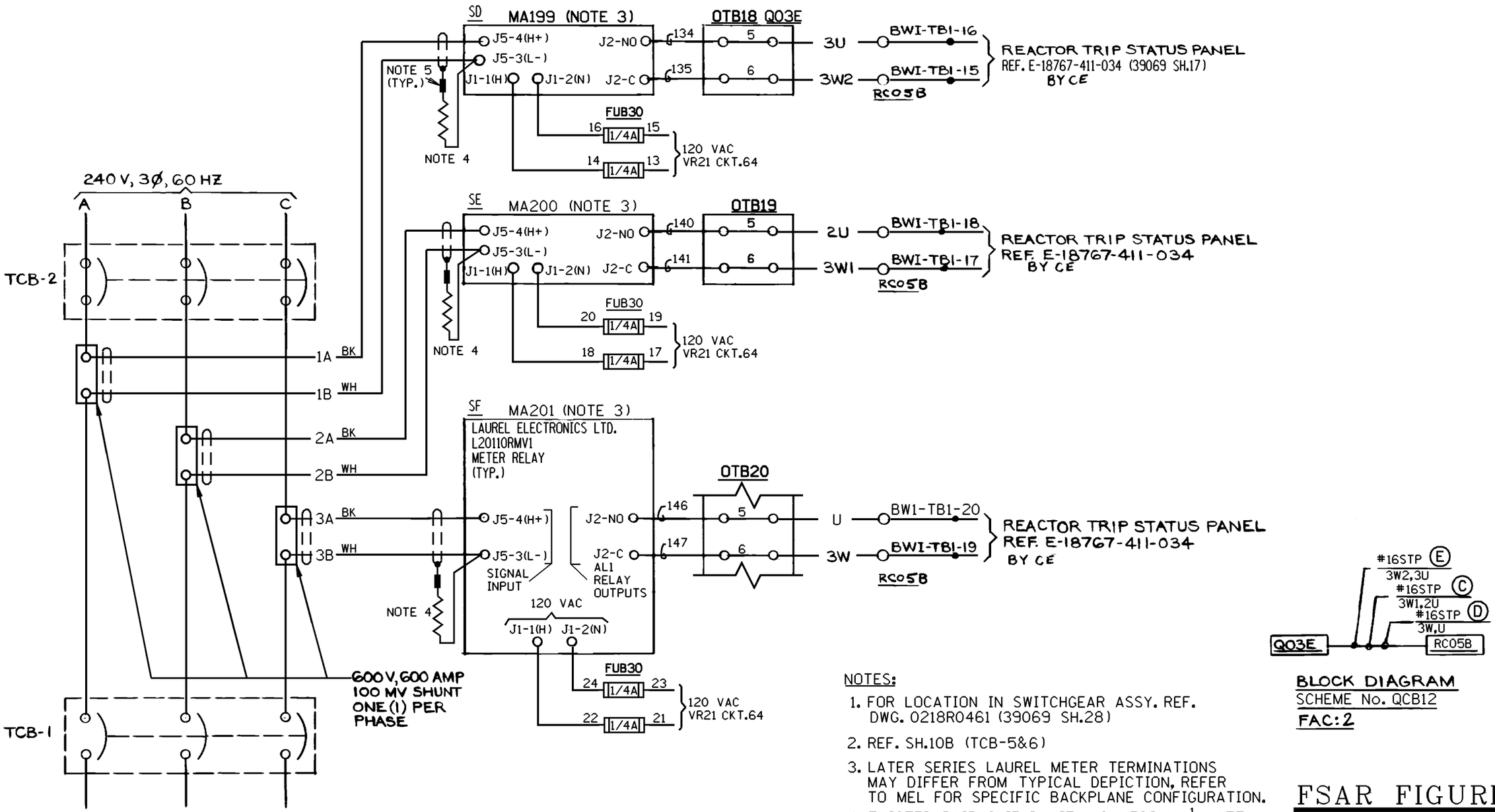
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
MANUAL TRIP PUSHBUTTON
COMPUTER INPUT
WATERFORD, CT.

BY JMM	CHKD. ——	APP. ——	APP. ——
DATE 8/8/96	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-32045 SH. 12		

$FILE$
$DATE$

$FILE$
$DATE$



BLOCK DIAGRAM
SCHEME No. QCB12
FAC:2

NOTES:

1. FOR LOCATION IN SWITCHGEAR ASSY. REF. DWG. 0218R0461 (39069 SH.28)
2. REF. SH.10B (TCB-5&6)
3. LATER SERIES LAUREL METER TERMINATIONS MAY DIFFER FROM TYPICAL DEPICTION, REFER TO MEL FOR SPECIFIC BACKPLANE CONFIGURATION.
4. TWISTED PAIR SHIELD WITH 10 MEGOHM, $\frac{1}{4}$ WATT RESISTOR DOES NOT TIE TO CABINET GROUND.
5. ━ DENOTES FIELD SPLICE.

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REFERENCE DRAWINGS

30022 SH.22 - VR21
39069 SH.46 - METER RELAY
39069 SH.50 - REMOTE CONNECTIONS
31105 - Q03 CONN. DIAG.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
8	7-28-11	INCORP DCN DM2-04-0183-09 PER DUR 11-800309	MKN	DLW	—	DLW

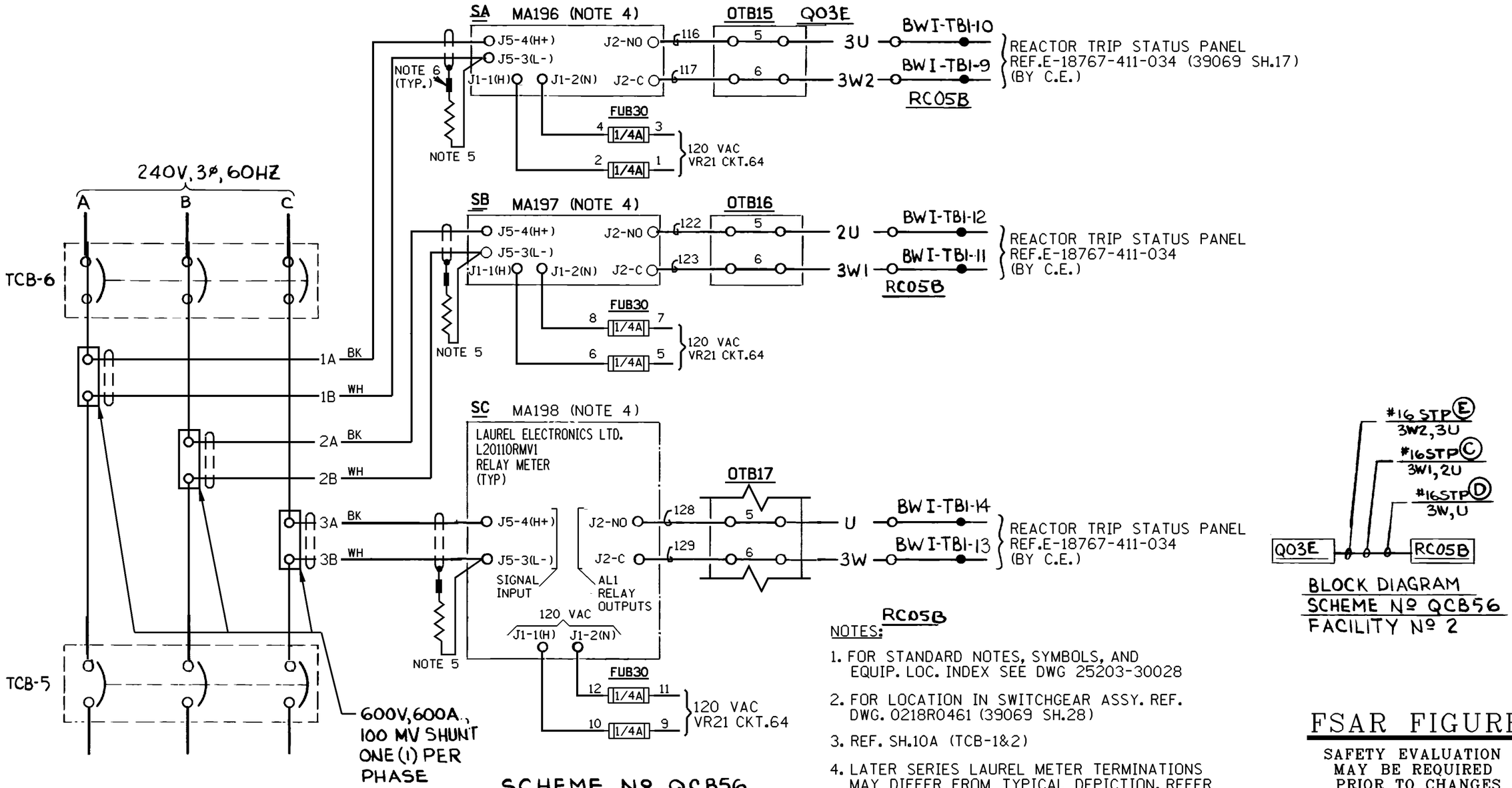
NON Q.A. | CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE MILLSTONE UNIT No.2
REACTOR TRIP SWITCHGEAR
UNDERCURRENT DEVICE (TCB-1&2)
WATERFORD, CT.

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO.	25203-32045 SH.10A	
N/A			



SCHEME Nº QCB56
FACILITY Nº 2

BLOCK DIAGRAM
SCHEME Nº QCB56
FACILITY Nº 2

RC05B

NOTES:

1. FOR STANDARD NOTES, SYMBOLS, AND EQUIP. LOC. INDEX SEE DWG 25203-30028
2. FOR LOCATION IN SWITCHGEAR ASSY. REF. DWG. 0218R0461 (39069 SH.28)
3. REF. SH.10A (TCB-1&2)
4. LATER SERIES LAUREL METER TERMINATIONS MAY DIFFER FROM TYPICAL DEPICTION, REFER TO MEL FOR SPECIFIC BACKPLANE CONFIGURATION.
5. TWISTED PAIR SHIELD WITH 10 MEGOHM, $\frac{1}{4}$ WATT RESISTOR DOES NOT TIE TO CABINET GROUND.
6. ━ DENOTES FIELD SPLICE.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

REFERENCE DRAWINGS

30022 SH.22 - VR21
39069 SH.45 - METER RELAY
39069 SH.50 - REMOTE CONNECTIONS
31105 - Q03 CONN. DIAG.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
7	7/28/11	INCORP DCN DM2-04-0183-09 PER DUR 11-800309	MKN	DLW	—	DLW

NON Q.A.

CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE
MILLSTONE UNIT No.2
REACTOR TRIP SWITCHGEAR
UNDERCURRENT DEVICE (TCB-5&6)
WATERFORD, CT.

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO. 25203-32045 SH.10B		
N/A			

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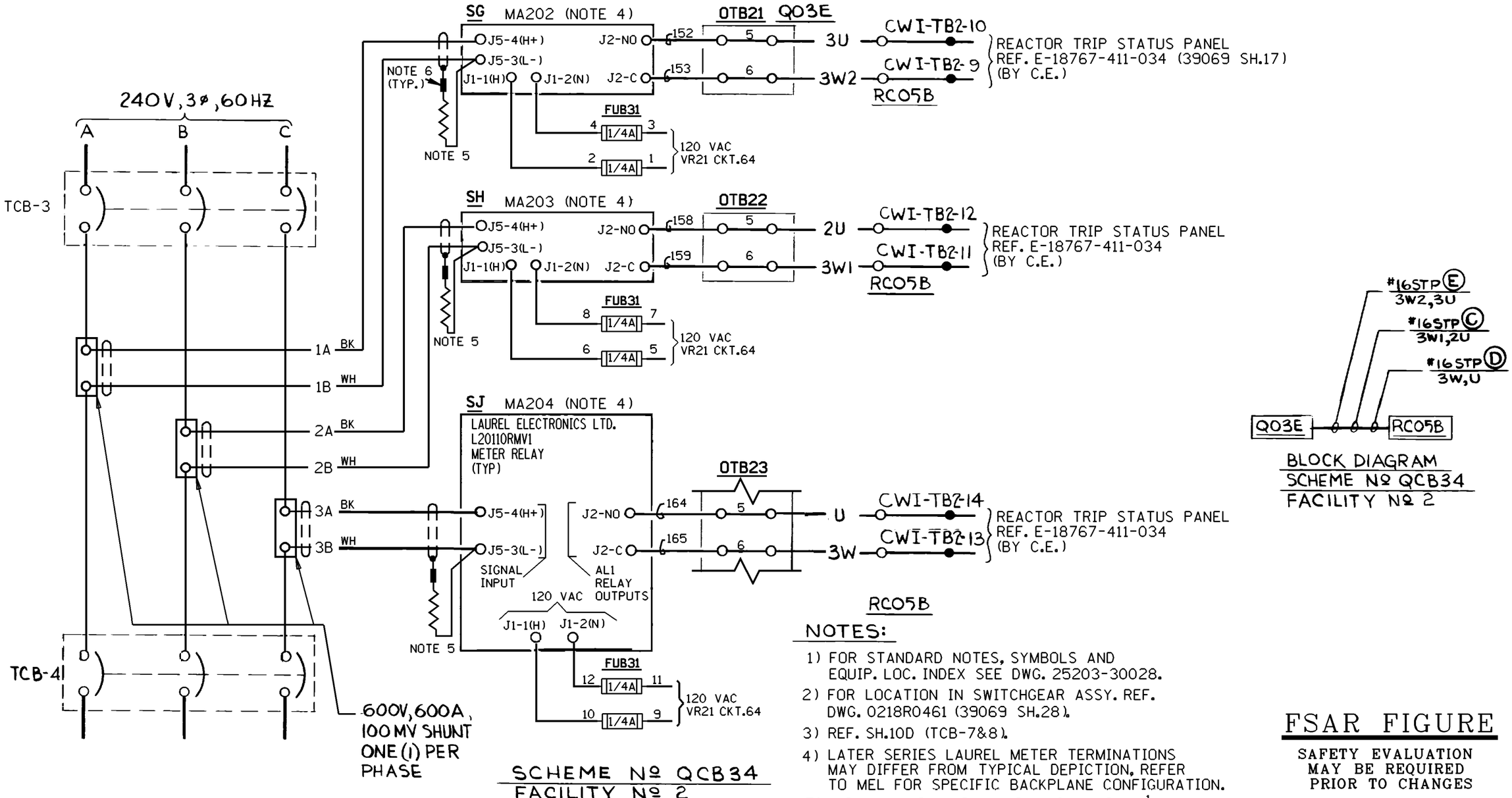

SCHEME Nº QCB34
FACILITY Nº 2

RC05B

NOTES:

1) FOR STANDARD NOTES, SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-30028.
2) FOR LOCATION IN SWITCHGEAR ASSY. REF. DWG. 0218R0461 (39069 SH.28).
3) REF. SH.10D (TCB-7&8).
4) LATER SERIES LAUREL METER TERMINATIONS MAY DIFFER FROM TYPICAL DEPICTION, REFER TO MEL FOR SPECIFIC BACKPLANE CONFIGURATION.
5) TWISTED PAIR SHIELD WITH 10 MEGOHM, $\frac{1}{4}$ WATT RESISTOR DOES NOT TIE TO CABINET GROUND.
6) ━ DENOTES FIELD SPLICE.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

REFERENCE DRAWINGS

30022 SH.22 - VR21
39069 SH.47 - METER RELAY
39069 SH.50 - REMOTE CONNECTIONS
31105 - Q03 CONN. DIAG.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
7	7/28/11	INCORP DCN DM2-04-0183-09 PER DUR 11-800309	MKN	DLW	—	DLW

NON Q.A. — CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

TITLE: MILLSTONE UNIT No.2
REACTOR TRIP SWITCHGEAR
UNDERCURRENT DEVICE (TCB-3&4)
WATERFORD, CT.

Dominion — Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO.	25203-32045 SH.10C	
N/A			

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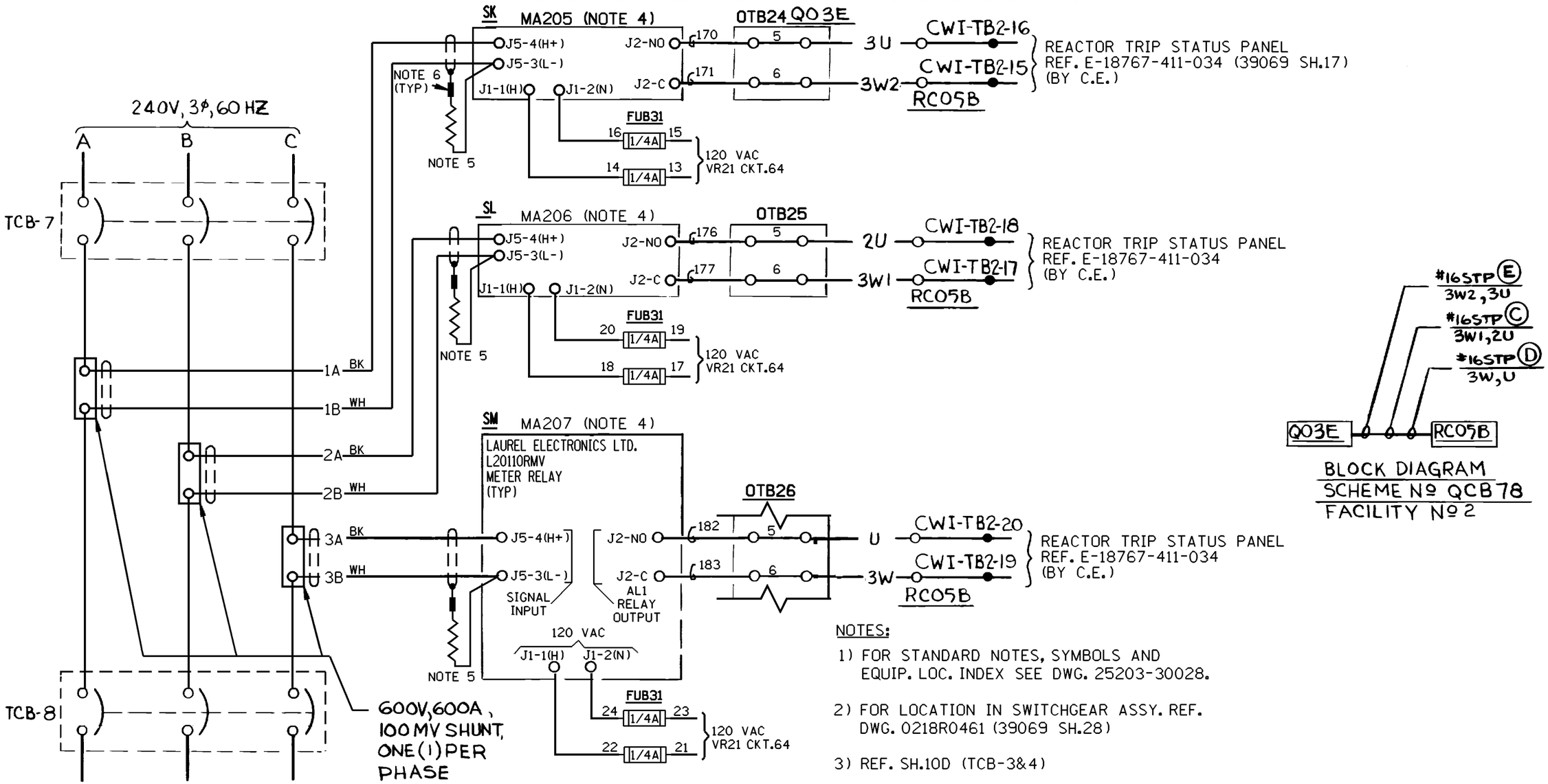


SCHEME Nº QCB78
FACILITY Nº 2

NOTES:

1) FOR STANDARD NOTES, SYMBOLS AND EQUIP. LOC. INDEX SEE DWG. 25203-30028.

2) FOR LOCATION IN SWITCHGEAR ASSY. REF. DWG. 0218R0461 (39069 SH.28)

3) REF. SH.10D (TCB-3&4)

4) LATER SERIES LAUREL METER TERMINATIONS MAY DIFFER FROM TYPICAL DEPICTION, REFER TO MEL FOR SPECIFIC BACKPLANE CONFIGURATION.

5) TWISTED PAIR SHIELD WITH 10 MEGOHM, $\frac{1}{4}$ WATT RESISTOR DOES NOT TIE TO CABINET GROUND.

6) ━ DENOTES FIELD SPLICE.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

REFERENCE DRAWINGS

30022 SH.22 - VR21
39069 SH.48 - METER RELAY
39069 SH.50 - REMOTE CONNECTIONS
31105 - Q03 CONN. DIAG.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
7	7/28/11	INCORP DCN DM2-04-0183-09 PER DUR 11-800309	MKN	DLW	—	DLW

NON Q.A.

CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

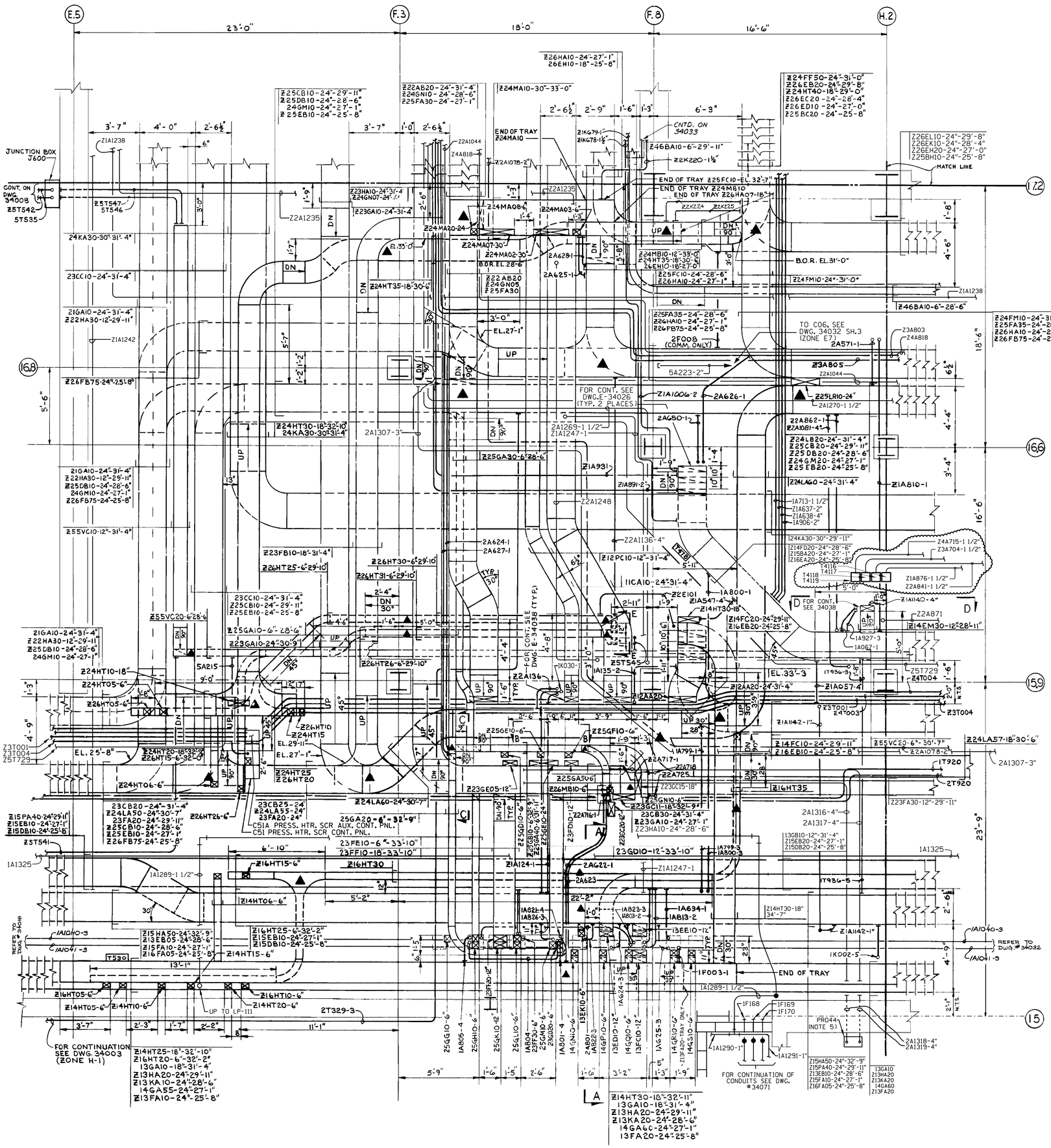
TITLE MILLSTONE UNIT No.2
REACTOR TRIP SWITCHGEAR
UNDERCURRENT DEVICE (TCB-7&8)
WATERFORD, CT.

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO.	25203-32045 SH.10D	
N/A			

$FILE$
$DATE$



NOTES:

1. FOR GENERAL NOTES SEE DWG. E-34026
2. FITTINGS FOR TRAYS 21GA10 & 11CA10 SHALL BE 24" RADIUS. ALL OTHER FITTINGS SHALL BE 18" RADIUS.
3. BOTTOM OF TRAY RISER IS ABBREVITED B.O.R.
4. UNLESS NOTED OTHERWISE B.O.R. SHALL BE AT EL. OF ITS ASSOCIATED TRAY. EXAMPLE: B.O.R. FOR 14GP10 IS EL. OF TRAY 14GA60, THUS EL. 27'-1"
5. REFERENCE 25203-34027 SH. 2 SECTION A-A.

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION UNIT No. 2
RACEWAY PLAN CABLE VAULT AREA-6, EL. 25'-6"

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C34031.MYB	
REVISED PER DC MP2-20-01108 & DUR 21-800277	DRAWING NO: 25203-34031	REV 17
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 0

MBLOGIN_anne694
18-JAN-2022 09:59
C34031.MYB

25203-34032 SH-0000000

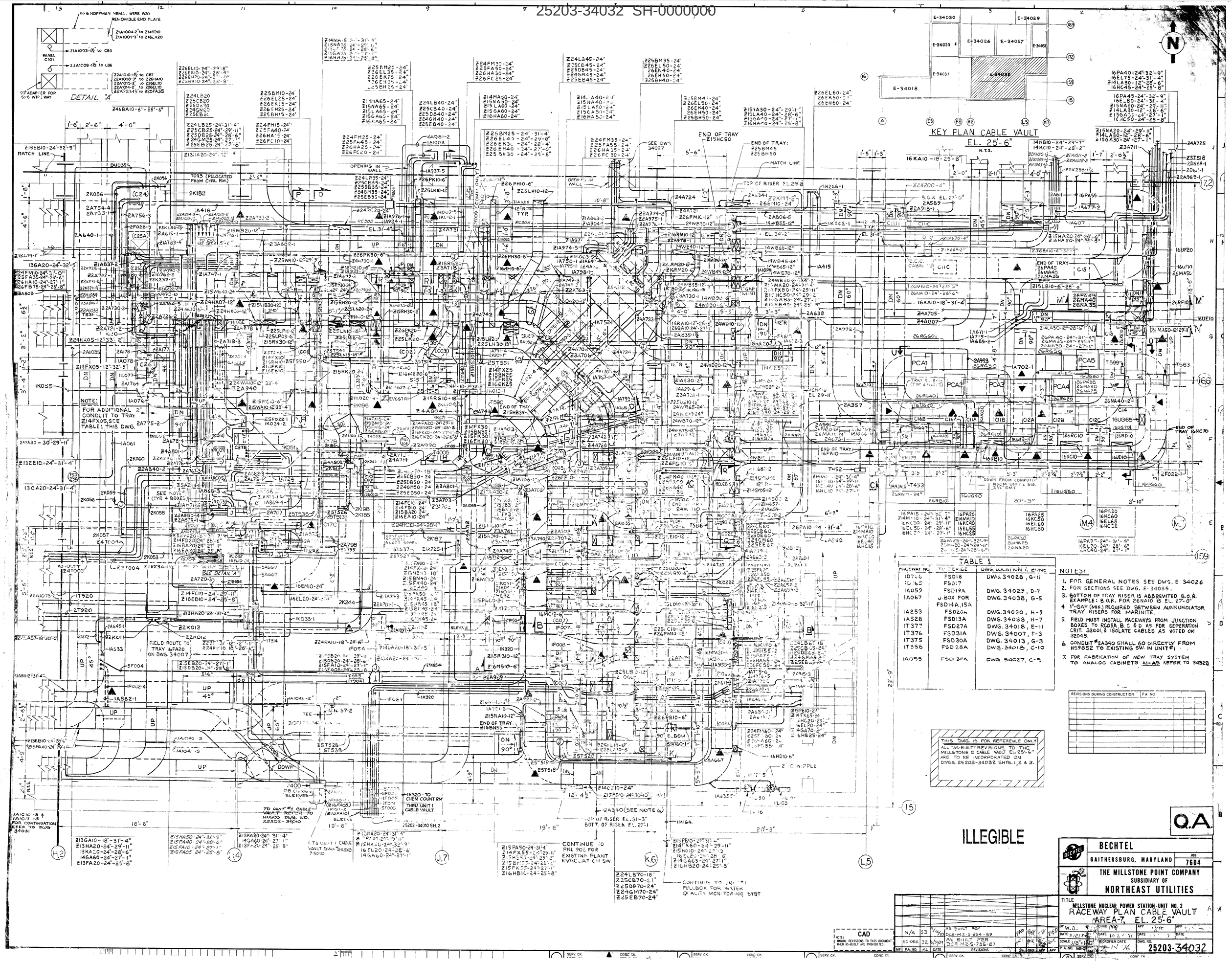


TABLE 1

RACEWAY NO.	TO DEVICE	DWG. LOCATION & ZONE
1D766	FSD18	DWG. 34028, G-11
1D765	FSD17	
1A059	FSD19A	DWG. 34029, D-7
1A067	J.BOX FOR FSD14A, 15A	DWG. 34038, G-5
1A253	FSD20A	DWG. 34030, H-9
1A528	FSD13A	DWG. 34038, H-7
1T377	FSD27A	DWG. 34018, E-11
1T376	FSD31A	DWG. 34007, F-3
1T375	FSD30A	DWG. 34013, G-3
1T388	FSD28A	DWG. 34018, C-10
1A053	FSD24A	DWG. 34027, C-5

NOTES:

1. FOR GENERAL NOTES SEE DWG. E 34026
2. FOR SECTIONS SEE DWG. E-34035.
3. BOTTOM OF TRAY RISER IS ABBREVITED B.O.R. EXAMPLE: B.O.R. FOR 26NA10 IS EL. 27'-0"
4. 1"-GAP (MIN.) REQUIRED BETWEEN AUNNUNCIATOR TRAY RISERS FOR MARINITE.
5. FIELD MUST INSTALL RACEWAYS FROM JUNCTION BOXES TO RC05A, B, C, & D AS PER SEPERATION CRIT. 33001, & ISOLATE CABLES AS NOTED ON 32045.
6. CONDUIT #2A340 SHALL GO DIRECTLY FROM HS9852 TO EXISTING SW. IN UNIT #1
7. FOR FABRICATION OF NEW TRAY SYSTEM TO ANALOG CABINETS A1-A5 REFER TO 34328

REVISIONS DURING CONSTRUCTION	P.A. NO.

THIS DWG. IS FOR REFERENCE ONLY
ALL "AS-BUILT" REVISIONS TO THE MILLSTONE II CABLE VAULT EL. 25'-6" ARE TO BE INCORPORATED ON DWGS. 25203-34032 SHTS. 1, 2 & 3.

ILLEGIBLE

Q.A.

BECHTEL
GAITHERSBURG, MARYLAND
JOB 7604

THE MILLSTONE POINT COMPANY
SUBSIDIARY OF
NORTHEAST UTILITIES

TITLE
MILLSTONE NUCLEAR POWER STATION-UNIT NO. 2
RACEWAY PLAN CABLE VAULT
AREA-7, EL. 25'-6"

MF	P.A. NO.	N.J.	DATE	REVISIONS	BY	CHK	APP	APP
	N/A	53	9/24/87	AS BUILT PER DCR-M2-S-254-89	FAB			
	85-082	32	6/30/89	AS BUILT PER DCR-M2-S-735-87				

CAD
NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

BY W.B. | CHKD DWG | APP | APP
DATE 2/2/87 | DATE 10/2/87 | DATE | DATE
SCALE 3/8"=1'-0" | MICROFILM DATE | DWG. NO. 25203-34032
P.A. NO. N69-0[illegible]

ENCLOSE BLD. WALL

CONTINUE CONDUITS TO EAST PENT. ROOM SEE DWG. 34029 (ZONE F-13)

SECTION O-O

KEY PLAN EL 14'-6"
N.T.S.

CONTAINMENT
AUX. BLDG.
TURBINE BLDG.
℄ TURB.

STEAM GENERATOR

PRESSURIZER

PRESSURIZER HTR. DIST. PNL'S.

℄ CONTAINMENT

(EAST) ℄ ELECT. PENETRATION ROW (TYP)

(WEST) ℄ ELECT. PENETRATION ROW (TYP)

SEE NOTE 4

SEE NOTE 1 (TOP TRAY ONLY)

WELL NUMBER (TYP)

NOTES:
1. ALL FITTINGS FOR THIS TRAY RUN SHALL BE 24" RADIUS.
2. FOR GENERAL NOTES AND REF. DWG'S. SEE DWG. 34040.
3. [E] DEVICES SHOWN WITH THIS SYMBOL ARE FIELD LOCATED, APPROX. LOCATION IS SHOWN.
4. TRAY IN THIS AREA TO HAVE TOP AND BOTTOM COVERS.
5. FOR FIRE DETECTION CONDUIT SEE DWG# - 34306 SH.1-6
6. EQUIPMENT IS ABANDONED IN PLACE.

FSAR FIGURE
SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION UNIT NO. 2
RACEWAY PLAN
CONTAINMENT AREA EL.14'-6"

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C34042.MYB	
REVISED PER DC MP2-14-01199 & DUR 15-800480	DRAWING NO: 25203-34042	REV 27
DSGN JSJ	SCALE: N/A	UNLESS OTHERWISE NOTED SH 0

PLAN VIEW MCC B-13 N.T.S.

PLAN VIEW MCC B-42 N.T.S.

SECTION A-A (H-10)

SECTION K-K (F-8) N.T.S.

KEY PLAN N.T.S.

SECTION W-W

SECTION E-E (H-13)

SECTION J-J (H-13)

SECTION H-H (H-8)

SECTION C-C (H-8)

SECTION D-D (F-7)

SECT 1-1 (B-7) & DETAIL N.T.S.

DETAIL "A" N.T.S.

SECTION B-B (G-11) N.T.S.

SECTION G-G (E-10)

SECTION F-F (F-12)

NOTES:

1. SEE DWG. #25027 FOR CONDUIT ASSOCIATED WITH CATHODIC PROTECTION EQUIPMENT.
2. PROVIDE MARINITE BARRIERS AS REQUIRED BY DWG. 25203-34001 RACEWAY NOTES SYMBOLS & DETAILS.
3. ALL TRAYS HAVE 18" RADIUS UNLESS OTHERWISE NOTED.
4. FOR LIGHTING, GROUNDING & COMM. PLAN SEE DWG. 35022.
4A. CONDUIT FROM AE9794 TO TRAY 24HA10 & ALL ASSOCIATED TERMINAL BOX LOCATIONS HAVE NOT BEEN VERIFIED.
5. FOR VENTILATION SYSTEM DUCT WORK SEE DWG. 24063
6. THESE PANELS TO BE MOUNTED AT EL. 24'-0" (BOT. OF PNL.)
7. FIELD TO FABRICATE MC & PLATES AS REQUIRED.
8. COND & TRAY SUPPORT LOCATION DETERMINED BY FLD. MAXIMUM SPACING BETWEEN SUPPORTS SHALL NOT EXCEED 6'-0", USE MIN OF 3 SUPPORTS.
9. DRILLED IN CONCRETE ANCHORS MUST CLEAR EXISTING REINFORCING DRILL HOLES IN CONCRETE TO GIVEN TOLERANCES & TEMPLATE BASE PLATE. ANY DEVIATION MUST HAVE STRUCTURAL ENGINEERING APPROVAL.
10. ADDED 2SWP-P1 (E-12) 2JB-6003 (H-12) & ASSOC'D CONDUITS DETAIL A(B-7), SECTION 1-1(B-10) CONDUITS 1W154, 1W155, 1W156 1W157, 1W158, NOTES 7, 8, 9 & 10 REVISED TRAY Z13VC10(F13)

NOTES: (CONT.)

REFERENCE DRAWINGS

1. 25203-11031 INTAKE STRUCTURE-ROOF PLAN AND SECTIONS.
2. 25203-20090 INTAKE STRUCTURE-PIPING PLAN EL. 14'-0".
3. 25203-20092 INTAKE STRUCTURE-PIPING DETAILS.
4. 25203-34062 RACEWAY PLAN OUTSIDE AREAS UNDERGROUND DUCT
5. 25203-35022 LIGHTING GROUNDING AND COMM PLAN INTAKE STRUCTURE
6. 25203-26008 P&ID
7. 25203-35027 CATHODIC PROTECTION INTAKE STRUCTURE
8. 25203-34001 RACEWAY NOTES, SYMBOLS & DETAILS.
9. 12552-EE-49F CONDUIT, GROUNDING, & LIGHTING (25203-34325) UNIT 2 STRAINER HOUSE

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE POWER STATION-UNIT NO.2
RACEWAY PLAN
INTAKE STRUCTURE

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C34060.MYB	
REVISED PER DC MPG-15-01053 & DUR 17-800234	DRAWING NO: 25203-34060	REV 25
DSGN MKN	SCALE: 1/4"=1'-0" UNLESS OTHERWISE NOTED	SH 0

$FILE$
$DATE$

RC30A
PT-102A
HIGH ALARM CONTACTS
(28500 SH.53A,B)

RC30B
PT-102B
HIGH ALARM CONTACTS
(28500 SH.54A,B)

RC30C
PT-102C
HIGH ALARM CONTACTS
(28500 SH.56A,B)

RC30D
PT-102D
HIGH ALARM CONTACTS
(28500 SH.57A,B)

RC30A
R.R.S. CH."X"
LOW ALARM CONTACT
(28500 SH.16)

RC30B
R.R.S. CH."Y"
LOW ALARM CONTACT
(28500 SH.16)

Ⓖ Ⓗ Ⓙ Ⓚ

C100
DIVERSE
SCRAM
MATRIX
(32012 SH.48)

Ⓕ Ⓐ

C05
DIVERSE SCRAM
RELAYS
94A DSS 94B DSS
(32012 SH.48)

C29
DSS
INTERPOSING
RELAY
94X/MG1
(32012 SH.48A)

✳

MG#1 (C29)
CONTACTOR TRIP
(39055 SH.15A)

C30
DSS
INTERPOSING
RELAY
94X/MG2
(32012 SH.48A)

▲

MG#2 (C30)
CONTACTOR TRIP
(39055 SH.15B)

Z1
AUX. FEED
INITIATION
(32012 SH.44)

Z2
AUX. FEED
INITIATION
(32012 SH.45)

SCHEME No. NM01
FACILITY Z5
(SEE NOTE 3)

NOTES:

1. ✳ REF. BLOCK DIAG. M-G SET No.1 37005,SH.21
2. ▲ REF. BLOCK DIAG. M-G SET No.2 37005,SH.22
3. REF. DWG. 32012,SH.46 FOR WIRE Nos.
4. FOR ANNUNCIATOR BLOCK DIAG. 32012,SH.48
5. ● REF. DWG. DIVERSE SCRAM 32012,SH.48A

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
4	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Q.A. CAD MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED

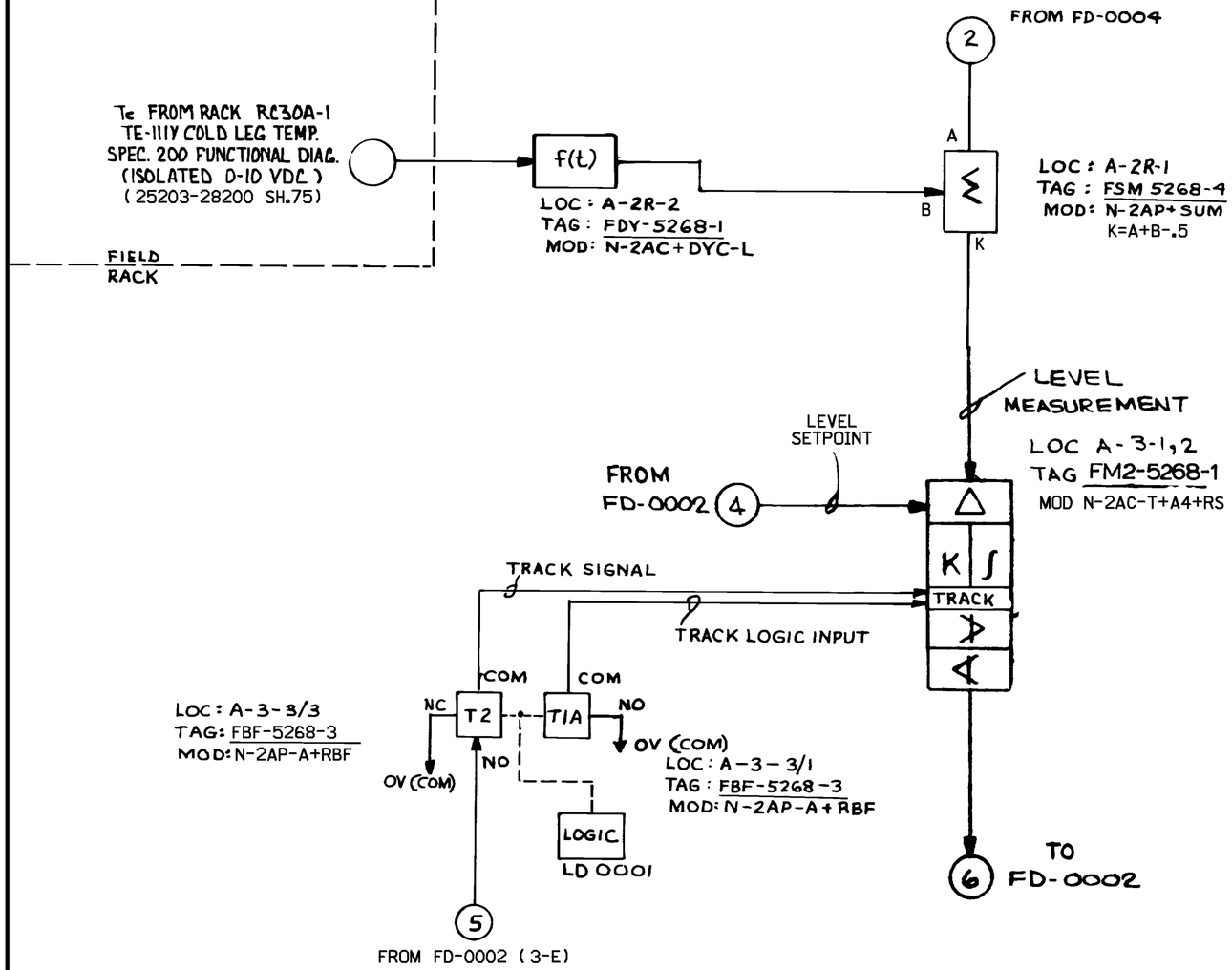
TITLE MILLSTONE UNIT 2
BLOCK DIAGRAM
DIVERSE SCRAM SYSTEM (A.T.W.S.)
WATERFORD, CONNECTICUT

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

BY T.A.SWIFT	CHKD. MEL	APP. MCH	APP.
DATE 7/21/87	DATE 11-12-87	DATE 11-13-87	DATE
SCALE NONE	DWG. NO. 25203-37005 SH.105		
80-082			

$FILE$
$DATE$



FSAR FIGURE

SAFETY EVALUATION
MAY BE REQUIRED
PRIOR TO CHANGES

NON Q.A.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

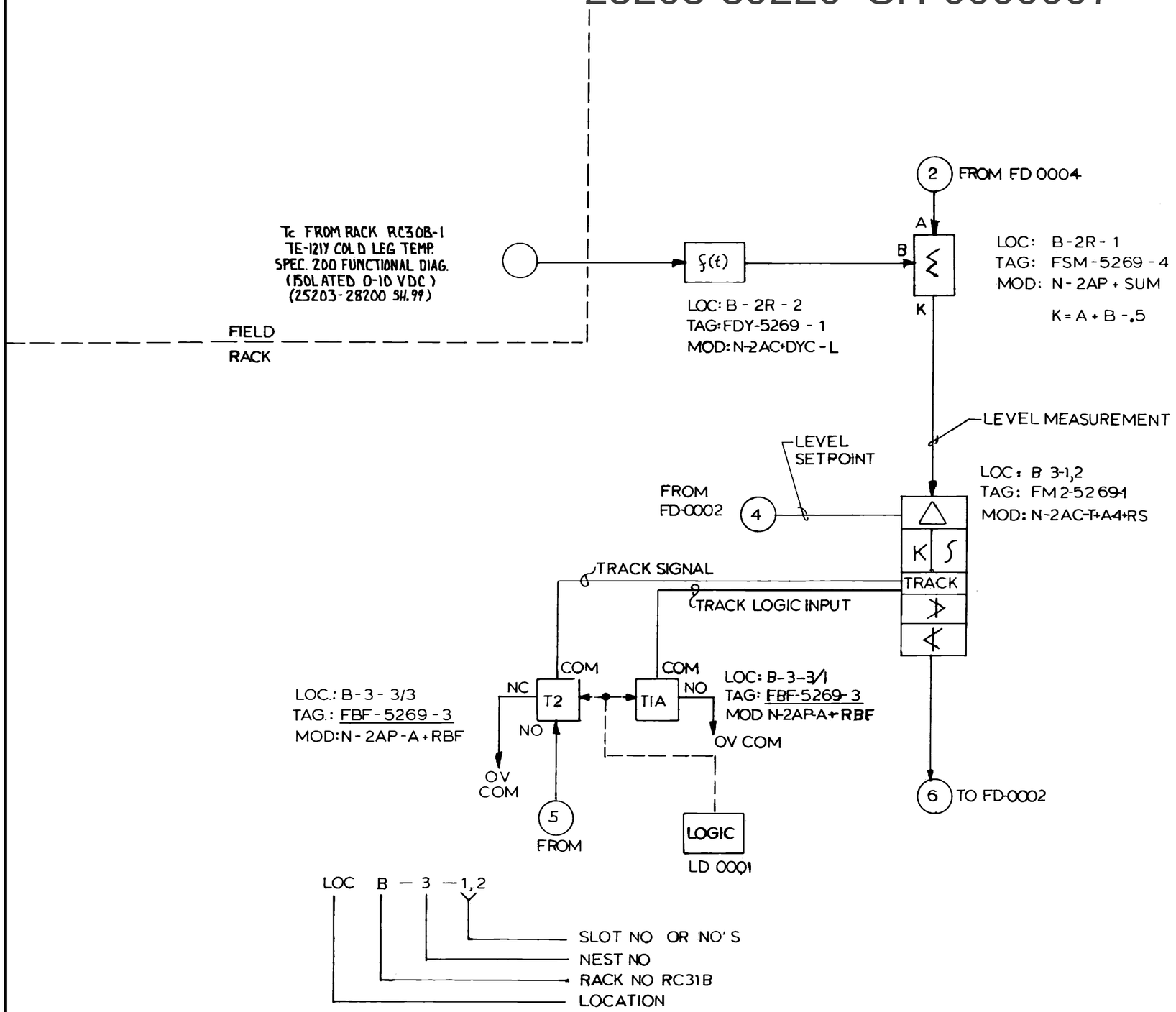
NO.	DATE	REVISIONS	BY	CHK	APP	APP
4	2-29-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "A" SINGLE ELEMENT CONTROL FUNCTIONAL DIAGRAM
WATERFORD. CT.

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO.	25203-39220 SH.7A	
N/A			

VENDOR DWG. No.
82N56704-FD-0001 REV.3



FSAR FIGURE

SAFETY EVALUATION
MAY BE REQUIRED
PRIOR TO CHANGES

NON Q.A.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
4	2-29-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "B" SINGLE ELEMENT CONTROL FUNCTIONAL DIAGRAM
WATERFORD, CT.

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO. 25203-39220 SH.7B		
N/A			

VENDOR DWG. No.
82N56704-FD-0001

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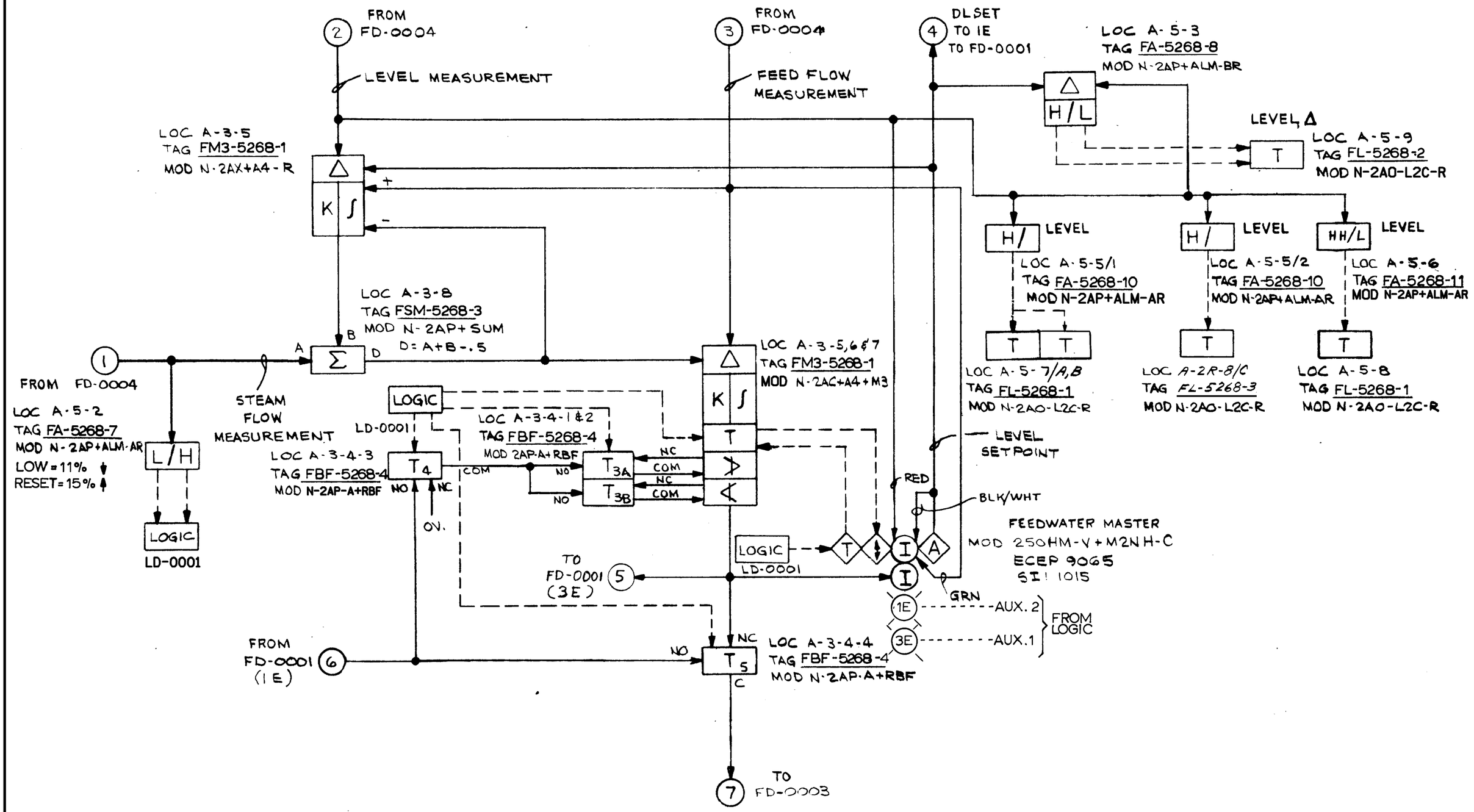


FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
EWR 2-93-00125	2	INCORP. DCN DM2-S-0321-94 PER DCR M2-S-0277-94	JMM	MM DME	JPP	JPP

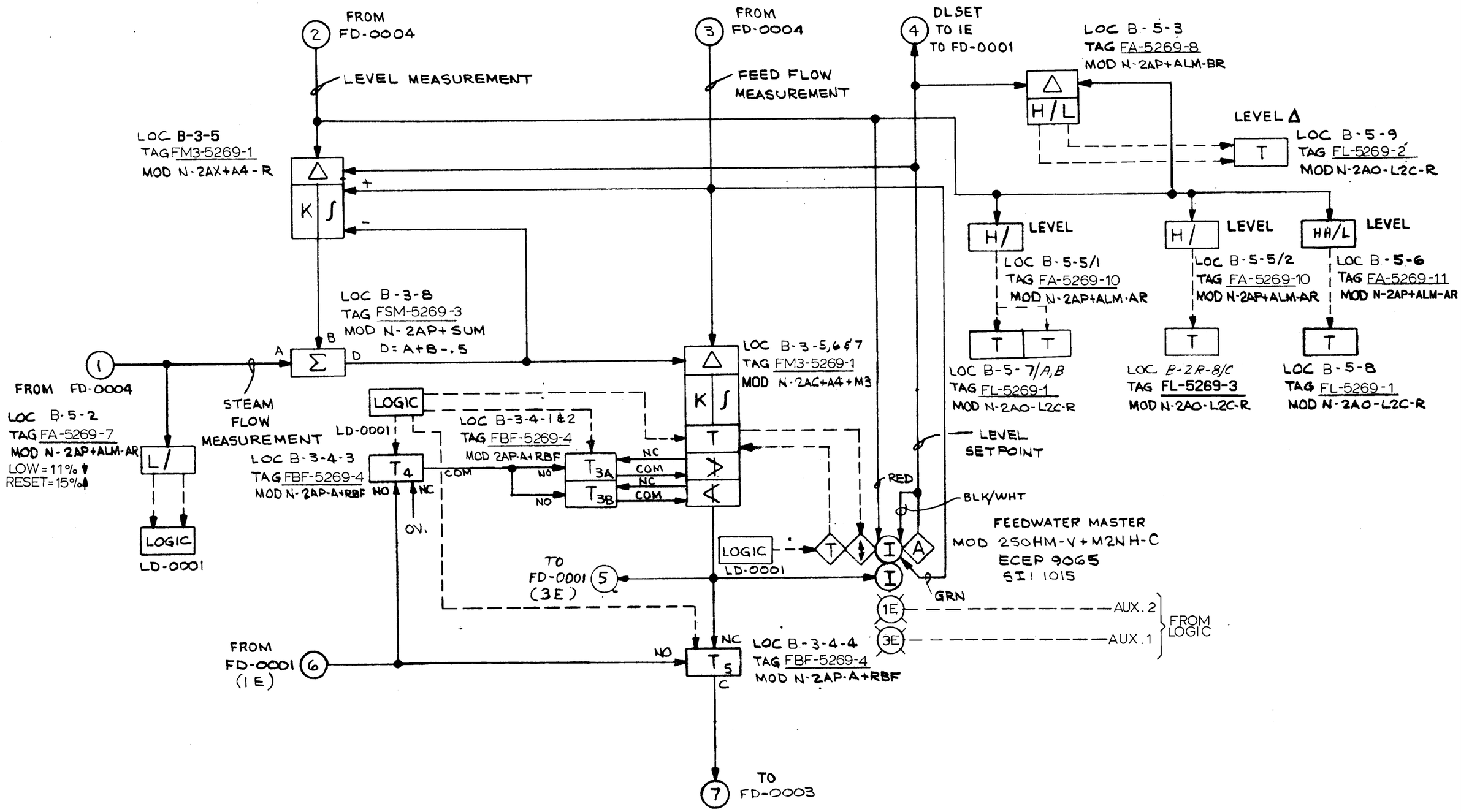
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
FEEDWATER CONTROL SYSTEM "A" THREE ELEMENT CONTROL FUNCTIONAL CONTROL
WATERFORD, CT.

BY JMM	CHKD. ——	APP. ——	APP. ——
DATE 5-2-95	DATE	DATE	DATE
SCALE N/A	DWG. NO. 25203-39220 SH. 8A		

$FILE$
$DATE$

$FILE$
$DATE$



FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

CAD

NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP
2 EWR 2-93-00125	5-2-95	INCORP. DCN DM2-S-0321-94 PER DCR M2-S-0277-94	JMM	MM DME	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT 2
FEEDWATER CONTROL SYSTEM "B" THREE ELEMENT CONTROL FUNCTIONAL CONTROL
WATERFORD, CT.

BY	CHKD.	APP.	APP.
JMM	—	—	—
DATE 5-2-95	DATE	DATE	DATE
SCALE N/A	DWG. NO.	25203-39220 SH. 8B	

FIELD

RACK

LOC A-4-4
TAG FBF-5268-5
MOD N-2AP-A+RBF

T10

RAMP DOWN
FOLLOW
100% 0%

LOC A-4-3
TAG FPG-5268-1
MOD N-2AP+RPG
RAMP CLOSE
ON TRIP

LOGIC
LD-0001

LOC A-4-4-1&2
TAG FBF-5268-5
MOD 2AP-A+RBF

NO NC T7 COM NO T6A NC COM NO T6B NC COM

0V.

LOC A-4-4-3
TAG FBF-5268-5
MOD N-2AP-A+RBF

0 C 0 10 100 LOAD

7 FROM FD-0002

K

LOC A-4-1
TAG FDL-5268-1
MOD N-2AP+DLS
K=1 (A-0.075)

LOC A-4-5,6&7
TAG FAM-5268-1
MOD N-2AC+M3+AM

A/M T A I

RED POINTER

FROM LOGIC AUX 2

R T I I

B/W POINTER

MAIN F.W.
TAG LIC-5268
MOD N-250HE+M2NH-F

LOC A-4-8
TAG FBF-5268-6
MOD N-2AP-A+RBF

T11

RAMP UP
FOLLOW

RAMP TO 40 % ON TRIP
NOTE: FOR JUMPER SELECTIONS SEE TABLE BELOW.

LOC A-4-9
TAG FPG-5268-2
MOD N-2AP+RPG

SET @ 4.0V

LOGIC
LD-0001

LOC A-4-8-1&2
TAG FBF-5268-6
MOD 2AP-A+RBF

NO NC T9 COM NO T8A NC COM NO T8B NC COM

0V.

LOC A-4-8-3
TAG FBF-5268-6
MOD N-2AP-A+RBF

LESS THAN OR EQUAL TO (-)0.5 VDC

0 C 0 10 LOAD

K

LOC A-4-1 RAMP OPEN
TAG FDL-5268-1
MOD N-2AP+DLS
K=2.71A

T12

HS
LOGIC
LD-0001

RAMP DOWN
100% 0%
RAMP CLOSE
ON HAND SWITCH SELECTION (HS-5215)

LOC A-4-2
TAG FSL-5268-2
MOD N-2AP+SSL

LOC A-5-1
TAG FPG-5268-3
MOD N-2AP+RPG

LOC A-4-7
TAG FAM-5268-1
MOD N-2AX+AM

A/M T A I

RED POINTER

BYPASS F.W.
TAG LIC-5215
MOD N-250HE+M2NH-F

T I I

B/W POINTER

LOC A-5-4
TAG FA-5268-9
MOD N-2AP+ALM-AR
HIGH = 108% (2.128V)
LOW = -8% (0.272V)

H/L

.4 TO 2.0 VDC
100 OHMS

V/I LOC: B-4-10 TAG: FV-5268-2 MOD: N-2AO-V2I

V/I LOC: A-6-4 TAG: FV-5268-4 MOD: N-2AO-VAI

HART FILTER TAG: EY-5268A

HART FILTER TAG: EY-5268B

100 OHMS
0.4 TO 2.0 VDC

H/L
LOC: A-1-4
TAG: FA-5268-13
MOD: N-2AP+ALM-AR
HIGH = 108% (2.128V)
LOW = -8% (0.272V)

LOC A-5-10
TAG FL-5268-2
MOD N-2AO-L2C-R

T

LOC: A-5-10
TAG: FL-5268-2
MOD: N-2AO-L2C-R

LOC A-4-10
TAG FV-5268-2
MOD N-2AO-V2I

V/I

HART FILTER TAG: EY-5215

RACK

FIELD

CONTACT OPEN ON SIGNAL FAILURE (VALVE LOCKUP) 32012 SH.31A

CONTACT OPEN ON SIGNAL FAILURE (VALVE LOCKUP) 32012 SH.31A

POSITION BYPASS MAIN OUTPUT
VALVE SEQUENCING

MSI & MSI OVERRIDE (32012 SH.31A)
4-20MA NF K6 43X SEE NOTE 1
NF

S F.O. A/S DIGITAL POSITIONER F.C.
BY-PASS FEED VALVE

(32012 SH.31A)
SEE NOTE 1 NF K6
4-20MA 4-20MA

S F.O. A/S F.C.
S F.O. A/S F.C.
S F.O. A/S F.C.
LOCK-UP SOLENOIDS

FV-5268-POS-B
FV-5268-POS-A
DIGITAL POSITIONERS
A/S
A/S
SWITCHOVER VALVE

REGULATING VALVE (FAIL AS IS) VALVE SHOWN OPEN

MSI 3-1-A
(32012 SH.31A)

FPG-5268-2 (JUMPER SELECTION)

JUMPER	FUNCTION		POSITION
J1	RAMP 1 UP OR DOWN:	RAMP UP	P1 TO P3
		RAMP DOWN	P1 TO P2
J2	RAMP 2 UP OR DOWN:	RAMP UP	P4 TO P5
		RAMP DOWN	P4 TO P6
J3	RAMP 1 RATE ADJUSTMENT:	INTERNAL	P7 TO P8
		EXTERNAL	P7 TO P9
J4	RAMP 2 RATE ADJUSTMENT:	INTERNAL	P10 TO P11
		EXTERNAL	P10 TO P12
J5	RAMP 1 RATE MULTIPLIER:	X1	P13 TO P16
		X10	P13 TO P15
		X100	P13 TO P14
J6	RAMP 2 RATE MULTIPLIER	X1	P17 TO P20
		X10	P17 TO P19
		X100	P17 TO P18
J7	OUTPUT RANGE SELECT:	RANGE 1	P21 TO P23
		RANGE 2	P22 TO P23
J8	HIGH-LIMIT ADJUSTMENT:	INTERNAL	P24 TO P26
		EXTERNAL	P24 TO P25
J9	LOW-LIMIT ADJUSTMENT:	INTERNAL	P27 TO P28
		EXTERNAL	P27 TO P29
J10	POWER SUPPLY FAULT RECOVERY:	ACTIVE	P30 TO P31
		INACTIVE	P31 TO P32

NOTE:

1. EXISTING CUSTOMER CURRENT FAIL TO SYSTEM. UPON RECEIPT OF MSI SIGNAL, MAIN FEED VALVE AND BYPASS FEED VALVE SIGNAL IS DRIVEN LOW BY GROUNDING THE INPUT TO FV-5268-2.

FOXBORO DWG. No. 82N56704 FD 0003 REV.03
CUST. ORDER No. 818040

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

Dominion Energy®

Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Waterford, CT.

TITLE
MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "A"
AUTO/MANUAL VALVE CONTROL
FUNCTIONAL DIAGRAM

REVISION DESCRIPTION (SIGNATURES ON FILE)	CAD NO: C392209A.HYB	
REVISED PER DC MP2-22-01088 & DUR 24-800176	DRAWING NO: 25203-39220	REV 9
DSGN JCV	SCALE: NONE	UNLESS OTHERWISE NOTED SH 9A

Authenticated By: MBULOGIN\jame694
20-NOV-2024 12:59
C392209A.HYB

FIELD

RACK

LOC B-4-4
TAG FBF-5269-5
MOD N-2AP-A+RBF

T10

FOLLOW | RAMP DOWN

LOC B-4-3
TAG FPG-5269-1
MOD N-2AP-A+RPG
RAMP CLOSE ON TRIP

LOGIC

LD-0001

LOC B-4-4-1&2
TAG FBF-5269-5
MOD 2AP-A+RBF

T7

0V.

LOC B-4-4-3
TAG FBF-5269-5
MOD N-2AP-A+RBF

0 C 0 10 100 LOAD

7 FROM FD-0002

K

LOC B-4-1
TAG FDL-5269-1
MOD N-2AP+DLS
K = 1(A-0.075)

LOC B-4-5,6&7
TAG FAM-5269-1
MOD N-2AC+M3+AM

A/M

T
I
I

RED POINTER

B/W POINTER

FROM LOGIC AUX 2

R

MAIN F.W.
TAG LIC-5269
MOD N-250HE+M2NH-F

LOC B-4-8
TAG FBF-5269-6
MOD N-2AP-A+RBF

T11

RAMP UP | FOLLOW

RAMP TO 40% ON TRIP
NOTE: FOR JUMPER SELECTIONS SEE TABLE BELOW.

LOC B-4-9
TAG FPG-5269-2
MOD N-2AP-A+RPG

SET @ 4.0V

LOGIC

LD-0001

LOC B-4-8-1&2
TAG FBF-5269-6
MOD 2AP-A+RBF

T9

T8A

T8B

0V.

LOC B-4-8-3
TAG FBF-5269-6
MOD N-2AP-A+RBF

LESS THAN OR EQUAL TO (-)0.5 VDC

0 C 0 10 LOAD

K

LOC B-4-1
TAG FDL-5269-1
MOD N-2AP+DLS
K=2.71A

T12

HS LOGIC LD-0001

RAMP OPEN | RAMP DOWN

100% 0%

RAMP CLOSE ON HAND SWITCH SELECTION (HS-5216)

LOC B-5-1
TAG FPG-5269-3
MOD N-2AP-A+RPG

LOC B-4-2
TAG FSL-5269-2
MOD N-2AP+SSL

LOC B-4-7
TAG FAM-5269-1
MOD N-2AX+AM

A/M

T
I
I

RED POINTER

BYPASS F.W.
TAG LIC-5216
MOD N-250HE+M2NH-F

B/W POINTER

LOC B-5-4
TAG FA-5269-9
MOD N-2AP+ALM-AR
HIGH = 108% (2.128V)
LOW = -8% (0.272V)

H/L

.4 TO 2.0 VDC

100 OHMS

LOC B-5-10
TAG FL-5269-2
MOD N-2AO-L2C-R

T

V/I LOC: B-4-10 TAG: FV-5269-2 MOD: N-2AO-V2I

V/I LOC: B-6-4 TAG: FV-5269-4 MOD: N-2AO-VAI

HART FILTER TAG: EY-5269A

HART FILTER TAG: EY-5269B

0.4 TO 2.0 VDC

100 OHMS

H/L LOC: B-1-4 TAG: FA-5269-13 MOD: N-2AP+ALM-AR
HIGH = 108% (2.128V)
LOW = -8% (0.272V)

T LOC: B-1-5 TAG: FL-5269-3 MOD: N-2AO-L2C-R

V/I LOC B-4-10 TAG FV-5269-2 MOD N-2AO-V2I

HART FILTER TAG: EY-5216

RACK

FIELD

CONTACT OPEN ON SIGNAL FAILURE (VALVE LOCKUP) 32012 SH.32A

CONTACT OPEN ON SIGNAL FAILURE (VALVE LOCKUP) 32012 SH.32A

POSITION: BYPASS, MAIN; OUTPUT
VALVE SEQUENCING

(32012 SH.32A)

4-20MA 4-20MA

SEE NOTE 1 NF K6

F.O. S A/S

F.O. S F.C.

F.O. S F.C. F.O. S F.C.

LOCK-UP SOLENOIDS

FV-5269-POS-B

A/S

DIGITAL POSITIONERS

A/S

FV-5269-POS-A

SWITCHOVER VALVE

REGULATING VALVE (FAIL AS IS) VALVE SHOWN OPEN

MSI 3-1-B

(32012 SH.32A)

MSI & MSI OVERRIDE
(32012 SH.32A)

4-20MA NF K6 43X NF SEE NOTE 1

S F.O. DIGITAL POSITIONER A/S F.C.

BY-PASS FEED VALVE

FPG-5269-2 (JUMPER SELECTION)

JUMPER	FUNCTION		POSITION
J1	RAMP 1 UP OR DOWN:	RAMP UP	P1 TO P3
		RAMP DOWN	P1 TO P2
J2	RAMP 2 UP OR DOWN:	RAMP UP	P4 TO P5
		RAMP DOWN	P4 TO P6
J3	RAMP 1 RATE ADJUSTMENT:	INTERNAL	P7 TO P8
		EXTERNAL	P7 TO P9
J4	RAMP 2 RATE ADJUSTMENT:	INTERNAL	P10 TO P11
		EXTERNAL	P10 TO P12
J5	RAMP 1 RATE MULTIPLIER:	X1	P13 TO P16
		X10	P13 TO P15
		X100	P13 TO P14
J6	RAMP 2 RATE MULTIPLIER	X1	P17 TO P20
		X10	P17 TO P19
		X100	P17 TO P18
J7	OUTPUT RANGE SELECT:	RANGE 1	P21 TO P23
		RANGE 2	P22 TO P23
J8	HIGH-LIMIT ADJUSTMENT:	INTERNAL	P24 TO P26
		EXTERNAL	P24 TO P25
J9	LOW-LIMIT ADJUSTMENT:	INTERNAL	P27 TO P28
		EXTERNAL	P27 TO P29
J10	POWER SUPPLY FAULT RECOVERY:	ACTIVE	P30 TO P31
		INACTIVE	P31 TO P32

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NOTE:

1. EXISTING CUSTOMER CURRENT FAIL TO SYSTEM. UPON RECEIPT OF MSI SIGNAL OR HIGH STEAM GEN. LEVEL SIGNAL, THE REG. VALVE SOV'S ENERGIZE TO RAPIDLY CLOSE THE VALVE WHEREAS THE BYPASS VALVE SOV IS DEENERGIZED TO CLOSE THE VALVE.

FOXBORO DWG. No. 82N56704 FD 0003 REV.03
CUST. ORDER No. 818040

Dominion Energy° | Dominion Nuclear Connecticut, Inc. Millstone Power Station Waterford, CT.

TITLE: MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "B"
AUTO/MANUAL VALVE CONTROL
FUNCTIONAL DIAGRAM

REVISION DESCRIPTION (SIGNATURES ON FILE) | CAD NO: C392209B.HYB

REVISED PER DC MP2-22-01088 & DUR 24-800176 | DRAWING NO: 25203-39220 | REV 9

DSGN JSJ | SCALE: NONE | UNLESS OTHERWISE NOTED | SH 9B

Authenticated By: MBULOGIN\jame694
20-NOV-2024 13:01
C392209B.HYB

STEAM FLOW
PRIME
BACKUP
FT
FT

STEAM GENERATOR LEVEL
PRIME
BACKUP
LT MOD N-E13DM
174.5" - 42.3"
LT MOD N-E13DM

FEEDWATER FLOW
PRIME
BACKUP
FT
FT

FIELD

RACK OR PANEL

I/V
LOC A-1-1
TAG FY-5268-1
MOD N-2AI-I2V

I/V
LOC A-1-3
TAG FY-5268-2
MOD N-2AI-I2V

I/V
LOC A-1-7
TAG FY-5268-3
MOD N-2AI-I2V

< LOC A-1-10
TAG FSL-5268-1
MOD N-2AP+SSL

PRIME STEAM FLOW
BACKUP STEAM FLOW
A Σ/2 B
LOC A-1-8
TAG FSM-5268-1
MOD N-2AP+SUM

FROM FD-0005
9
T
PRIME/AVG/BACKUP SELECT
MOD N-2AP-A+RBF (3)
LOC A-2-10
TAG FBF-5268-2
V/I
TO STM FLOW REC
TO SGFP SPD CNTRL(C531A)
LOC A-6-4
TAG FV-5268-3
MOD 2AO-VAI

V/I
LOC: A-6-9
TAG: FV-5268-4
MOD: N-2AO-VAI
TO SGFP SPD CNTRL (C531B)

1 TO FD-0002

PRIME LEVEL
LOC A-2-8
TAG FBF-5268-1
MOD N-2AP-A+RBF
T
BACKUP LEVEL TO LEVEL REC
FROM FD-0005
8
PRIME/LOW/BACKUP SELECT
V/I
LOC A-2-9
TAG FV-5268-1
MOD N-2AO-VAI
TO
2 FD-0001
FD-0002

LOC. A-2-10
TAG FBF-5268-2
MOD N-2AP-A+RBF
B Σ/2 A
LOC: A-1-9
TAG: FSM-5268-2
MOD: N-2AP+SUM
T
TAG FBF-5268-2
MOD N-2AP-A+RBF(3)
LOC A-2-10
FROM FD-0005
10
PRIME/AVG/BACKUP SELECT
V/I
TO F/W REC
LOC A-2-9
TAG FV-5268-1
MOD 2AO-VAI
3 TO FD-0002

H/L
LOC A-2-1
TAG FA-5268-1
MOD N-2AP+ALM-AR
XMTR FAIL

H/L
LOC A-2-3
TAG FA-5268-3
MOD N-2AP+ALM-AR
XMTR FAIL

H/L
LOC A-2-2
TAG FA-5268-2
MOD N-2AP+ALM-AR
XMTR FAIL

H/L
LOC A-2-6
TAG FA-5268-5
MOD N-2AP+ALM-AR
XMTR FAIL

H/L
LOC A-2-5
TAG FA-5268-4
MOD N-2AP+ALM-AR
XMTR FAIL

H/L
LOC A-2-7
TAG FA-5268-6
MOD N-2AP+ALM-AR
XMTR FAIL

DSI
LOC A-2-4
TAG FDS-5268-1
MOD N-2AX+DSI

TO PANEL INDICATING LIGHTS (NF)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NON Q.A.

CAD
NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
5	5-10-05	INCORP. DCN DM2-00-0431-03	JSJ			
4	10-15-03	INCORP. DCN DM2-00-0251-01, DM2-00-0280-00, DM2-00-0853-99	DLW	WBZ	-	DED
3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion
Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE MILLSTONE UNIT NO. 2
FEEDWATER CONTROL SYSTEM "A"
FUNCTIONAL DIAGRAM
WATERFORD, CT

BY D.MINELLA/JSJ	CHKD.	APP.	APP.
DATE 11-20-99	DATE	DATE	DATE
SCALE -H-	DWG. NO. 25203-39220 SH. 10A		

FOXBORO DWG. NO. 82N56704FD0004

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25203-39220 SH-0000010

STEAM FLOW | STEAM GENERATOR LEVEL | FEEDWATER FLOW

PRIME | BACKUP | PRIME | BACKUP | PRIME | BACKUP

FT | FT | LT MOD N-E13DM 174.5" - 42.3" | LT MOD N-E13DM | FT | FT

FIELD

RACK OR PANEL

I/V LOC B-1-1 TAG FY-5269-1 MOD N-2AI-I2V

I/V LOC B-1-3 TAG FY-5269-2 MOD N-2AI-I2V

I/V LOC B-1-7 TAG FY-5269-3 MOD N-2AI-I2V

< LOC B-1-10 TAG FSL-5269-2 MOD N-2AP+SSL

PRIME STEAM FLOW (A) Σ/2 (B) BACKUP STEAM FLOW

LOC B-1-8 TAG FSM-5269-1 MOD N-2AP+SUM

FROM FD-0005 (9)

PRIME/AVG/BACKUP SELECT

T MOD N-2AP-A+RBF (3) LOC B-2-10 TAG FBF-5269-2

V/I TO STM FLOW REC

TO SGFP SPD CNTRL(C531B)

PRIME/AVG/BACKUP SELECT

LOC B-6-4 TAG FV-5269-3 MOD 2AO-VAI

1 TO FD-0002

V/I LOC: B-6-9 TAG: FV-5269-4 MOD: N-2AO-VAI

TO SGFP SPD CNTRL (C531A)

PRIME LEVEL

T LOC B-2-8 TAG FBF-5269-1 MOD N-2AP-A+RBF

FROM FD-0005 (8)

BACKUP LEVEL TO LEVEL REC

PRIME/LOW/BACKUP SELECT

V/I LOC B-2-9 TAG FV-5269-1 MOD N-2AO-VAI

2 TO FD-0001 FD-0002

B Σ/2 A

LOC B-2-10 TAG FBF-5269-2 MOD N-AP-A+RBF

LOC: B-1-9 TAG: FSM-5269-2 MOD: N-2AP+SUM

T TAG FBF-5269-2 MOD N-2AP-A+RBF(3) LOC B-2-10

FROM FD-0005 (10)

PRIME/AVG/BACKUP SELECT

V/I TO F/W REC

LOC: B-2-9 TAG: FV-5269-1 MOD: N-2AO-VAI

3 TO FD-0002

H/L LOC B-2-1 TAG FA-5269-1 MOD N-2AP+ALM-AR XMTR FAIL

H/L LOC B-2-3 TAG FA-5269-3 MOD N-2AP+ALM-AR XMTR FAIL

H/L LOC B-2-2 TAG FA-5269-2 MOD N-2AP+ALM-AR XMTR FAIL

H/L LOC B-2-6 TAG FA-5269-5 MOD N-2AP+ALM-AR XMTR FAIL

H/L LOC B-2-5 TAG FA-5269-4 MOD N-2AP+ALM-AR XMTR FAIL

H/L LOC B-2-7 TAG FA-5269-6 MOD N-2AP+ALM-AR XMTR FAIL

DSI LOC B-2-4 TAG FDS-5269-1 MOD N-2AX+DSI

TO PANEL INDICATING LIGHTS (NF)

FSAR FIGURE

SAFETY EVALUATION MAYBE REQUIRED PRIOR TO CHANGES

NOTE: CAD
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

NON Q.A.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
5	5-10-05	INCORP. DCN DM2-00-0431-03	JSJ	WBZ	-	JPP
4	10-15-03	INCORP. DCN DM2-00-0251-01, DM2-00-0280-00, DM2-00-0853-99	DLW	WBZ	-	DED
3	2-7-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion — Dominion Nuclear Connecticut, Inc. Millstone Power Station

TITLE: MILLSTONE UNIT NO. 2
FEEDWATER CONTROL SYSTEM "B"
FUNCTIONAL DIAGRAM
WATERFORD, CT

BY D.MINELLA/JSJ	CHKD.	APP.	APP.
DATE 11-20-99	DATE	DATE	DATE
SCALE	DWG. NO.	25203-39220 SH. 10B	

FOXBORO DWG. NO. 82N56704FD0004

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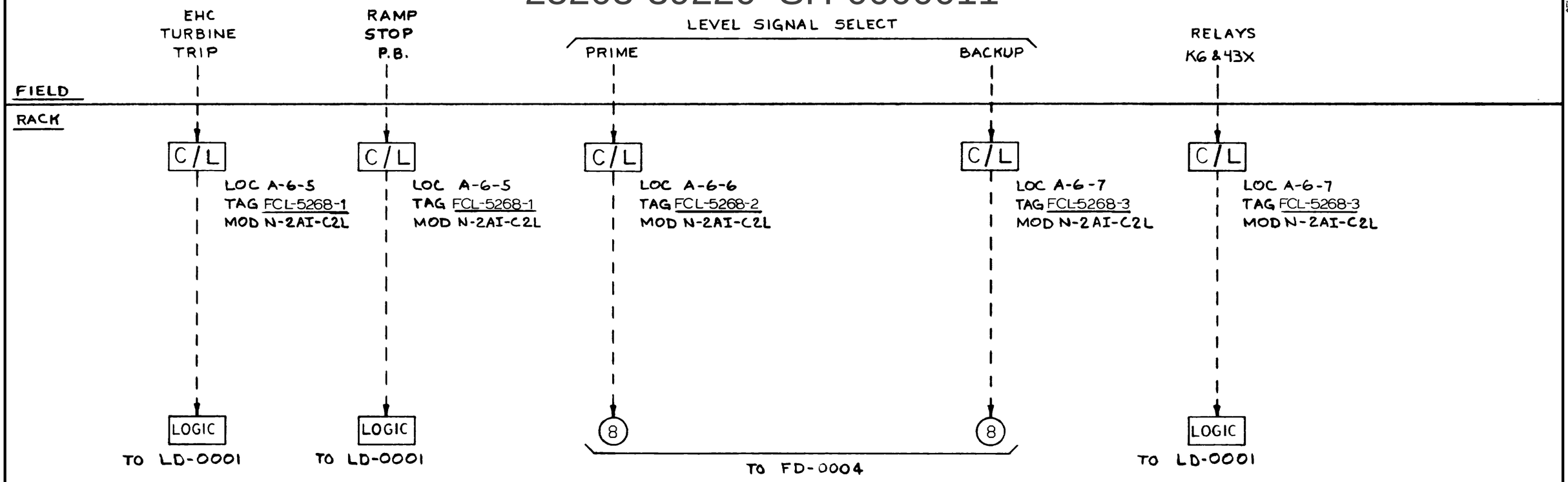
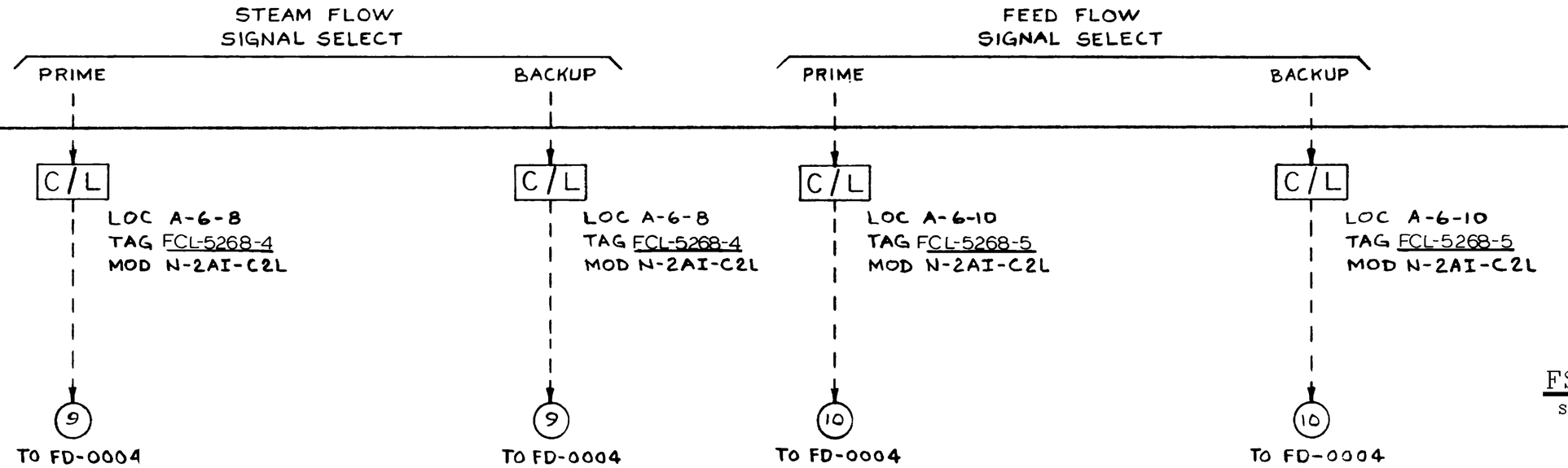
$DATE$





FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

NON Q.A.

CAD

NOTE: MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
3	2-29-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

VENDOR DWG. No.
82N56704-FD-0005 REV.2

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "A"
FUNCTIONAL DIAGRAM
WATERFORD. CT.

BY JSJ/JMM	CHKD. ——	APP. JPP	APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00	DATE 2-29-00
SCALE N/A	DWG. NO.	25203-39220 SH.11A	
N/A			

$FILE$
$DATE$

FIELD

RACK

EHC TURBINE TRIP

C/L

LOC B-6-5
TAG FCL-5269-1
MOD N-2AI-C2L

LOGIC

TO LD-0001

RAMP STOP P.B.

C/L

LOC B-6-5
TAG FCL-5269-1
MOD N-2AI-C2L

LOGIC

TO LD-0001

LEVEL SIGNAL SELECT

PRIME

C/L

LOC B-6-6
TAG FCL-5269-2
MOD N-2AI-C2L

(B)

BACKUP

C/L

LOC B-6-7
TAG FCL-5269-3
MOD N-2AI-C2L

(B)

TO FD-0004

RELAYS K6&43X

C/L

LOC B-6-7
TAG FCL-5269-3
MOD N-2AI-C2L

LOGIC

TO LD-0001

FIELD

RACK

STEAM FLOW SIGNAL SELECT

PRIME

C/L

LOC B-6-8
TAG FCL-5269-4
MOD N-2AI-C2L

(9)

TO FD-0004

BACKUP

C/L

LOC B-6-8
TAG FCL-5269-4
MOD N-2AI-C2L

(9)

TO FD-0004

FEED FLOW SIGNAL SELECT

PRIME

C/L

LOC B-6-10
TAG FCL-5269-5
MOD N-2AI-C2L

(10)

TO FD-0004

BACKUP

C/L

LOC. B-6-10
TAG FCL-5269-5
MOD N-2AI-C2L

(10)

TO FD-0004

FSAR FIGURE

SAFETY EVALUATION MAY BE REQUIRED PRIOR TO CHANGES

NON Q.A.

CAD

NOTE:
MANUAL REVISIONS TO THIS DOCUMENT WHEN AS-BUILT ARE PROHIBITED.

REVISIONS DURING CONSTRUCTION

NO.	DATE	REVISIONS	BY	CHK	APP	APP
3	2-29-00	INCORP. DCN DM2-00-0025-00	JSJ	WBZ	JPP	JPP

Dominion

Dominion Nuclear Connecticut, Inc.
Millstone Power Station

TITLE
MILLSTONE UNIT No.2
FEEDWATER CONTROL SYSTEM "B"
FUNCTIONAL DIAGRAM
WATERFORD, CT.

BY JSJ/JMM	CHKD. ——	APP. JPP / APP. JPP
DATE 2-3-00	DATE ——	DATE 2-29-00 / DATE 2-29-00
SCALE N/A / N/A	DWG. NO.	25203-39220 SH.11B

VENDOR DWG. No.
82N56704-FD-0005 REV.2

Millstone Power Station Unit 2
Technical Requirements Manual

(Incorporated by Reference)

THIS IS A CONTROLLED COPY OF THE UNIT 2 TECHNICAL REQUIREMENTS
MANUAL INCLUDING CHANGE NO. 191
UPDATED BY TECHNICAL PUBLISHING

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1.0.A.4.3	"	4.3.3.7.3	
	"		
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1.0.C.4.1.b.3	"	4.7.9.2.b.3	
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1.0.F.4.1.c	"	4.7.10.c	
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9.5.3	Cask (SFP) Crane SR	4.9.7	
10.0.1	SFP Doors LCO	3.9.15 3/4.9.15	
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3.0.3	Table 1 Appendix R SSR SR's SR	7.1.1- SR7.1.26(*)	26 SR tbls

*Refer to separate Appendix R sections (26 total) for LCO & SR breakdowns

Table Nos: LCO's are Table 7.1.1-1 thru 7.1.26-1.

Table Nos: SR's are Table 7.1.1-2 thru 7.1.26-2.

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Old TRM LCO -	SR	Tbl 1 Item	Component ID	System Description	New TRM LCO	New TRM SR
3.0.1 -	3.0.3	27	P18A, 18B. 18C	Charging Pumps	7.1.1	SR 7.1 .I
3.0.1 -	3.0.3	28	CH-089	Regen HX Outlet Is0	7.1.1	SR 7.1.1
3.0.1 -	3.0.3	29	CH-518,519	Charging to RCS loops	7.1 .I	SR 7.1.1
3.0.1 -	3.0.3	30	CH-517	Aux Spray to PZR (RCS)	7.1.1	SR 7.1.1
3.0.1 -	3.0.3	31	CH-429	Charging Pump Header Is0 MOV	7.1 .I	SR 7.1 .I
3.0.1 -	3.0.3	32	CH-501	VCT outlet MOV	7.1.1	SR 7.1.1
3.0.1 -	3.0.3	33	CH-504	Chrg Pps suct fm RWST MOV	7.1.1	SR 7.1.1
3.0.1 -	3.0.3	34	CH-192	Chrg Pps suct fm RWST AOV	7.1.1	SR 7.1.1
3.0.1 -	3.0.3	57A	LI-206A, 208A	Instr, Monitor local BAST level	7.1.2	SR 7.1.2
3.0.1 -	3.0.3	8A	MS-265B	MS Low Pt Drain -ESF-MSI	7.1.3	SR 7.1.3
3.0.1 -	3.0.3	88	MS-266B	MS Low Pt Drain -ESF-MSI	7.1.3	SR 7.1.3
3.0.1 -	3.0.3	50	PT- 1 02B- 1	Instrument, Monitoring Pzr Press	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	54	PT-10238	Instr, Monitor SG Press	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	56	QEXC2PDA. 4PDA	Instr, Monitor NI, wide rang	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	56A	QEXCl PDA, 3PDA	Instr, Monitor NI, wide range	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	58A	PT-103,103-1	Instr, Monitor Pzr Press low rng	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	58F	LT-103. 1 lox, 110Y	Instr, monitor Pzr level, cold, nor	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	58C	LT-1114A. 11 148	Instr, monitor SG 1 level,wide mg	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	58B	LT-11238	Instr, Monitor SG 2 level, nar mg	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	588	LT-1124A. 1 124B	Instr, Monitor SG 2 level,wide rng	7.1.4	SR 7.1.4
3.0.1 -	3.0.3	51	TE-121X	Instr, Monitoring hot leg temp	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	52	TE-125	Instr. Monitoring cold leg temp	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	51A	TE-111X	Instr, Monitoring hot leg temp	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	52A	TE-115	Instr, Monitoring cold leg temp	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	58 B	FT-5278A. 5278B	Instr, Monitor AFW flow to SG 2	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	58C	FT-5277A. 52778	Instr, Monitor AFW flow to SG 1	7.1.5	SR 7.1.5
3.0.1 -	3.0.3	69A	Bus 25A (Hi)	6900 V breakers for RCPs	7.1.6	SR 7.1.6
3.0.1 -	3.0.3	69B	Bus 25B (H2)	6900 V breakers for RCPs	7.1.6	SR 7.1.6
3.0.1 -	3.0.3	21	SI-306	SDC HX bypass	7.1.7	SR 7.1.7
3.0.1 -	3.0.3	22	SI-651,652	SDC suct MOVs	7.1.7	SR 7.1.7
3.0.1 -	3.0.3	580	TE-351 X, 351 Y	SDC sup, ret temp monitors	7.1.7	SR 7.1.7
3.0.1 -	3.0.3	58G	TE-303X, 303Y	SDC HXs outlet temps indication	7.1.7	SR 7.1.7
3.0.1 -	3.0.3	19	RC-402/ 404	RCS- PORVS	7.1.8	SR 7.1.8
3.0.1 -	3.0.3	26D	SI-614,624,634,644	SIT Outlet MOVs	7.1.9	SR 7.1.9
3.0.1 -	3.0.3	26E	SI-613,623,633,643	SIT Nitrogen vent valves	7.1.9	SR 7.1.9
3.0.1 -	3.0.3	20	SI-615,625,635,645	LPSl flow MOVs	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	24	Sl-657	SDC HX outlet	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	23A	Sl-449.450	SDC min flow alt	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	26A1	SI-432.444	LPSl suct fm RWST	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	26A2	SI440,441	LPSl suct fm RCS	7.1.10	SR 7.1 .I0
3.0.1 -	3.0.3	26B	S1452,453	LPSl disch to SDC HX	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	26C	Sl-456,457	SDC HX outlet	7.1.10	SR 7.1.10
3.0.1 -	3.0.3	36	CS-13.1B	RSWT outlet MOV	7.1.11	SR7.1.11
3.0.1 -	3.0.3	35	CH-508,509	Boric Acid Grav feed MOVs	7.1.12	SR 7.1.12
3.0.1 -	3.0.3	58E	TE-8108	Ctmt Ambient Temp monitoring	7.1.13	SR 7.1.13
3.0.1 -	3.0.3	25A	CS-4A, 4B	Ctmt Spray Stop Valves	7.1.14	SR 7.1.14

CROSS INDEX: OLD APPX R TABLE COMPONENTS TO NEW APPX R LCO AND SRS SECTIONS

Old TRM LCO -	SR	Tbl 1 Item	Component ID	System Description	New TRM LCO	New TRM SR
3.0.1 -	3.0.3	1	SV-4188	AFW Turbine	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	2	P4 & H-21	AFW Turbine	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	7	MS-202	AFW Turbine steam stop MOV	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	8	P9A, P9B	AFW pumps, motor driven	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	10	FW-44	AFW P4 Discharge Is0 MOV	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	12	FW-43N43B	AFW flow control valves	7.1.15	SR 7.1.15
3.0.1 -	3.0.3	55	LT-5282	Instr, Monitor CST level	7.1.16	SR 7.1.16
3.0.1 -	3.0.3	55A	LIS-5489	Instr, Local CST level Ind	7.1.16	SR 7.1.16
3.0.1 -	3.0.3	3	MS-64N64 B	MSlVs	7.1.17	SR 7.1.17
3.0.1 -	3.0.3	4	MS-65N65B	MSlV Bypass	7.1.17	SR 7.1.17
3.0.1 -	3.0.3	5	MS-1 SON1 90B	AOVs	7.1.18	SR7.1 .I8
3.0.1 -	3.0.3	6	MS-220N220B	SG Blowdown Valves	7.1.19	SR 7.1.19
3.0.1 -	3.0.3	39	RB-13.1A, 13.18	RBCCW out of SDC HXs	7.1.20	SR 7.1.20
3.0.1 -	3.0.3	40	RB-21 lA, 21 1B	RBCCW Pp A Inlet AOVs	7.1.20	SR 7.1 20
3.0.1 -	3.0.3	41	RB-21 lC, 21 1D	RBCCW Pp 6 Inlet AOV's	7.1 20	SR 7.1.20
3.0.1 -	3.0.3	42	RB-211E,211F	RBCCW Pp C Inlet AOVs	7.1 -20	SR 7.1.20
3.0.1 -	3.0.3	43	RB-4.1A. 4.18	RBCCW Hx A outlet AOVs	7.1 -20	SR 7.1.20
3.0.1 -	3.0.3	44	RB4.1C. 4.10	RBCCW Hx B outlet AOVs	7.1.20	SR 7.1.20
3.0.1 -	3.0.3	45	RB-4.1 E, 4.1 F	RBCCW Hx C outlet AOVs	7.1.20	SR 7.1.20
3.0.1 -	3.0.3	46	PllA, P l l B , P l l C	RBCCW pumps	7.1.20	SR 7.1.20
3.0.1 -	3.0.3	47	P5A, P5C, LlA, L1C	RBCCW pp disch x-tie AOVs	7.1.20	SR 7.1.20
3.0.1 -	3.0.3	13	RB-251A. 251B	SW pumps 8 Strainers	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	14	SW-97AI 978	SW Disch hdr x-tie valves	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	15	SW-3.1N 3.16	SW inlet to RBCCW HXs	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	14A	SW-2A. 2B, 2C	SW pump disch valves	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	16A	SW-9N 9B/ 9C	SW outlet fm RBCCW HXs	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	18A	SW-5N 5BI 5C	SW inlet to TBCCW HXs	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	188	SW-12BI 12C	SW is0 to EDG A & B	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	18C	SW-296	Drain on SW to EDG B	7.1.21	SR 7.1.21
3.0.1 -	3.0.3	62	SBO DG	Alternate AC diesel gen	7.1.22	SR 7.1.22
3.0.1 -	3.0.3	59	Bus 24C (A3)	4160 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	60	Bus 240 (A4)	4160 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	61	Bus 24E (A5)	4160 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	63	Bus 22E (805)	480 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	64	Bus 22F (B06)	480 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	65	851 (mcc-22-1 E)	MCC B51	7.1 2 3	SR 7.1.23
3.0.1 -	3.0.3	66	852 (mcc-22-2E)	MCC B52	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	67	B61 (mcc-22-1 F)	MCC B61	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	68	862 (mcc-22-2F)	MCC B62	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	70	ELUs	Em. Light Units	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	62A	Bus 24A (Al)	4160 volt switchgear	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	62B	Bus 248 (A2)	4160 volt switchgear	7.1 :23	SR 7.1.23
3.0.1 -	3.0.3	62C	Bus 22A (BOl)	480 volt load center	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	62D	Bus 228 (802)	480 volt load center	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	69K	VA 20	120 VAC panel VIAC-2	7.1.23	SR 7.1.23
3.0.1 -	3.0.3	69	DC3	201 C swing batt charger	7.1 24	SR 7.1.24
3.0.1 -	3.0.3	69C	Bus DO1	125 VDC Bus 201A	7.1 24	SR 7.1.24

CROSS INDEX: OLD APPX R TABLE COMPONENTS TO NEW APPX R LCO AND SRS SECTIONS

Old TRM LCO -	SR	Tbl 1 Item	Component ID	System Description	New TRM LCO	New TRM SR
3.0.1 -	3.0.3	69D	Bus DO2	125 VDC Bus 201 B	7.1.24	SR 7.1 24
3.0.1 -	3.0.3	69E	DV 10	125 VDC panel 201A-1V	7.1.24	SR 7.1 2 4
3.0.1 -	3.0.3	69F	DV 20	125 VDC panel 201B-1V	7.1 24	SR 7.1.24
3.0.1 -	3.0.3	69G	D11	125 VDC panel 201A-1	7.1.24	SR 7.1 24
3.0.1 -	3.0.3	69H	Dl2	125 VDC panel 201A-2	7.1.24	SR 7.1.24
3.0.1 -	3.0.3	69 I	D2 1	125 VDC panel 201 B-1	7.1.24	SR 7.1.24
3.0.1 -	3.0.3	69J	D22	125 VDC panel 201 B-2	7.1.24	SR 7.1.24
3.0.1 -	3.0.3	71	FIRE-258, 259, 559, 560	Fire Supp system	7.1.25	SR 7.1.25
3.0.1 -	3.0.3	18D	Fire Suppression sys	Alt(Fire Pro) cooling to EDGs	7.1.25	SR 7.1.25
3.0.1 -	3.0.3	72	Radio Comm Sys	Radio Communication system	7.1 2 6	SR 7.1.26
3.0.1 -	3.0.3	74	HSD Storage Box	Box Inventory items available	7.1 2 6	SR 7.1.26
3.0.1 -	3.0.3	75	FSD Storage Box	Box Inventory items available	7.1.26	SR 7.1.26
3.0.1 -	3.0.3	76	CSD Repair Material	Materials available for repairs	7.1.26	SR 7.1.26
3.0.1 -	3.0.3	81	F156A, B, C, D	fans-East &West DC rooms	7.1.26	SR 7.1.26
3.0.1 -	3.0.3	82	F157A. B, C, D	fans-East & West 480 V rooms	7.1 2 6	SR 7.1.26
3.0.1 -	3.0.3	83	Bottle 3A, 3B,3C	bottles and valves	7.1 2 6	SR 7.1 26

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Dominion

Memorandum

JAN 30 2002

RA-02-015

To: Millstone Unit No.2 Technical Requirements Manual Controlled Copy Holders

From: C. Schwarz
Director Nuclear Station Operations & Maintenance

RE-Issuance of Millstone Unit No.2 Technical Requirements Manual

Please find attached the Millstone Unit No.2 Technical Requirements Manual (TRM).This reissued TRM has been written in the Standard Technical Specifications format. The TRM contains the re-located requirements from Technical Specifications and certain material that warrants administrative controls (e.g. , Appendix R -Safe Shutdown Requirements) and does not contain and will not contain any clarification of a particular LCO or Surveillance. The information in the TRM will be controlled in a manner consistent with the Technical Specifications and by use of the Master Manual, License Basis Management Program, MP-03-LBM-PRG, and specifically by procedure MP-03-LBM-SAP04.

It is suggested that the TRM be maintained next to the Millstone Unit No.2 Technical Specifications. Distribution and control of the TRM will be similar to that of The Technical Specifications. If you have any questions concerning this manual ,please call Rick Bonner at 5230 or Ravi Joshi at 2080.

cc: Memo File
Nuclear Records

1.0 DEFINITIONS

The DEFINED TERMS appear in capitalized type throughout these TECHNICAL REQUIREMENTS. These capitalized terms are defined in Section 1.0 of the Technical Specifications.

In addition to the terms defined in the Technical Specifications, the following defined terms and acronyms appear in the TECHNICAL REQUIREMENTS. These defined terms and acronyms also appear in capitalized type throughout the TECHNICAL REQUIREMENTS.

CURRENT LICENSING BASIS

1.1 The CURRENT LICENSING BASIS (CLB) is the set of NRC requirements applicable to a specific plant, plus a licensee's docketed and currently effective written commitments for ensuring compliance with, and operation within, applicable NRC requirements and the plant-specific DESIGN BASIS, including all modifications and additions to such commitments over the life of the facility operating license.

The set of NRC requirements applicable to a specific plant CLB include:

a. NRC regulations in 10 CFR Parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, and 100 and appendices thereto
b. Commission orders
c. license conditions
d. exemptions
e. technical specifications (TSs)
f. plant-specific DESIGN BASIS information defined in 10 CFR 50.2 and documented in the most recent Updated Final Safety Analysis Report (UFSAR) (as required by 10 CFR 50.71)
g. licensee commitments remaining in effect that were made in docketed licensing correspondence (such as licensee responses to NRC bulletins, Licensee Event Reports, generic letters, and enforcement actions)
h. licensee commitments documented in NRC safety evaluations

DESIGN BASIS

1.2 Design bases information, defined by 10 CFR 50.2, is documented in the UFSAR as required by 10 CFR 50.71. The DESIGN BASIS of safety-related structures, systems, and components (SSCs) is established initially during the original plant licensing and relates primarily to the accident prevention or mitigation functions of safety-related SSCs. The DESIGN BASIS of a safety-related SSC is a subset of the CLB.

FUNCTIONAL – FUNCTIONALITY

1.3 FUNCTIONALITY is an attribute of SSCs that is not controlled by TSs. An SSC is FUNCTIONAL or has FUNCTIONALITY when it is capable of performing its SPECIFIED FUNCTION, as set forth in the CLB. FUNCTIONALITY does not apply to SPECIFIED SAFETY FUNCTIONS, but does apply to the ability of non-TS SSCs to perform other SPECIFIED FUNCTIONS that have a necessary support function.

SPECIFIED FUNCTION/SPECIFIED SAFETY FUNCTION

1.4 The SPECIFIED FUNCTION(S) of the system, subsystem, train, component or device (hereafter referred to as system) is that SPECIFIED SAFETY FUNCTION(S) in the CLB for the facility. In addition to providing the SPECIFIED SAFETY FUNCTION, a system is expected to perform as designed, tested and maintained. When system capability is degraded to a point where it cannot perform with reasonable expectation or reliability, the system should be judged inoperable/nonfunctional, even if at this instantaneous point in time the system could provide the SPECIFIED SAFETY FUNCTION.

TECHNICAL REQUIREMENT

1.5 A TECHNICAL REQUIREMENT represents the lowest functional capability or performance level allowed for the specified equipment or parameter.

TECHNICAL SURVEILLANCE REQUIREMENT

1.6 TECHNICAL SURVEILLANCE REQUIREMENTS are requirements related to test, calibration, or inspection to verify that the associated TECHNICAL REQUIREMENTS are met.

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

For the purpose of the TRs, the Technical Requirements Manual (TRM) terms specified below are considered similar to the listed Technical Specification terms:

Technical Specification Term	Technical Requirements Manual Term
Technical Specifications (TS)	Technical Requirements Manual (TRM)
Limiting Condition for Operation (LCO)	TECHNICAL REQUIREMENT (TR)
Surveillance Requirement (SR)	TECHNICAL SURVEILLANCE REQUIREMENT (TSR)
Specification	Requirement

TECHNICAL REQUIRMENT

TR 3.0.1 Compliance with the TRs contained in the succeeding requirements is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the TR, the associated ACTION requirements shall be met.

TR 3.0.2 Noncompliance with a requirement shall exist when the requirements of the TR and associated ACTION requirements are not met within the specified time intervals, except as provided in TR 3.0.6. If the TR is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

TR 3.0.3 When a TR is not met, except as provided in the associated ACTION requirements, the unit shall be placed in a safe condition as determined by the Operational Decision Making process. Corrective action shall be initiated in accordance with the corrective action program.

TR 3.0.4 When a TR is not met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL MODE or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing nonfunctional systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Requirement are stated in the individual Requirements, or

c. When an allowance is stated in the individual value, parameter, or other Requirement.

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL REQUIRMENT

This Requirement shall not prevent changes in OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

TR 3.0.5 When a system, subsystem, train, component or device is determined to be nonfunctional solely because its emergency power source is nonfunctional, or solely because its normal power source is nonfunctional, it may be considered FUNCTIONAL for the purpose of satisfying the requirements of its applicable TR, provided: (1) its corresponding normal or emergency power source is FUNCTIONAL; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are FUNCTIONAL, or likewise satisfy the requirements of this requirement. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in a safe condition as determined by the Operational Decision Making process. Corrective action shall be initiated in accordance with the corrective action program.

This TR is not applicable in MODES 5 or 6.

TR 3.0.6 Equipment removed from service or declared nonfunctional to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its FUNCTIONALITY or the FUNCTIONALITY of other equipment. This is an exception to TR 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate FUNCTIONALITY.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 4.0.1 TSRs shall be met during the OPERATIONAL MODES or other conditions specified for individual TRs unless otherwise stated in an individual TSR.

Failure to meet a TSR, whether such failure is experienced during the performance of the TSR or between performances of the TSR, shall be failure to meet the TR. Failure to perform a TSR within the specified TSR interval shall be failure to meet the TR except as provided in TSR 4.0.3. TSRs do not have to be performed on nonfunctional equipment or variables outside specified limits.

TSR 4.0.2 Each TSR shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance time interval.

TSR 4.0.3 If it is discovered that a TSR was not performed within its specified surveillance interval, then compliance with the requirement to declare the TR not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the TSR. A risk evaluation shall be

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

performed for any TSR delayed greater than 24 hours and the risk impact shall be managed.

If the TSR is not performed within the delay period, the TR must immediately be declared not met, and the applicable ACTION(s) must be entered.

When the TSR is performed within the delay period and the TSR is not met, the TR must immediately be declared not met, and the applicable ACTION(s) must be entered.

TSR 4.0.4 Entry into an OPERATIONAL MODE or other specified condition in the Applicability of a TR shall only be made when the TRs Technical Surveillance Requirements have been met within their specified Frequency, except as provided by TSR 4.03. When a TR is not met due to TSRs not having been met, entry into an OPERATIONAL MODE or other specified condition in the Applicability shall only be made in accordance with TR 3.0.4.

This provision shall not prevent entry into OPERATIONAL MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES:

TR 3.0.1 through 3.0.4 establish the general requirements applicable to TRs.

TR 3.0.1 establishes the Applicability statement within each individual TR as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the TR is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a TR are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility that is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the TR must be met. This time limit is the allowable outage time to restore a nonfunctional system or component to FUNCTIONAL status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, the unit shall be placed in a safe condition as determined by the Operational Decision Making process and corrective action shall be initiated in accordance with the corrective action program.

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

The specific time limits of the ACTION requirements are applicable from the point in time it is identified that a TR is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. The ACTIONS for not meeting a single TR adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4). Individual TRs may include a specified time limit for the completion of a TSR when equipment is removed from service. In this case, the allowable outage time limits of ACTION requirements are applicable when this limit expires if the TSR has not been completed.

TR 3.0.2 establishes that noncompliance with a TR exists when the requirements of the TR are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this TR is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a TR and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a TR is restored within the time interval specified in the associated ACTION requirements.

TR 3.0.3 establishes the ACTION requirements that must be implemented when a TR is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this TR is to delineate the ACTIONS when plant operation cannot be maintained within the limits of safe operation defined by the TR and its ACTION requirements. Planned entry into TR 3.0.3 should be avoided. If it is not practicable to avoid planned entry into TR 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into TR 3.0.3 should have less effect on plant safety than other practicable alternatives.

TR 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when a TR is not met. It allows placing the unit in a MODE or other specified condition slated in that Applicability (e.g., the Applicability desired to be entered) when unit conditions are such that the requirements of the TR would not be met, in accordance with TR 3.0.4.a, TR 3.0.4.b, or TR 3.0.4.c.

TR 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the TR not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with required ACTIONS that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

the required ACTIONS.

TR 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with the TR not met after performance of a risk assessment addressing nonfunctional systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate.

The risk assessment may use quantitative, qualitative, or blended approaches, and the risk assessment will be conducted using the plant program, procedures, and criteria in place to implement 10 CFR 50.65(a)(4), which requires that risk impacts of maintenance activities be assessed and managed. The risk assessment, for the purposes of TR 3.0.4(b), must take into account all nonfunctional Technical Requirements Manual equipment regardless of whether the equipment is included in the normal 10 CFR 50.65(a)(4) risk assessment scope. The risk assessments will be conducted using the procedures and guidance endorsed by Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Regulatory Guide 1.160 endorses the guidance in NUMARC 93-01, “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” These documents address general guidance for conduct of the risk assessment, quantitative, and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to pan and conduct other activities in a manner that controls overall risk, increased risk awareness by shift and management personnel, actions to reduce the duration of the condition, actions to minimize the magnitude of risk increases (establishment of backup success paths or compensatory measures), and determination that the proposed MODE change is acceptable. Consideration should also be given to the probability of completing restoration such that the requirements of the TR would be met prior to the expiration of ACTIONS Completion Times that would require exiting the Applicability.

TR 3.0.4.b may be used with single, or multiple systems and components unavailable. NUMARC 93-01 provides guidance relative to consideration of simultaneous unavailability of multiple systems and components.

The results of the risk assessment shall be considered in determining the acceptability of entering the MODE or other specified condition in the Applicability, and any corresponding risk management actions. The TR 3.0.4.b risk assessments do not have to be documented.

The Technical Requirements Manual allows continued operation with equipment unavailable in MODE 1 for the duration of the Completion Time. Since this is allowable, and since in general the risk impact in that particular MODE bounds the risk of transitioning into and through the applicable MODES or other specified conditions in the Applicability of the TR, the use of the TR 3.0.4.b allowance should be generally

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TECHNICAL SURVEILLANCE REQUIREMENTS

acceptable, as long as the risk is assessed and managed as stated above. However, there is a small subset of systems and components that have been determined to be more important to risk and use of the TR 3.0.4.b allowance is prohibited. The TRs governing these system and components contain Notes prohibiting the use of TR 3.0.4.b by stating that TR 3.0.4 and/or TR 4.0.4 are not applicable.

TR 3.0.4.c allows entry into a MODE or other specified condition in the Applicability with the TR not met based on a Note in the Requirement which states TR 3.0.4.c is applicable. These specific allowances permit entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific required ACTION of a Requirement. The risk assessments performed to justify the use of TR 3.0.4.b usually only consider systems and components. For this reason, TR 3.0.4.c is typically applied to Requirements which describe values and parameters and may be applied to other Requirements based on plant-specific approval.

The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to FUNCTIONAL status before entering as associated MODE or other specified condition in the Applicability.

The provisions of TR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3, to MODE 4, and MODE 4 to MODE 5.

Upon entry into a MODE or other specified condition in the Applicability with the TR not met, TR 3.0.1 and TR 3.0.2 require entry into the applicable Conditions and required ACTIONS until the Condition is resolved, until the TR is met, or until the unit is not within the Applicability of the Technical Requirement.

Technical Surveillance Requirements do not have to be performed on the associated nonfunctional equipment (or on variables outside the specified limits), as permitted by TSR 4.0.1. Therefore, utilizing TR 3.0.4 is not a violation of TSR 4.0.1 or TSR 4.0.4 for any Technical Surveillance Requirements that have not been performed on nonfunctional equipment. However, TSRs must be met to ensure FUNCTIONALITY prior to declaring the associated equipment FUNCTIONAL (or variable within limits) and restoring compliance with the affected TSR.

TR 3.0.5 delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is nonfunctional. It specifically prohibits operation when one

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

division is nonfunctional because its normal or emergency power source is nonfunctionaland a system, subsystem, train, component or device in another division is nonfunctional for another reason.

The provisions of this TR permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the TR for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be nonfunctional solely because of the nonfunctionality of its normal emergency power source.

In MODES 5 and 6 TR 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable TR in these MODES must be adhered to.

TR 3.0.6 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared nonfunctional to comply with ACTIONS. The sole purpose of this TR is to provide an exception to TR 3.0.2 (e.g., to not comply with the applicable required ACTION(s)) to allow the performance of TSRs to demonstrate:

a. The FUNCTIONALITY of the equipment being returned to service; or

b. The FUNCTIONALITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed TSRs. The TR does not provide time to perform any other preventive or corrective maintenance.

TSR 4.0.1 through 4.0.4 establish the general requirements applicable to TSRs.

TSR 4.0.1 establishes the requirement that TSRs must be met during the OPERATIONAL MODES or other conditions for which the requirements of the TR apply unless otherwise stated in an individual TSR. The purpose of this requirement is to ensure that TSRs are performed to verify the FUNCTIONALITY of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated TR are applicable. Failure to meet a TSR within the specified TSR interval, in accordance with TSR 4.0.2 constitutes a failure to meet a TR.

Systems and components are assumed to be FUNCTIONAL when the associated TSRs have been met. Nothing in this requirement, however, is to be construed as implying that systems or components are FUNCTIONAL when either:

a. The system or components are known to be nonfunctional, although still meeting the TSRs or

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

b. The requirements of the TSR(s) are known to be not met between required TSR performances.

TSRs do not have to be performed when the facility is in an OPERATIONAL MODE or other specified conditions for which the requirements of the associated TR do not apply unless otherwise specified. The TSRs associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a TR.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given TSR. In this case, the unplanned event may be credited as fulfilling the performance of the TSR. This allowance includes those TSRs whose performance is normally precluded in a given MODE or other specified condition.

TSRs, including TSRs invoked by ACTION requirements, do not have to be performed on nonfunctional equipment because the ACTIONS define the remedial measures that apply. TSRs have to be met and performed in accordance with TSR 4.0.2, prior to returning equipment to FUNCTIONAL status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment FUNCTIONAL. This includes ensuring applicable TSRs are not failed and their most recent performance is in accordance with TSR 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered FUNCTIONAL provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

TSR 4.0.2 establishes the limit for which the specified time interval for TSRs may be extended. It permits an allowable extension of the normal TSR interval to facilitate TSR scheduling and consideration of plant operating conditions that may not be suitable for conducting the TSR, e.g., transient conditions or other ongoing TSR or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for TSRs that are performed at each REFUELING outage and are specified with an 18-month TSR interval. It is not intended that this provision be used repeatedly as a convenience to extend TSR intervals beyond that specified for TSRs that are not performed during REFUELING outages. The limitation of TSR 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular TSR being performed is the verification of conformance with the TSR. This provision is sufficient to ensure that the reliability ensured through TSR activities is not significantly degraded beyond that obtained from the specified TSR interval.

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 4.0.3 establishes the flexibility to defer declaring affected equipment nonfunctional or an affected variable outside the specified limits when a TSR has not been completed within the specified TSR interval. A delay period of up to 24 hours or up to the limit of the specified TSR interval, whichever is greater, applies from the point in time that it is discovered that the TSR has not been performed in accordance with TSR 4.0.2,and not at the time that the specified TSR interval was not met.

This delay period provides adequate time to complete TSRs that have been missed. This delay period permits the completion of a TSR before complying with ACTION requirements or other remedial measures that might preclude completion of the TSR. The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the TSR, the safety significance of the delay in completing the required TSR, and the recognition that the most probable result of any particular TSR being performed is the verification of conformance with the requirements.

When a TSR, with a TSR interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations, (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, TSR 4.0.3 allows for the full delay period of up to the specified TSR interval to perform the TSR. However, since there is not a time interval specified, the missed TSR should be performed at the first reasonable opportunity.

TSR 4.0.3 provides a time limit for, and allowances for the performance of, TSRs that become applicable as a consequence of MODE changes imposed by ACTION requirements.

Failure to comply with specified TSR intervals for the TSRs is expected to be an infrequent occurrence. Use of the delay period established by TSR 4.0.3 is a flexibility that is not intended to be used as an operational convenience to extend TSR intervals. While up to 24 hours or the limit of the specified TSR interval is provided to perform the missed TSR, it is expected that the missed TSR will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the TSR as well as any plant configuration changes required) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the TSR. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.” This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed TSR should be treated as an emergent condition as discussed in

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed TSRs will be placed in the licensee's Corrective Action Program.

If a TSR is not completed within the allowed delay period, then the equipment is considered nonfunctional or the variable is considered outside the specified limits and the entry into the ACTION requirements for the applicable TR begins immediately upon expiration of the delay period. If a TSR is failed within the delay period, then the equipment is nonfunctional, or the variable is outside the specified limits and entry into the ACTION requirements for the applicable TR begins immediately upon the failure of the TSR.

Completion of the TSR within the delay period allowed by this TSR, or within the Allowed Outage Time of the applicable ACTIONS, restores compliance with TSR 4.0.1.

TSR 4.0.4 establishes the requirement that all applicable TSRs must be met before entry into a MODE or other specified condition in the Applicability.

This Requirement ensures that system and component FUNCTIONALITY requirements are variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to FUNCTIONAL status before entering an associated MODE or other specified condition in the Applicability.

A provision is included to allow entry into a MODE or other specified condition in the Applicability when a TR is not met due to the Technical Surveillance Requirement not being met in accordance with TR 3.0.4.

However, in certain circumstances, failing to meet a TSR will not result in TSR 4.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is non-functional or outside its specified limits, the associated TSR(s) are not required to be performed, per TSR 4.0.1, which states that Technical Surveillances Requirements do not have to be performed on nonfunctional equipment. When equipment is nonfunctional, TSR 4.0.4 does not apply to the associated TSR(s) since the requirement for the TSR(s) to be performed is removed. Therefore, failing to perform the Technical Surveillance Requirement(s) within the specified Frequency does not result in a TSR 4.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the TR is not met in this instance, TR 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. TSR 4.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Technical Surveillance Requirement has not been performed within the specified Frequency, provided the requirement to declare the TR not met has been delayed in accordance with TSR 4.0.3.

3/4.0 TECHNICAL REQUIREMENT (TR) AND TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TECHNICAL SURVEILLANCE REQUIREMENTS

The provisions of TSR 4.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TSR 4.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, MODE 3 to MODE 4, and MODE 4 to MODE 5.

3/4.1 REACTIVITY CONTROL SYSTEM

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

TECHNICAL REQUIRMENT

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be FUNCTIONAL:

a. A flow path with a piping temperature of greater than 55°F from the boric acid storage tank via either a boric acid pump or a gravity feed connection and a charging pump to the Reactor Coolant System if only the boric acid storage tank in TECHNICAL REQUIREMENT 3.1.2.7.a is FUNCTIONAL, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in TECHNICAL REQUIREMENT 3.1.2.7.b is FUNCTIONAL.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths FUNCTIONAL, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification 3.1.1.1 in MODE 5 or Technical Specification 3.9.1 in MODE 6 until at least one injection path is restored to FUNCTIONAL status.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated FUNCTIONAL:

a. At least once per 31 days by verifying each manual, power operated, and automatic valve in the boron injection flow path that is not locked, sealed or otherwise secured in position, is in the correct position.

b. At least once per 24 hours by verifying that the boric acid piping temperature is greater than 55°F.

3/4.1 REACTIVITY CONTROL SYSTEM

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and operations that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration as described in the Technical Specifications in the event the single injection system becomes nonfunctional. The MODES 5 and 6 ACTION requirement to suspend positive reactivity additions does not preclude completion of actions to establish a safe conservative plant condition, or to maintain or increase reactor vessel inventory provided the boron concentration of the makeup water source is greater than or equal to the boron concentration for the required SHUTDOWN MARGIN (SDM).

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.1.a verifies the correct alignment for manual, power operated, and automatic valves in the boration flow paths to provide assurance that the proper flow paths will exist to establish boric acid flow to the Reactor Coolant System (RCS). This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency has been shown to be acceptable through operating experience.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.1.b verifies the temperature of the boric acid piping to ensure FUNCTIONALITY of the boration flow path. Ambient air temperature in the vicinity of the boric acid piping on elevations (-)5’ -0” and (-)25’ -6” is a valid indication of piping temperature. The 24 hour frequency is sufficient to identify temperature changes that approach the limit.

REFERENCE:

1. License Amendment 283
2. License Amendment 293

3/4.1 REACTIVITY CONTROL SYSTEM

FLOW PATHS - OPERATING

TECHNICAL REQUIREMENT

3.1.2.2 The following boron injection flowpaths to the RCS via the charging pump(s) shall be FUNCTIONAL:

a. At least one of the following combinations:

1. One boric acid storage tank, with the tank contents in accordance with Figure 3.1-1 and a piping temperature greater than 55°F, its associated gravity feed valve, and boric acid pump.

2. Two boric acid storage tanks, with the weighted average of the combined contents of the tanks in accordance with Figure 3.1-1 and a piping temperature greater than 55°F, their associated gravity feed valves, and boric acid pumps.

3. Two boric acid storage tanks, each with contents in accordance with Figure 3.1-1 and a piping temperature greater than 55°F, at least one gravity feed valve, and at least one boric acid pump.

b. The flow path from a FUNCTIONAL refueling water storage tank, as per TECHNICAL REQUIREMENT 3.1.2.8.b.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With fewer than the above required boron injection flow paths to the Reactor Coolant System FUNCTIONAL, manage the risk impact in accordance with the requirements of 10 CFR 50.65(a)(4).

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated FUNCTIONAL:

a. At least once per 31 days by verifying each manual, power operated, and automatic valve in the boron injection flow path that is not locked, sealed or otherwise secured in position, is in the correct position.

b. At least once per 24 hours by verifying that the boric acid piping temperature is greater than 55°F.

3/4.1 REACTIVITY CONTROL SYSTEM

FLOW PATHS - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

c. At least once per 36 months, by verifying each valve in the required boron injection flowpaths that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection flowpaths are provided to ensure single FUNCTIONAL capability in the event an assumed failure of a pump or valve renders one of the flowpaths nonfunctional. Redundant flow paths from the boric acid storage tanks are achieved through boric acid pumps, gravity feed lines and charging pumps. Redundant flow paths from the refueling water storage tank are achieved through charging pump flow path guaranteed by TECHNICAL REQUIREMENT 3.1.2.2 and the HPSI flow path guaranteed by Technical Specification 3.5.2 and 3.5.3.

Paragraph (a)(4) of 10CFR50.65 “Maintenance Rule” requires that licensees assess and manage the increase in risk that may result from proposed maintenance activities, including out of service Structures, Systems, and Components (SSCs). Management of risk involves consideration of incremental and aggregate impacts. The incremental risk increase is predominantly assessed using quantitative PRA methods, which measure the effect on core damage frequency and large early release frequency associated with the plant configuration. The risk increase is then managed in accordance with action thresholds defined by NUMARC 93-01. The aggregate impact is controlled by meeting Maintenance Rule requirements for establishing and meeting SSC performance criteria. These requirements include consideration of the risk significance of SSCs in establishing performance goals. Significant risk impacts are documented in the corrective action program and assessed as appropriate.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.2.a verifies the correct alignment for manual, power operated, and automatic valves in the boration flow paths and provides assurance that the proper flow paths will exist to establish boric acid flow to the Reactor Coolant System (RCS). This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve

3/4.1 REACTIVITY CONTROL SYSTEM

FLOW PATHS - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31-day frequency has been shown to be acceptable through operating experience.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.2.b verifies the temperature of the boric acid piping to ensure FUNCTIONALITY of the boration flow path. Ambient air temperature in the vicinity of the boric acid piping on elevations (-)5’ -0” and (-)25’ -6” is a valid indication of piping temperature. The 24 hour frequency is sufficient to identify temperature changes that approach the limit.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.2.c verifies the ability of the valves in the boration injection flowpath to actuate to the correct position on an actual or simulated actuation signal. The 36 month frequency is consistent with other Millstone Power Station Unit 2 actuation testing requirements and with standard industry practices and guidelines.

REFERENCE:

1. License Amendment 283

3/4.1 REACTIVITY CONTROL SYSTEM

CHARGING PUMP - SHUTDOWN

TECHNICAL REQUIREMENT

3.1.2.3 At least one charging pump in the boron injection flow path required FUNCTIONAL pursuant to TECHNICAL REQUIREMENT 3.1.2.1 shall be FUNCTIONAL.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump FUNCTIONAL, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification 3.1.1.1 in MODE 5 or Technical Specification 3.9.1 in MODE 6 until one charging pump is restored to FUNCTIONAL status.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.3 The above required charging pump shall be demonstrated FUNCTIONAL by verifying the delivered flow of the charging pump at the required discharge pressure is greater than or equal to the required flow when tested in accordance with the Inservice Testing (IST) program.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and operations that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration as described in the technical specifications in the event the single injection system becomes nonfunctional. The MODES 5 and 6 ACTION requirement to suspend positive reactivity additions does not preclude completion of actions to establish a safe conservative plant condition, or to maintain or increase reactor vessel inventory provided the boron concentration of the makeup water source is greater than or equal to the boron concentration for the required SHUTDOWN MARGIN (SDM).

3/4.1 REACTIVITY CONTROL SYSTEM

CHARGING PUMP - SHUTDOWN

TECHNICAL SURVEILLANCE REQUIREMENTS

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.3, which verifies charging pump flow and discharge pressure, is a normal test of charging pump performance required by the ASME OM Code. The frequency for this surveillance is a Code requirement governed by the IST program. Such inservice testing detects component degradation and incipient failures.

REFERENCE:

1. License Amendment 283

2. License Amendment 293

3/4.1 REACTIVITY CONTROL SYSTEM

CHARGING PUMPS - OPERATING

TECHNICAL REQUIREMENT

3.1.2.4 At least two charging pumps shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump FUNCTIONAL, manage the risk impact in accordance with the requirement of 10 CFR 50.65(a)(4).

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.4.1 Two charging pumps shall be demonstrated FUNCTIONAL by verifying the delivered flow of the charging pump at the required discharge pressure is greater than or equal to the required flow when tested in accordance with the Inservice Testing (IST) program.

4.1.2.4.2 Two charging pumps shall be demonstrated FUNCTIONAL at least once per 36 months by verifying that each charging pump NOT in Pull-To-Lock starts automatically on an actual or simulated actuation signal.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS average temperature above 200 °F, a minimum of two separate and redundant boron injection flowpaths are provided to ensure single FUNCTIONAL capability in the event an assumed failure of a pump or valve renders one of the flowpaths nonfunctional. Redundant flow paths from the boric acid storage tanks are achieved through Boric Acid Pumps, gravity feed lines and Charging Pumps. Redundant flow paths from the Refueling Water Storage Tank are achieved through Charging Pump flow path guaranteed by TECHNICAL REQUIREMENT 3.1.2.2 and the HPSI flow path guaranteed by Technical Specification 3.5.2 and 3.5.3.

3/4.1 REACTIVITY CONTROL SYSTEM

CHARGING PUMPS - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

Paragraph (a)(4) of 10CFR50.65 “Maintenance Rule” requires that licensees assess and manage the increase in risk that may result from proposed maintenance activities, including out of service Structures, Systems, and Components (SSCs). Management of risk involves consideration of incremental and aggregate impacts. The incremental risk increase is predominantly assessed using quantitative PRA methods, which measure the effect on core damage frequency and large early release frequency associated with the plant configuration. The risk increase is then managed in accordance with action thresholds defined by NUMARC 93-01. The aggregate impact is controlled by meeting Maintenance Rule requirements for establishing and meeting SSC performance criteria. These requirements include consideration of the risk significance of SSCs in establishing performance goals. Significant risk impacts are documented in the corrective action program and assessed as appropriate.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.4.1, which verifies charging pump flow at the design discharge pressure, is a normal test of charging pump performance required by the ASME OM Code. The frequency for this surveillance is a Code requirement governed by the IST program. Such inservice testing detects component degradation and incipient failures.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.4.2 verifies the ability of the charging pumps which are NOT in Pull-To-Lock to automatically start on an actual or simulated actuation signal. The 36 month frequency is consistent with other Millstone Power Station Unit 2 actuation testing requirements and with standard industry practices and guidelines.

REFERENCE:

1. License Amendment 283

2. DCR M2-07-002, “Replacement of the Unit 2 Charging Pumps Discharge Pulsation Dampeners”

3/4.1 REACTIVITY CONTROL SYSTEM

BORIC ACID PUMPS - SHUTDOWN

TECHNICAL REQUIREMENT

3.1.2.5 At least one boric acid pump shall be FUNCTIONAL if only the flow path through the boric acid pump in TECHNICAL REQUIREMENT 3.1.2.1.a is FUNCTIONAL.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump FUNCTIONAL as required to complete the flow path of TECHNICAL REQUIREMENT 3.1.2.1.a, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification 3.1.1.1 in MODE 5 or Technical Specification 3.9.1 in MODE 6 until at least one boric acid pump is restored to FUNCTIONAL status.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.5 One boric acid pump shall be demonstrated FUNCTIONAL by verifying the developed head of the boric acid pump at the flow test point is greater than or equal to the required developed head when tested in accordance with the Inservice Testing (IST) program.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and operations that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration as described in the Technical Specifications in the event the single injection system becomes nonfunctional. The MODES 5 and 6 ACTION requirement to suspend positive reactivity additions does not preclude completion of actions to establish a safe conservative plant condition, or to maintain or increase reactor vessel inventory provided the boron concentration of the makeup water source is greater than or equal to the boron concentration for the required SHUTDOWN MARGIN (SDM).

3/4.1 REACTIVITY CONTROL SYSTEM

BORIC ACID PUMPS - SHUTDOWN

TECHNICAL SURVEILLANCE REQUIREMENTS

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.5, which verifies boric acid pump design flow, is a normal test of boric acid pump performance required by the ASME OM Code. A quarterly frequency for such tests is a Code requirement. Such inservice testing detects component degradation and incipient failures.

REFERENCE:

1. License Amendment 283

2. License Amendment 293

3/4.1 REACTIVITY CONTROL SYSTEM

BORIC ACID PUMPS - OPERATING

TECHNICAL REQUIREMENT

3.1.2.6 The boric acid pump(s) in the boron injection flow path(s) required FUNCTIONAL pursuant to TECHNICAL REQUIREMENT 3.1.2.2.a shall be FUNCTIONAL if the flow path through the boric acid pump in TECHNICAL REQUIREMENT 3.1.2.2.a is FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the boric acid pump(s) required for the boron injection flow path(s) pursuant to TECHNICAL REQUIREMENT 3.1.2.2.a nonfunctional, manage the risk impact in accordance with the requirements of 10CFR50.65(a)(4).

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.6.1 The boric acid pump(s) shall be demonstrated FUNCTIONAL by verifying the developed head of each boric acid pump at the flow test point is greater than or equal to the required developed head when tested in accordance with the Inservice Testing (IST) program.

4.1.2.6.2 Two boric acid pumps shall be demonstrated FUNCTIONAL at least once per 36 months by verifying that each boric acid pump starts automatically on an actual or simulated actuation signal.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection flowpaths are provided to ensure single FUNCTIONAL capability in the event an assumed failure of a pump or valve renders one of the flowpaths nonfunctional. Redundant flow paths from the boric acid storage tanks are achieved through boric acid pumps, gravity feed lines and charging pumps. Redundant flow paths from the refueling water storage tank are achieved through charging pump flow path guaranteed by TECHNICAL REQUIREMENT 3.1.2.2 and the HPSI flow path guaranteed by Technical Specification 3.5.2 and 3.5.3.

3/4.1 REACTIVITY CONTROL SYSTEM

BORIC ACID PUMPS - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

Paragraph (a)(4) of 10CFR50.65 “Maintenance Rule” requires that licensees assess and manage the increase in risk that may result from proposed maintenance activities, including out of service Structures, Systems, and Components (SSCs). Management of risk involves consideration of incremental and aggregate impacts. The incremental risk increase is predominantly assessed using quantitative PRA methods, which measure the effect on core damage frequency and large early release frequency associated with the plant configuration. The risk increase is then managed in accordance with action thresholds defined by NUMARC 93-01. The aggregate impact is controlled by meeting Maintenance Rule requirements for establishing and meeting SSC performance criteria. These requirements include consideration of the risk significance of SSCs in establishing performance goals. Significant risk impacts are documented in the corrective action program and assessed as appropriate.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.6.1, which verifies boric acid pump developed head at the flow test point, is a normal test of boric acid pump performance required by the ASME OM Code. The frequency for this surveillance is a Code requirement governed by the IST Program. Such inservice testing detects component degradation and incipient failures.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.6.2 verifies the ability of the boric acid pumps to automatically start on an actual or simulated actuation signal. The 36 month frequency is consistent with other Millstone Power Station Unit 2 actuation testing requirements and with standard industry practices and guidelines.

REFERENCE:

1. License Amendment 283

3/4.1 REACTIVITY CONTROL SYSTEM

BORATED WATER SOURCES - SHUTDOWN

TECHNICAL REQUIREMENT

3.1.2.7 As a minimum, one of the following borated water sources shall be FUNCTIONAL:

a. One boric acid storage tank with:

1. A concentration between 2.5 and 3.5 weight percent boric acid.
2. A minimum volume of 3750 gallons, and
3. A minimum boric acid storage tank temperature of 55°F.

or b. The refueling water storage tank with:

1. A minimum contained volume of 57,300 gallons,
2. A minimum boron concentration of 1720 ppm when in MODE 5,
3. A minimum boron concentration as defined in Technical Specification 3.9.1 when in MODE 6.
4. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources FUNCTIONAL, suspend all operations involving CORE ALTERATIONS and positive reactivity additions that could cause introduction of coolant into the RCS with boron concentration less than required to meet Technical Specification 3.1.1.1 in MODE 5 or Technical Specification 3.9.1 in MODE 6 until at least one borated water source is restored to FUNCTIONAL status.

3/4.1 REACTIVITY CONTROL SYSTEM

BORATED WATER SOURCES - SHUTDOWN

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated FUNCTIONAL:

a. By performing the following as appropriate for the tank(s) being credited to satisfy this TRM requirement:

1. Verify boron concentration of the BASTs at least once per 31 days and within 12 hours after each solution volume increase of **>** 1% of indicated tank volume.

2. Verify boron concentration of the RWST at least once per 7 days.

3. Verify water level of both BASTs and the RWST at least once per 7 days.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the RWST ambient air temperature is <35°F.

c. At least once per 24 hours by verifying that the Boric Acid Storage Tank temperature is greater than 55°F when it is the source of borated water.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source as well as the analysis assumption to determine the boration requirement to ensure adequate SHUTDOWN MARGIN.

The boron capability required below 200°F is based upon providing a SHUTDOWN MARGIN (SDM) within the limit specified in the CORE OPERATING LIMITS REPORT at 140°F after xenon decay. This condition requires either 3750 gallons of 2.5% boric acid solution from the boric acid storage tanks or 57,300 gallons of 1720 ppm borated water from the refueling water storage tank.

The maximum boron concentration requirement (3.5%) and the minimum temperature requirement (55°F) for the Boric Acid Storage Tank ensure that boron does not precipitate in the Boric Acid System. The daily surveillance requirement provides sufficient assurance that the temperature of the tank will be maintained higher than 55°F at all times.

3/4.1 REACTIVITY CONTROL SYSTEM

BORATED WATER SOURCES - SHUTDOWN

BASES:

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source as well as the analysis assumption to determine the boration requirement to ensure adequate SHUTDOWN MARGIN.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.7.a verifies the boron concentration of the water and the water level of the tank. This assures the availability of the borated water source for boron dilution incidents and other transients using the RWST as a borated water source.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.7.b verifies the RWST temperature is greater than 35°F when it is the source of borated water and the RWST ambient air temperature is less than 35°F.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.7.c verifies the Boric Acid Storage Tank temperature is greater than 55°F when it is the source of borated water. This may be accomplished by verifying that the ambient air temperature in the vicinity of the BAST is greater than 55°F.

REFERENCE:

1. License Amendment 283

2. License Amendment 293

3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

TECHNICAL REQUIREMENT

3.1.2.8 Both of the following borated water sources shall be FUNCTIONAL:

a. At least one of the following Boric Acid Storage Tank(s) (BASTs) combinations:

1. One boric acid storage tank, with the tank contents in accordance with Figure 3.1-1 and a minimum temperature of 55°F, its associated gravity feed valve, and boric acid pump, or
2. Two boric acid storage tanks, with the weighted average of the combined contents of the tanks in accordance with Figure 3.1-1 and a minimum temperature of 55°F, their associated gravity feed valves, and boric acid pumps, or
3. Two boric acid storage tanks, each with contents in accordance with Figure 3.1-1 and a minimum temperature of 55°F, at least one gravity feed valve, and at least one boric acid pump.

and b. The refueling water storage tank with:

1. A minimum contained volume of 370,000 gallons of water,
2. A minimum boron concentration of 1720 ppm,
3. A minimum solution temperature of 50°F when in MODES 1 and 2, and
4. A minimum solution temperature of 35°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source FUNCTIONAL, manage the risk impact in accordance with the requirements of 10CFR50.65(a)(4).

TECHNICAL SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated FUNCTIONAL:

a. By performing the following as appropriate for the tank(s) being credited to satisfy this TRM requirement:

3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

1. Verify boron concentration of the BASTs at least once per 31 days and within 12 hours after each solution volume increase of ≥ 1% of indicated tank volume.
2. Verify boron concentration of the RWST at least once per 7 days.
3. Verify water level of both BASTs and the RWST at least once per 7 days.

b. When in MODES 3 and 4, at least once per 24 hours by verifying the RWST temperature is ≥35°F when the RWST ambient air temperature is <35°F.

c. When in MODES 1 and 2, at least once per 24 hours by verifying the RWST temperature is ≥50°F when the RWST ambient air temperature is <50°F.

d. At least once per 24 hours by verifying that the boric acid storage tank temperatures are greater than 55°F.

BASES:

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, and 4) boric acid pumps.

The minimum boration capability is sufficient to provide a SHUTDOWN MARGIN within the limits specified in the CORE OPERATING LIMITS REPORT at all temperatures above 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires an equivalent of 4900 gallons of 3.5% boric acid solution from the boric acid storage tanks plus 15,000 gallons of 1720 ppm borated water from the refueling water storage tank. The refueling water storage tank can also be used alone by feed-and-bleed using well under the 370,000 gallons of 1720 ppm borated water required.

The requirement for a minimum contained volume of 370,000 gallons of borated water in the refueling water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Technical Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

3/4.1 REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

TECHNICAL SURVEILLANCE REQUIREMENTS

The analysis to determine the boration requirements assumed that the Reactor Coolant System is borated concurrently with cooldown. In the limiting situation when letdown is not available, the cooldown is assumed to be initiated within 26 hours and cooldown to 220°F, is completed in the next 28 hours.

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source as well as the analysis assumption to determine the boration requirement to ensure adequate SHUTDOWN MARGIN.

Paragraph (a)(4) of 10CFR50.65 “Maintenance Rule” requires that licensees assess and manage the increase in risk that may result from proposed maintenance activities, including out of service Structures, Systems, and Components (SSCs). Management of risk involves consideration of incremental and aggregate impacts. The incremental risk increase is predominantly assessed using quantitative PRA methods, which measure the effect on core damage frequency and large early release frequency associated with the plant configuration. The risk increase is then managed in accordance with action thresholds defined by NUMARC 93-01. The aggregate impact is controlled by meeting Maintenance Rule requirements for establishing and meeting SSC performance criteria. These requirements include consideration of the risk significance of SSCs in establishing performance goals. Significant risk impacts are documented in the corrective action program and assessed as appropriate.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.8.a verifies the boron concentration in each water source and the water level in each water source. This assures the availability of the borated water source for boron dilution incidents and other transients using the BAST and RWST as a borated water sources.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.8.b verifies the RWST temperature is greater than 35°F when in MODES 3 and 4 and the RWST ambient air temperature is less than 35°F.

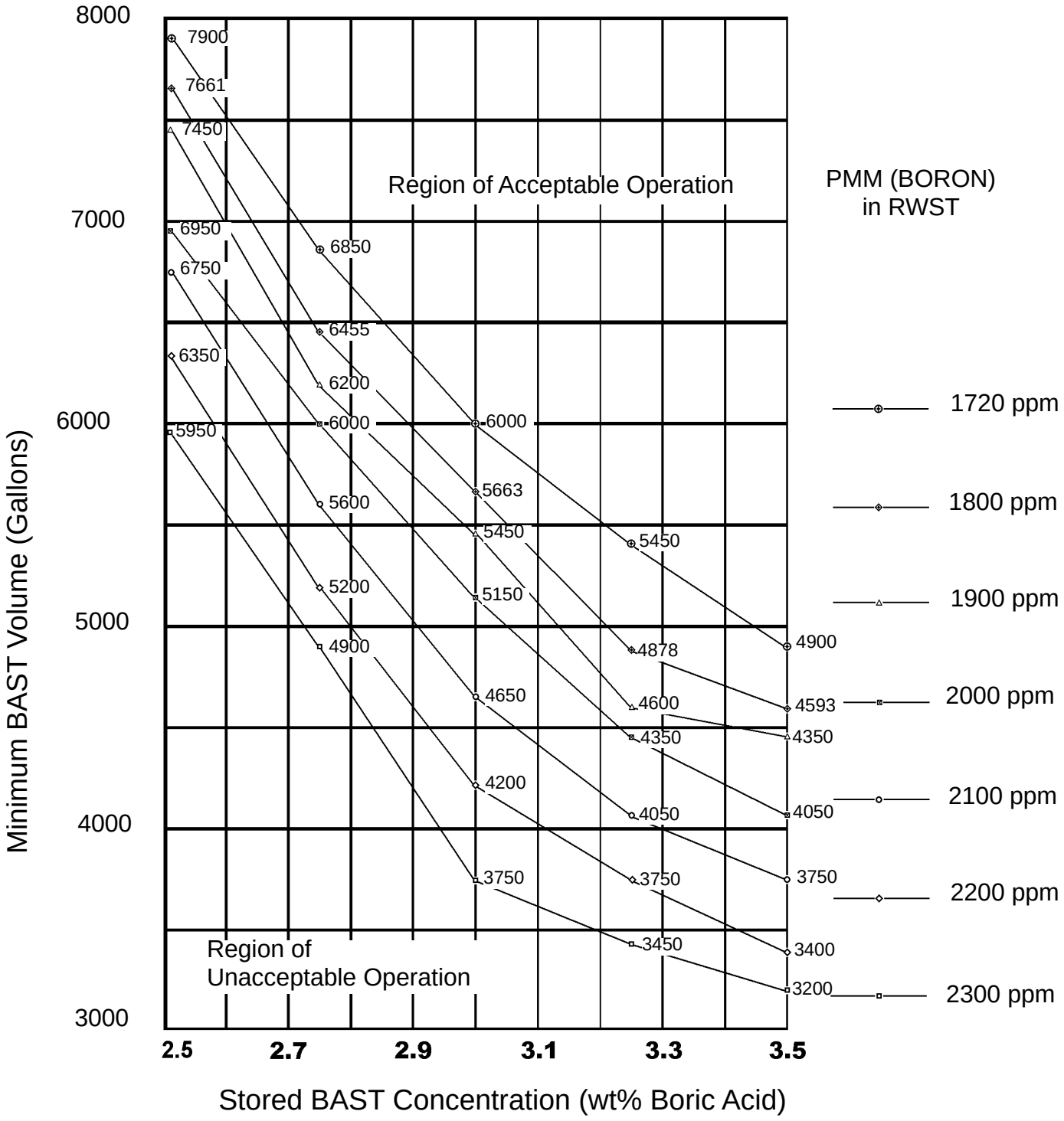
TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.8.c verifies the RWST temperature is greater than 50°F when in MODES 1 and 2 and the RWST ambient air temperature is less than 50°F.

TECHNICAL SURVEILLANCE REQUIREMENT 4.1.2.8.d verifies the BAST temperature is greater than 55°F. This may be accomplished by verifying that the ambient air temperature in the vicinity of the BAST is greater than 55°F.

REFERENCE:

1. License Amendment 283

FIGURE 3.1-1
MINIMUM VOLUME VS
STORED BAST CONCENTRATION
(WT% BORIC ACID)



3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.1.1 In accordance with Technical Specification LCO 3.3.1.1, Reactor Protective Instrumentation.

APPLICABILITY:

In accordance with Technical Specification LCO 3.3.1.1.

ACTION:

In accordance with Technical Specification LCO 3.3.1.1.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit as listed in TRM Table 3.3-2, in accordance with the test frequency specified in TS SR 4.3.1.1.3.

BASES:

In accordance with Technical Specification Bases section 3/4.3.1.

REFERENCE:

1. CR M3-00-02659.
2. Technical Specification section 3/4.3.1

Table 3.3-2
Reactor Protective Instrumentation Response Times

FUNCTIONAL Unit No.	FUNCTIONAL Unit Description	Response Time
1	Manual Reactor Trip	≤ 2.0 seconds
2	Power Level - High	
	2.a Neutron Flux power	≤ 0.4 seconds
	2.b Delta T Power	≤ 8.4 seconds[#] and ≤8.0 seconds[##]
3	Reactor Coolant Flow - Low	≤ 0.65 seconds
4	Pressurizer Pressure - High	≤ 0.90 seconds
5	Containment Pressure - High	≤ 0.90 seconds
6	Steam Generator Pressure - Low	≤ 0.90 seconds
7	Steam Generator Water Level - Low	≤ 0.90 seconds
8	Local power Density - High	
	8.a Neutron Flux Power	≤ 0.40 seconds*
	8.b Delta T Power	≤ 8.4 seconds[#] and ≤ 8.0 seconds[##]
9	Thermal Margin/Low Pressure	
	9.a Pressurizer Pressure/Neutron Flux Power	≤ 0.90 seconds
	9.b Delta T Power	≤ 0.89 seconds[#] and ≤ 8.0 seconds[##]
10	Loss of Turbine - Hydraulic Fluid Pressure	Not Applicable

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the signal shall be measured from the detector output or input of first electronic component in channel.

\# Response time includes contribution of RTDs

RTD response time only. This value is equivalent to the time interval required for the RTD's output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

PRESSURIZER HIGH PRESSURE CHANNELS

TECHNICAL REQUIREMENT

3.3.1.1.1 All pressurizer high pressure reactor protection channels shall be FUNCTIONAL.

APPLICABILITY:

In accordance with Technical Specification LCO 3.3.1.1 Applicability.

ACTION:

Restore any nonfunctional pressurizer high pressure reactor protection channel to FUNCTIONAL status within 30 days of placing the channel in the tripped condition, or be in MODE 3 within the next 6 hours with the failed channel in the bypassed condition.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

PRESSURIZER HIGH PRESSURE CHANNELS

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.1.1.4 In accordance with Technical Specification LCO 3.3.1.1 Surveillance Requirements.

BASES:

Technical Specification 3.3.1.1, “Instrumentation - Reactor Protective Instrumentation” ACTION 2, requires an inoperable reactor protection channel to be placed in the tripped condition if not restored to the OPERABLE status within 48 hours. With a pressurizer high pressure channel in the tripped condition, the high failure of a second pressurizer pressure channel would initiate a reactor trip and open both pressurizer power operated relief valves (PORVs). Opening the pressurizer PORVs would result in an undesired loss of primary coolant. To minimize the risk of a loss of primary coolant associated with a second failure, a 30 day restriction on plant operation with a failed high pressurizer pressure channel in the tripped condition is necessary. If the channel is not restored to OPERABLE status the plant shall be shutdown to MODE 3, where this function is not required by Technical Specifications. After entering MODE 3, the failed channel shall be placed in the bypassed condition.

REFERENCE:

1. Condition Report M3-00-02659

3/4.3 INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.2.1 In accordance with Technical Specification LCO 3.3.2.1, Engineered Safety Feature Actuation System Instrumentation.

APPLICABILITY:

In accordance with Technical Specification LCO 3.3.2.1.

ACTION:

In accordance with Technical Specification LCO 3.3.2.1.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit as listed in TRM Table 3.3-5, in accordance with the test frequency specified in TS SR 4.3.2.1.3.

BASES:

In accordance with Technical Specification Bases section 3/4.3.2.

REFERENCE:

1. CR M3-00-2659
2. Technical Specification section 3/4.3.2
3. NRC Generic Letter 96-01, “Testing of Safety Related Logic Circuits”

Table 3.3-5
Engineered Safety Features Response Times

Initiating Signal and Function	Response Time in Seconds	Equipment Tested
1. Manual		
a. SIAS		
Safety Injection (ECCS)	Not Applicable	Not Applicable
Containment Isolation	Not Applicable	Not Applicable
Enclosure Building Filtration System	Not Applicable	Not Applicable
b. CSAS		
Containment Spray	Not Applicable	Not Applicable
c. CIAS		
Containment Isolation	Not Applicable	Not Applicable
d. SRAS		
Containment Sump Recirculation	Not Applicable	Not Applicable
e. EBFAS		
Enclosure Building Filtration System	Not Applicable	Not Applicable
f. Auxiliary Feedwater Initiation	Not Applicable	Not Applicable
g. Main Steam Isolation	Not Applicable	Not Applicable
2. Pressurizer Pressure - Low		
a. Safety Injection (ECCS)		
1) High Pressure Safety Injection	≤ 25.0*/10.0**	See Table 3.3-5A
2) Low Pressure Safety Injection	≤ 45.0*/30.0**	See Table 3.3-5B
3) Charging Pump	≤ 35.0*/35.0**	See Table 3.3-5C
4) Containment Air Recirculation System	≤ 26.0*/15.0**	See Table 3.3-5D
b. Containment Isolation	≤ 7.5	See Table 3.3-5E
c. Enclosure Building Filtration System	≤ 45.0*/45.0**	See Table 3.3-5F

Table 3.3-5
Engineered Safety Features Response Times

Initiating Signal and Function	Response Time in Seconds	Equipment Tested
3. Containment Pressure - High		
a. Safety Injection (ECCS)		
1) High Pressure Safety Injection	≤ 25.0*/10.0**	See Table 3.3-5A
2) Low Pressure Safety Injection	≤ 45.0*/30.0**	See Table 3.3-5B
3) Charging Pumps	≤ 35.0*/35.0**	See Table 3.3-5C
4) Containment Air Recirculation System	≤ 26.0*/15.0**	See Table 3.3-5D
b. Containment Isolation	≤ 7.5	See Table 3.3-5E
c. Enclosure Building Filtration System	≤ 45.0*/45.00**	See Table 3.3-5F
d. Main Steam Isolation	≤ 6.9	See Table 3.3-5G
e. Feedwater Isolation	≤ 14	See Table 3.3-5H
4. Containment Pressure -- High-High		
a. Containment Spray	≤ 35.6*(1)/16.0**(1)	See Table 3.3-5I
5. DELETED		
6. Steam Generator Pressure - Low		
a. Main Steam Isolation	≤ 6.9	See Table 3.3-5G
b. Feedwater Isolation	≤ 14	See Table 3.3-5H
7. Refueling Water Storage Tank - Low		
a. Containment Sump Recirculation	≤ 120	See Table 3.3-5K
8. Steam Generator Level - Low		
a. Auxiliary Feedwater System	≤ 240/ > 180	See Table 3.3-5L
b. Steam Generator Blooding Isolation	≤ 10	

Table 3.3-5
Engineered Safety Features Response Times

Table Notation

* Diesel generator starting and sequence loading delays included.

** Diesel generator starting and sequence loading delays not included.

(1) Header fill time not included.

Table 3.3-5A
High Pressure Safety Injection

TECHNICAL REQUIREMENT	ESF Actuation Signal	Tested Equipment	Equipment Action
2a1, 3a1	SIAS	P41A	Start
2a1, 3a1	SIAS	P41B	Start
2a1, 3a1	SIAS	P41C	Start
2a1, 3a1	SIAS	2-SI-616	Open
2a1, 3a1	SIAS	2-SI-617	Open
2a1, 3a1	SIAS	2-SI-626	Open
2a1, 3a1	SIAS	2-SI-627	Open
2a1, 3a1	SIAS	2-SI-636	Open
2a1, 3a1	SIAS	2-SI-637	Open
2a1, 3a1	SIAS	2-SI-646	Open
2a1, 3a1	SIAS	2-SI-647	Open
2a1, 3a1	SIAS	2-SI-618	Close
2a1, 3a1	SIAS	2-SI-628	Close
2a1, 3a1	SIAS	2-SI-638	Close
2a1, 3a1	SIAS	2-SI-648	Close

Notes for Table 3.3-5A:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. HPSI Pumps - The pump under test is aligned with its recirculation valves open and its facility stop valve closed. Timing for the component actuation portion of the response time begins with the pump start signal and ends when the required Technical Specification pump PSID is developed.

3. HPSI Valves - Timing for the component actuation portion of the response time begins with the valve open (or close) signal and ends when the control board valve position lamps indicate full travel.

Table 3.3-5B
Low Pressure Safety Injection

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
2a2, 3a2	SIAS	P42A	Start
2a2, 3a2	SIAS	P42B	Start
2a2, 3a2	SIAS	2-SI-615	Open
2a2, 3a2	SIAS	2-SI-625	Open
2a2, 3a2	SIAS	2-SI-635	Open
2a2, 3a2	SIAS	2-SI-645	Open

Notes for Table 3.3-5B:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. LPSI Pumps - The pump under test is aligned with its recirculation valves open and its facility stop valve closed. Timing for the component actuation portion of the response time begins with the pump start signal and ends when the required Technical Specification pump PSID is developed.

3. LPSI Valves - Timing for the component actuation portion of the response time begins with the valve open (or close) signal and ends when the control board valve position lamps indicate full travel.

Table 3.3-5C
Charging Pump

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
2a3, 3a3 note 6	SIAS	P18A	Start
2a3, 3a3 note 6	SIAS	P18B	Start
2a3, 3a3 note 6	SIAS	P18C	Start
2a3, 3a3 note 3, 4	SIAS P19A	Start	
2a3, 3a3 note 3, 4	SIAS P19B	Start	
2a3, 3a3 note 3, 5	SIAS	2-CH-501	Close
2a3, 3a3 note 3, 5	SIAS	2-CH-510	Close
2a3, 3a3 note 3, 5	SIAS	2-CH-511	Close
2a3, 3a3 note 3, 5	SIAS	2-CH-512	Close
2a3, 3a3 note 3, 5	SIAS	2-CH-508	Open
2a3, 3a3 note 3, 5	SIAS	2-CH-509	Open
2a3, 3a3 note 3, 5	SIAS	2-CH-514	Open

Notes for Table 3.3-5C:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. Charging Pumps - Initially, one charging pump is running. The pump under test is initially off. Timing for the component actuation portion of the response time begins with the pump start signal and ends when charging header flow has increased by the required GPM.

Table 3.3-5C
Charging Pump

3. Response time testing of this component is outside the scope of Technical Specification Surveillance 4.3.2.1.3. However, verification of the response time of this component is necessary to ensure that the equipment is capable of supporting the safety function within the response times assumed in the accident analyses. If the overall response time of this component does not meet the required response time, FUNCTIONALITY and Reportability Determinations should be performed to ensure the accident analyses assumptions are met.

4. Boric acid pumps - Timing for the component actuation portion of the response time begins with the pump start signal and ends when the required Technical Specification pump PSID is developed.

5. The component actuation time of this component is tested under the IST program. The acceptance criteria for the component actuation time is established conservatively, such that the overall response time criteria will be met if the component actuation time is met.

6. Any charging pump with it’s control switch positioned to Pull-To-Lock (PTL) will be precluded from automatic operation under both normal and abnormal (e.g., SIAS) signal.

Table 3.3-5D
Containment Air Recirculation System

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
2a4, 3a4	SIAS	CAR F14A	Start Slow Speed
2a4, 3a4	SIAS	CAR F14B	Start Slow Speed
2a4, 3a4	SIAS	CAR F14C	Start Slow Speed
2a4, 3a4	SIAS	CAR F14D	Start Slow Speed
2a4, 3a4 note 3, 4	SIAS	2-RB-210 (HV-6739)	Close
2a4, 3a4 note 3, 4	SIAS	2-RB-28.3A (HV-6080)	Open
2a4, 3a4 note 3, 4	SIAS	2-RB-28.3B (HV-6084)	Open
2a4, 3a4 note 3, 4	SIAS	2-RB-28.3C (HV-6088)	Open
2a4, 3a4 note 3, 4	SIAS	2-RB-28.3D (HV-6092)	Open
2a4, 3a4 note 3, 4	SIAS	2-RB-8.1A (HV-6315)	Close
2a4, 3a4 note 3, 4	SIAS	2-RB-8.1B (HV-6316)	Close
2a4, 3a4 note 3, 4	SIAS	2-RB-68.1A (HV-6731)	Open
2a4, 3a4 note 3, 4	SIAS	2-RB-68.1B (HV-6735)	Open
2a4, 3a4 note 3, 4	CIAS	2-RB-402	Close
2a4, 3a4 note 3, 4	SIAS	2-SW-8.1A (HV-6308)	Open
2a4, 3a4 note 3, 4	SIAS	2-SW-8.1B (HV-6307)	Open
2a4, 3a4 note 3, 4	SIAS	2-SW-8.1C (HV-6706)	Open
2a4, 3a4 note 3, 4	SIAS	2-SW-3.2A (HV-6438)	Close
2a4, 3a4 note 3, 4	SIAS	2-SW-3.2B (HV-6439)	Close

Notes for Table 3.3-5D:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. The CAR fans are individually started and stopped in slow speed. Timing for the component actuation portion of the response time begins with the fan start signal and ends when the CAR fan motor current decays to the normal steady state, slow speed motor current (Amperes) following the initial motor in-rush current.

Table 3.3-5D
CONTAINMENT AIR RECIRCULATION SYSTEM

3. Response time testing of this component is outside the scope of Technical Specification Surveillance 4.3.2.1.3. However, verification of the response time of this component is necessary to ensure that the equipment is capable of supporting the safety function within the response times assumed in the accident analyses. If the overall response time of this component does not meet the required response time, FUNCTIONALITY and Reportability Determinations should be performed to ensure the accident analyses assumptions are met.

4. The component actuation time of this component is tested under the IST program. The acceptance criteria for the component actuation time is established conservatively, such that the overall response time criteria will be met if the component actuation time is met.

Table 3.3-5E
Containment Isolation

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
2b, 3b	CIAS	2-AC-12 (HV-8122)	Close
2b, 3b	CIAS	2-AC-15 (HV-8124)	Close
2b, 3b	CIAS	2-AC-20 (HV-8656)	Close
2b, 3b	CIAS	2-AC-47 (HV-8121)	Close
2b, 3b	CIAS	2-CH-089 (HV-2525)	Close
2b, 3b	CIAS	2-CH-198	Close
2b, 3b	CIAS	2-CH-505	Close
2b, 3b	CIAS	2-CH-506	Close
2b, 3b	CIAS	2-CH-516	Close
2b, 3b	CIAS	2-EB-100 (HV-8378)	Close
2b, 3b	CIAS	2-EB-88 (HV-8150)	Close
2b, 3b	CIAS	2-EB-89 (HV-8151)	Close
2b, 3b	CIAS	2-EB-91 (HV-8380)	Close
2b, 3b	CIAS	2-EB-92 (HV-8379)	Close
2b, 3b	CIAS	2-EB-99 (HV-8377)	Close
2b, 3b	CIAS	2-GR-11.1 (HV-9125)	Close
2b, 3b	CIAS	2-GR-11.2 (HV-9126)	Close
2b, 3b	CIAS	2-LRR-43.1 (HV-9015)	Close
2b, 3b	CIAS	2-LRR-43.2 (HV-9016)	Close
2b, 3b	CIAS	2-LRR-61.1 (HV-9230)	Close
2b, 3b	CIAS	2-MS-191A (HV-4250)	Close
2b, 3b	CIAS	2-MS-191B (HV-4251)	Close
2b, 3b	CIAS	2-MS-220A (HV-4246)	Close
2b, 3b	CIAS	2-MS-220B (HV-4248)	Close
2b, 3b	CIAS	2-PMW-43 (HV-7311)	Close
2b, 3b	CIAS	2-RC-001 (HV-1064)	Close
2b, 3b	CIAS	2-RC-002 (HV-1062)	Close
2b, 3b	CIAS	2-RC-003 (HV-1060)	Close
2b, 3b	CIAS	2-RC-45 (HV-7690)	Close
2b, 3b	CIAS	2-SI-312 (HV-7312)	Close
2b, 3b	CIAS	2-SSP-16.1 (HV-9151)	Close
2b, 3b	CIAS	2-SSP-16.2 (HV-9150)	Close

Table 3.3-5E
CONTAINMENT ISOLATION

Notes for Table 3.3-5E:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. Containment Isolation Valves - Timing for the component actuation portion of the response time begins with the valve close signal and ends when the control board valve position lamps indicate full travel.

Table 3.3-5F
Enclosure Building Filtration

The EBFAS fan being tested is aligned with its suction damper and header isolation damper open. Timing begins with a fan start signal and ends when EBFAS train D/P is stable.

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
2c, 3c	EBFAS	F25A	Start
2c, 3c	EBFAS	F25B	Start

Notes for Table 3.3-5F:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/ differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. The EBFAS fan under tested is aligned with its suction damper and header isolation damper open. Timing for the component actuation portion of the response time begins with the fan start signal and ends when EBFAS train D/P is stable.

Table 3.3-5G
Main Steam Isolation

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
3d, 6a	MSI	MSIV HV-4217A (MS-64A)	Close
3d, 6a	MSI	MSIV HV-4217B (MS-64A)	Close
3d, 6a	MSI	MSIV HV-4221A (MS-64B)	Close
3d, 6a	MSI	MSIV HV-4221B (MS-64B)	Close

Notes for Table 3.3-5G:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. The MSIV being tested is aligned open with its bypass valve open. Timing for the component actuation portion of the response time begins with the valve close signal and ends when the MSIV indicates closed.

Table 3.3-5H
Feedwater Isolation

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
3e, 6b	MSI	2-FW-38A (HV-5245)	Close
3e, 6b	MSI	2-FW-38B (HV-5247)	Close
3e, 6b	MSI	2-FW-41A (HV-5215)	Close
3e, 6b	MSI	2-FW-41B (HV-5216)	Close
3e, 6b	MSI	2-FW-42A (HV-5263)	Close
3e, 6b	MSI	2-FW-42B (HV-5264)	Close
3e, 6b	MSI	2-FW-51A (HV-5268)	Close
3e, 6b	MSI	2-FW-51B (HV-5269)	Close
3e, 6b	MSI	2-MS-265B (HV-4193)	Close
3e, 6b	MSI	2-MS-266B (HV-4209)	Close
3e, 6b	MSI	SGFP Turb H5A	Stop
3e, 6b	MSI	SGFP Turb H5B	Stop

Notes for Table 3.3-5H:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

Table 3.3-5I
Containment Spray

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
4a	CSAS	CS P43A	Start
4a	CSAS	CS P43B	Start
4a	CSAS	2-CS-4.1A (HV-3021)	Open
4a	CSAS	2-CS-4.1B (HV-3022)	Open

Notes for Table 3.3-5I:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. CS Pumps - The pump being tested is aligned with its minimum flow recirculation valves open. If Shutdown Cooling (SDC) is in operation, the CS header isolation valve is closed. If SDC is not in operation, the CS pump discharge valve is closed. Timing for the component actuation portion of the response time begins with the pump start signal and ends when the required pump differential pressure (psid) is established and stable.

3. CS Valves - Timing for the component actuation portion of the response time begins with the valve open signal and ends when the control board valve position lamps indicate full travel.

Table 3.3-5J

Deleted

Table 3.3-5K
Containment Sump Recirculation

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
7a	SRAS	2-CS-16.1A (HV-3008)	Open
7a	SRAS	2-CS-16.1B (HV-3009)	Open

Notes for Table 3.3-5K:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. Containment Sump Outlet Isolation Valves - Timing for the component actuation portion of the response time begins with the valve open signal and ends when the control board valve position lamps indicate full travel.

Table 3.3-5L
Auxiliary Feedwater System

TECHNICAL REQUIREMENT	ESF Actuation Signal	Equipment Tested	Equipment Action
8a	AFAIS	2-FW-43A (HV-5276)	Open
8a	AFAIS	2-FW-43B (HV-5279)	Open
8a	AFAIS	AFW P9A	Start
8a	AFAIS	AFW P9B	Start

Notes for Table 3.3-5L:

1. Response time is defined as the time interval from when the monitored process parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge/differential pressures reach their required values, etc.). The response time includes the signal processing time, ESAS actuation time, diesel generator starting and sequencing time (when applicable), and component actuation time (e.g., valve opening or closing stroke time, motor and pump acceleration time).

2. The AFW pump being tested is aligned for recirculation. Timing for the component actuation portion of the response time test begins with the start logic satisfied and ends when the aux feedwater pump develops the required differential pressure and valves HV-5276/FW-43A (HV-5279/FW-43B) are open.

Table 3.3-5M
Actuated Components Not Response Time Tested

The following components receive ESF Actuation Signals, but do not affect the response time of the ESF functions listed in Table 3.3-5. Therefore, the components listed below are not periodically time response tested

SF Actuation Signal	Equipment Not Tested	Equipment Action	Basis
SIAS	2-CH-196	Close	The timing for primary make-up water isolation does not affect the charging volumetric flow function.
SIAS	2-CH-515	Close	Timing for let down isolation does not affect the charging function.
SIAS	2-CHW-11 (HV-8847)	Close	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-CHW-12 (HV-8854)	Close	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-CHW-13 (HV-8850)	Close	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-CHW-14 (HV-8855)	Close	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-CHW-3 (HV-8846)	Open	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-CHW-33 (HV-8853)	Open	The timing for closure of this chilled water valve is not critical to establishing the safety function assumed in the accident analysis.
SIAS	2-SI- 634	Open	The SIT Valves are disabled in the open position during heat up.
SIAS	2-SI-614	Open	The SIT Valves are disabled in the open position during heat up.
SIAS	2-SI-624	Open	The SIT Valves are disabled in the open position during heat up.
SIAS	2-SI-644	Open	The SIT Valves are disabled in the open position during heat up.
SIAS	F112A	Inhibit	The timing of the inhibit of Battery Room Vent Fan ’A’ does not impact the initiation of the safety injection function.

Table 3.3-5M
Actuated Components Not Response Time Tested

SIAS	F112B	Inhibit	The timing of the inhibit of Battery Room Vent Fan 'B' does not impact the initiation of the safety injection function.
SIAS	F13A	Inhibit	CEDM cooling fans do not support completion of the safety injection function.
SIAS	F13B	Inhibit	CEDM cooling fans do not support completion of the safety injection function.
SIAS	F13C	Inhibit	CEDM cooling fans do not support completion of the safety injection function.
SIAS	F15A	Start	The response time for the start of the ESF room recirculation fans is not critical to the safety injection function.
SIAS	F15B	Start	The response time for start of the ESF room recirculation fans is not critical to the safety injection function.
SIAS	F54A	Start	The response time for start of the DC vital switchgear room cooling fans is not critical to the safety injection function.
SIAS	F54B	Start	The response time for start of the DC vital switchgear room cooling fans is not critical to the safety injection function.
SIAS	H7A	Start	The diesel generators are tested per the Tech Specs. The diesel starting and load sequencing time is included in the ESF Response Time when required.
SIAS	H7B	Start	The diesel generators are tested per the Tech Specs. The diesel starting and load sequencing time is included in the ESF Response Time when required.

Table 3.3-5M
Actuated Components Not Response Time Tested

SIAS	P5A	Start	The SW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.
SIAS	P5B	Start	The SW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.

Table 3.3-5M
Actuated Components Not Response Time Tested

SIAS	P5C	Start	The SW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.
SIAS	P11A	Start	The RBCCW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.

Table 3.3-5M
Actuated Components Not Response Time Tested

SIAS	P11B	Start	The RBCCW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.
SIAS	P11C	Start	The RBCCW pumps are not ESF equipment. It is impractical to response time test the component actuation time for these pumps. This is considered acceptable because the component actuation time (i.e., motor and pump acceleration time) is small compared with the time for signal processing, ESAS actuation, and diesel starting and load sequencing. The time for signal processing, ESAS actuation, and diesel starting and load sequencing are response time tested and the acceptance criteria are set consistent with the response time assumed in the accident analysis, including an allowance for motor and pump acceleration time. Periodic surveillance testing and the IST will detect any changes in the performance characteristics of the pump.
SIAS	P122A	Start	The timing for start of the vital switchgear chilled water pumps is not critical to establishing the safety function assumed in the accident analysis.
SIAS	P122B	Start	The timing for start of the vital switchgear chilled water pumps is not critical to establishing the safety function assumed in the accident analysis.
SIAS	X169A	Start	The timing for start of the DC Swgr Room Chillers is not critical to establishing the safety function assumed in the accident analysis.
SIAS	X169B	Start	The timing for start of the DC Swgr Room Chillers is not critical to establishing the safety function assumed in the accident analysis.

Table 3.3-5M
Actuated Components Not Response Time Tested

CIAS	2-AC-1 (HV-8050)	Close	Enclosure building damper, not a containment isolation valve. FUNCTIONAL testing verifies that the damper goes to the required position. The component actuation time is not critical to the accident analysis assumption. Therefore response time testing is not required.
CIAS	2-AC-3 (HV-8081)	Open	Enclosure building damper, not a containment isolation valve. FUNCTIONAL testing verifies that the damper goes to the required position. The component actuation time is not critical to the accident analysis assumption. Therefore response time testing is not required.
CIAS	2-AC-8 (HV-8127)	Open	Enclosure building damper, not a containment isolation valve. FUNCTIONAL testing verifies that the damper goes to the required position. The component actuation time is not critical to the accident analysis assumption. Therefore response time testing is not required.
CIAS	2-AC-11 (HV-8128)	Close	Enclosure building damper, not a containment isolation valve. FUNCTIONAL testing verifies that the damper goes to the required position. The component actuation time is not critical to the accident analysis assumption. Therefore response time testing is not required.
CIAS	2-AC-527 (HV-8262AB1)	Close	This valve isolates Radiation Monitor RM-8262 to prevent leakage from the radiation monitors when the containment isolation valves in the sample lines are re-opened manually to align the hydrogen analyzers and PASS after an accident. This valve is subject to the leak requirements of Appendix J and ISI/ IST requirements. The timing for closure for this valve is not critical to the accident analysis assumptions regarding containment isolation. The function of this valve is to protect the radiation monitor after the containment isolation valves have performed their functions and are opened manually to provide flow to the H2 analyzer and for PASS

Table 3.3-5M
Actuated Components Not Response Time Tested

CIAS	2-AC-528 (HV-8262AB2)	Close	This valve isolates Radiation Monitor RM-8262 to prevent leakage from the radiation monitors when the containment isolation valves in the sample lines are re-opened manually to align the hydrogen analyzers and PASS after an accident. This valve is subject to the leak requirements of Appendix J and ISI/ IST requirements. The timing for closure for this valve is not critical to the accident analysis assumptions regarding containment isolation. The function of this valve is to protect the radiation monitor after the containment isolation valves have performed their functions and are opened manually to provide flow to the H2 analyzer and for PASS
CIAS	2-AC-529 (HV-8123AB1)	Close	This valve isolates Radiation Monitor RM-8123 to prevent leakage from the radiation monitors when the containment isolation valves in the sample lines are re-opened manually to align the hydrogen analyzers and PASS after an accident. This valve is subject to the leak requirements of Appendix J and ISI/ IST requirements. The timing for closure for this valve is not critical to the accident analysis assumptions regarding containment isolation. The function of this valve is to protect the radiation monitor after the containment isolation valves have performed their functions and are opened manually to provide flow to the H2 analyzer and for PASS
CIAS	2-AC-530 (HV-8123AB2)	Close	This valve isolates Radiation Monitor RM-8123 to prevent leakage from the radiation monitors when the containment isolation valves in the sample lines are re-opened manually to align the hydrogen analyzers and PASS after an accident. This valve is subject to the leak requirements of Appendix J and ISI/ IST requirements. The timing for closure for this valve is not critical to the accident analysis assumptions regarding containment isolation. The function of this valve is to protect the radiation monitor after the containment isolation valves have performed their functions and are opened manually to provide flow to the H2 analyzer and for PASS

Table 3.3-5M
Actuated Components Not Response Time Tested

CIAS	F23	Stop	The containment PURGE supply fan does not perform a containment isolation function. The response time for stopping the fan on a CIAS is not critical to the safety function assumed in the accident analysis.
CIAS	F39A	Stop	The containment radiation monitor fan does not perform a containment isolation function. The response time for stopping the fan on a CIAS is not critical to the safety function assumed in the accident analysis.
CIAS	F39B	Stop	The containment radiation monitor fan does not perform a containment isolation function. The response time for stopping the fan on a CIAS is not critical to the safety function assumed in the accident analysis.
CIAS	P33A	Stop	Containment sump pumps are stopped on a CIAS for equipment protection. The time it takes to stop the pumps does not effect the completion of the containment isolation function.
CIAS	P33B	Stop	Containment sump pumps are stopped on a CIAS for equipment protection. The time it takes to stop the pumps does not effect the completion of the containment isolation function.
EBFAS	2-EB-40 (HV-8063)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-41 (HV-8070)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-42 (HV-8073)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-50 (HV-8153)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-51 (HV-8074)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.

Table 3.3-5M
Actuated Components Not Response Time Tested

EBFAS	2-EB-52 (HV-8079)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-55 (HV-8654)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-56 (HV-8695)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-60 (HV-8143)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-61 (HV-8062)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-72 (HV-8650)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-EB-73 (HV-8651)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-106 (HV-8133)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-107 (HV-8247)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-116 (HV-8249)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.

Table 3.3-5M
Actuated Components Not Response Time Tested

EBFAS	2-HV-117 (HV-8248)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-202 (HV-8361)	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	2-HV-207 (HV-8002)	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	2-HV-208 (HV-8003B)	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	2-HV-210 (HV-8005)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. This damper is disable in the open position Therefore, individual component time response testing is not required.
EBFAS	2-HV-211 (HV-8004)	Close	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-212A (HV-8006)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	2-HV-212B (HV-8007)	Open	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.

Table 3.3-5M
Actuated Components Not Response Time Tested

EBFAS	2-HV-495	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	2-HV-496	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	2-HV-497	Close	Control room isolation damper. The current TRM requirements do not include a response time for this function. Condition Report, CR M2-99-0282 was issued to investigate the need for response time testing of this component in accordance with the assumptions in the radiological accident analysis.
EBFAS	F32A	Start	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	F32B	Start	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	X61A	Start	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
EBFAS	X61B	Start	The EBFAS time response test is a system level test. Timing begins with a fan start signal and ends when EBFAS train D/P is stable. Therefore, individual component time response testing is not required.
MSIAS	2-MS-65A (HV-4218)	Close	The MSIV bypass valve is normally closed. It is only opened for a short time to equalize the pressure across the MSIV during plant start-up. Therefore response time testing is not required.

Table 3.3-5M
Actuated Components Not Response Time Tested

MSIAS	2-MS-65B (HV-4222)	Close	The MSIV bypass valve is normally closed. It is only opened for a short time to equalize the pressure across the MSIV during plant start-up. Therefore response time testing is not required.
SRAS	2-RB-13.1A (HV-6050)	Open	This RBCCW valve opens on an SRAS to direct RBCCW flow to the shutdown cooling heat exchanger. FUNCTIONAL testing of this valve is adequate to demonstrate that the safety function of this valve is met. Therefore, response time testing is not required.
SRAS	2-RB-13.1B (HV-6055)	Open	This RBCCW valve opens on an SRAS to direct RBCCW flow to the shutdown cooling heat exchanger. FUNCTIONAL testing of this valve is adequate to demonstrate that the safety function of this valve is met. Therefore, response time testing is not required.
SRAS	2-SI-659	Close	Closure of the RWST minflow valves prevents highly radioactive water from being sent to the RWST following an SRAS. FUNCTIONAL testing of this valve is adequate to demonstrate that the safety function of this valve is met. Therefore, response time testing is not required.
SRAS	2-SI-660	Close	Closure of the RWST minflow valves prevents highly radioactive water from being sent to the RWST following an SRAS. FUNCTIONAL testing of this valve is adequate to demonstrate that the safety function of this valve is met. Therefore, response time testing is not required.
SRAS	P42A	Stop	LPSI pumps, P42A and P42B are not time response tested because a failure of the pump to trip within the required SRAS response time would be considered a equipment failure issue as opposed to a time response issue.
SRAS	P42B	Stop	LPSI pumps, P42A and P42B are not time response tested because a failure of the pump to trip within the required SRAS response time would be considered a equipment failure issue as opposed to a time response issue.
LOAD SHED	52-A302	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.

Table 3.3-5M
Actuated Components Not Response Time Tested

LOAD SHED	52-A304	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	52-A312	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	52-A401	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	52-A410	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	52-A411	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	52-A505	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	L105	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	L106	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	LO46A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	LO46B L	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.

Table 3.3-5M
Actuated Components Not Response Time Tested

LOAD SHED	P11A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P11B	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P11C	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P41A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P41B	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P41C	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P42A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P42B	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P43A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P43B	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.

Table 3.3-5M
Actuated Components Not Response Time Tested

LOAD SHED	P5A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P5B	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P5C	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P98	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
LOAD SHED	P9A	Load Shed	Response time testing of the load shed function is not required. The load shed function is tested implicitly during the Loss of Normal Power (LNP) test.
AEAS	2-EB-40	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-41	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-42	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-50	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-51	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-52	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-55	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-56	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation

Table 3.3-5M
Actuated Components Not Response Time Tested

AEAS	2-EB-60	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-61	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-72	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-EB-73	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-165	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-170	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-171	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-202	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-207	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-208	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-210	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-211	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-212A	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation

Table 3.3-5M
Actuated Components Not Response Time Tested

AEAS	2-HV-212B	Open	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-495	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-496	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	2-HV-497	Close	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	F20	Stop	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	F25A	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	F25B	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	F32A	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	F32B	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	X61A	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation
AEAS	X61B	Start	The fuel handling accident analysis does not credit any AEAS related ACTION for accident mitigation

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.2 INCORE DETECTOR INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.3.2 The incore detection system shall be FUNCTIONAL with:

a. At least 34 (75% of 45) incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant, and

c. A minimum of 9 FUNCTIONAL incore detector segments at each detector segment level.

Where;

A FUNCTIONAL incore detector segment shall consist of a FUNCTIONAL rhodium detector constituting one of the segments in a fixed detector string.

A FUNCTIONAL incore detection location shall consist of a string in which at least three of the four incore detector segments are FUNCTIONAL.

APPLICABILITY:

When the incore detection system is used for:

a. Monitoring the AZIMUTHAL POWER TILT,

b. Recalibration of the excore neutron flux detection system, or

c. Monitoring the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR or the linear heat rate.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.2 INCORE DETECTOR INSTRUMENTATION

TECHNICAL REQUIREMENT

ACTION:

Either Action a **AND** b (if applicable) **OR** Action c must be satisfied.

a. With less than 34 (75% of 45) incore detector locations FUNCTIONAL and with Technical Requirement 3.3.3.2.b and 3.3.3.2.c satisfied, the system can be used for the above applicable monitoring or calibration functions provided that the number of FUNCTIONAL incore detector locations remain at least 27 (60% of 45 incore detection locations) and initiate corrective action to restore at least 34 (75% of 45 incore detector locations) FUNCTIONAL,

AND

b. If less than 34 (75% of 45) incore detector locations are FUNCTIONAL during the first 30% power flux map following a core alteration, then limit power to 30% or less until a cycle specific fuel misload analysis has been performed and implemented,

OR

c. Declare the incore detection system nonfunctional. With the incore detection system nonfunctional, do not use the system for the above applicable monitoring or calibration functions.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.2 INCORE DETECTOR INSTRUMENTATION

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated FUNCTIONAL:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

    1. Monitoring the AZIMUTHAL POWER TILT.

    2. Recalibration of the excore neutron flux detection system, or

    3. Monitoring the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR or the linear heat rate.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

BASES:

The FUNCTIONALITY of the incore detectors with the specified minimum compliment of equipment ensures that the measurements from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

Analyses performed by Framatome and summarized in Reference 2, performed in accordance with the NRC approved methodology (Reference 1), conclude that the uncertainty limits on Linear Heat Rate and F^T_r, 7% and 6% respectively, are met with at least 27 (60% of 45) incore detector locations FUNCTIONAL. Reference 2 also demonstrates that the ASI uncertainty with 60% FUNCTIONAL detector locations is accommodated in the FSAR Chapter 14 Safety Analyses. In addition, when Action b is applicable, a cycle-specific analysis as was done in support of Cycle 26 (Reference 2), must be performed to demonstrate the ability of the incore detection system to detect a fuel misload that could result in exceedance of a specified acceptable fuel design limit.

REFERENCE:

1. Siemens Report No. EMF-96-029(P)(A), dated January 1997, volumes 1 and 2, “Reactor Analysis System for PWRs, Volume 1-Methodology Description, Volume 2 - Benchmarking Results.”

2. LBDCR 18-MP2-007, “Functionality Definition Change for the Incore Detector Instrumentation.”

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.3 SEISMIC INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.3.3. The seismic monitoring instrumentation channels shown in TRM Table 3.3-7 shall be FUNCTIONAL.

APPLICABILITY:

All MODES.

ACTION:

a. With the number of FUNCTIONAL seismic monitoring channels less than required by TRM Table 3.3-7, restore the nonfunctional channel(s) to FUNCTIONAL status within 30 days or otherwise enter the condition into the corrective action program.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.3 SEISMIC INSTRUMENTATION

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.3.3 Each of the above seismic monitoring instrumentation channels shall be Demonstrated FUNCTIONAL by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in TRM Table 4.3-4.

BASES:

The FUNCTIONALITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design of the facility.

REFERENCE:

1. Condition Report M3-00-02659

Table 3.3-7
Seismic Monitoring Instrumentation Response Times

INSTRUMENT CHANNEL	SENSOR LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS FUNCTIONAL
1. TIME HISTORY ACCELEROGRAPHS			
a. Containment Base Slab	El. -24’0” @ 215° outside of containment	.001 to 1 g	1
b. Containment Structure	El. 75’0” @ 215° outside of containment	.001 to 1 g	1
c. Auxiliary Building	El. 14’6” in maintenance work area	.001 to 1 g	1
d. Intake Structure	El. 18’0” on South wall	.001 to 1 g	1
e. Free Field	El. 14’6” ground level on pad 139’ South-East of Condensate Storage Tank	.001 to 1 g	1
2. PEAK ACCELEROGRAPHS			
a. Containment Base Slab	El. -24’0” outside of containment	0 to 1 g	1
b. Steam Generator Support	El. -0’7” S/G #1	0 to 1 g	1
c. Pressurizer Support	El. 14’6”	0 to 1 g	1
d. Safety Injection Tank Support	El. 38’6”	0 to 1 g	1
3. SEISMIC TRIGGER			
a. Containment Base Slab	El. -24’0” @ 215° outside of containment	.005 to .02 g	1
4. RESPONSE SPECTRUM RECORDER			
a. Containment Base Slab	El. -24’0” outside of containment	-	1

Table 4.3-4
Seismic Monitoring Instrumentation Technical Surveillance Requirements

INSTRUMENT CHANNEL	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. TIME HISTORY ACCELEROGRAPHS			
a. Containment Base Slab	M	R	SA
b. Containment Structure	M	R	SA
c. Auxiliary Building	M	R	SA
d. Intake Structure	M	R	SA
e. Free Field	M	R	SA
2. PEAK ACCELEROGRAPHS			
a. Containment Base Slab	N.A.	R	N.A.
b. Steam Generator Support	N.A.	R	N.A.
c. Pressurizer Support	N.A.	R	N.A.
d. Safety Injection Tank Support	N.A.	R	N.A.
3. SEISMIC TRIGGER			
a. Containment Base Slab	N.A.	R	SA
4. RESPONSE SPECTRUM RECORDER			
a. Containment Base Slab	N.A.	R	N.A.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.3.4 The meteorological monitoring instrumentation channels shown in TRM Table 3.3-8 shall be FUNCTIONAL.

APPLICABILITY:

All MODES.

ACTION:

a. With the number of FUNCTIONAL meteorological monitoring channels less than required by TRM Table 3.3-8, suspend all releases of gaseous radioactive material from the radwaste gas decay tanks until the nonfunctional channel(s) is restored to FUNCTIONAL status.

b. With one or more required meteorological monitoring channels nonfunctional for more than 7 days, enter condition into the corrective action program.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.4 METEREOLOGICAL INSTRUMENTATION

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.3.4 Each meteorological monitoring instrumentation channel shall be demonstrated FUNCTIONAL by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in TRM Table 4.3-5.

BASES:

The FUNCTIONALITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23, “Onsite Meteorological Programs.”

If the ACTIONS of the TECHNICAL REQUIREMENT cannot be met, the condition is documented in the corrective action program and assessed, as appropriate.

REFERENCE:

1. Condition Report M3-00-02659

Table 3.3-8
Meteorological Monitoring InstrumentatioN

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM CHANNELS FUNCTIONAL
1. WIND SPEED			
a. Nominal Elev.	142 ft.	± 0.22 m/sec*	1
b. Nominal Elev.	374 ft.	± 0.22 m/sec*	1
2. WIND DIRECTION			
a. Nominal Elev.	142 ft.	± 5°	1
b. Nominal Elev.	374 ft.	± 5°	1
3. AIR TEMPERATURE - DELTA T			
a. Nominal Elev.	142 ft.	± 0.18°F	1
b. Nominal Elev.	374 ft.	± 0.18°F	1

* Starting speed of anemometer shall be < 0.45 m/sec.

Table 4.3-5
Meteorological Monitoring Instrumentation Technical Surveillance Requirements

INSTRUMENT CHANNEL	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 142	D	SA
b. Nominal Elev. 374	D	SA

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3 3.7 As a minimum, the fire detection instrumentation for each fire detection zone in TRM Table 3.3-10 shall be FUNCTIONAL.

APPLICABILITY:

Whenever equipment in that fire detection zone is required to be FUNCTIONAL.

ACTION:

With the number of FUNCTIONAL fire detection instrument(s) less than the minimum number of FUNCTIONAL requirements of TRM Table 3.3-10:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the nonfunctional instrument(s) at least once per hour unless the instrument(s) is located inside the containment. Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005.

   If the instrument(s) are located inside the containment, then inspect the associated containment area at least once per 8 hours or monitor and record the average containment air temperatures manually once an hour or rely on the plant process computer to continuously monitor associated area containment temperatures with an alarm setpoint of 150°F using the following temperature elements:

   - TE8098 (located 30’ elev. C and D RCP’s)
   - TE9770 (located 18’ elev. West Penetration)
   - TE9771 (located 18’ elev. East Penetration)
   - TE8097 (located 30’ elev. A and B RCPS)

b. Restore the nonfunctional equipment to FUNCTIONAL status within 14 days, or develop a plan and schedule for restoring the equipment to FUNCTIONAL status.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments, which are accessible during plant operation, shall be demonstrated FUNCTIONAL at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated FUNCTIONAL by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The circuitry associated with the supervision of the above fire detection instruments and circuits, per NFPA 72-D, shall be demonstrated FUNCTIONAL at least once per 6 months.

BASES:

FUNCTIONALITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility Fire Protection program.

In the event that a portion of the fire detection instrumentation is nonfunctional, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the nonfunctional instrumentation is restored to FUNCTIONALITY.

Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005.

Fire detection in containment is limited to the areas above the reactor coolant pumps and above the electrical penetration areas. The plant process computer (PPC) continuously monitors air temperatures throughout containment to compute technical specification containment average air temperature. However monitoring the average temperature is not ideal for detecting incipient stage fires in the reactor coolant pump and the cable penetration areas.

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

BASES: (Continued)

Individual temperature monitoring elements located in containment are set to alarm at 150°F. When the PPC alarms it will display any appropriate AOP's associated with the alarm. The PPC must display the procedure for fire response A0P2559, "Fire" to credit these points as an appropriate compensatory measure for a nonfunctional fire detection system in containment.

The temperature monitoring elements indicated were chosen based on their proximity to the existing fire detection systems. In the event the temperature element associated with the nonfunctional fire detector or the PPC fails, it is acceptable to monitor and record average containment air temperatures manually once an hour.

Based on the ability of the PPC to produce historic records of temperatures for the locations specified, this compensatory measure would provide an acceptable means of monitoring for fire conditions and would document regulatory adherence.

REFERENCE:

1. Condition Report M3-00-2659

3/4.3 INSTRUMENTATION

3/4.3.3.8 ACCIDENT RADIATION MONITORING INSTRUMENTATION

TECHNICAL REQUIREMENT

3.3.3.8 The mid and high range channels of the Wide Range Gas Monitor (WRGM) shall be FUNCTIONAL with an alarm setpoint set less than the EAL Alert Classification level.

APPLICABILITY:

MODES 1, 2, 3, 4, 5, and 6.

ACTION:

a. With an alarm setpoint of the mid or high range channel(s) of the WRGM exceeding the EAL Alert Classification level, adjust the setpoint to within the required value within twenty-four hours or declare the channel(s) nonfunctional.

b. With mid or high range channel(s) of WRGM nonfunctional, within 72 hours, initiate an alternate method of monitoring the appropriate parameters (not required if the nonfunctionality is only due to alarm setpoint)

   AND

   Restore required channel(s) to FUNCTIONAL status within 30 days, or otherwise initiate corrective actions.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.3.3.8 The mid and high range channels of the WRGM shall be demonstrated FUNCTIONAL by the performance of the following surveillances at the frequencies shown:

   a. CHANNEL CHECK once per 12 hours.

   b. CHANNEL FUNCTIONAL TEST once per 31 days.

   c. CHANNEL CALIBRATION once per 18 months.

3/4.3 INSTRUMENTATION

3/4.3.3.8 ACCIDENT RADIATION MONITORING INSTRUMENTATION

BASES:

When the Unit 1 stack high range radiation monitor was deactivated and removed, Unit 2 had to install a new high range radiation monitor (the Wide Range Gas Monitor, or WRGM, RM-8169) to monitor its release to the Millstone stack during an accident.

FUNCTIONALITY and TECHNICAL SURVEILLANCE REQUIREMENT for RM-8169 are needed to replace the equivalent requirements applicable to the Unit 1 high range monitor. FUNCTIONALITY and TECHNICAL SURVEILLANCE REQUIREMENT for the WRGM are consistent with technical specifications which have similar requirements for the Unit 2 vent high range radiation monitor (RM-8168). These requirements will assure that radiation monitoring is available during an accident with releases to the environment for assessment purposes.

The alternate method of monitoring specified in TR ACTION b. is contained in procedures. It includes use of the lower range (channel 1) of the WRGM, if FUNCTIONAL and on scale.

REFERENCE:

1. CR-01-01975, “New Wide Range Gas Monitor Does Not Appear to Meet NUREG-0737 Requirement”

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

TECHNICAL REQUIREMENT

3.4.7 The reactor coolant system chemistry shall be maintained within the limits specified in TRM Table 3.4-1.

APPLICABILITY:

All MODES.

ACTION:

MODES 1, 2, 3 and 4.

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, implement established plant chemistry guidelines.

b. With any one or more chemistry parameter in excess of its Transient Limit, implement established plant chemistry guidelines.

MODES 5 and 6.

With the concentration of either chloride or fluoride in the reactor coolant system in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤500 psia, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.7 The reactor coolant system chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in TRM Table 4.4-1.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

BASES:

The limitations on reactor coolant system contaminants ensure that corrosion of the reactor coolant system is minimized and reduce the potential for reactor coolant system leakage or failure due to stress corrosion. Maintaining the concentrations of the contaminants within the Steady State Limits shown on TRM Table 3.4-1 provides adequate corrosion protection to ensure the structural integrity of the reactor coolant system over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent.

Established plant chemistry guidelines are contained in MP-PROC-CH-CP 2802A, “Primary Chemistry Control.”

The TECHNICAL SURVEILLANCE REQUIREMENTS provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

REFERENCE:

1. License Amendment No. 266

Table 3.4-1
Reactor Coolant System
Chemistry Limits

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN	≤ 0.10 ppm*	≤ 1.00 ppm*
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.10 ppm	≤ 1.00 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

Table 4.4-1
Reactor Coolant System
Chemistry Limits Technical Surveillance Requirements

PARAMETER	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

*Not required with $T_{avg} \leq 250°F$.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

TECHNICAL REQUIREMENT

3.4.9.1 In accordance with Technical Specification LCO 3.4.9.1, Pressure/ Temperature Limits.

APPLICABILITY:

In accordance with Technical Specification 3.4.9.1.

ACTION:

In accordance with Technical Specification 3.4.9.1.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.9.1

a. In accordance with Technical Specifications Surveillance Requirement 4.4.9.1a.

b. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-3.

c. The results of the examinations in TECHNICAL SURVEILLANCE REQUIREMENT 4.4.9.1b shall be used to update Technical Specifications Table 3.4-2 and Technical Specifications Figures 3.4-2a and 3.4-2b.

Table 4.4-3
Reactor Vessel Material
Irradiation Surveillance Schedule

CAPSULE	WITHDRAWAL SCHEDULE (EFPY)
W-97	3.0
W-97 (Flux Monitor)	10.0
W-104	10.0
W-83	15.3
W-277	EOL
W-263	Spare
W-284	Spare

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

BASES:

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50. Removal of reactor vessel irradiation surveillance specimens does not constitute a CORE ALTERATION per Technical Specification 1.12.

REFERENCE:

1. License Amendment No. 272

2. NRC safety evaluation for approval of changes to Table 4.4-3, “Millstone Power Station, Unit No. 2 - Changes to the Reactor Pressure Vessel Surveillance Capsule Withdrawal Schedule (TAC No. MB9149),” dated October 1, 2003

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9.2 PRESSURIZER

TECHNICAL REQUIREMENT

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY:

MODES 1, 2, 3, 4 and 5.

ACTION:

With any of the above limits exceeded, perform the following:

a. Restore the temperature to within limit within 30 minutes.

AND

b. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the pressurizer and determine that the pressurizer remains acceptable for continued operation within 72 hours. Otherwise, be in at least MODE 3 within the next 6 hours and reduce pressurizer pressure to less than 500 psia within the following 30 hours.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperature and spray water temperature differential shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9.2 PRESSURIZER

TECHNICAL REQUIREMENT

3.4.9.2.1 The pressurizer spray and aux spray piping shall be limited to a maximum spray water temperature differential of 350°F.

APPLICABILITY:

MODES 1, 2, 3, 4 and 5.

ACTION:

With the above limit exceeded, perform the following:

a. Minimize cycling of pressurizer spray and aux spray until spray water temperature differential can be reduced below 350°F.

AND

b. Write a CR to notify Engineering to track the number of thermal transients.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.9.2.1

a. The pressurizer spray water temperature differential shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

b. When pressurizer main spray is unavailable, and the pressurizer aux spray is in use, the water temperature differential shall be determined to be within the limits at lease once per 30 minutes.

REFERENCE:

1. Specification 7604-M-290
2. DM2-00-0019-99, Revised Design Specification for Pressurizer Spray and Auxiliary Spray Piping

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9.2 PRESSURIZER

BASES:

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. Verification that pressurizer temperature conditions are within the limits of TECHNICAL REQUIREMENT 3.4.9.2, at least once per 30 minutes, is required when undergoing planned changes of ≥10°F. The 30 minute time interval permits assessment and correction for temperature deviations within a reasonable time.

REFERENCE:

1. License Amendment No. 266

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9.4 SHUTDOWN COOLING OVERPRESSURE PROTECTION

TECHNICAL REQUIREMENT

3.4.9.4 Shutdown Cooling System overpressure protection shall be provided as follows: (1)

a. All HPSI pumps shall be prevented from automatically injecting into the RCS when the RCS is not vented through a vent of ≥ 2.2 in^2.

b. A maximum of one HPSI pump may be capable of automatically injecting into the RCS when the RCS is vented through a vent of ≥ 2.2 in^2.

APPLICABILITY:

When the Shutdown Cooling System is connected to the RCS and the head is on the reactor vessel.

ACTION:

With more than the maximum allowed number of HPSI pumps capable of automatically injecting into the RCS, take immediate ACTION to comply with the TECHNICAL REQUIREMENT 3.4.9.4 above.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.9.4 Verify all required HPSI pumps are prevented from automatically injecting into the RCS at least once per 12 hours by:(2)

a. Racking down the motor circuit breaker from the power supply circuit; or

b. Shutting and tagging the discharge valve with the key lock on the control panel (2-51-654 or 2-SI-656); or

c. Placing the pump control switch in the pull-to-lock position.

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(1) Refer to Technical Specification 3.4.9.3 for additional restrictions concerning HPSI pump operation.

(2) In MODE 4, Technical Specification 3.5.3 requires at least one HPSI pump to be OPERABLE. The only acceptable method to provide Shutdown Cooling System overpressure protection, without affecting the OPERABILITY of this HPSI pump, is to place the pump control switch in the pull-to-lock position.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.9.4 SHUTDOWN COOLING OVERPRESSURE PROTECTION

BASES:

An inadvertent start of a HPSI pump could result in the overpressurization of the SDC System. To prevent this, it is necessary to disable HPSI pump automatic injection capability into the RCS whenever the SDC System is connected to the RCS. If the RCS is not vented, the automatic injection capability of all HPSI pumps must be prevented. This restriction may be relaxed if the RCS is adequately vented. If a vent of ≥ 2.2 in^2 has been established, a maximum of one HPSI may be capable of automatically injecting into the RCS. If the reactor vessel head has been removed, there is no restriction on HPSI pump automatic injection capability.

If a loss of RCS inventory or reduction in SHUTDOWN MARGIN event occurs, the appropriate response will be to correct the situation by starting RCS makeup pumps. If the loss of inventory or SHUTDOWN MARGIN is significant, this may necessitate the use of HPSI pumps that are being maintained not capable of automatically injecting into the RCS in accordance with this requirement. The use of the HPSI pump(s) to restore RCS inventory or SHUTDOWN MARGIN will require entry into the associated ACTION statement. The ACTION statement requires immediate ACTION to comply with the requirements. The restoration of RCS inventory or SHUTDOWN MARGIN can be considered to be part of the immediate ACTION to restore the HPSI pump(s) to a not capable of automatically injecting status. While recovering RCS inventory or SHUTDOWN MARGIN, RCS pressure will be maintained below SDC design pressure. After RCS inventory or SHUTDOWN MARGIN has been restored, the HPSI pump(s) should be immediately made not capable of automatically injecting and the ACTION statement exited.

REFERENCE:

1. Technical Specification LCO 3.4.9.3

2. Technical Specification LCO 3.5.3

3. Condition Report M3-00-02659

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2, 3 COMPONENTS

TECHNICAL REQUIREMENT

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with TECHNICAL SURVEILLANCE REQUIREMENT 4.4.10.

APPLICABILITY:

MODES 1, 2, 3, 4, 5, and 6, except when a ASME Code Class 1, 2, or 3 component is not required to be FUNCTIONAL.

ACTION:

a. With one or more ASMECode Class 1, 2 and 3 component(s) in a degraded or nonconforming condition(s), perform the following ACTIONS within 72 hours:*

   1. Determine that structural integrity is still maintained in the degraded or nonconforming condition;

      or

   2. Isolate the affected component(s) from service.

b. If the above ACTION is not completed within 72 hours, immediately declare the affected component(s) nonfunctional.

* - For ASME Code Class 1 and 2 components which have PRESSURE BOUNDARY LEAKAGE, the affected component(s) must be isolated from service per the requirements of TS 3.4.6.2, if applicable or declared nonfunctional.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.10 The structural integrity of ASME Code Classes 1, 2 and 3 components shall be inspected in accordance with the Inservice Inspection Program.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2, 3 COMPONENTS

BASES:

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a.

This TR requires that the structural integrity of ASME Code Class 1, 2 and 3 components be maintained. Structural integrity is the ability to withstand specified loading, with an acceptable margin, without collapse, rupture, brittle fracture, or unstable flaw growth.

Components that may experience unanalyzed loads, have a deficient original design analysis, or have pressure boundary wall loss, but nevertheless can be shown to meet the minimum requirements of the original construction Code or ASME Section III Appendix F, are considered to retain their structural integrity. For components with degraded material properties or relevant conditions as defined in ASME Section XI, the criteria for structural integrity are contained or referenced in Section XI.

Alternative evaluation criteria may be used to determine structural integrity when approved by the NRC. In addition to the initial discovery, supplemental NDE may be required to fully characterize the degraded condition. Preliminary judgments regarding structural integrity may be expressed considering the available information and the expectation that a final determination of structural integrity will result once all characterization, evaluation and documentation are complete.

For ASME Code Class 1 or 2 components which have PRESSURE-BOUNDARY LEAKAGE, the structural integrity of the component is considered not to be maintained and the affected component(s) must be isolated from service per the requirements of TS 3.4.6.2, if applicable. If the affected component(s) cannot be isolated from service, the affected component(s) is immediately declared nonfunctional and the required ACTIONS of the applicable TS and TRs apply. For all other structural integrity (i.e. non pressure boundary leakage) degraded or nonconforming conditions of Class 1 and Class 2 components, the structural integrity of the affected component will be evaluated consistent with the requirements of this TR.

If the required determination of structural integrity cannot be accomplished within 72 hours or the results are indeterminate or structural integrity is not maintained, or the affected component is not isolated, or cannot be isolated, ACTION must proceed to declare the affected component(s) nonfunctional, as specified in ACTION statement 'b.' The appropriate TS or TR shall be considered not met, as appropriate.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY - ASME CODE CLASS 1, 2, 3 COMPONENTS

BASES: (Continued)

Surveillance Requirement 4.4.10 requires performing inservice inspections of ASME Section XI Code Class 1, 2 and 3 components in accordance with TS 4.0.5.

Inservice inspection of ASME Code Class 1, 2 and 3 components are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda, as required by 10 CFR 50.55a(g), to ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(a)(3) and (g)(6)(i). The surveillance intervals specified in Section XI of the ASME Code apply.

REFERENCE:

1. Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," November 7, 1991, and Revision 1, October 8, 1997.

2. 10 CFR 50.55a(b)(2)(xiii), "Flaws in Class 3 Piping," Federal Register (Vol. 64, No. 183), September 22,1999.

3. Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1,2 and 3 Piping," June 15,1990; and August 16,1990.

4. Code Case N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping."

5. Code Case N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping."

6. Elinor G. Adensam, U.S. NRC to Martin L. Bowling, "Millstone Nuclear Power Station, Unit Nos. 2 and 3 - Request for an Alternative to the Requirements of the ASME Code," TAC Nos. MA3889 and MA3884, dated February 23,1999.

7. License Amendment No. 264. LBDCR 02-2-11 September 13, 2002.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

TECHNICAL REQUIREMENT

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series capable of being powered from emergency buses shall be FUNCTIONAL at each of the following locations:

a. Reactor vessel head

b. Pressurizer steam space

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

a. With any one of the PORVs or PORV block valves of the pressurizer steam space vent path nonfunctional in MODES 1, 2 or 3, follow the ACTION requirements of Technical Specification 3.4.3.

b. With any one of the PORVs or PORV block valves of the pressurizer steam space vent path nonfunctional in MODE 4, follow the ACTION requirements of Technical Specification 3.4.9.3.

c. With the reactor vessel head vent path nonfunctional, STARTUP and/or POWER OPERATION may continue provided that the nonfunctional vent path is maintained closed with power removed from the valve actuator of all the valves in the nonfunctional vent path. Restore the reactor vessel head vent path to FUNCTIONAL status within 30 days or otherwise enter the condition into the corrective action program.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated FUNCTIONAL at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during COLD SHUTDOWN or REFUELING.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

BASES:

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The FUNCTIONALITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The flow test verifies that each flowpath for the reactor vessel head vent and the pressurizer steam space vent is FUNCTIONAL. This verification can be performed by using a series of overlapping tests to ensure flow is verified through all parts of the system.

REFERENCE:

1. License Amendment No. 266
2. DCR M2-02006
3. DCR M2-05002

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - $T_{AVG} \geq 300°F$

TECHNICAL REQUIREMENT

3.5.2 In accordance with Technical Specification LCO 3.5.2, ECCS Subsystems - T_{avg} >300°F

APPLICABILITY: In accordance with Technical Specification LCO 3.5.2.

ACTION:

In accordance with Technical Specification LCO 3.5.2.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated FUNCTIONAL:

a. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

b. By conducting a flow balance verification immediately prior to returning to service any portion of a subsystem after the completion of a modification that could alter system flow characteristics. The injection leg flow rate shall be as follows:

1. HPSI Headers - the sum of the three lowest injection flows must be $\geq$ 471 gpm. The sum of the four injection flows must be $\leq$ 675 gpm.

2. LPSI Header - the sum of the three lowest injection flows must be $\geq$ 2850 gpm. The sum of the four injection flows must be

$$\leq 4500 + \left[\frac{\text{RWST level (\%)} - 10(\%)}{90\%} \times 200\right]$$

3/4.5 EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{AVG} \geq 300°F$

TECHNICAL SURVEILLANCE REQUIREMENTS

BASES:

TECHNICAL SURVEILLANCE REQUIREMENT 4.5.2.a verifies by a visual inspection that no loose debris (rags, trash, clothing, etc.) is present in the containment that could be transported to the containment sumps and cause restriction of the pump suction during LOCA conditions. This is a good housekeeping item, which is an integral part of any maintenance or surveillance activity. It does not verify FUNCTIONALITY of the ECCS or any ECCS functions assumed in the safety analysis. This approach is consistent with NUREG-1432, which does not contain a requirement to inspect the containment sump prior to establishing CONTAINMENT INTEGRITY. In addition, the containment sump will continue to be inspected every 18 months as required by Technical Specification SR 4.5.2.j.

TECHNICAL SURVEILLANCE REQUIREMENT 4.5.2.b verifies proper flow distribution following any modifications that could alter system flow characteristics. The purpose of the ECCS throttle valve surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Verification of the correct position for the mechanical and/or electrical valve stops can be performed by either of the following methods:

1. Visually verify the valve opens to the designated throttled position; or

2. Manually position the valve to the designated throttled position and verify that the valve does not move when the applicable valve control switch is placed to “OPEN.”

Post maintenance testing associated with a system modification, which is controlled by plant procedures, will include verification of proper flow distribution if the associated modification could adversely affect the flow distribution. Without this verification, the respective system could not be declared FUNCTIONAL.

REFERENCE:

1. License Amendment 283

3/4.6 CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

TECHNICAL REQUIREMENT

3.6.3.1 The Containment Isolation Valves listed in TRM Table 3.6-1 Containment Isolation Valve List, shall be FUNCTIONAL.

APPLICABILITY:

In accordance with Technical Specification 3.6.3.1.

ACTION:

In accordance with Technical Specification 3.6.3.1.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.6.3.1 In accordance with Technical Specification 3.6.3.1 Surveillance Requirements.

BASES:

In accordance with the Technical Specifications Bases section 3/4.6.3 (Containment Isolation Valves).

In addition, the attached TRM Table 3.6-1 provides a list of containment isolation valves which are a subset of those valves identified by the FSAR, Table 5.2-11, “Containment Structure Isolation Valve Information.” The containment isolation valves identified are main process isolation valves that are within the containment penetration boundary and conform to the requirements of 10 CFR 50 Appendix A (GDC 55, 56, or 57). Manual test, vent or drain valves within the containment penetration boundary are also identified. An explanation of the several notes included on TRM Table 3.6-1 follows:

Column 1 provides the containment penetration number and containment isolation valve description.

Column 2 provides the component local identification.

Column 3 provides the stroke time closure requirements.

Note “a” provides the extended allowed outage time of 72 hours for those isolation valves that are located on penetrations for closed systems as defined by the Standard Review Plan (SRP) 6.2.4. The 72 hour allowed outage time also applies to the RBCCW containment isolation valves, which are located on a closed system as defined by Millstone Unit 2 original design basis.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

Note “b” indicates the small (1 inch or less) valves used as Test, Drain or Vent valves, which are normally closed.

REFERENCE:

The attached TRM Table has been compiled from a review of the P&ID drawings, the FSAR Chapter 5 and has been supplemented by plant inspections of the listed penetrations.

1. MP2 Final Safety Analysis Report (FSAR) Sections 5.2.8.1.2, 5.2.8.2.1 and Table 5.2-11.b.

2. Condition Report M3-00-02659

3. Amendment 278 to Millstone Unit 2 Technical Specifications

Table 3.6-1
Containment Isolation Valve List

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 1 - Demin Water to Ctmt Iso - Inner - check	2-PMW-3	NA
No. 1 - Demin Water to Ctmt Iso - Outer - AOV, CIAS	2-PMW-43	5
No. 1 - Demin Water to Ctmt Iso - Outer – manual	2-PMW-165	NA
No. 2 - RCS Letdown Iso - Inner - AOV, SIAS	2-CH-515	5
No. 2 - RCS Letdown Iso - Inner - AOV, CIAS	2-CH-516	5
No. 2 - RCS Letdown Iso - Outer- AOV, CIAS	2-CH-089	5
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-763	NA
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-658	NA
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-260	NA
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-082	NA
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-083	NA
No. 2 - RCS Letdown Isolation - Outer - manual – b	2-CH-067	NA
No. 2 - RCS Letdown Isolation - Inner - manual – b	2-CH-991	NA
No. 3 - RCS Charging Isolation - Inner - AOV	2-CH-517	NSR
No. 3 - RCS Charging Isolation - Inner - AOV	2-CH-518	NSR
No. 3 - RCS Charging Isolation - Inner - AOV	2-CH-519	NSR
No. 3 - RCS Charging Isolation - Outer - MOV	2-CH-429	NSR
No. 3 - RCS Charging Isolation - Inner - manual	2-CH-434	NA
No. 3 - RCS Charging Isolation - Inner - manual – b	2-CH-001	NA
No. 3 - RCS Charging Isolation - Inner - manual – b	2-CH-002	NA
No. 3 - RCS Charging Iso - Inner - manual – b	2-CH-003	NA
No. 3 - RCS Charging Iso - Inner - manual – b	2-CH-004	NA
No. 3 - RCS Charging Iso - Inner - manual – b	2-CH-443	NA
No. 3 - RCS Charging Iso - Outer - manual – b	2-CH-710	NA
No. 3 - RCS Charging Iso - Inner - manual – b	2-CH-714	NA
No. 3 - RCS Charging Iso - Inner - manual – b	2-CH-661	NA
No. 3 - RCS Charging Iso - Outer - Relief	2-CH-986	NA
No. 3 - RCS Charging Iso - Inner - manual	2-RC-071	NA
No. 4 - Containment Spray Iso - Outer - MOV-CSAS	2-CS-4.1A	NSR
No. 4 - Containment Spray Iso - Inner - check	2-CS-5A	NA
No. 4 - Containment Spray Iso - Outer - manual – b	2-CS-049A	NA
No. 4 - Containment Spray Iso - Outer - manual – b	2-CS-049C	NA

Table 3.6-1
Containment Isolation Valve List

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 5 - Containment Spray Iso - Outer - MOV-CSAS	2-CS-4.1B	NSR
No. 5 - Containment Spray Iso - Inner - check	2-CS-5B	NA
No. 5 - Containment Spray Iso - Outer - manual – b	2-CS-101	NA
No. 6 - Safety Inj. Isolation - Inner - check	2-SI-706D	NA
No. 6 - Safety Inj. Isolation - Outer – manual – b	2-SI-095	NA
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-041F	NA
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-742D	NA
No. 6 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-645	NSR
No. 6 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-646	NSR
No. 6 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-647	NSR
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-734	NA
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-735	NA
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-848	NA
No. 6 - Safety Inj. Isolation - Outer - manual	2-SI-160	NA
No. 6 - Safety Inj. Isolation - Outer - manual	2-SI-161	NA
No. 6 - Safety Inj. Isolation - Outer - manual – b	2-SI-163	NA
No. 7 - Safety Inj. Isolation - Inner – check	2-SI-706A	NA
No. 7 - Safety Inj. Isolation - Outer - manual – b	2-SI-742A	NA
No. 7 - Safety Inj. Isolation - Outer - manual – b	2-SI-107	NA
No. 7 - Safety Inj. Isolation - Outer - manual – b	2-SI-715	NA
No. 7 - Safety Inj. Isolation - Outer - manual – b	2-SI-717	NA
No. 7 - Safety Inj. Isolation - Outer -MOV SIAS	2-SI-615	NSR
No. 7 - Safety Inj. Isolation - Outer -MOV SIAS	2-SI-616	NSR
No. 7 - Safety Inj. Isolation - Outer -MOV SIAS	2-SI-617	NSR
No. 7 - Safety Inj. Isolation - Outer -manual – b	2-SI-041A	NA
No. 7 - Safety Inj. Isolation - Outer -manual – b	2-SI-716	NA
No. 7 - Safety Inj. Isolation - Outer -manual – b	2-SI-718	NA
No. 8 - Safety Inj. Isolation - Inner -check	2-SI-706C	NA
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-733	NA
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-041D	NA
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-742C	NA
No. 8 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-635	NSR
No. 8 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-636	NSR
No. 8 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-637	NSR
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-041E	NA
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-110	NA
No. 8 - Safety Inj. Isolation - Outer - manual – b	2-SI-846	NA

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 9 - Safety Inj. Isolation - Inner - check	2-SI-706B	NA
No. 9 - Safety Inj. Isolation - Outer - manual – b	2-SI-722	NA
No. 9 - Safety Inj. Isolation - Outer - manual – b	2-SI-721	NA
No. 9 - Safety Inj. Isolation - Outer - manual – b	2-SI-742B	NA
No. 9 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-625	NSR
No. 9 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-626	NSR
No. 9 - Safety Inj. Isolation - Outer - MOV - SIAS	2-SI-627	NSR
No. 9 - Safety Inj. Isolation - Outer - manual – b	2-SI-723	NA
No. 9 - Safety Inj. Isolation - Outer - manual – b	2-SI-720	NA
No. 10 - RCS-SDC Suction Isolation - Inner - MOV	2-SI-651	NSR
No. 10 - RCS-SDC Suction Isolation - Inner - manual – b	2-SI-043A	NA
No. 10 - RCS-SDC Suction Isolation - Outer - manual – b	2-SI-102A	NA
No. 10 - RCS-SDC Suction Isolation - Outer - manual – b	2-SI-101A	NA
No. 10 - RCS-SDC Suction Isolation - Outer - manual	2-SI-709	NA
No. 11 - SIT Recirc & Test Line Isolation - Outer - manual – a	2-SI-463	NA
No. 12 - Ctmt Sump SI Recirc Line Iso - Outer - MOV-SRAS	2-CS-16.1A	NSR
No. 12 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-130	NA
No. 12 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-132	NA
No. 12 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-133	NA
No. 12 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-134	NA
No. 13 - Ctmt Sump SI Recirc Line Iso - Outer - MOV-SRAS	2-CS-16.1B	NSR
No. 13 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-135	NA
No. 13 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-137	NA
No. 13 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-138	NA
No. 13 - Ctmt Sump SI Recirc Line Iso - Outer - manual – b	2-CS-139	NA
No. 14 - Ctmt Sump Drain Line Iso - Inner - AOV-CIAS	2-SSP-16.1	5
No. 14 - Ctmt Sump Drain Line Iso - Outer - AOV-CIAS	2-SSP-16.2	5
No. 14 - Ctmt Sump Drain Line Iso - Outer - manual – b	2-SSP-51	NA
No. 14 - Ctmt Sump Drain Line Iso - Outer - manual – b	2-SSP-73	NA

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 15 - Main Fd Iso - Outer - Check; Air Assist Open - AOV-MSI – a	2-FW-5A	NA
No. 15 - Aux Fd Iso - Outer - Check; Air Assist Open - AOV – a	2-FW-12A	NA
No. 15 - Main Fd Iso - Outer - manual – b, a	2-FW-86	NA
No. 15 - Main Fd Iso - Outer - manual - b, a	2-FW-261A	NA
No. 15 - Main Fd Iso - Outer - manual - Hydrazine supply – a	2-FW-15A	NA
No. 16 - Main Fd Iso - Outer - Check; Air Assist Open - AOV-MSI – a	2-FW-5B	NA
No. 16 - Main Fd Iso - Outer - Check; Air Assist Open - AOV – a	2-FW-12B	NA
No. 16 - Main Fd Iso - Outer - manual - Hydrazine supply – a	2-FW-15B	NA
No. 16 - Main Fd Iso - Outer - manual - b, a	2-FW-182	NA
No. 17 - Personnel Airlock LLRT Iso – b	2-AC-520	NA
No. 17 - Personnel Airlock LLRT Iso – b	2-AC-521	NA
No. 17 - Personnel Airlock LLRT Iso – b	2-AC-522	NA
No. 17 - Personnel Airlock Equalizing - Outer - Ball	2-AC-523	NA
No. 17 - Personnel Airlock Equalizing - Inner - Ball	2-AC-524	NA
No. 19 - Main Stm Iso - Outer -AOV - Steam dump valve – a	2-MS-190A	NSR
No. 19 - Main Stm Iso - Outer - manual – b, a	2-MS-371	NA
No. 19 - Main Stm Iso - Outer - MOV-Stm to AFW pump – a	2-MS-201	NSR
No. 19 - Main Stm Iso - Outer - manual bypass, a	2-MS-459	NA
No. 19 - Main Stm Iso - Outer -manual - b, a	2-MS-297	NA
No. 19 - Main Stm Iso - Outer - AOV - MSI closes valve – a	2-MS-265B	6
No. 19 - Main Stm Iso - Outer - Relief valves(8) – a	2-MS-247 thru 254	NA
No. 19 - Main Stm Iso - Outer - MSI - air open - check – a	2-MS-64A	6
No. 19 - Main Stm Iso - Outer - MOV - MSI closes – a	2-MS-65A	NSR
No. 19 - Main Stm Iso - Outer - manual – b, a	2-MS-255	NA
No. 20 - Main Stm Iso - Outer - AOV - Steam dump valve, – a	2-MS-190B	NSR
No. 20 - Main Stm Iso - Outer - manual - b, a	2-MS-369	NA
No. 20 - Main Stm Iso - Outer - MOV - Stm to AFW pump – a	2-MS-202	NSR
No. 20 - Main Stm Iso - Outer - manual bypass – a	2-MS-458	NA
No. 20 - Main Stm Iso - Outer - manual - b, a	2-MS-296	NA
No. 20 - Main Stm Iso - Outer - AOV - MSI closes valve – a	2-MS-266B	6
No. 20 - Main Stm Iso - Outer - Relief valves (8), a	2-MS-239 thru 246	NA
No. 20 - Main Stm Iso - Outer - MSI - air open - check – a	2-MS-64B	6

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 20 - Main Stm Iso - Outer - MOV - MSI closes – a	2-MS-65B	NSR
No. 20 - Main Stm Iso - Outer - manual - b, a	2-MS-258	NA
No. 21 - RCS Loop 1 smpl Iso. - Inner - AOV-CIAS	2-RC-001	5
No. 21 - RCS Pzr Surge Line smpl Iso - Inner - AOV-CIAS	2-RC-002	5
No. 21 - RCS Pzr Stm Space smpl Iso - Inner - AOV-CIAS	2-RC-003	5
No. 21 - RCS - PDT&QT smpl Iso - Inner - AOV-CIAS	2-LRR-61.1	5
No. 21 - RCS Sample Line Iso - Inner - sample valve - manual – b	2-RC-434	NA
No. 21 - RCS Sample Line Iso - Outer - AOV-CIAS	2-RC-045	5
No. 21 - RCS Sample Line Iso - Inner - sample valve - manual – b	2-RC-435	NA
No. 22 - SG No. 1 Blwdwn Iso - Outer - AOV-CIAS, AFAIS, Hi Rad – a	2-MS-220A	5
No. 23 - SG No. 2 Blwdwn Iso - Outer - AOV-CIAS, AFAIS, Hi Rad – a	2-MS-220B	5
No. 24 - RBCCW Sply to RCPs Iso - Outer - MOV – a	2-RB-30.1A	NSR
No. 24 - RBCCW Sply to RCPs - Outer - manual - b, a	2-RB-289	NA
No. 25 - RBCCW Sply to CAR coolers - Outer - AOV – a	2-RB-28.1D	NSR
No. 25 - RBCCW Sply to CAR coolers - Outer - manual - b, a	2-RB-282	NA
No. 26 - RBCCW Sply to CAR coolers - Outer - AOV – a	2-RB-28.1B	NSR
No. 26 - RBCCW Sply to CAR coolers - Outer - manual - b, a	2-RB-283	NA
No. 26 - RBCCW Sply to CAR coolers - Outer - manual - b, a	2-RB-345	NA
No. 27 - RBCCW Sply to CAR coolers - Outer - AOV – a	2-RB-28.1A	NSR
No. 27 - RBCCW Sply to CAR coolers - Outer - manual - b, a	2-RB-236	NA
No. 28 - RBCCW Sply to CAR coolers - Outer - AOV - a	2-RB-28.1C	NSR
No. 28 - RBCCW Sply to CAR coolers - Outer - manual - b, a	2-RB-237	NA
No. 29 - RBCCW Return fm RCPs - Outer - MOV – a	2-RB-37.2A	NSR
No. 29 - RBCCW Return fm RCPs - Outer - manual - b, a	2-RB-298	NA
No. 29 - RBCCW Return fm RCPs - Outer - manual - b, a	2-RB-297A	NA
No. 30 - RBCCW Ret fm CAR cooler - Outer - AOV – a	2-RB-28.2D	NSR

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 30 - RBCCW Ret fm CAR cooler - Outer – SIAS AOV – a	2-RB-28.3D	NSR
No. 31 - RBCCW Ret fm CAR cooler - Outer - AOV – a	2-RB-28.2B	NSR
No. 31 - RBCCW Ret fm CAR cooler - Outer – SIAS AOV – a	2-RB-28.3B	NSR
No. 32 - RBCCW Ret fm CAR cooler - Outer - AOV – a	2-RB-28.2A	NSR
No. 32 - RBCCW Ret fm CAR cooler - Outer - SIAS AOV – a	2-RB-28.3A	NSR
No. 33 - RBCCW Ret fm CAR cooler - Outer - AOV – a	2-RB-28.2C	NSR
No. 33 - RBCCW Ret fm CAR cooler - Outer - SIAS AOV – a	2-RB-28.3C	NSR
No. 34 - Nitrogen Sply to Ctmt Iso - Outer - AOV-CIAS	2-SI-312	5
No. 34 - Nitrogen Sply to Ctmt - Outer - manual – b	2-SI-045	NA
No. 35 - Prim Drain Tank Disch - Inner Iso - AOV-CIAS	2-LRR-43.1	5
No. 35 - Prim Drain Tank Disch - Outer Iso - AOV-CIAS	2-LRR-43.2	5
No. 35 - Prim Drain Tank Disch - Outer Iso - manual – b	2-LRR-291	NA
No. 35 - Prim Drain Tank Disch - Outer Iso - manual – b	2-LRR-293	NA
No. 35 - Prim Drain Tank Disch - Outer Iso - manual – b	2-LRR-295	NA
No. 36 - Instr Air BU to Ctmt - Inner Iso check	2-IA-569	NA
No. 36 - Instr Air BU to Ctmt - Inner - manual – b	2-IA-572	NA
No. 36 - Instr Air BU to Ctmt - Outer Iso - manual	2-IA-566	NA
No. 37 - Instr Air Sply to Ctmt - Inner - Iso - check	2-IA-43	NA
No. 37 - Instr Air Sply to Ctmt - Outer Iso - AOV	2-IA-27.1	NSR
No. 37 - Instr Air Sply to Ctmt - Outer - Manual – b	2-IA-40	NA
No. 38 - Station Air Sply to Ctmt - Inner - manual – b	2-SA-28	NA
No. 38 - Station Air Sply to Ctmt - Outer Iso - manual	2-SA-19	NA
No. 38 - Station Air Sply to Ctmt - Inner Iso - check	2-SA-22	NA
No. 39 - Ctmt Air PURGE Sply - Outer Iso - AOV-HiRad	2-AC-4	NSR
No. 39 - Ctmt Air PURGE Sply - Inner Iso - AOV-HiRad	2-AC-5	NSR
No. 39 - Ctmt Air PURGE Sply - Outer Iso - manual – b	2-AC-21	NA

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 40 - Ctmt Air PURGE Sply - Outer Iso - AOV-HiRad	2-AC-7	NSR
No. 40 - Ctmt Air PURGE Sply - Inner Iso - AOV-HiRad	2-AC-6	NSR
No. 40 - Ctmt Air PURGE Sply - Inner - manual – b	2-AC-31	NA
No. 42 - Fuel Transfer Tube - Inner - manual	Blank Flange	NA
No. 42 - Fuel Transfer Tube Flange Test - Inner - manual – b	2-RW-31	NA
No. 42 - Fuel Transfer Tube - Inner - manual – b	2-RW-291	NA
No. 42 - Sleeve Drain Valve - Inner - manual – b	2-RW-292	NA
No. 43 - RCP seals bleedoff Inner AOV-CIAS	2-CH-506	5
No. 43 - RCP seals bleedoff Outer AOV-CIAS	2-CH-505	5
No. 43 - RCP seals bleedoff Outer AOV-CIAS	2-CH-198	5
No. 43 - RCP seals bleedoff Line Outer - manual - b	2-CH-744	NA
No. 43 - RCP seals bleedoff Line Outer - manual - b	2-CH-768	NA
No. 43 - RCP seals bleedoff Line Outer - manual - b	2-CH-758	NA
No. 43 - RCP seals bleedoff Line Outer - manual - b	2-CH-701	NA
No. 49 - Fire Header Ctmt Sply - Outer Iso - manual	2-FIRE-108	NA
No. 49 - Fire Header Ctmt Sply - Outer - manual – b	2-FIRE-125	NA
No. 49 - Fire Header Ctmt Sply - Inner Iso - check	2-FIRE-109	NA
No. 51 - Waste Gas fm Ctmt - Inner Iso - AOV-CIAS	2-GR-11.1	5
No. 51 - Waste Gas fm Ctmt - Outer - manual – b	2-GR-63	NA
No. 51 - Waste Gas fm Ctmt - Outer Iso - AOV-CIAS	2-GR-11.2	5
No. 53 - RBCCW Sply to RCPs - Outer - MOV -a	2-RB-30.1B	NSR
No. 53 - RBCCW Sply to RCPs - Outer - manual - b, a	2-RB-291	NA
No. 54 - RBCCW Return fm RCPs - Outer - MOV – a	2-RB-37.2B	NSR
No. 54 - RBCCW Return fm RCPs - Outer - manual - b, a	2-RB-300	NA
No. 54 - RBCCW Return fm RCPs - Outer - manual - b, a	2-RB-299A	NA
No. 61 - Ctmt Air Monitor Sply - Inner Iso - AOV, CIAS	2-EB-88	5
No. 61 - Ctmt Air Monitor Sply - Outer - manual – b	2-AC-101	NA
No. 61 - Ctmt Air Monitor Sply - Outer Iso - AOV, CIAS	2-AC-12	5

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 62 - Ctmt Air Monitor Rtrn - Inner Iso - check	2-AC-54	5
No. 62 - Ctmt Air Monitor Rtrn - Outer Iso - manual – b	2-AC-103	NA
No. 62 - Ctmt Air Monitor Rtrn - Outer Iso - AOV, CIAS	2-AC-15	5
No. 63 - Ctmt ILRT Instrument Test - Inner Iso - manual	2-AC-117	NA
No. 63 - Ctmt ILRT Instrument Test - Outer Iso - manual	2-AC-114	NA
No. 63 - Ctmt ILRT Instr. - Outer Test Flange	Blank Flange	NA
No. 64 - Ctmt ILRT Instrument Test - Inner Iso - manual	2-AC-116	NA
No. 64 - Ctmt ILRT Instrument Test - Outer Iso - manual	2-AC-112	NA
No. 64 - Ctmt ILRT Instr. - Outer Test Flange	Blank Flange	NA
No. 65 - SG #1 Bldn Sample - Outer Iso - AOV, CIAS a	2-MS-191A	5
No. 67 - Refueling Water Purif. Rtrn - Inner Iso - manual	2-RW-232	NA
No. 67 - Refueling Water Purif. Rtrn - Outer Iso - manual	2-RW-21	NA
No. 67 - Refueling Water Purif. Rtrn - Outer - manual – b	2-RW-158	NA
No. 68 - Refueling Water Purif. Sply - Inner Iso - manual	2-RW-154	NA
No. 68 - Refueling Water Purif. Sply - Outer Iso - manual	2-RW-63	NA
No. 68 - Refueling Water Purif. Sply - Outer - manual – b	2-RW-159	NA
No. 72 - SG #2 Bldn Sample - Outer Iso - AOV, CIAS a	2-MS-191B	5
No. 82 - Hyd. PURGE - Inner Iso - AOV - CIAS, HiRad	2-EB-91	5
No. 82 - Hyd. PURGE - Outer Iso - AOV - CIAS, HiRad	2-EB-92	5
No. 82 - Hyd. PURGE -Outer - manual – b	2-EB-120	NA
No. 82 - Hyd. PURGE - Inner - manual	2-EB-86	NA
No. 83 - Hyd. PURGE - Inner Iso - AOV - CIAS, HiRad	2-EB-100	5
No. 83 - Hyd. PURGE - Outer Iso - AOV - CIAS, HiRad	2-EB-99	5
No. 83 - Hyd. PURGE -Outer - manual – b	2-EB-121	NA
No. 85 - Ctmt ILRT Pressurization - Outer Iso. - manual	SF-01 Sp. Flnge	NA
No. 85 - Ctmt ILRT Pressurization - Outer - manual – b	2-AC-107	NA
No. 85 - Ctmt ILRT Pressurization - Inner Iso Flange	Blank Flange	NA

Table 3.6-1
Containment Isolation Valve LisT

Pen. No. and Function Description	Valve No.	Maximum IsolationTime (seconds)
No. 86 - Ctmt Air Monitor Sply - Inner Iso - AOV, CIAS	2-EB-89	5
No. 86 - Ctmt Air Monitor Sply - Outer - manual – b	2-AC-102	NA
No. 86 - Ctmt Air Monitor Sply - Outer Iso - AOV, CIAS	2-AC-47	5
No. 87 - Ctmt Air Monitor Rtrn - Inner Iso - check	2-AC-55	NA
No. 87 - Ctmt Air Monitor Rtrn - Outer - manual – b	2-AC-104	NA
No. 87 - Ctmt Air Monitor Rtrn - Outer Iso - AOV, CIAS	2-AC-20	5

a = The Allowed Outage Time = 72 hours

b = Vent, Drain, or Test Connection

NA = Closure Time is Not Applicable to Manual Valves, Relief Valves, Blind Flanges and Check Valves

NSR = No Stroke time Required. This is applicable for valves that are open, go open, open during an accident or valves that are normally shut and stay shut during an accident.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

TECHNICAL REQUIREMENT

3.6.4.1 Two independent containment hydrogen monitors shall be FUNCTIONAL.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor nonfunctional, restore the nonfunctional monitor to FUNCTIONAL status within 30 days or otherwise initiate corrective actions.

b. With both hydrogen monitors nonfunctional, restore at least one monitor to FUNCTIONAL status within 72 hours or otherwise initiate corrective actions.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated FUNCTIONAL:

a. By the performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. By the performance of a CHANNEL CALIBRATION at least once per 92 days on a STAGGERED TEST BASIS.

BASES:

The containment hydrogen monitors are used to assess the degree of core damage during beyond-design-basis accidents and to confirm that random or deliberate ignition of hydrogen in the containment atmosphere has taken place. Each containment hydrogen monitor is required to be FUNCTIONAL within 90 minutes after initiation of safety injection. The requirement that monitors be FUNCTIONAL within 90 minutes is based on the NRC's expectation outlined in Regulatory Guide 1.7, Revision 3, "Control of Combustible Gas Concentration in Containment Following a LOCA." If an explosive gas mixture threatens CONTAINMENT INTEGRITY during beyond-design-basis events, then severe accident management guidelines use the hydrogen monitors to implement strategies that protect the integrity of the containment boundary.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

BASES: (Continued)

The equipment for monitoring containment hydrogen levels must be FUNCTIONAL, reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following beyond-design-basis accidents to facilitate accident management, including emergency planning. Since these monitors are not required to mitigate the effects of design basis events, entry into other operational MODES is permitted while taking ACTIONS to restore these monitors to service.

The hydrogen monitors are connected directly to the containment boundary and are an extension of the containment boundary during post accident conditions. When placed into service, any leakage from these monitors would bypass the enclosure building filtration region and discharge to atmosphere. Therefore, pressure boundary integrity is subject to Technical Specification 6.13 "Systems Integrity," which establishes a program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels.

The TECHNICAL SURVEILLANCE REQUIREMENT To perform a CHANNEL CALIBRATION at least every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two-point calibration, utilizing the following sample gases:

a. One volume percent hydrogen, balance nitrogen

b. Four volume percent hydrogen, balance nitrogen

The ACTION requirement to initiate corrective actions means that the problem is entered into the Millstone Corrective Action Program and addresses why the hydrogen monitor(s) was (were) not restored to FUNCTIONAL status within the specified time. Use of the corrective action process in this situation will ensure management attention and oversight to minimize the additional time the containment hydrogen monitor(s) is (are) nonfunctional. ACTIONS shall be taken in a timely manner to establish alternate methods for determining hydrogen concentration levels as needed and determined by management.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

REFERENCE:

1. Amendment No. 287, dated June 29, 2005

2. DCR M2-04009, “Implementation of Revised Combustible Gas (Hydrogen) Control Requirements in Containment”

3. Regulatory Commitment RCR-42915 as described in Letter Serial No. 04-386, dated September 8, 2004

4. Consolidated Line Item Improvement Process Notice of Availability, published September 25, 2003 (68 FR 55416) in support of TSTF-447

3/4.7 PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

TECHNICAL REQUIREMENT

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY:

All MODES.

ACTION:

With the requirements of the above specification not satisfied:

a. Immediately reduce the steam generator pressure to ≤ 200 psig, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be determined to be > 70°F at least once per hour when pressures in the steam generators are > 200 psig and T_{avg} is < 200°F.

3/4.7 PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

BASES:

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

REFERENCE:

1. License Amendment No. 272

3/4.7 PLANT SYSTEMS

3/4.7.5 FLOOD LEVEL

TECHNICAL REQUIREMENT

3.7.5.1 At least one FUNCTIONAL service water pump motor shall be protected against flooding to a minimum elevation of 28 feet Mean Sea Level USGS datum if either:

a. The water level, including wave crest height, is exceeding plant grade level (14.0 feet Mean Sea level USGS datum), or

b. Three or more of the following conditions are occurring simultaneously:

1. The center of circulation of a storm with sustained winds greater than 60 miles per hour, as determined by radar, or reconnaissance, is presently located within the critical area as defined on Figure 3.7-1.

2. The projected track of a storm, with sustained winds greater than 60 miles per hour, which is expected to strike the facility within 12 hours as determined by radar, reconnaissance, or forecasted track projection, is from between 130° clockwise to 350°.

3. The central pressure of the storm expected to strike the facility within 12 hours is or is forecasted to be $\leq$ 28.0 in. Hg; or the measured 15 minute average wind speed at nominal elevation 33 feet from on site meteorological instrumentation exceeds 60 miles per hour.

4. A storm with sustained winds greater than 60 miles per hour is: expected to strike the facility within 12 hours; and the 15 minute average wind direction at nominal elevation 33 feet from on site meteorological instrumentation is from the sector 150° clockwise to 300°.

APPLICABILITY:

ALL MODES.

ACTION:

With the water level, including wave crest height, exceeding either plant grade or with three or more of the above specified meteorological conditions being exceeded simultaneously, immediately initiate ACTION to protect at least one service water pump motor against flooding to a minimum elevation of 28 feet; complete this protective ACTION within 2 hours.

3/4.7 PLANT SYSTEMS

3/4.7.5 FLOOD LEVEL

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.5.1.1 The water level, including wave crest height, shall be determined to be below plant grade at least once per hour when the center of circulation of a storm with sustained winds greater than 60 miles per hour is expected to strike the facility within 12 hours.

4.7.5.1.2 The following conditions shall be determined at least once per 2 hours when the center of circulation of a storm with sustained winds greater than 60 miles per hour is expected to strike the facility within 12 hours. The meteorological conditions shall be determined from weather service forecasts and/or from the site meteorological instrumentation:

- location of center of storm
- forecasted time storm is expected to strike
- storm track (actual and forecasted)
- storm approach angle
- central pressure of storm
- 33 feet 15 minute average wind speed
- 33 feet 15 minute average wind direction

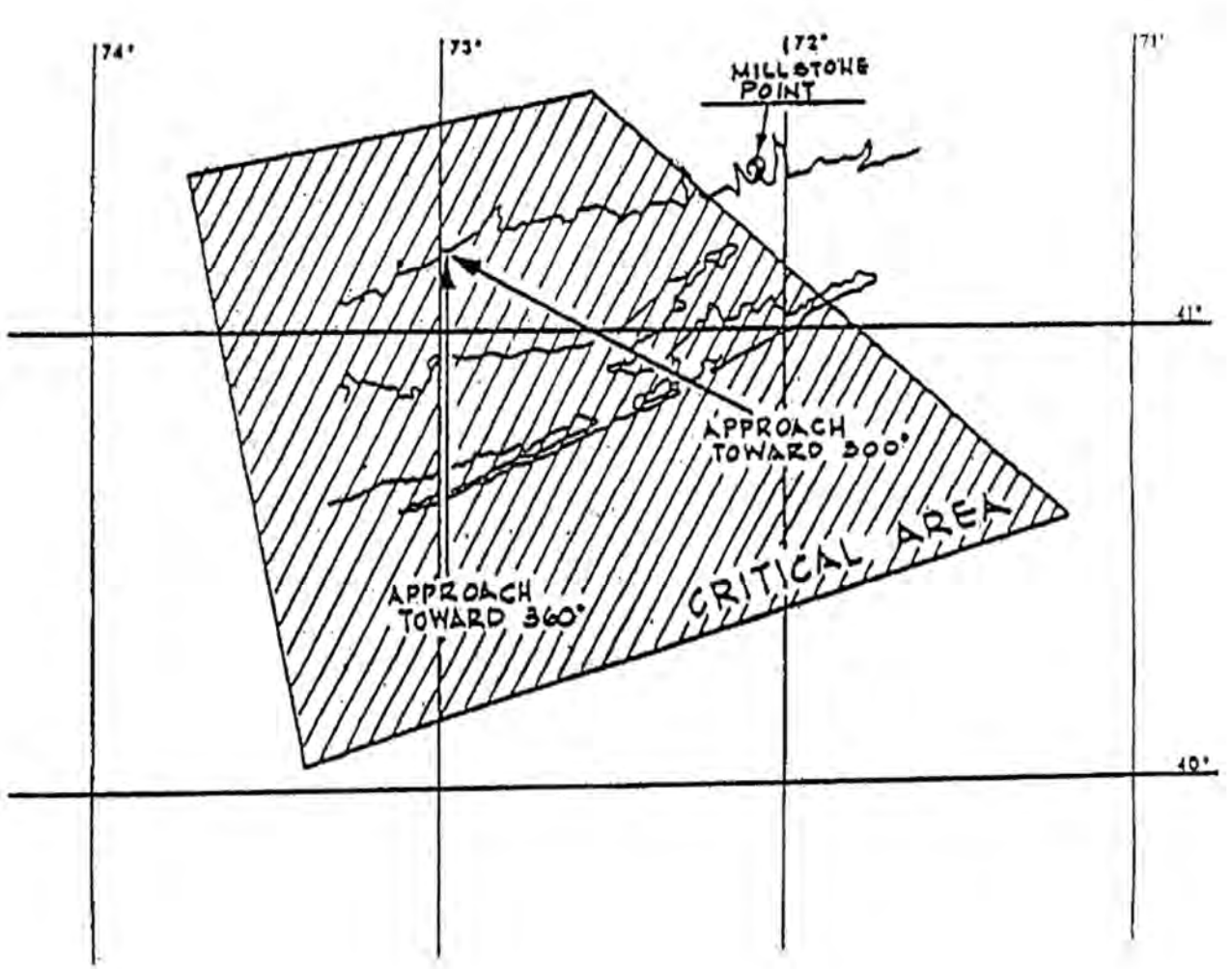

FIGURE 3.7-1
CRITICAL AREA

3/4.7 PLANT SYSTEMS

3/4.7.5 FLOOD LEVEL

BASES:

Water level must rise above 22 feet above mean sea level before reaching the base of the service water pump motors. Therefore service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet. This will ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure accessibility to the intake structure, ACTION to provide pump motor protection will be initiated when the water level including wave crest height reaches plant grade level. Entry into the ACTION to protect one service water pump from the flooding precludes any need to perform surveillance requirements.

REFERENCE:

1. License Amendment No. 272

3/4.7 PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

TECHNICAL REQUIREMENT

3.7.7.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY:

AT ALL TIMES.

ACTION:

a. Each sealed source having removable contamination in excess of the above limit shall be immediately withdraw from use and:

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

b. The provisions of TECHNICAL REQUIREMENT 3.0.3 are not applicable.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.7.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.7.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequencies described below.

a. Sources in use (excluding STARTUP sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive materials:

3/4.7 PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

TECHNICAL SURVEILLANCE REQUIREMENTS (Continued)

1) With a half-life greater than 30 days (excluding Hydrogen 3), and

2) In any form other than gas.

b. Stored sources not in use - Each sealed source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. STARTUP sources - Each sealed STARTUP source shall be tested prior to being subjected to core flux and following repair or maintenance to the source.

3/4.7 PLANT SYSTEMS

3/4.7.7 SEALED SOURCE CONTAMINATION

BASES:

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with TECHNICAL SURVEILLANCE REQUIREMENTS commensurate with the probability of damage to a source in that group. Those sources, which are not frequently handled, are required to be tested more often than those which are not. Sealed sources, which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices), are considered to be stored and need not be tested unless they are removed from the shield mechanism.

REFERENCE:

1. License Amendment No. 272

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.1 FIRE SUPPRESSION WATER SYSTEM

TECHNICAL REQUIREMENT

3.7.9.1 The fire suppression water system shall be FUNCTIONAL with:

a. Three high pressure fire pumps, each with a capacity of at least 1,800 gpm, with pump discharge aligned to the fire suppression header,

b. Two water supplies, each with a minimum contained volume of 200,000 gallons, and

c. A FUNCTIONAL flow path capable of taking suction from the fire water tanks and transferring the water through distribution piping with FUNCTIONAL sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser required to be FUNCTIONAL per TECHNICAL REQUIREMENTS 3.7.9.2 (Spray/Sprinkler TECHNICAL REQUIREMENT) and 3.7.9.3 (Fire Hose Stations TECHNICAL REQUIREMENT).

APPLICABILITY:

At all times.

ACTION:

a. With one pump and/or one water supply nonfunctional:

1. Restore the nonfunctional equipment to FUNCTIONAL status within 7 days **OR**

2. Provide an alternate backup pump or water supply within 24 hours **AND** develop a plan and schedule, within 14 days, for restoring the system to FUNCTIONAL status.

b. With two pumps nonfunctional:

1. Establish a continuous fire watch in the Turbine Building with backup suppression within 1 hour **AND**

2. Provide an alternate backup pump within 24 hours **AND**

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.1 FIRE SUPPRESSION WATER SYSTEM

TECHNICAL SURVEILLANCE REQUIREMENTS

3. Restore the nonfunctional equipment to FUNCTIONAL status within 7 days **OR** develop a plan and schedule, within 14 days, for returning the equipment to FUNCTIONAL status.

c. With the Fire Suppression Water System otherwise nonfunctional:

1. Establish a continuous fire watch in the Turbine Building with backup suppression within 1 hour **AND**

2. Restore the fire suppression water distribution system to FUNCTIONAL within 24 hours **OR**

3. Establish a backup Fire Suppression Water System within 24 hours **AND** develop a plan and schedule, within 14 days, for restoring the system to FUNCTIONAL status.

4.7.9.1 The fire suppression water system shall be demonstrated FUNCTIONAL:

a. At least once per 7 days, by verifying the contained water supply volume.

b. At least once per 31 days, on a STAGGERED TEST BASIS, by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days, by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

d. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months, by performing a system FUNCTIONAL test which includes simulated automatic actuation of the system throughout its operating sequence,

and:

1. Verifying that each pump develops at least 1800 gpm at a pump differential pressure of 100 psid, and

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.1 FIRE SUPPRESSION WATER SYSTEM

TECHNICAL SURVEILLANCE REQUIREMENTS

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that each pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 75 psig.

f. At least once per 3 years, by performing a flow test of the system in accordance with Chapter 5, Section 11, of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

g. The fire pump diesel engine shall be demonstrated FUNCTIONAL:

1. At least once per 31 days, by verifying:

a. The fuel storage tank contains at least 125 gallons of fuel; and

b. The engine starts from ambient conditions and operates for at least 30 minutes, while loaded with the fire pump on recirculation flow.

2. By verifying:

a. At least once per 92 days, a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water, and sediment.

b. At least once per 18 months, by subjecting the engine to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

3. The diesel engine starting 12-volt batteries and charger shall be demonstrated FUNCTIONAL:

a. At least once per 7 days, by verifying:

1. The electrolyte level of each battery cell is above the plates.

2. The voltage of each battery is greater than or equal to 12 volts.

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.1 FIRE SUPPRESSION WATER SYSTEM

b. At least once per 92 days, by verifying the specific gravity is appropriate for continued service of the batteries.

c. At least once per 18 months, by verifying:

1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration; and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

BASES:

1. The FUNCTIONALITY of the fire suppression system ensures adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the high pressure fire pumps, water supply tanks, fire hydrants, fire hose stations, distribution piping and valves, spray and/or sprinkler systems and Halon systems.

   In the event that portions of the fire suppression systems are nonfunctional, alternate backup fire fighting equipment is required to be made available in the affected areas until the nonfunctional equipment is restored to service.

   In the event the fire suppression water system becomes nonfunctional, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

2. It is the intent of monthly diesel engine fire pump testing to conduct this test during a different week than electric motor pump testing. Nominal monthly (“once per 31 days”) and 18 month surveillance requirements are consistent with the Unit 1 and 3 surveillance requirements. The surveillance procedures for each of the three fire pumps are common to Units 1, 2, and 3.

   (See AR 97026324-03 for additional information).

REFERENCE:

1. Condition Report M3-00-2659

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.2 SPRAY AND/OR SPRINKLER SYSTEMS

TECHNICAL REQUIREMENT

3.7.9.2 The following spray and/or sprinkler systems shall be FUNCTIONAL:

a. Diesel Generator Rooms
b. Diesel Generator Day Tank Rooms
c. Cable Vault (Aux. Building)
    1. Sprinkler (in tray)
    2. Sprinkler (ceiling level)
d. Cable Vault (Turbine Building)
    1. Automatic Wet Pipe Sprinkler System (45’0”)
    2. Automatic Deluge System (vertical cable shaft and elevation 25’0” cable vault)
e. Hydrogen Seal Oil Unit
f. Turbine Building Northeast Corner
g. Turbine Building 31’6”/14’6” - North
h. Turbine Building 31’6”/14’6” - South
i. Lube Oil Room
j. Aux. Building (-45’6”) General Area
k. Aux. Building (14’6”) Truck Access
l. Turbine Bearing
m. Steam Generator Feed Pumps
n. Aux. Bldg. (14’6”) at MCC B-61
o. Aux. Bldg. (-25’6”) at Charging Pump Cubicle
p. Aux. Bldg. (14’6”) General Area

APPLICABILITY:

Whenever equipment in the spray/sprinkler protected areas is required to be FUNCTIONAL.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems nonfunctional, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour. Restore the nonfunctional equipment to FUNCTIONAL status within 14 days, or develop a plan and schedule for restoring the equipment to FUNCTIONAL status.

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.2 SPRAY AND/OR SPRINKLER SYSTEMS

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.9.2 Each of the above required spray and/or sprinkler systems shall be demonstrated FUNCTIONAL:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months:

1. By performing a system FUNCTIONAL test which includes simulated automatic actuation of the system; and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By inspection of the spray headers to verify their integrity, and

3. By inspection of each nozzle to verify no blockage.

c. At least once per 3 years by performing an air or water flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.2 SPRAY AND/OR SPRINKLER SYSTEMS

BASES:

The FUNCTIONALITY of the fire suppression system ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are nonfunctional, alternate backup fire fighting equipment is required to be made available in the affected areas until the nonfunctional equipment is restored to service.

In the event the fire suppression water system becomes nonfunctional, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

REFERENCE:

1. Condition Report M3-00-2659

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.3 FIRE HOSE STATIONS

TECHNICAL REQUIREMENT

3.7.9.3 The fire hose stations shown in TRM Table 4.7-5 shall be FUNCTIONAL.

APPLICABILITY:

Whenever equipment in the areas protected by the fire hose stations is required to be FUNCTIONAL*.

ACTION:

With one or more of the fire hose stations shown in TRM Table 4.7-5 nonfunctional, route an additional equivalent capacity fire hose to the unprotected area(s) from an FUNCTIONAL hose station within 1 hour or establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s). If the nonfunctional hose station(s) is not the primary means of fire suppression, then route the additional fire hose(s) or establish a continuous fire watch with fire suppression equipment within 24 hours.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.9.3 Each of the fire hose stations shown in TRM Table 4.7-5 shall be demonstrated FUNCTIONAL.

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station. The exception to the above will be the containment hose stations. The equipment will be located outside containment except when the unit is in COLD SHUTDOWN.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and
   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve FUNCTIONALITY and no flow blockage.
   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

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* Containment hose stations shall be FUNCTIONAL in MODE 5 when required to support maintenance activities and MODE 6.

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.3 FIRE HOSE STATIONS

BASES:

The FUNCTIONALITY of the fire suppression system ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are nonfunctional, alternate backup fire fighting equipment is required to be made available in the affected areas until the nonfunctional equipment is restored to service.

In the event the fire suppression water system becomes nonfunctional, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

REFERENCE:

1. Condition Report M3-00-2659

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.4 HALON FIRE SUPPRESSION SYSTEM

TECHNICAL REQUIREMENT

3.7.9.4 The following Halon 1301 fire suppression systems shall be FUNCTIONAL with an intact gas boundary, a FUNCTIONAL activation system, and a container having a net weight of not less than 95% of full charge weight at 325 psig minimum (corrected to 70°F).

a. West D.C. Switchgear Room.

b. East D.C. Switchgear Room.

APPLICABILITY:

At all times.

ACTION:

With one or more of the above systems nonfunctional, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour. Restore the system to FUNCTIONAL status within 14 days, or develop a plan and schedule for restoring the system to FUNCTIONAL status.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.9.4 Each of the above halon fire suppression systems shall be demonstrated FUNCTIONAL.

a. At least once per 6 months:

   1. By performing a system FUNCTIONAL test which includes simulated automatic operation of the system; and,

      a. Simulated manual actuation of the system.

      b. Verifying that the storage containers have a net weight of not less than 95% of full charge weight at 325 psig (corrected to 70°F).

      c. Verifying the associated dampers close.

b. At least once per 18 months;

   1. By performing a visual inspection of the discharge nozzles to assure no blockage.

   2. By performing a visual inspection to assure the gas boundary is intact.

3/4.7 PLANT SYSTEMS

3/4.7.9 FIRE PROTECTION SYSTEMS

3/4.7.9.4 HALON FIRE SUPPRESSION SYSTEM

BASES:

The FUNCTIONALITY of the fire suppression system ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, halon fire suppression system and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are nonfunctional, alternate backup fire fighting equipment is required to be made available in the affected areas until the nonfunctional equipment is restored to service.

REFERENCE:

1. Condition Report M3-00-2659

3/4.7 PLANT SYSTEMS

3/4.7.10 PENETRATION FIRE BARRIERS

TECHNICAL REQUIREMENT

3.7.10 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals) shall be FUNCTIONAL.

APPLICABILITY:

At all times unless otherwise determined that the separation of safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area is not required based on the MODE of operation.

ACTION:

With one or more of the above required fire rated assemblies and/or penetration sealing devices nonfunctional, within 1 hour:

1. Verify the fire areas/zones on at least one side of the affected fire rated assembly contains a FUNCTIONAL fire detection or automatic suppression system at the fire barrier and establish a fire watch patrol that inspects both areas at least once per hour. Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005, or

2. Establish a continuous fire watch on at least one side of the affected fire rated assembly and/or penetration seal, or

3. Temporarily repair the nonfunctional fire rated assembly and/or sealing device and classify it as temporary.

All temporary repairs of nonfunctional fire rated assemblies and/or sealing devices shall be permanently repaired within 30 days, or implement ACTION 1 or 2 above.

3/4.7 PLANT SYSTEMS

3/4.7.10 PENETRATION FIRE BARRIERS

TECHNICAL SURVEILLANCE REQUIREMENTS

4.7.10 The above required fire rated assemblies and penetration sealing devices shall be verified to be FUNCTIONAL by a visual inspection:

a. At least once per 18 months for fire doors.

b. At least once per 18 months for fire barrier penetration seals, on at least 10% of the total number of penetration seals. If any of the penetration seals in the inspection sample are found to be nonfunctional, then an additional 10% sample of the total number of penetration seals shall be visually inspected. Sampling and inspection shall continue until all of the seals in a sample are found FUNCTIONAL or 100% of the seals are inspected.

c. At least once per 18 months for fire dampers, on at least 10% of the total number of fire dampers. If any of the fire dampers in the inspection sample are found to be nonfunctional, then an additional 10% sample of the total number of dampers shall be inspected. Sampling and inspection shall continue until all the fire dampers in a sample are found FUNCTIONAL or 100% of the fire dampers are inspected.

d. Prior to returning a fire rated assembly and/or penetration sealing device to FUNCTIONAL

e. status following repairs or maintenance.

BASES:

The FUNCTIONAL integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During a period of time when a barrier is not FUNCTIONAL, alternate measures are taken to prevent the possible spread of fire. These measures include verifying the FUNCTIONALITY of fire detection or suppression systems on one side of the affected barrier and establishing a fire watch patrol, or posting a continuous, fire watch in the vicinity of the affected barrier, or installation of a temporary fire stop pending restoration of the permanent seal.

3/4.7 PLANT SYSTEMS

3/4.7.10 PENETRATION FIRE BARRIERS

BASES: (Continued)

Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005.

REFERENCE:

1. Condition Report M3-00-2659
2. TRM Table 7-1: Barrier Fire Watch Matrix

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

TECHNICAL REQUIREMENT

3.8.1.1.c The diesel oil storage tank (T-148) shall contain at least 18,620 total gallons of fuel.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

a. With the diesel oil storage tank volume less than 18,620 total gallons, restore diesel oil storage tank level within 48 hours or declare one emergency diesel generator nonfunctional.

b. With the diesel oil storage tank fuel oil volume less than 13,357 total gallons, declare one emergency diesel generator nonfunctional.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.8.1.1.3 Verify the diesel oil storage tank contains at least 18,620 total gallons of fuel at least once per 31 days.

BASES:

There are two emergency diesel generators (EDG) for Millstone Nuclear Power Station Unit No. 2. Each EDG is provided with a separate fuel oil supply system, which consists of a diesel oil supply tank and associated piping, valves, and instrumentation controls. Technical Specifications require a minimum of 12,000 gallons of fuel oil to be stored in each diesel oil supply tank (T-48A and T-48B).

A cross-tie with two locked-closed valves for the diesel oil supply tanks is provided so that a total inventory of 24,000 gallons would be available to either one of the EDGs. In addition, a 25,000 gallon above ground fuel oil storage tank (T-148), which is common for both EDGs, is provided to receive fuel oil from delivery trucks prior to placing fuel oil in the diesel oil supply tanks.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

BASES: (Continued):

ETE-MP-2020-1002,(1) determined that a total useable fuel oil volume of 42,106 gallons is necessary to support initial operation of both EDGs for 24 hours and then allow one of the EDGs to continue operation at its rated load for a total of 7 days. This 42,106 gallon requirement for total minimum fuel oil inventory is greater than the combined usable capacity of 24,000 gallons available from the two supply tanks.

The EDG supply tanks require an additional 18,106 gallons(2) to satisfy the 42,106 gallon total minimum inventory of useable fuel oil. The 18,106 gallons of additional volume of fuel oil is supplied from the common storage tank T-148. A minimum volume of 18,106 useable gallons equates to a fuel oil level of approximately 80 inches above the bottom of tank T-148. To ensure a usable volume of 18,106 gallons is maintained in tank T-148, a tank low-level alarm based on maintaining 18,620 total gallons is necessary to annunciate whenever the stored volume decreases below this value.

TR ACTION Statement “b” requires the operator to declare one of the EDG’s nonfunctional if the total fuel oil inventory in tank T-148 decreases to 13,357 total gallons (i.e. six day supply) or less. A minimum volume of 13,357 total gallons (12,843 usable gallons) equates to a fuel oil level of approximately 63.3 inches above the bottom of the tank T-148. The Plant Process Computer (PPC) point L7004 will provide a low-low level alarm function based on the 13,357 gallon requirement whenever the stored volume decreases below this value.

Due to equipment uncertainties and oil density fluctuations plus an additional volume allowing operators time to respond to the alarm, the low-level alarm setpoint corresponding to the 7-day requirement must be set to approximately 88 inches above the bottom of tank T-148 to ensure the low-level alarm occurs prior to actual tank level falling below the minimum fuel oil requirement. In terms of volume indication, the nominal setpoint corresponds to 19,628 gallons. Additionally, due to the same uncertainties and oil density fluctuations plus an additional volume allowing operators time to respond to the alarm, the low-low-level alarm setpoint corresponding to the 6-day requirement must be set to approximately 66.9 inches above the bottom of tank t-148 to ensure the low-low level alarm occurs prior to the actual tank level falling below the

(1) ETE-MP-2020-1002 “Increasing the TRM required Diesel Fuel Oil Storage for T-148.”

(2) The 18,620 gallon minimum fuel oil volume requirement determined in ETE-MP-2020-1002 assumed fuel is initially stored at 24°F (delivered) and experiences a temperature elevation of 86°F. This assumption was made in order to address the effects elevated temperature has on volume, and consequently, the heating value of the oil.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

BASES: (Continued):

minimum fuel oil requirement. In terms of volume indication, the nominal setpoint corresponds to 14,300 gallons.

Alternate measures may be taken for verifying the minimum fuel oil volume requirement given failure of the level instrumentation or annunciator circuitry. Fuel oil volume may be verified by manual means by use of an oil dipstick or through the use of PPC point L7004.

A digital indicator, LI-7004A, is located on Control Room panel CO6F to provide the operator with a continuous indication of the fuel oil level in EDG storage tank T-148. This indicator is a dual bar graph style with the right side providing height of fuel in inches above the bottom of the tank and the left side providing net volume in gallons. The visible scale range of fuel height is 0-120 inches and the scale range for volume is 0-25 (x1000) gallons. Above each bar graph is a digital display of the scale reading. Also, the PPC has the capability of providing the operator with EDG storage tank T-148 fuel oil level and volume in terms of instrument range.

The above ground fuel oil storage tank T-148 contains a temperature element for monitoring the fuel oil bulk temperature. A low temperature alarm will annunciate when the fuel oil temperature falls below 40°F indicating the possible failure of the tank heating system and the need for compensatory ACTIONS to maintain the fuel oil temperature above the established cloud point. Should the fuel oil temperature fall below the cloud point, the quality of the fuel within the storage tank T-148 becomes questionable due to wax formation and the operator is to declare one of the EDG's nonfunctional in accordance with ACTION Statement "b" since the storage tank design volume cannot be considered available. The fuel oil temperature is available on the PPC to provide the operator with continuous temperature monitoring.

ETE-MP-2020-1002 "Increasing the TRM required Diesel Fuel Oil Storage for T-148" and calculation 97-ENG-1804E2, Rev 4,(3) provide the basis for establishing the following correlation between components of level monitoring lop L7004:

TRM Table 3.8.1.1.c-1 provides a correlation between tank level in inches from the bottom of tank to volume indication for both the process minimum limit and the instrumentation setpoint associated with the 18,620 gallons requirement.

TRM Table 3.8.1.1.c-2 provides a correlation between tank level in inches from bottom of tank to volume indication for both the process minimum limit and the instrumentation setpoint associated with the 13,357 total gallons requirement.

(3) Calculation 97-ENG-1804E2, Rev. 4, "Diesel Oil Storage Tank (T-148) Level Loop Uncertainty and Alarm Setpoint, L-7004."

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

BASES: (Continued):

TRM TABLE 3.8.1.1.C-1
18,620 Total Gallons Requirement Corresponding Level Values

18,620 Total Gallons Requirement (18,106 Useable Gallons) (7 day supply)	T-148 level (inches above bottom of tank)	T-148 VOLUME GALLONS
Low-level Alarm Setpoint	≥ 88 inches	≥ 19,628
Fuel Volume @ minimum TRM limit of	83.8 inches	18,620

TRM TABLE 3.8.1.1.C-2
13,357 Total Gallons Requirement Corresponding Level Values

13,357 Total Gallons Requirement (12,842 Useable Gallons) (6 day supply)	T-148 Level (inches above bottom of tank)	T-148 VOLUME GALLONS
Low-Low Level Alarm Setpoint	≥ 66.9 inches	≥ 14,300
Fuel Volume @ minimum TRM limit	63.3 inches	13,357

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

TECHNICAL REQUIREMENT

BASES: (Continued):

Following a loss of coolant accident, both EDGs will operate initially at the continuous rated loads of 2750 kW for 24 hours at which time (assuming no fuel oil replenishment is provided) one of the EDGs may be secured and the other EDG may continue operation at its rated load for an additional six (6) days. The total volume of fuel oil required for two EDGs to initially operate for 24 hours, and then allow one of the EDGs to continue to operate at its rated load for six additional days, is 42,106 useable gallons out of which 24,000 gallons are stored in diesel oil supply tanks (T-48A and T-48B). Therefore, the extra requirement of fuel oil of approximately 18,106 useable gallons will be stored in the diesel oil storage tank T-148. The value of 13,357 total gallons (12,842 useable gallons) corresponds to a reduced level of stored fuel oil equivalent to 6 days.(1)

The TECHNICAL REQUIREMENT cannot be met if less than the seven (7) days diesel oil supply for the EDG is available. However, the ACTION statement allows for temporarily maintaining a reduced diesel oil supply level for a maximum time duration of 48 hours before declaring one EDG nonfunctional. This temporary reduction in diesel oil supply level may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations which may be necessitated by increasing particulate levels or any number of other oil quality degradations. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG nonfunctional. This period is acceptable based on the remaining capacity (>6 days), the fact that ACTIONS will be initiated to obtain replenishment, and the low probability of an event during this brief period.

The TECHNICAL SURVEILLANCE REQUIREMENT provides verification that there is an adequate inventory of diesel fuel oil in the storage tank to support the operation of EDGs at the continuous rated load of 2750 kW for 24 hours, at which time (assuming no fuel oil replenishment is provided) one of the EDGs may be secured and the other EDG may continue operation at its rated load for an additional six (6) days. The seven (7) day period is sufficient time to place the unit in a safe condition and to bring in replenishment fuel from an offsite location. The 31 day frequency is adequate to ensure that a sufficient supply of oil is available, since low-level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

The above requirement is consistent with NUREG-1432, “Standard Technical Specification 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air.”

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

REFERENCE:

1. ETE-MP-2020-1002 “Increasing the TRM required Diesel Fuel Oil Storage for T-148.”
2. Calculation 97-ENG-1804E2, Rev. 4, “Diesel Oil Storage Tank (T-47A) Level Loop Uncertainty and Alarm Setpoint, L-7004.”
3. Condition Report M3-00-02659.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OFFSITE LINE POWER SOURCES

TECHNICAL REQUIREMENT

3.8.1.2 The offsite lines to the Millstone Switchyard: 310, 348 (includes 3252 line), 371 (includes 364 line), and 383, shall be FUNCTIONAL.

APPLICABILITY:

When Millstone Power Station (MPS) electrical output exceeds 1650 MWe net.

ACTION:

a. With one offsite line nonfunctional, perform the following or reduce total station output to ≤ 1650 MWe net within the next 6 hours:

   1. Restore FUNCTIONALITY of the affected offsite line within 72 hours, or

   2. Establish the following ACTION requirements AND restore FUNCTIONALITY within 7 days for Lines 310, 348/3252, and 383, or within 14 days for Line 371/364.

      a. Once per shift, verify the remaining offsite lines to the MPS switchyard are FUNCTIONAL.

      b. Perform an initial weather assessment for the scheduled line outage duration and then once per shift thereafter.

         1. If the assessment predicts adverse or inclement weather will exist while the offsite line is nonfunctional (i.e., out of service), reduce total station output to ≤ 1650 MWe net prior to arrival of the adverse or inclement weather.

      c. Within one hour prior to or after entering this condition and at least once per 24 hours thereafter, verify that both the Millstone Unit 2 EDGs and the Millstone Unit 3 EDGs are OPERABLE and the Millstone Unit 3 SBO diesel generator is available. Restore any inoperable Millstone Unit 2 EDG or Millstone Unit 3 EDG to OPERABLE status and/or the Millstone Unit 3 SBO to available status within 72 hours or reduce total station output to ≤ 1650 MWe net within the next 6 hours.

b. With two offsite lines nonfunctional, reduce total station output to ≤ 1650 MWe net within the next 30 minutes.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

TECHNICAL SURVEILLANCE REQUIREMENTS

4.8.1.2 The four offsite lines to the Millstone Switchyard shall be determined to be FUNCTIONAL at least once per 7 days when station output exceeds 1650 MWe net.

BASES:

TR 3.8.1.2 requires that all four offsite 345 kV transmission lines are FUNCTIONAL when MPS electrical output exceeds 1650 MWe net. TR 3.8.1.2 contains ACTIONS that must be performed when one offsite 345 kV transmission line is nonfunctional.

TR 3.8.1.2 provides flexibility to preclude plant downpowers due to planned and unplanned offsite transmission line outages. The TRM requirements provide this flexibility while also maintaining adequate defense-in-depth to ensure grid reliability and stability are preserved and the ability of the plants to respond to design basis accidents is not adversely affected.

With one offsite line nonfunctional, ACTION a.1 allows 72 hours to restore FUNCTIONALITY. This 72-hour allowed outage time (AOT) is conservatively based on the loss of one offsite line being equal to the risk of losing one onsite connection to the offsite power system (i.e., TS 3.8.1.1).

With one offsite line nonfunctional, ACTION a.2 contains the provision to allow up to 7 days for Lines 310, 348/3252, and 383, or 14 days for Line 371/364 to restore FUNCTIONALITY if ACTIONS a.2.a, a.2.b, and a.2.c are met. The applicable AOT reduces the risk of a plant perturbation as a result of having to downpower the unit for short duration line outages and provides flexibility for conducting maintenance and improves operational safety margin by the following:

- Minimizes the number of plant downpowers for short duration 345 kV transmission line outages.
- Reduces the likelihood of a loss of offsite power event by establishing an AOT with additional defense-in-depth measures to minimize the potential for a double circuit failure scenario which can result in grid instability.

During the applicable AOT, ACTION a.2.a ensures there are no known issues that could threaten the reliability of the remaining 345 kV transmission lines. Verifying the remaining 345 kV offsite transmission lines to the MPS switchyard are FUNCTIONAL, increases confidence that the remaining 345 kV lines will remain FUNCTIONAL during the planned outage of the affected 345 kV line.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

EMERGENCY DIESEL GENERATOR FUEL OIL SUPPLY

BASES: (Continued):

During the applicable AOT, the action to perform weather assessments as required by ACTION a.2.b ensures appropriate actions are taken to minimize the potential for adverse or inclement weather event to impact grid reliability with one line nonfunctional. If adverse or inclement weather is predicted, the nonfunctional 345 kV line would be restored to FUNCTIONAL status or station output would be reduced to ≤ 1650 MWe net prior to the arrival of the adverse or inclement weather.

During the applicable AOT, ACTION a.2.c (i.e., both the Millstone Unit 2 and Millstone Unit 3 EDGs are operable and the Millstone Unit 3 SBO is available) provides assurance that AC power will be available to support required safety-related equipment in the unlikely event of a complete loss of offsite power during the time one of the 345 kV lines is nonfunctional. This ACTION ensures that electrical power will be available in a timely manner to perform the required functions to maintain cooling to the reactor core in the unlikely event a loss of offsite power was to occur during the AOT.

For the condition where the applicable AOT is in use for one nonfunctional transmission line and one or more of the following components is out-of-service:

- a Millstone Unit 2 EDG
- a Millstone Unit 3 EDG
- the Millstone Unit 3 SBO diesel generator

72 hours is allowed for restoration of the out-of-service component. If any one of these components is not restored within 72 hours, reduce total station output to ≤ 1650 MWe net within the next 6 hours.

The allowed outage times for Lines 310, 348/3252, 371/364, and 383 are based on the configuration of the transmission lines at Hunts Brook Junction where Lines 383 and 310 cross over Line 371/364 and Line 348/3252 runs to the west of the crossover. With Line 348/3252, 310, or 383 nonfunctional, the possibility exists that either Line 383 or 310 could drop on Line 371/364 and result in three lines nonfunctional. This condition would impact grid stability and therefore, a 7-day AOT is allowed with the specified ACTION requirements in place. When Line 371/364 is nonfunctional, if either Line 310 or 383 drops, two transmission lines remain FUNCTIONAL. Therefore, a 14-day AOT is allowed with the specified ACTION requirements in place.

REFERENCE:

1. License Amendment 328

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

TECHNICAL REQUIREMENT

3.8.2.1.b The ventilation trains for the following switchgear rooms shall be FUNCTIONAL:

a. Upper and Lower 4160 VAC Switchgear Rooms.

b. East and West 480 VAC Switchgear Rooms.

c. B51 and B61 MCC Cubicles.

d. East and West DC Switchgear Rooms.

APPLICABILITY:

When associated switchgear is required to be OPERABLE.

ACTION:

Note: Separate ACTION entry is allowed for each switchgear/MCC cubicle ventilation train.

- Upper and Lower 4160 VAC Switchgear Rooms,
- East and West 480 VAC Switchgear Rooms, and
- East and West DC Switchgear Rooms.

a. With a switchgear ventilation train nonfunctional, immediately implement the applicable switchgear ventilation system compensatory ACTIONS in accordance with the applicable operating procedure or enter the applicable Technical Specification ACTION Statement for the associated switchgear. If compensatory ACTIONS have been implemented, within 30 days, either restore the affected switchgear ventilation train to FUNCTIONAL status or prepare a FUNCTIONALITY assessment, in accordance with station procedures. Applicable time constraints for continued operation with a switchgear ventilation train nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

TECHNICAL REQUIREMENT

ACTION: (Continued)

b. With both trains of a switchgear ventilation system nonfunctional, immediately implement the applicable switchgear ventilation system compensatory ACTIONS in accordance with the applicable operating procedure or enter the applicable Technical Specification ACTION Statement for the associated switchgear. If compensatory actions have been implemented, within 7 days, either restore at least one train of the switchgear ventilation system to FUNCTIONAL status or prepare a FUNCTIONALITY assessment, in accordance with station procedures. Applicable time constraints for continued operation with both switchgear ventilation trains nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

- B51 and B61 MCC Cubicles.

Note: The compensatory ACTIONS contained in the applicable operating procedure for B51 and B61 MCC Cubicles switchgear ventilation trains may adversely affect enclosure building integrity. Evaluate and enter Technical Specification ACTION Statement 3.6.5.2, “Containment Systems - Enclosure Building,” as necessary.

c. With either B51 or B61 MCC switchgear ventilation train nonfunctional, immediately implement the applicable switchgear ventilation system compensatory ACTIONS in accordance with the applicable operating procedure and enter Technical Specification ACTION Statement 3.6.5.2, if necessary. Otherwise, enter the applicable Technical Specification ACTION Statement for the associated switchgear. If entry into Technical Specification ACTION Statement 3.6.5.2 is required as a result of the use of compensatory ACTIONS, restore the affected switchgear ventilation train to FUNCTIONAL status within 24 hours, or terminate the use of compensatory measures per the applicable operating procedure and enter the applicable Technical Specification ACTION Statement for the associated switchgear. If entry into Technical Specification ACTION Statement 3.6.5.2 is NOT required and compensatory ACTIONS have been implemented, within 30 days, either restore the affected switchgear ventilation train to FUNCTIONAL status or prepare a FUNCTIONALITY assessment, in accordance with station procedures. Applicable time constraints for continued operation with a switchgear ventilation train nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

TECHNICAL SURVEILLANCE REQUIREMENTS

d. With both B51 and B61 MCC switchgear ventilation trains nonfunctional and entry into Technical Specification ACTION Statement 3.6.5.2 is required, restore both trains of the switchgear ventilation system to FUNCTIONAL status and operation within 24 hours, or terminate the use of compensatory measures per the applicable operating procedure and enter the applicable Technical Specification ACTION Statement for the associated switchgear. With both B51 and B61 MCC switchgear ventilation trains nonfunctional and entry into Technical Specification ACTION Statement 3.6.5.2 is NOT required, within 7 days, either restore at least one train of the switchgear ventilation system to FUNCTIONAL status or prepare a FUNCTIONALITY assessment, in accordance with station procedures. Applicable time constraints for continued operation with both switchgear ventilation trains nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

4.8.2.1b Normal daily plant equipment operator rounds verify the switchgear ventilation systems are FUNCTIONAL.

BASES:

This TECHNICAL REQUIREMENT provides guidance to address switchgear ventilation systems that are nonfunctional. If any switchgear ventilation train is nonfunctional the compensatory measures contained in the applicable operating procedure must be implemented immediately (pursued without delay and in a controlled manner) or the associated switchgear shall be declared INOPERABLE and the appropriate Technical Specification ACTION Statement (TSAS) entered. Since the compensatory measures contained in the applicable operating procedure will ensure the associated switchgear will remain OPERABLE for all design basis events, it is acceptable to address each switchgear ventilation train separately.

The ACTION requirements are preceded by a note that states separate ACTION entry for each switchgear/MCC cubicle ventilation train is allowed. This is acceptable since the required ACTIONS for each condition provide appropriate compensatory ACTIONS for each nonfunctional ventilation train.

- Upper and Lower 4160 VAC Switchgear Rooms,
- East and West 480 VAC Switchgear Rooms, and
- East and West DC Switchgear Rooms.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

BASES: (Continued)

a. If one switchgear ventilation train for any switchgear is under compensatory measures a 30 day allowed outage time (AOT) is specified. The use of compensatory ACTIONS for the AOT or other applicable time constraint is acceptable since the compensatory measures ensure the associated switchgear will remain OPERABLE for all design basis events.

b. If both switchgear ventilation trains for any switchgear are under compensatory measures, a 7 day AOT is specified to restore at least one of the switchgear ventilation trains to FUNCTIONAL status.

Within the specified AOT either restore the switchgear ventilation trains to FUNCTIONAL status or a FUNCTIONALITY assessment will be prepared, in accordance with station procedures. Applicable time constraints for continued operation with both switchgear ventilation trains nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

- B51 and B61 MCC Cubicles:

The ACTION requirements are preceded by a note that states the compensatory ACTIONS contained in the applicable operating procedure for B51 and B61 MOC Cubicles switchgear ventilation trains may adversely affect enclosure building integrity. If enclosure building integrity is adversely affected, entry into Technical Specification ACTION Statement 3.6.5.2, “Containment Systems - Enclosure Building,” may be necessary. This ACTION statement requires restoration of enclosure building integrity within 24 hours or a plant shutdown to MODE 5 will be required.

a. If one switchgear ventilation train for either B51 or B61 switchgear is under compensatory measures and the compensatory measures result in a loss of enclosure building integrity, the ventilation train shall be restored to FUNCTIONAL status and operation within 24 hours. The use of a 24 hour AOT is specified to be consistent with the AOT for Technical Specification 3.6.5.2. If the ventilation train is not restored to FUNCTIONAL status within 24 hours, the use of compensatory measures shall be terminated to restore enclosure building integrity and the appropriate TSAS shall be entered for the affected switchgear. If the use of compensatory ACTIONS does not result in a loss of enclosure building integrity, a 30 day AOT is specified. The use of compensatory ACTIONS for the AOT or other applicable time constraint is acceptable since the compensatory measures ensure the associated switchgear will remain OPERABLE for all design basis events.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

BASES: (Continued)

b. If both B51 and B61 switchgear ventilation trains are under compensatory measures, and the compensatory measures result in a loss of enclosure building integrity, the ventilation trains shall be restored to FUNCTIONAL status within 24 hours. The use of a 24 hour AOT is specified to be consistent with the AOT for Technical Specification 3.6.5.2. If the ventilation trains are not restored to FUNCTIONAL status within 24 hours, the use of compensatory measures shall be terminated to restore enclosure building integrity and the appropriate TSAS shall be entered for the affected switchgear. If the use of compensatory ACTIONS does not result in a loss of enclosure building integrity, a 7 day AOT is specified to restore at least one of the switchgear ventilation trains to FUNCTIONAL status.

   Within the specified AOT either restore the switchgear ventilation trains to FUNCTIONAL status or a FUNCTIONALITY assessment will be prepared, in accordance with station procedures. Applicable time constraints for continued operation with both switchgear ventilation trains nonfunctional and the compensatory ACTIONS in place shall be identified and justified in the assessment.

   - Upper and Lower 4160 VAC Switchgear Rooms
   - East and West 480 VAC Switchgear Rooms
   - B51 and B61 MCC Cubicles
   - East and West DC Switchgear Rooms.

a. Periodic testing of system valves and fans may interrupt cooling flow to one of these rooms for a short period of time (< 5 minutes). The ventilation for the affected room remains FUNCTIONAL during this testing. However, if the component being tested cannot be restored, i.e. ventilation/ventilation cooling cannot be restored, then the applicable ACTION statement for the affected ventilation train/system will be entered.

REFERENCE:

1. Technical Specification Section 3/4.8.2, Onsite Power Distribution Systems.
2. Technical Evaluation M2-EV-99-0093, Rev. 04.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER AC DISTRIBUTION SYSTEMS

ELECTRICAL SWITCHGEAR ROOM VENTILATION

REFERENCE: (Continued)

3. Regulatory Issue Summary - RIS-05-020, Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, information to Licensees regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability.
4. NRR Staff Analysis response to Clinton Power Station regarding Technical Specification Requirements which were based on NRC Inspection Manuel, Part 9900, STS10 OP STS.
5. CR M2-98-2661, was initiated against OP2315 SD to re-evaluate compensatory measures established to enhance identified weaknesses. This TRM addition is a corrective action for CR M2-98-2661.
6. 10 CFR Part 50.59 Safety Evaluation S2-EV-99-0090.
7. Condition Report M3-00-02659.

3/4.9 REFUELING OPERATIONS

3/4.9.3.2 SPENT FUEL POOL TEMPERATURE

TECHNICAL REQUIREMENT

3.9.3.2 The spent fuel pool bulk temperature shall be maintained at less than or equal to 150°F.

APPLICABILITY:

Whenever irradiated fuel is stored in the spent fuel pool.

ACTION:

With the above conditions not satisfied:

a. Immediately initiate ACTIONS to restore the spent fuel pool temperature to less than or equal to 150°F, and

b. Within one hour, suspend all fuel movement in the spent fuel pool, and

c. At least once per 4 hours, record the spent fuel pool bulk temperatures.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.9.3.2 The spent fuel pool bulk temperature shall be verified to be less than or equal to 150°F at least once per 12 hours.

BASES:

The requirement that the spent fuel pool bulk temperature be maintained at or below 150°F ensures that the bulk water temperature will not exceed 200°F with the loss of spent fuel pool cooling assumed following a design basis LOCA. This ensures the post-LOCA function of the spent fuel pool cooling system.

The requirements that the spent fuel pool bulk temperature be maintained at or below 150°F ensures that the temperature and humidity above the pool are compatible with personnel comfort and safety requirements. The maximum temperature limit of 150°F in the spent fuel pool ensures that the design temperature of the spent fuel pool cooling system, liner/building structures, and racks are not exceeded. The heat removal capacity of the spent fuel pool cooling system, and/or shutdown cooling system (in MODES 5 or 6, or with core defueled) is sufficient to maintain the spent fuel pool bulk water temperature to less than or equal to 150°F.

REFERENCE:

1. Condition Report M3-00-02659

2. DCR M2-01001, TRMCR 01-2-5

3/4.9 REFUELING OPERATIONS

3/4.9.3.3 SPENT FUEL POOL DECAY TIME

TECHNICAL REQUIREMENT

3.9.3.3 The reactor shall be maintained in MODE 5 or 6, or core defueled until either:

a. 616 hours has passed since sub-criticality for the fuel bundles remaining in the spent fuel pool which were discharged from the previous REFUELING*, of less than or equal to 81 fuel bundles, or

b. The heat load in the spent fuel pool is less than 10.16×10^6 BTU/hr.

APPLICABILITY:**

MODES 5 and 6, and core defueled.

ACTION:

With the above conditions not satisfied, immediately initiate ACTIONS to restore the reactor to MODE 5 or 6, or core defueled.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.9.3.3 Prior to entry into MODE 4 from MODE 5, verify that either:

a. 616 hours has passed since sub-criticality for the fuel discharged from the previous REFUELING,* of less than or equal to 81 fuel bundles, or

b. The heat load in the spent fuel pool is less than 10.16×10^6 BTU/hr.

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* For the purpose of this requirement, the fuel bundles remaining in the spent fuel pool which were discharged from the previous REFUELING, is defined as the fuel bundles discharged at the end of the most recent REFUELING, that will not be reloaded into the next fuel cycle.

** When either TECHNICAL REQUIREMENT, Item (a) or (b) has been satisfied following a REFUELING, the TECHNICAL REQUIREMENT is satisfied for the remainder of the fuel cycle, until the next REFUELING.

3/4.9 REFUELING OPERATIONS

3/4.9.3.3 SPENT FUEL POOL DECAY TIME

BASES:

The requirement for the reactor to remain in MODE 5 or 6, or core defueled until either:

a. 616 hours has passed since sub-criticality for the fuel bundles remaining in the SFP which were discharged from the previous REFUELING of less than or equal to 81 fuel bundles, or

b. The heat load in the SFP is less than 10.16×10^6 BTU/hr. At a SFP heat load of 10.16×10^6 BTU/hr, adequate cooling is available to maintain the SFP bulk temperature less than or equal to 150°F should a single failure occur in the SFP cooling system.

This ensures that adequate cooling is available to maintain the SFP bulk temperature less than or equal to 150°F should a single failure occur in the SFP cooling system.

If either TECHNICAL REQUIREMENT Item (a) or (b) are met, then the SFP heat load will be less than 10.16×10^6 BTU/hr. Once the SFP heat load is below this value, the SFP cooling system can accommodate a single failure and maintain the SFP bulk water temperature to less than 150°F. The decay time of 616 hours for recently discharge fuel is the bounding decay time for a discharge of less than or equal to 81 fuel bundles to assure a SFP decay heat load of less than or equal to 10.16×10^6 BTU/hr.

If the spent fuel pool decay heat load is greater than 10.16×10^6 BTU/hr, then entry into MODE 4 from MODE 5 is not allowed. The reason for this restriction on entering MODE 4 with this heat load, is that one train of shutdown cooling must be available to either supplement or replace SFP cooling, and thus could not also be aligned to support MODE 4 ECCS requirements. One train of the shutdown cooling system is capable of cooling both the core and the SFP, should a failure occur in the SFP cooling system, and maintain the SFP water temperature to less than 150°F.

REFERENCE:

1. Condition Report M3-00-02659
2. DCR M2-01001, TRMCR 01-2-5

3/4.9 REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

TECHNICAL REQUIREMENT

3.9.5 Direct communications shall be maintained between the control room and personnel at the REFUELING station during fuel or CEA movement within the reactor pressure vessel.

APPLICABILITY:

MODE 6.

ACTION:

When direct communications between the control room and personnel at the REFUELING station cannot be maintained, suspend all operations involving fuel or CEA movement within the reactor pressure vessel.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the REFUELING station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during fuel or CEA movement within the reactor pressure vessel.

BASES:

The requirement for communications capability ensures that REFUELING station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during fuel or CEA movement within the reactor pressure vessel.

REFERENCE:

1. Condition Report M3-00-02659

3/4.9 REFUELING OPERATIONS

3/4.9.6 CONTAINMENT BUILDING REFUELING MACHINE

TECHNICAL REQUIREMENT

3.9.6 The containment building REFUELING machine used for movement of fuel assemblies shall be FUNCTIONAL with:

a. A minimum capacity of 1600 pounds in the “fuel only” region,

b. An overload cutoff limit of less than or equal to 1700 pounds in the “fuel only” region.

c. A minimum capacity of 2900 pounds in the “fuel plus hoist” region, and

d. An overload cutoff limit of less than or equal to 3590 pounds in the “fuel plus hoist” region.

APPLICABILITY:

During movement of fuel assemblies.

ACTION:

With the requirements of TECHNICAL REQUIREMENT 3.9.6 listed above not satisfied, suspend all REFUELING machine operations involving movement of fuel assemblies.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.9.6 The containment building REFUELING machine used for movement of fuel assemblies shall be demonstrated FUNCTIONAL within 72 hours prior to the start of moving fuel assemblies by performing applicable load tests of at least 1600 and 2900 pounds and demonstrating applicable automatic load cutoffs of less than or equal to 1700 and 3590 pounds.

3/4.9 REFUELING OPERATIONS

3/4.9.6 CONTAINMENT BUILDING REFUELING MACHINE

BASES:

The fuel hoist assembly is located on the REFUELING platform and is positioned over a specific core location using the joysticks located on the console shelf panel. The hoist and grapple assembly is lowered to a position over the fuel assembly (Upper Grapple Operate Zone - UGOZ). The operator turns the grapple selector switch to open the grapple at the bottom of the hoist assembly. The grapple is lowered over the fuel assembly to the Lower Grapple Operating Zone (LGOZ) and the operator turns the grapple selector switch to close the grapple, engaging the top fitting of the fuel assembly. The grapple and fuel assembly are raised using the joystick on the console shelf panel, as the fuel assembly is hoisted out of the core it is protected by an overload cutoff provided in the Programmable Logic Controller (PLC) software until it is safely inside the hoist box (fuel only region). This overload limit protects the fuel assembly, core internals and pressure vessel from excessive lifting force in the event they are inadvertently engaged during lifting operations.

After the fuel is inside the hoist box, the fuel and hoist box are lifted into the trolley mast with the loads carried by the hoist assembly (fuel plus hoist region). Overload cutoff settings are provided by the PLC software and an external hard wire setting in the load weighing system to prevent the forces from exceeding the maximum design lifting capacity of the refuel machine.

The basis for TECHNICAL REQUIREMENT 3/4.9.6 is to ensure that: 1) the REFUELING machine has sufficient load capacity to lift a fuel element, 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations, and 3) the REFUELING machine is protected from excessive lifting forces that may exceed its design rated maximum load capacity.

REFERENCE:

1. Condition Report M3-00-2659.
2. TRMCR 00-2-19.
3. PaR Nuclear, Software Requirements Specification (SRS) Refueling Machine Millstone Power Station - Unit 2, Procedure No. 70587896.
4. DNC drawing 25203-39277 SH 354, Drive Assembly Hoist, Refuel Machine.

3/4.9 REFUELING OPERATIONS

3/4.9.7 SPENT FUEL POOL CASK CRANE TRAVEL LIMITS

TECHNICAL REQUIREMENT

3.9.7 Loads in excess of 1800 pounds shall be prohibited from travel over irradiated fuel assemblies in the storage pool spent fuel racks, with the following exceptions:

a. Heavy Dummy Fuel Assembly,

b. Consolidated Fuel Storage Box,

c. Single-failure-proof lifts in accordance with the guidelines of NUREG-0612.

APPLICABILITY:

With irradiated fuel assemblies in the storage pool spent fuel racks.

ACTION:

With the requirements of the above specification not satisfied, place cask crane load in a safe condition.

TECHNICAL SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and/or physical stops which prevent cask crane travel with loads in excess of 1800 pounds over irradiated fuel assemblies shall be demonstrated FUNCTIONAL at least once per calendar quarter.

BASES:

The restriction on movement of loads in excess of the combined nominal weight of a fuel assembly, CEA and associated handling tool over irradiated fuel assemblies in the storage pool spent fuel racks using the spent fuel pool cask crane ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analysis. Specific analysis has been performed for the drop of a consolidated fuel storage box on an intact fuel assembly. The results of this analysis determined that the impacted assembly would not fail. The drop of the heavy dummy fuel assembly is bounded by the consolidated fuel storage box drop analysis.

3/4.9 REFUELING OPERATIONS

3/4.9.7 SPENT FUEL POOL CASK CRANE TRAVEL LIMITS

TECHNICAL REQUIREMENT

BASES: continued

With the cask crane design and application of single-failure-proof lifting devices that satisfy the increased design margins of NUREG-0612, a load drop onto irradiated fuel assemblies is not credible and need not be postulated. For other cranes capable of travel with loads in excess of 1800 pounds over irradiated fuel assemblies, administrative controls are in place to ensure compliance with the heavy loads guidelines of NUREG-0612.

REFERENCE:

1. Condition Report M3-00-02659.

6.0 ADMINISTRATIVE CONTROLS

6.2.2 FIRE PROTECTION ADMINISTRATIVE REQUIREMENTS

The following administrative requirements for staffing and training are applicable to support the Fire Protection System TECHNICAL REQUIREMENTS in the below listed TRM sections:

3/4.3.3.7 Fire Detection Instrumentation

3/4.7.9.1 Fire Suppression Water System

3/4.7.9.2 Spray and/or Sprinkler Systems

3/4.7.9.3 Fire Hose Stations

3/4.7.9.4 Halon Fire Suppression System

3/4.7.10 Penetration Fire Barriers

ADMINISTRATIVE REQUIREMENTS

a. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include two members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

b. A training program for the Fire Brigade shall be maintained and shall meet or exceed the requirements of NFPA 27 - 1975, except for Fire Brigade training sessions which shall be held at least quarterly.

* The Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate ACTION is taken to fill the required positions.

6.0 ADMINISTRATIVE CONTROLS

6.13 SYSTEM INTEGRITY REQUIREMENTS

The licensee shall implement a program to reduce leakage from systems outside containment that would, or could, contain highly radioactive fluids during a serious transient, or accident, to as low as practical levels. This program shall include the following:

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at a frequency not to exceed REFUELING cycle intervals.

PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT LEAKAGE REDUCTION PROGRAM

1. PURPOSE:

This program provides controls to minimize LEAKAGE from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at REFUELING cycle intervals or less.

2. SCOPE:

The systems, or portions of the systems, included in this program are limited to those which penetrate containment and may contain highly radioactive fluids after a serious transient or accident. The applicable systems are:

a. Containment Spray System;

b. Low Pressure Safety Injection;

c. Shutdown Cooling System;

d. Chemical and Volume Control System (boron precipitation control only);

e. High Pressure Safety Injection System;

f. Reactor Coolant System Post Accident Sample System;

g. Containment Air Post Accident Sample System; and

h. Hydrogen Sampling System.

6.0 ADMINISTRATIVE CONTROLS

6.13 SYSTEMS INTEGRITY

PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT LEAKAGE REDUCTION PROGRAM

3. PREVENTIVE MAINTENANCE:

Each applicable system, or portion of the system is included in the plant Preventive Maintenance Program. Since LEAKAGE from these systems would have to penetrate welded or passive barriers (i.e. gaskets, packing, pump seals, etc.), and is subject to the visual inspections described below, the normal plant Preventive Maintenance Program satisfies the intent of this program.

4. PERIODIC VISUAL INSPECTIONS:

Each applicable system, or portion of the system is subject to periodic visual inspections. Periodic visual inspection are performed during such activities as ASME Section XI pressure tests, operator rounds, health physics surveillances, and for systems not routinely in service during FUNCTIONALITY surveillance testing. Any LEAKAGE or residual boric acid buildup is identified using a trouble report and corrected in accordance with the Corrective Action Program.

5. INTEGRATED LEAK TEST REQUIREMENTS:

Integrated leak test requirements for each system are performed at REFUELING cycle intervals, or less. This testing will verify the external leak tightness of the applicable systems, or portions of the systems, by inspecting the system or subsystem boundary, measuring visible LEAKAGE from mechanical joints, summing the total LEAKAGE, and comparing the individual and total LEAKAGE results with the acceptance criteria. In addition, testing is performed to verify system-to-system LEAKAGE is within the acceptance criteria. This testing is performed by the following procedures.

Procedure	Title	Frequency
SP 21157	HPSI System and ECCS Suction Piping Leakage Test	REFUELING
SP 21162	Volume Control Portion of the CVCS Leakage Test	REFUELING
SP 21198	Containment Spray System Leakage Test	REFUELING
SP 21199	LPSI System and Shutdown Cooling Heat Exchangers Leakage Test	REFUELING
SP 2849	Surveillance Test for Reactor Coolant PASS	REFUELING
SP 2608G	Leak Test of the Containment Air Post-Accident Sample System	REFUELING
SP 2608H	Leak Check of the Hydrogen Sampling System	REFUELING

6.0 ADMINISTRATIVE CONTROLS

6.13 SYSTEMS INTEGRITY

PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT LEAKAGE REDUCTION PROGRAM

6. REFERENCES:

1. NUREG 0737, “Clarification of TMI Action Plan Requirements,” Section Ill.D.1.1, “Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized Water Reactors.”
2. NRC Information Notice 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere,” dated September 19, 1991
3. Millstone Unit No. 2 Technical Specification 6.13, “Systems Integrity.”
4. Condition Report M3-00-02659.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: As Identified In Each TECHNICAL REQUIREMENT

TECHNICAL REQUIREMENT IS AS SPECIFIED IN THE FOLLOWING COMPONENT SECTIONS.

APPLICABILITY:

Is as specified in the following component sections.

ACTION:

With an Appendix R Safe Shutdown Related (ARSR) component listed in the following component tables nonfunctional (unable to meet its intended Appendix R shutdown function), perform the following:

a. Restore FUNCTIONALITY of the affected component(s) within 14 days*, or

b. Establish the following compensatory measures to satisfy the FUNCTIONALITY requirements of ARSR components listed in the following component tables unless other compensatory measures are stated:

1. Establish a one-hour roving fire watch for all affected areas identified in the following component tables (except the control room, containment, waste gas decay tanks room, spent resin tank area, ion exchange room, spent fuel pool and cask laydown area and service water cable tunnels). Refer to the Appendix R Compliance Report for fire watch patrol areas. Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005.

2. Verify the FUNCTIONALITY of all fire suppression and detection systems, note required transient combustibles, and verify the lack of substantial new transient combustibles, in the affected areas (except containment) on a Daily Basis. Bring any changes to the attention of the Shift Manager. Establishment of the above compensatory measures will allow continued operation for up to sixty (60) days pending restoration of the component(s) FUNCTIONALITY, or

* Compensatory measures will be established prior to the end of the 14 day window when it is apparent that the 14 day restoration period cannot be met.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: As Identified In Each TECHNICAL REQUIREMENT

TECHNICAL REQUIREMENT IS AS SPECIFIED IN THE FOLLOWING COMPONENT SECTIONS. (Continued)

ACTION: (continued)

c. Prepare a FUNCTIONALITY assessment, in accordance with station procedures, if warranted, for the function described in the Appendix R Compliance Report, if component FUNCTIONALITY cannot be established within 60 days per ACTION Statement b. Applicable time constraints for continued operation with ARSR component(s) nonfunctional will be identified and justified in the assessment.

d. Restore FUNCTIONALITY of the affected Risk Significant component within 72 hours, or

e. Establish the following risk management action (RMA) Defense-In-Depth compensatory measures for the next 30 days to limit transient combustible material and ignition sources identified in the following component tables:

1. Review existing transient combustible material permits for the affected areas.
2. Review existing ignition source permits for the affected areas.
3. Perform an area walkdown of the affected areas.
4. Evaluate all work requiring transient combustibles, hot work, or racking of breakers in the affected areas.
5. On a daily basis, brief the shift operators and the fire brigade lead on the significance of a fire in the affected areas.
6. Restore FUNCTIONALITY of the affected Risk Significant component within the 30 days or prepare a FUNCTIONALITY assessment in accordance with station procedures for the Risk Significant component out of service.

f. For ARSR components requiring prompt action, establish compensatory measures within the time specified in the component table.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: As Identified In Each TECHNICAL REQUIREMENT

ADMINISTRATIVE REQUIREMENT:

During Modes 1, 2, 3, or 4, each on duty operating shift shall have one licensed or nonlicensed operator in addition to the minimum shift crew composition required by Section 6.2.2 of the Millstone Unit 2 Safety Technical Specifications. The shift manager shall call in personnel to meet this requirement, if needed. If this requirement cannot be met within eight hours, TECHNICAL REQUIREMENT ACTION Statement “b” shall be implemented for Appendix R Fire Areas R-1, R-2, R-3, R-14, and R-16 within the next four hours.

BASES:

1. ARSR equipment are those components used in plant systems to achieve and maintain HOT STANDBY and COLD SHUTDOWN when a fire disables any area or system(s) at Millstone Unit 2. This Appendix R Safe Shutdown TECHNICAL REQUIREMENTS section addresses the FUNCTIONALITY of ARSR equipment not addressed in Millstone Unit 2 Technical Specifications. Specifically, this document:

   a. Identifies ARSR equipment not addressed in Technical Specifications required to support plant safe shutdown following a fire.

   b. Identifies the FUNCTIONALITY requirements for the ARSR equipment.

   c. Identifies the conditions (such as local or remote) and plant location (fire areas) in which ARSR equipment FUNCTIONALITY is required.

   d. Provides compensatory measures in the event ARSR equipment becomes nonfunctional.

   e. Provides TECHNICAL SURVEILLANCE REQUIREMENTS for ARSR equipment.

   f. Provides a basis for establishing administrative controls over ARSR equipment.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

2. The consequences of a fire at a nuclear plant are considered to be less severe than those for Design Basis Accidents. The 14-day window for restoration of ARSR component FUNCTIONALITY, the establishment of compensatory measures, and the once per cycle TECHNICAL SURVEILLANCE REQUIREMENT reflect this decreased level of risk. Compensatory measures will be established prior to the 14-day window when it is apparent that the 14-day restoration period cannot be met.

3. ARSR equipment which are addressed in more restrictive technical specifications are not addressed in this document. Exceptions include shutdown process instrumentation which are allowed to be out of service by technical specifications based upon the availability of redundant channels. If the redundant channel(s) are not ARSR equipment, the redundant channel cannot be credited as “ARSR redundant” and the instrumentation is included in this document for further administrative control.

4. The compensatory measures are intended to ensure that either an alternative component is available to fulfill the shutdown function after a fire, or to minimize the possibility that a fire will occur in an area of the plant which relies on the particular component for safe shutdown. Roving fire watches must monitor the area of the device in question, as a minimum, within the specified time frame, plus or minus 25% of the time interval specified in the ACTION statement for periodic roving fire watches. The 25% extension of the time interval specified does not degrade the reliability that results from performing the rove at the specified interval, based on plant experience, and Fire Protection Engineering analysis as documented in Technical Evaluation M3-EV-02-2005.

5. The lack of fire watch requirements for the control room is based upon continuous staffing by trained operators. The lack of fire watch requirements for containment is based upon the administrative control of transient combustibles and the low potential of introducing transient combustibles or ignition sources, since containment is not normally accessible during POWER OPERATION. The lack of fire watch requirements for the waste gas decay tanks room, spent resin tank area and ion exchange area is based upon the fact that the areas contain negligible combustible loading and are normally inaccessible, limiting the potential of introducing combustibles or ignition sources. The lack of fire watch requirements for the spent fuel pool and cask laydown area is based upon the fact that the area is underwater. Further, these areas contain no ARSR components or cables. The lack of fire watch requirements for the service water cable tunnels is based upon the low potential of introducing transient combustibles or ignition sources, since these tunnels are normally inaccessible. While this area contains ARSR cables, alternate shutdown methods are available were a fire to affect both cable tunnels.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

6. The FUNCTIONALITY Assessment, in accordance with station procedures, if warranted, for continued plant operation is intended to ensure the nonfunctionality of the ARSR equipment has been thoroughly evaluated in accordance with existing engineering guidelines and that the appropriate level of plant management is cognizant of the status of the equipment.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

7. Entry into MODE 4 or higher is permitted with ARSR components nonfunctional. The ACTIONS identified in the “Compensatory Measures” column of the component FUNCTIONALITY Requirements and Compensatory Measures tables shall be implemented at the time of entry into MODE 4 for components found to be nonfunctional during MODES 5 or 6. Compensatory ACTIONS shall be implemented when the component is declared nonfunctional for components found to be nonfunctional during MODES 1, 2, 3, or 4.

8. TECHNICAL SURVEILLANCE REQUIREMENT of the Turbine Driven Auxiliary Feedwater Pump (P4) cannot be implemented before entering MODE 3. If the previous TECHNICAL SURVEILLANCE REQUIREMENT was satisfactory, and P4 has not been declared nonfunctional for other reasons, MODES 3 and 4 can be entered without restriction. This approach is consistent with Technical Specification 4.7.1.2.a.2.b which allows entry into MODE 3, even though the Technical Specification surveillance for the Turbine Driven Auxiliary Feedwater Pump is not performed prior to MODE 3 entry.

9. Electrical Breaker TECHNICAL SURVEILLANCE REQUIREMENT Testing

   a. TECHNICAL SURVEILLANCE REQUIREMENT Frequency

      TECHNICAL SURVEILLANCE REQUIREMENT frequency are as specified in the Preventive Maintenance Program. Changes in the frequency for breaker PM’s, as allowed by the Preventive Maintenance Program, continue to fulfill the Appendix R TECHNICAL SURVEILLANCE REQUIREMENT and prove FUNCTIONALITY of these breakers. The electrical breaker TECHNICAL SURVEILLANCE REQUIREMENTS may be performed in any MODE when preventive maintenance on the breakers is allowed by either the Preventive Maintenance Program or by breaker availability, such as during a train outage.

   b. Breaker Manual Cycling

      Manual cycling of a breaker is defined as cycling the breaker via the mechanical trip and close buttons.

   c. ARSR Equipment Isolation

      When ARSR equipment (not a breaker) is removed from service for maintenance, repairs, testing, etc. and its associated power supply breaker(s) and/or control power breaker(s) are intentionally opened, no additional Appendix R compensatory ACTIONS specifically for the opened breakers are required since removal of the ARSR equipment from service already has adequate compensatory ACTIONS.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

d. NON-ARSR Equipment Isolation

When equipment not listed as ARSR equipment is removed from service but its associated power supply breaker(s) and/or control power breaker(s) is listed as ARSR equipment, no Appendix R compensatory ACTION is required for the open breaker(s) since the component is not credited for operation in any Appendix R shutdown scenario.

e. When D0105 is out for maintenance, D0111 will be used to jumper out D0105 to supply panel DV10.

When D0108 is out for maintenance, D0111 will be used to jumper out D0108 to supply panel DV30.

When D0205 is out for maintenance, D0211 will be used to jumper out D0205 to supply panel DV40.

When D0208 is out for maintenance, D0211 will be used to jumper out D0208 to supply panel DV20.

10. Existing compensatory ACTION(s) for a nonfunctional fire suppression and/or detection system may be utilized in lieu of FUNCTIONALITY for that suppression and/or detection system when equal compensatory ACTIONS are required for additional ARSR equipment TECHNICAL REQUIREMENT for the same fire area.

11. ACTION c requires that the impact be managed by performance of a FUNCTIONALITY Assessment, in accordance with station procedures, if warranted, if ARSR component FUNCTIONALITY cannot be established within 60 days. The use of a FUNCTIONALITY Assessment, in accordance with station procedures, if warranted, assures the FUNCTIONALITY of ARSR components through evaluation or the use of compensatory measures if needed.

12. The intent of limiting transient combustibles as a compensatory measure is to create a heightened awareness of potential fire hazards (combustibles), limit to the extent possible the introduction of new hazards, and limit existing hazards through removal as work is completed. This would be accomplished by the fire watch, and the administrate controls of the permitting process. It is not the intent to prevent the introduction of materials required for repair of the system(s) in question, or fire protection systems used as compensatory measures.

(See AR 06000479-02 for additional information)

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

13. Risk Significant Components

According to Regulatory Guide 1.160, Revision 3 “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” the NRC determined additional guidance was necessary to adequately assess and manage the risk from internal fires in the conduct of activities required by MRule (a)(4). Revision 4A of NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” addresses this concern by providing methods licensees can use to identify equipment important to mitigation of risk of core damage from fire initiators. The revision also describes approaches to developing and implementing fire Risk Management Actions (RMAs) and the tools for effective implementation of the guidance.

The risk associated with the performance of maintenance activities having the potential to cause a fire (e.g., welding, use of cutting and grinding tools, transient combustibles, etc.), removal of fire detection or suppression equipment from service, and removal or impairment of fire barriers (e.g., opening of fire doors to facilitate maintenance, removal of protective barriers on cable trays or conduit, etc.) are all addressed by the existing Appendix R program. No additional action is warranted under MRule (a)(4) for these items. However, under the new guidance, removal of equipment from service important to core damage mitigation requires additional action to ensure adequate risk assessment for the impact of fire risk.

According to NUMARC 93-01, Revision 4A, the identification of equipment important for mitigating core damage resulting from fire initiating events can come from either: 10 CFR 50.58/Appendix R, a screening analysis (e.g. Fire Induced Vulnerability Evaluation), or a fire Probability Risk Assessment (PRA). MPS3 does not have fire PRAs, so Appendix R was used as the basis to identify the equipment within the existing MRule (a)(4) scope that is found to have appreciable impact on core damage mitigation for fire initiators.

The Fire Related Safe Shutdown Component list was compared with the PRA Internal Events model list of components which are risk significant for core damage frequency. Those components that were common to both lists, comprised the population of components considered for inclusion in the MRule (a)(4) program for fire risk impact. This component list was further refined down to a risk-informed list through site-specific working groups and a fleet expert panel. Once the appropriate scope of components was determined, RMAs to compensate for the reduction of the fire Safe Shutdown Capability when the component is removed from service, were developed based on guidance provided in NUMARC 93-01, Section 11.3.7.5.

The RMAs will be implemented using the method detailed in NUMARC 93-01, Section 11.3.7.3, Item 2. If a component important for mitigating core damage

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

BASES (Continued)

13. (continued)

resulting from fire initiating events is out of service for 72 hours, RMAs must be implemented. Within 30 days, the component is to be returned to service. If the component is not returned to service within 30 days, additional risk evaluations to determine additional actions must be performed.

It is important to note, the RMAs developed under the new guidance only address incremental risk from fire events. It is not appropriate to utilize the fire risk RMA implementation methods to address aggregate risk from fire and internal events. Risk from internal events will continue to be evaluated under the current MRule (a)(4) program.

REFERENCE:

1. Condition Report M3-00-02659.
2. RP-5, Operability Determinations.

Appendix R Safe Shutdown TECHNICAL REQUIREMENTS Index

TR No.	System Description	Component ID	ADDITIONAL REQUIREMENTS	TRM 7.1 Page No.
7.1.1	Charging Pumps	P18A, 18B, 18C	TR 3.1.2.4	10
7.1.1	Regen HX outlet iso	CH-089	TR 3.1.2.4	10
7.1.1	Charging to RCS loops	CH-518, 519	TR 3.1.2.4	11
7.1.1	Aux Spray to PZR (RCS)	CH-517	TR 3.1.2.4	11
7.1.1	Charging pump header iso MOV	CH-429	TR 3.1.2.4	11
7.1.1	VCT outlet MOV	CH-501	TR 3.1.2.4	11
7.1.1	Chrg Pps suct fm RWST MOV	CH-504	TR 3.1.2.4	12
7.1.1	Chrg Pps suct fm RWST AOV	CH-192	TR 3.1.2.4	12
7.1.2	Instr, Monitor local BAST level	LI-206A, 208A	TR 3.1.2.8.a	16
7.1.3	MS Low Pt Drain -ESF-MSI	MS-265B	TS 3.3.2.1	19
7.1.3	MS Low Pt Drain -ESF-MSI	MS-266B	TS 3.3.2.1	19
7.1.4	Instr, Monitor SG Press	PT-1023B	TS 3.3.3.5	22
7.1.4	Instrument, Monitoring Pzr Press	PT-102B-1	TS 3.3.3.5	22
7.1.4	Instr, Monitor NI, wide rang	QEXC2PDA, 4PDA	TS 3.3.3.5	22
7.1.4	Instr, Monitor NI, wide range	QEXC1PDA, 3PDA	TS 3.3.3.5	22
7.1.4	Instr, Monitor Pzr Press low rng	PT-103, 103-1	TS 3.3.3.5	22
7.1.4	Instr, monitor Pzr level, cold, nor	LT-103, 110X, 110Y	TS 3.3.3.5	23
7.1.4	Instr, Monitor SG 2 level, nar rng	LT-1123B	TS 3.3.3.5	23
7.1.4	Instr, Monitor SG 2 level,wide rng	LT-1124A, 1124B	TS 3.3.3.5	23
7.1.4	Instr, monitor SG 1 level,wide rng	LT-1114A, 1114B	TS 3.3.3.5	24
7.1.5	Instr, Monitoring hot leg temp	TE-121X	TS 3.3.3.8	27
7.1.5	Instr, Monitoring hot leg temp	TE-111X	TS 3.3.3.8	27
7.1.5	Instr, Monitoring cold leg temp	TE-125	TS 3.3.3.8	27
7.1.5	Instr, Monitoring cold leg temp	TE-115	TS 3.3.3.8	27
7.1.5	Instr, Monitor AFW flow to SG 2	FT-5278A. 5278B	TS 3.3.3.8	28
7.1.5	Instr, Monitor AFW flow to SG 1	FT-5277A, 5277B	TS 3.3.3.8	28
7.1.6	6900 V breakers for RCPs Bus 25A	(H1)	TS 3.4.1.2	31
7.1.6	6900 V breakers for RCPs Bus 25B	(H2)	TS 3.4.1.2	31
7.1.7	SDC HX bypass	SI-306	TS 3.4.1.3	34
7.1.7	SDC suct MOVs	SI-651, 652	TS 3.4.1.3	34
7.1.7	SDC sup, ret temp monitors	TE-351X, 351Y	TS 3.4.1.3	34
7.1.7	SDC HXs outlet temps indication	TE-303X, 303Y	TS 3.4.1.3	35
7.1.8	RCS- PORVs	RC-402/ 404	TS 3.4.3	38
7.1.9	SIT Nitrogen vent valves	SI-613,623,633,643	TS 3.5.1.a	41
7.1.9	SIT Outlet MOVs	SI-614,624,634,644	TS 3.5.1.a	41
7.1.10	LPSI flow MOVs	SI-615,625,635,645	TS 3.4.1.3	44
7.1.10	SDC min flow alt	SI-449, 450	TS 3.4.1.3	44
7.1.10	SDC HX outlet	SI-657	TS 3.4.1.3	44
7.1.10	LPSI suct fm RWST	SI-432, 444	TS 3.4.1.3	44
7.1.10	LPSI suct fm RCS	SI-440, 441	TS 3.4.1.3	44
7.1.10	LPSI disch to SDC HX	SI-452, 453	TS 3.4.1.3	45
7.1.10	SDC HX outlet	SI-456, 457	TS 3.4.1.3	45
7.1.11	RWST outlet MOV	CS-13.1B	n/a	48

Appendix R Safe Shutdown TECHNICAL REQUIREMENTS Index (Continued)

TR No.	System Description	Component ID	ADDITIONAL REQUIREMENTS	TRM 7.1 Page No.
7.1.12	Boric Acid Grav feed MOVs	CH-508, 509	TR 3.1.2.2.a	51
7.1.13	Ctmt Ambient Temp monitoring	TE-8108	TS 3.6.1.5	54
7.1.14	Ctmt Spray Stop Valves	CS-4A, 4B	TS 3.6.2.1	57
7.1.15	AFW Turbine	SV-4188	TS 3.7.1.2	60
7.1.15	AFW Turbine	P4 & H-21	TS 3.7.1.2	60
7.1.15	AFW Turbine steam stop	MOV MS-202	TS 3.7.1.2	60
7.1.15	AFW pumps, motor driven	P9A, P9B	TS 3.7.1.2	60
7.1.15	AFW P4 Discharge Iso	MOV FW-44	TS 3.7.1.2	61
7.1.15	AFW flow control valves	FW-43A/43B	TS 3.7.1.2	61
7.1.16	Instr, Monitor CST level	LT-5282	TS 3.7.1.3	64
7.1.16	Instr, Local CST level Ind	LIS-5489	TS 3.7.1.3	64
7.1.17	MSIVs	MS-64A/64B	TS 3.7.1.5	67
7.1.17	MSIV Bypass M	S-65A/65B	TS 3.7.1.5	67
7.1.18	ADVs	MS-190A/190B	TS 3.7.1.7	70
7.1.19	SG Blowdown Valves	MS-220A/220B	TS 3.7.1.8	73
7.1.20	RBCCW out of SDC HXs	RB-13.1A, 13.1B	TS 3.7.3.1	76
7.1.20	RBCCW Pp A Inlet AOVs	RB-211A, 211B	TS 3.7.3.1	76
7.1.20	RBCCW Pp B Inlet AOV's	RB-211C, 211D	TS 3.7.3.1	77
7.1.20	RBCCW Pp C Inlet AOVs	RB-211E, 211F	TS 3.7.3.1	77
7.1.20	RBCCW Hx A outlet AOVs	RB-4.1A, 4.1B	TS 3.7.3.1	78
7.1.20	RBCCW Hx B outlet AOVs	RB-4.1C, 4.1D	TS 3.7.3.1	78
7.1.20	RBCCW Hx C outlet AOVs	RB-4.1E, 4.1F	TS 3.7.3.1	79
7.1.20	RBCCW pumps	P11A, P11B, P11C	TS 3.7.3.1	79
7.1.20	RBCCW pp disch x-tie AOVs	RB-251A, 251B	TS 3.7.3.1	79a
7.1.21	SW pumps & Strainers	P5A, P5C, L1A, L1C	TS 3.7.4.1	82
7.1.21	SW Disch hdr x-tie valves	SW-97A/ 97B	TS 3.7.4.1	82
7.1.21	SW pump disch valves	SW-2A, 2B, 2C	TS 3.7.4.1	83
7.1.21	SW inlet to RBCCW HXs	SW-3.1A/ 3.1B	TS 3.7.4.1	83
7.1.21	SW outlet fm RBCCW HXs	SW-9A/ 9B/ 9C	TS 3.7.4.1	84
7.1.21	SW inlet to TBCCW HXs	SW-5A/ 5B/ 5C	TS 3.7.4.1	84
7.1.21	SW iso to EDG A & B	SW-12B/ 12C	TS 3.7.4.1	84
7.1.21	Drain on SW to EDG B	SW-296	TS 3.7.4.1	84a
7.1.22	Alternate AC diesel gen	SBO DG	TS 3.8.1.1.a	87
7.1.23	4160 volt switchgear	Bus 24C (A3)	TS 3.8.2.1	90
7.1.23	4160 volt switchgear	Bus 24D (A4)	TS 3.8.2.1	91
7.1.23	4160 volt switchgear	Bus 24E (A5)	TS 3.8.2.1	91
7.1.23	4160 volt switchgear	Bus 24A (A1)	TS 3.8.2.1	92
7.1.23	4160 volt switchgear	Bus 24B (A2)	TS 3.8.2.1	92
7.1.23	480 volt load center	Bus 22A (B01)	TS 3.8.2.1	92
7.1.23	480 volt load center	Bus 22B (B02)	TS 3.8.2.1	92
7.1.23	480 volt switchgear	Bus 22E (B05)	TS 3.8.2.1	93
7.1.23	480 volt switchgear	Bus 22F (B06)	TS 3.8.2.1	93
7.1.23	MCC B51	B51 (MCC-22-1E)	TS 3.8.2.1	93

Appendix R Safe Shutdown TECHNICAL REQUIREMENTS Index (Continued)

TR No.	System Description	Component ID	ADDITIONAL REQUIREMENTS	TRM 7.1 Page No.
7.1.23	MCC B52	B52 (MCC-22-2E)	TS 3.8.2.1	93
7.1.23	MCC B61	B61 (MCC-22-1F)	TS 3.8.2.1	94
7.1.23	MCC B62	B62 (MCC-22-2F)	TS 3.8.2.1	94
7.1.23	120 VAC panel VIAC-2	VA 20	TS 3.8.2.1	94
7.1.23	Em. Light Units	ELUs	TS 3.8.2.1	95
7.1.24	125 VDC Bus 201A	Bus D01	TS 3.8.2.3	99
7.1.24	125 VDC Bus 201B	Bus D02	TS 3.8.2.3	99
7.1.24	125 VDC panel 201A-1V	DV 10	TS 3.8.2.3	100
7.1.24	125 VDC panel 201B-1V	DV 20	TS 3.8.2.3	101
7.1.24	125 VDC panel 201A-1	D11	TS 3.8.2.3	101
7.1.24	125 VDC panel 201A-2	D12	TS 3.8.2.3	101
7.1.24	125 VDC panel 201B-1	D21	TS 3.8.2.3	102
7.1.24	125 VDC panel 201B-2	D22	TS 3.8.2.3	102
7.1.25	Alt (Fire Pro) cooling to EDGs	FIRE-258,259,559,560	n/a	105
7.1.25	Fire Supp System	Fire Suppression sys	n/a	105
7.1.25	Cross-tie to AFW suction	2-FIRE-34	n/a	105
7.1.26	Radio Communication System	Hand-held Radios	n/a	108
7.1.26	Box Inventory items available	HSD Storage Box	n/a	108
7.1.26	Box Inventory items available	FSD Storage Box	n/a	108
7.1.26	Materials available for repairs	CSD Repair Material	n/a	109
7.1.26	Fans-East & West DC rooms	F156A, B, C, D	n/a	109
7.1.26	Fans-East & West 480 V rooms	F157A, B, C, D	n/a	109
7.1.26	Bottles & Valves	Bottle 3A, 3B, 3C	n/a	110
7.1.27	Emergency Diesel Generators	H7A, H7B	TS 3.8.1.1	113

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL REQUIREMENT: 3.1.2.4 - Charging Pumps

TECHNICAL REQUIREMENT

7.1.1 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.1-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.1 The capability of equipment listed in TRM Table 7.1.1-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.1-2.

Table 7.1.1-1
CVCS Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see Action "d")
A	P18A Charging Pump	Pump must be FUNCTIONAL from control room	With P18A nonfunctional perform ACTIONS b.1, b.2 for Fire Area R-2.	With P18A pump nonfunctional, Risk Mitigation Areas are: R-2 (FHA A-8B, A-8D, A-13 and T-10), R-3 (FHA T-1A, T-1C and T-1F), R-5 (FHA A-8A), R-8 (FHA A-16), R-10 (FHA A-21 and A-23) and R-11 (FHA A-28)
B	P18B Charging Pump	Pump must be controllable from panel C10	1. With P18B nonfunctional from C10 ensure P18C is FUNCTIONAL from C10 or perform ACTIONS b.1, b.2 for Fire Areas R-1 & R-13. 2. With P18B nonfunctional from panel C10 perform ACTIONS b.1, b.2 for fire area R-4.	N/A
C	P18C Charging Pump	1. Pump must be controllable from panel C10. 2. Pump must be FUNCTIONAL from control room	1. With P18C nonfunctional from C10 ensure P18B is FUNCTIONAL from C10 or perform ACTIONS b.1, b.2 for Fire Areas R-1 & R-13. 2. With P18C nonfunctional from CR perform ACTIONS b.1, b.2 for Fire Area R-4.	N/A
D	Z2 Powered Charging Pump	At least one pump must be powered from facility Z2.	N/A	With Z2 pump nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-1B, A-24 and A-25), R-4 (FHA A-6A), R-7 (FHA A-15), R-9 (FHA A-20 and A-22), R-13 (FHA T-6, R-14 (FHA T-7 and T-9) and R-17 (FHA A-10B)
E	CH-089 Regen HX Outlet Iso	Valve must be controllable from panel C10	Perform ACTIONS b.1, b.2 for Fire Area R-1	N/A

Table 7.1.1-1
CVCS Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see Action "d")
F	CH-518 Charging to RCS	Valve must be capable of being cycled by local manual hand-wheel	Verify that valve CH-518 & 519 can be cycled from the CR and BU air bottles are available or perform ACTIONS b.1, b.2 for all fire areas except R-1.	N/A
G	CH-519 Charging to RCS	1. Valve must be capable of being cycled by local manual hand-wheel. 2. Valve must be controllable from panel C10.	1. Verify that valve CH-518 & 519 can be cycled from the CR and BU air bottles are available or perform ACTIONS b.1, b.2 for all fire areas except R-1. 2. Perform ACTIONS b.1, b.2 for fire area R-1.	N/A
H	CH-517 Auxiliary Spray to Pressurizer AOV	1. Valve must be capable of being cycled by local manual hand-wheel. 2. Valve must be capable of being cycled using BU air bottles by remote CR operation. 3. Valve must be controllable from panel C10 using BU air bottles. 4. Valve must fail closed when solenoid is de-energized.	1. With hand-wheel nonfunctional perform ACTIONS b.1, b.2 for fire areas R-2, R-10 & R-15. 2. Verify valve can be cycled by local manual hand-wheel or perform ACTIONS b.1, b.2 for ALL fire areas except R-1. 3. With valve nonfunctional perform ACTIONS b.1, b.2 for Fire area R-1. 4. With valve nonfunctional perform ACTIONS b.1, b.2 for Fire areas R-2 & R-15	N/A
I	CH-429 Charging Pp Hdr Iso MOV	1. Valve must be capable of being cycled by local manual hand-wheel. 2. Valve must be controllable from panel C10	1. With valve nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1 & R-2. 2. With valve nonfunctional, ensure manual capability is available.	N/A

Table 7.1.1-1
CVCS Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see Action “d”)
J	CH-501 VCT Outlet MOV	Valve must be capable of being cycled by local manual hand-wheel.	With valve nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1,R-4, R-7, R-9, R-13, R-14, R-16 & R-17.	N/A
K	CH-504 Chrg Pps Suct fm RWST	Valve must be capable of being cycled by local manual hand-wheel.	With valve nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1.	N/A
L	CH-192 Charging Pps Suct fm RWST AOV	Valve must be capable of being cycled by local manual hand-wheel.	1. With the hand-wheel nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1, R-4, R-7, R-9, R-13, R-14, R-16 & R-17; and 2. Verify valve can be operated using BU air or perform ACTIONS b.1, b.2 for fire areas R-2, R-3,R-5, R-6, R-8, R-10, R-11, R-12 & R-15.	N/A

Table 7.1.1-2
CVCS Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR No.	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	P18A Charging Pump	Verify P18A can be operated from the control room.	Once per REFUELING cycle
B	P18B Charging Pump	Verify control of P18B can be isolated from the control room and that it can be operated from C10.	Once per REFUELING cycle
C	P18C Charging Pump	1. Verify control of P18C can be isolated from the control room and that it can be operated from C10. 2. Verify P18C can be operated from the control room.	Once per REFUELING cycle
D	Z2 Powered Charging Pump	N/A	N/A
E	CH-089 Regen HX Outlet Iso	Verify control can be isolated from the control room and operated room C10.	Once per REFUELING cycle
F	CH-518 Charging to RCS	Verify valve can be manually cycled using its hand-wheel.	Once every 48 months.
G	CH-519 Charging to RCS	1. Verify valve can be manually cycled using its hand- wheel. 2. Verify control of 2-CH-519 can be isolated from the control room and operated from panel C10.	1. Once every 48 months. 2. Once per REFUELING cycle.
H	CH-517 Auxiliary Spray to Pressurizer	1. Verify valve can be manually cycled using its hand-wheel. 2. Verify valve can be operated from control room using backup air bottles. 3. Verify valve control can be isolated from control room and operated from C10, using backup air bottles. 4. Verify valve fails closed on loss of control power.	1. Once every 48 months. 2. Once per REFUELING cycle 3. Once per REFUELING cycle 4. Once per REFUELING cycle

Table 7.1.1-2
CVCS Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR No.	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
I	CH-429 Charging Pp Hdr Iso MOV	1. Verify valve can be manually cycled using its hand-wheel. 2. Verify valve control can be isolated from control room and operated from Panel C10.	1. In accordance with the MOV surveillance frequency established by the PM Program. 2. Once per REFUELING cycle
J	CH-501 VCT Outlet MOV	Verify valve can be manually cycled using its hand-wheel.	In accordance with the MOV surveillance frequency established by the PM Program.
K	CH-504 Chrg Pps Suct fm RWST	Verify valve can be manually cycled using its hand-wheel	In accordance with the MOV surveillance frequency established by the PM Program
L	CH-192 Chrg Pps Suct fm RWST AOV	Verify valve can be manually cycled using its hand-wheel	Once every 48 months.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL REQUIREMENT: 3.1.2.8.a - Boric Acid Storage Tanks

TECHNICAL REQUIREMENT

7.1.2 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.2-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.2 The capability of equipment listed in TRM Table 7.1.2-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.2-2.

Table 7.1.2-1
BAST Local Level Instruments
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	LI-206A Local BAST Level Ind	Local level indicator must be FUNCTIONAL.	With local level indication nonfunctional, perform ACTIONS b.1, b.2 listed for Fire Areas R-2 and R-10.
B	LI-208A Local BAST Level Ind	Local level indicator must be FUNCTIONAL.	With local level indication nonfunctional, perform ACTIONS b.1, b.2 listed for Fire Areas R-2 and R-10.

Table 7.1.2-2
BAST Local Level Indicators
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	LI-206A Local BAST Level Ind	Verify the instrument is calibrated.	Once per REFUELING cycle
B	LI-208A Local BAST Level Ind	Verify the instrument is calibrated.	Once per REFUELING cycle

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.3.2.1 Engineered Safety Feature Actuation System Instrumentation

TECHNICAL REQUIREMENT

7.1.3 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.3-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.3 The capability of equipment listed in TRM Table 7.1.3-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.3-2.

Table 7.1.3-1
MSI Components
Functionality Requirements And Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-MS-265B Mn Steam Low Point Drain	Valve must be capable of failing closed when the solenoid is de-energized.	With valve nonfunctional, perform ACTIONS b.1, b.2 for Fire Areas R-1 and R-17.
B	2-MS-266B Mn Steam Low Point Drain	Valve must be capable of failing closed when the solenoid is de-energized.	With valve nonfunctional, perform ACTIONS b.1, b.2 for Fire Areas R-1 and R-2.

Table 7.1.3-2
MSIV Components Technical Surveillance Requirements

TSR No.	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-MS-265B Mn Steam Low Point Drain	Verify valve will fail closed on loss of control power	Once per REFUELING cycle
B	2-MS-266B Mn Steam Low Point Drain	Verify valve will fail closed on loss of control power	Once per REFUELING cycle

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.3.3.5 REMOTE SHUTDOWN INSTR'S

TECHNICAL REQUIREMENT

7.1.4 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.4-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.4 The capability of equipment listed in TRM Table 7.1.4-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.4-2.

Table 7.1.4-1
Remote Shutdown Instruments, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	PT-102B-1 Pressurizer Pressure	Instrument loop and pressure indicator must be FUNCTIONAL at Panel C10.	With loop nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1.
B	PT-1023B Stm Gen # 2 Pressure	Instrument loop must be FUNCTIONAL at Panel C10.	With loop nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1.
C	QEXC2PDA (WR-DET-B), QEXC4PDA (WR-DET-D), QEXC1PDA (WR-DET-A), QEXC3PDA (WR-DET-C)	Instrument loops must be FUNCTIONAL from the Control Room.	1. With both B and D loops nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-7, R-9, R-14, R-16, and R-17. 2. With loop D nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire area R-13 3. With both A and C loops nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-2 and R-10 4. With all loops (A,B,C,D) nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-4, R-5, R-6, R-7, R-8, R-9, R-10, R-11, R-12, R-13, R-14, R-15, R-16 and R-17.
D	PT-103, PT-103-1 Pressurizer Pressure Low Range	Instrument loops must be FUNCTIONAL from the Control Room.	1. With 103 loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-7, R-9, R-13, R-14, R-16 and R-17. 2. With 103-1 loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, and R-10. 3. With loops 103 and 103-1 nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-4, R-5, R-6, R-8, R-11, and R-12.

Table 7.1.4-1
Remote Shutdown Instruments, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
E	LT-103 (Cold) LT-110X LT-110Y Pressurizer Level	1. Loop LT-103 must be FUNCTIONAL from Panel C10. 2. Loop LT-110X must be FUNCTIONAL from Control Room. 3. Loop LT-110Y must be FUNCTIONAL from Control Room.	1. With loop LT-103 nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1. 2. With loop LT-110X nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-2, R-10, and R-15. 3. With loop LT-110Y nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-7, R-9, R-14, R-15, R-16, and R-17.
F	LT-1123B SG 2 Water Level Narrow Range LT-1124A LT-1124B SG2 Water Level Wide Range	1. Loops L-1123B and F-5278B must be FUNCTIONAL from Panel C10. 2. SG WR Level, or AFW Flow and NR Level for same SG must be FUNCTIONAL from Control Room.	1. With either loop F-5278B or L-1123B nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1. 2. With loop L-1124A nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-2. 3. With loops L-1124A and F-5278A nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-10. 4. With Loop L-1124B nonfunctional from Control Room, perform ACTIONS b.1, b.2 for Fire Area R-13. 5. With loops L-1124B & F-5278B nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-7, 9, 14, 16 and R-17.

Table 7.1.4-1
Remote Shutdown Instruments, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
G	LT-1114A LT-1114B SG 1 Water Level Wide Range	SG Wide Range Water Level or AFW Flow must be FUNCTIONAL from Control Room	1. With loop L-1114A nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-2. 2. With loops L-1114A and F-5277A nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-10. 3. With loops L-1114B and F-5277B nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-7, 9, 13, 14, 16 & R-17.

Table 7.1.4-2
Remote Shutdown Instruments, Appendix R Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	PT-102B-1 Pressurizer Pressure	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
B	PT-1023B Stm Gen # 2 Pressure	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
C	QEXC2PDA (WR-DET-B), QEXC4PDA (WR-DET-D), QEXC1PDA (WR-DET-A), QEXC3PDA (WR-DET-C)	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
D	PT-103, PT-103-1 Pressurizer Pressure Low Range	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
E	LT-103 (Cold) LT-110X LT-110Y Pressurizer Level	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
F	LT-1123B LT-1124A SG 2 Water Level Wide Range LT-1124B	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.
G	LT-1114A LT-1114B SG 1 Water Level Wide Range	Perform a CHANNEL CALIBRATION to verify that the instrument is calibrated.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.3.3.8 Accident Monitor Instr's

TECHNICAL REQUIREMENT

7.1.5 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.5-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.5 The capability of equipment listed in TRM Table 7.1.5-1 to meet its Appendix R shutdown related FUNCTIONALity requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.5-2.

Table 7.1.5-1
Accident Monitoring Instrumentation, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	TE-121X RCS Hot Leg 2 Temperature	1. Loop must be FUNCTIONAL at Panel C10. 2. Loop must be FUNCTIONAL at Control Room.	1. With loop nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1. 2. With loop nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for all fire areas EXCEPT R-1, R-2, and R-10.
B	TE-111X RCS Hot Leg 1 Temperature	Loop must be FUNCTIONAL at Control Room.	With loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-4, R-5, R-6, R-8, R-10, R-11, R-12, and R-15.
C	TE-125 RCS Cold Leg 2 Temperature	1. Loop must be FUNCTIONAL at Panel C10. 2. Loop must be FUNCTIONAL at Control Room.	1. With loop nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1. 2. With loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for all fire areas EXCEPT R-1, R-2, and R-10.
D	TE-115 RCS Cold Leg 1 Temperature	Loop must be FUNCTIONAL at Control Room.	With loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-4, R-5, R-6, R-8, R-10, R-11, R-12, and R-15.

Table 7.1.5-1
Accident Monitoring Instrumentation, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
E	FT-5278A, FT-5278B AFW Flow to SG 2.	1. Loops F-5278B and L-1123B must be FUNCTIONAL from Panel C10. 2. SG WR Level, or AFW Flow and NR Level for same SG must be FUNCTIONAL from Control Room.	1. With either loop F-5278B or L-1123B nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire area R-1. 2. With loop L-1124A nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-2. 3. With loops L-1124A and F-5278A nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-10. 4. With Loop L-1124B nonfunctional from Control Room, perform ACTIONS b.1, b.2 for Fire Area R-13. 5. With loops L-1124B & F-5278B nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-7, 9, 14, 16 & R-17.
F	FT-5277A, FT-5277B AFW Flow to SG 1	SG Wide Range Water Level or AFW Flow must be FUNCTIONAL from Control Room	1. With loop L-1114A nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire area R-2. 2. With loops L-1114A and F-5277A nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-10. 3. With loops L-1114B and F-5277B nonfunctional from Control room perform ACTIONS b.1, b.2 for fire area R-7, 9, 13, 14, 16 & R-17.

Table 7.1.5-2
Accident Monitoring Instrumentation, Appendix R Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	TE-121X RCS Hot Leg 2 Temperature	1. Perform CHANNEL CALIBRATION to verify the instrument is calibrated. 2. Perform CHANNEL CALIBRATION as presently stated.	Once per REFUELING cycle.
B	TE-111X RCS Hot Leg 1 Temperature	Perform CHANNEL CALIBRATION to verify the instrument is calibrated.	Once per REFUELING cycle.
C	TE-125 RCS Cold Leg 2 Temperature	1. Perform CHANNEL CALIBRATION to verify the instrument is calibrated. 2. Perform CHANNEL CALIBRATION as presently stated.	Once per REFUELING cycle.
D	TE-115 RCS Cold Leg 1 Temperature	Perform CHANNEL CALIBRATION to verify the instrument is calibrated.	Once per REFUELING cycle.
E	FT-5278A, FT-5278B AFW Flow to SG 2.	Perform a CHANNEL CALIBRATION to verify that the instruments are calibrated.	Once per REFUELING cycle.
F	FT-5277A, FT-5277B AFW Flow to SG 1	Perform a CHANNEL CALIBRATION to verify that the instruments are calibrated.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.4.1.2 Coolant Loops and Circulation

TECHNICAL REQUIREMENT

7.1.6 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.6-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.6 The capability of equipment listed in TRM Table 7.1.6-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.6-2.

Table 7.1.6-1
6900 Volt Bus 25A (H1) and 25B (H2) Switchgear - RCP Breakers
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	6900 Volt RCP Breaker H104	RCP Breaker must be capable of being cycled (tripped) by local manual operation.	With local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-9 and R-17.
B	6900 Volt RCP Breaker H105	RCP Breaker must be capable of being cycled (tripped) by local manual operation.	With local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-9 and R-17.
C	6900 Volt RCP Breaker H201	RCP Breaker must be capable of being cycled (tripped) by local manual operation.	With local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-10.
D	6900 Volt RCP Breaker H202	RCP Breaker must be capable of being cycled (tripped) by local manual operation.	With local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-10.

Table 7.1.6-2
6900 Volt Bus 25A (H1) and 25B (H2) Switchgear - RCP Breakers
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	6900 Volt RCP Breaker H104	Verify that the circuit breaker can be cycled (tripped) by local manual operation.	In accordance with the surveillance frequency established by the Preventive Maintenance Program: See Bases item No. 9.
B	6900 Volt RCP Breaker H105	Verify that the circuit breaker can be cycled (tripped) by local manual operation.	In accordance with the surveillance frequency established by the Preventive Maintenance Program: See Bases item No. 9.
C	6900 Volt RCP Breaker H201	Verify that the circuit breaker can be cycled (tripped) by local manual operation.	In accordance with the surveillance frequency established by the Preventive Maintenance Program: See Bases item No. 9.
D	6900 Volt RCP Breaker H202	Verify that the circuit breaker can be cycled (tripped) by local manual operation.	In accordance with the surveillance frequency established by the Preventive Maintenance Program: See Bases item No. 9.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.4.1.3 Coolant Loops and Circulation

TECHNICAL REQUIREMENT

7.1.7 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.7-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.7 The capability of equipment listed in TRM Table 7.1.7-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.7-2.

Table 7.1.7-1
Shutdown Cooling Components, Hot Shutdown
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-SI-306 SDC HX Bypass AOV	Valve must be capable of being cycled by local manual hand-wheel operation.	With the valve nonfunctional, perform ACTIONS b.1, b.2 for all fire areas.
B	2-SI-651,2-SI-652 SDC Suction Isolation MOVs	1. Each valve must be closed and electrically disconnected or made nonfunctional. Valve 2-SI-651 shall have manual disconnect switch NSI651 locked open. Valve 2-SI-652 shall have manual disconnect switch 89-SI-652 open. 2. Each valve must be capable of being electrically opened. 3. Each valve must be capable of being opened by local manual hand-wheel operation.	1. N/A NOTES: For plant operation in MODES 1 - 4, Technical Specification LCO 3.6.3.1, Containment Isolation Valves, also applies. Thus for MODES 1 - 4, valves must be OPERABLE to satisfy Technical Specifications and FUNCTIONAL to satisfy Appendix R requirements. 2. With one valve nonfunctional, perform ACTIONS b.1, b.2 fire area R-1. 3. With one valve nonfunctional, perform ACTIONS b.1, b.2 for fire area R-2, R-3, R- 7, R-8, R-9, R-10, R-11,R-13, R-14, R-15, R-16, R-17.
C	TE-351X, TE-351Y SDC combined suction/ return header temperature	Both instrument loops must be FUNCTIONAL from Control Room.	With either instrument loop nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-5, R-6, R-8, R-10, R-11, R-12, and R-15.

Table 7.1.7-1
Shutdown Cooling Components, Hot Shutdown
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION “a”)
D	TE-303X, TE-303Y SDC Heat Exchanger Outlets Temperature	Both instrument loops must be FUNCTIONAL from the Control Room.	1. With instrument loop TE-303X nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-4, R-6, R-8, R-10, and R-11. 2. With instrument loop TE-303Y nonfunctional from the Control Room, perform ACTIONS b.1, b.2 for fire areas R-5, R-7, R-9, R-13, R-14, R-16, and R-17.

Table 7.1.7-2
Shutdown Cooling Components, Hot Shutdown
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-SI-306 SDC HX Bypass AOV	Verify that the valve can be manually cycled using its hand-wheel.	Once every 48 months.
B	2-SI-651, 2-SI-652 SDC Suction Isolation MOVs	1. Administrative control over manual disconnect switch NSI651 locked open, for 2-SI-651, and manual disconnect switch 89-SI-652 operation for 2-SI-652 are provided in plant heat-up and cool-down procedures.	8. N/A
		2. Verify that each valve can be remotely cycled from the Control Room. Valves are electrically cycled when entering shutdown cooling and when securing from shutdown cooling, during plant heat-up. No additional surveillance testing needs to be performed.	9. Once per REFUELING cycle.
		3. Verify that each valve can be manually cycled using its hand-wheel.	10. In accordance with the MOV surveillance frequency established by the PM Program.
C	TE-351X, TE-351Y SDC combined suction/return header temperature	Perform CHANNEL CALIBRATION to verify each instrument loop is calibrated.	Once per REFUELING cycle.
D	TE-303X, TE-303Y SDC Heat Exchanger Outlets Temperature	Perform CHANNEL CALIBRATION to verify each instrument loop is calibrated.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.4.3 RCS Relief Valves

TECHNICAL REQUIREMENT

7.1.8 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.8-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.8 The capability of equipment listed in TRM Table 7.1.8-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.8-2.

Table 7.1.8-1
RCS Relief Valves - PORVs
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-RC-402, Pressurizer Relief Valve (PORV)	Valve 2-RC-402 must be capable of closure from Panel C70A.	With 2-RC-402 nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1, R-9, and R-15.
B	2-RC-404, Pressurizer Relief Valve (PORV)	Valve 2-RC-404 must be capable of closure from Panel C70B.	With 2-RC-404 nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1, R-10, and R-15.

Table 7.1.8-2
RCS Relief Valves - PORVs
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-RC-402, Pressurizer Relief Valve (PORV)	Verify that valve 2-RC-402 can be closed at Panel C70A.	Once per REFUELING cycle.
B	2-RC-404, Pressurizer Relief Valve (PORV)	Verify that valve 2-RC-404 can be closed at Panel C70b.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.5.1.a SIT Isolation Valves

TECHNICAL REQUIREMENT

7.1.9 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.9-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.9 The capability of equipment listed in TRM Table 7.1.9-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.9-2.

Table 7.1.9-1
Safety Injection Tank Isolation Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-SI-614 2-SI-624 2-SI-634 2-SI-644 SIT Block Valve (MOVs)	Each valve must be capable of being closed by local manual hand-wheel operation.	With one valve nonfunctional, perform ACTIONS b.1, b.2 for all fire areas EXCEPT R-1, R- 2, R-15, R-17.
B	2-SI-613 2-SI-623 2-SI-633 2-SI-643 SIT Vent Valve (AOVs)	Each valve must be capable of being electrically opened.	With any valve nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1.

Table 7.1.9-2
Safety Injection Tank Isolation Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-SI-614 2-SI-624 2-SI-634 2-SI-644 SIT Block Valve (MOVs)	Verify the MOV can be manually cycled.	In accordance with the MOV surveillance frequency established by the PM Program.
B	2-SI-613 2-SI-623 2-SI-633 2-SI-643 SIT Vent Valve (AOVs)	Verify that each valve can be remotely opened from the Control Room.	Once per REFUELING cycle

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.4.1.3 Coolant Loops and Circulation

TECHNICAL REQUIREMENT

7.1.10 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.10-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.10 The capability of equipment listed in TRM Table 7.1.10-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.10-2.

Table 7.1.10-1
ECCS - LPSI Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-SI-615 2-SI-625 2-SI-635 2-SI-645 LPSI Flow MOV	Each valve must be capable of being cycled by local manual hand-wheel operation.	1. With either 2-SI-615 or 2-SI-625 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R- 2,R-3, R-7, R-9, R-13, R-14,R-16,and R-17. 2. With either 2-SI-635 or 2-SI-645 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R- 2,R-8, R-10, and R-11.
B	2-SI-449 2-SI-450 LPSI (SDC) Min Flow AOV	Each valve must be capable of being closed by local manual hand-wheel operation.	With one valve nonfunctional, perform ACTIONS b.1, b.2 for all fire areas.
C	2-SI-657 (LPSI) SDC HX Outlet AOV	Valve must be capable of being cycled by local manual hand-wheel operation.	With the valve nonfunctional, perform ACTIONS b.1, b.2 for all fire areas.
D	2-SI-432 2-SI-444 LPSI Suction from RWST AOV	Each valve must be capable of being closed by local manual hand-wheel operation.	1. With 2-SI-432 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-5, R-7, R-9, R-13, R- 14, R-16,and R-17. 2. With 2-SI-444 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-6, R-8, R-10, and R-11.
E	2-SI-440 2-SI-441 LPSI/SDC Suction from RCS AOV	Each valve must be capable of being opened by local manual hand-wheel operation.	1. With 2-SI-441 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-6, R-8, R-10, and R-11. 2. With 2-SI-440 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-5, R-7, R-9, R-13, R-14,R-16, and R-17.

Table 7.1.10-1
ECCS - LPSI Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
F	2-SI-452 2-SI-453 LPSI disch to SDC HX AOV	Each valve must be capable of being opened by local manual hand-wheel operation.	1. With 2-SI-452 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-6, R-8, R-10, and R-11. 2. With 2-SI-453 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-5, R-7, R-9, R-13, R-14,R-16, and R-17.
G	2-SI-456 2-SI-457 SDC HX outlet AOV	Each valve must be capable of being opened by local manual hand-wheel operation.	1. With 2-SI-456 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-6, R-8, R-10, and R-11. 2. With 2-SI-457 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-5, R-7, R-9, R-13, R-14,R-16, and R-17.

Table 7.1.10-2
ECCS - LPSI Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-SI-615 2-SI-625 2-SI-635 2-SI-645 LPSI Flow MOV	Verify that each valve can be manually cycled using its hand-wheel.	In accordance with the MOV surveillance frequency established by the PM Program.
B	2-SI-449 2-SI-450 LPSI (SDC) Min Flow AOV	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling, and securing from shutdown cooling during plant heat-up. No additional surveillance testing needs to be performed.	Once every 48 months.
C	2-SI-657 (LPSI) SDC HX Outlet AOV	Verify that the valve can be manually cycled using its hand-wheel.	Once every 48 months.
D	2-SI-432 2-SI-444 LPSI Suction from RWST AOV	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling. No additional surveillance testing needs to be performed.	Once every 48 months.
E	2-SI-440 2-SI-441 LPSI/SDC Suction from RCS AOV	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling. No additional surveillance testing needs to be performed.	Once every 48 months.
F	2-SI-452 2-SI-453 LPSI disch to SDC HX AOV	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling. No additional surveillance testing needs to be performed.	Once every 48 months.
G	2-SI-456 2-SI-457 SDC HX outlet AOV	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling. No additional surveillance testing needs to be performed.	Once every 48 months.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: None

TECHNICAL REQUIREMENT

7.1.11 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.11-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.11 The capability of equipment listed in TRM Table 7.1.11-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.11-2.

Table 7.1.11-1
ECCS - RWST Outlet Valve
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
A	2-CS-13.1B RWST Stop (outlet) MOV	Valve must be capable of being opened by local manual handwheel operation.	With the valve nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, and R-2.	With valve nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-1G, A-12A and A-24)

Table 7.1.11-2
ECCS - RWST Outlet Valve
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-CS-13.1B RWST Stop (outlet) MOV	Verify that the valve can be manually cycled using its hand-wheel.	In accordance with the MOV surveillance frequency established by the PM Program.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL REQUIREMENT: 3.1.2.2.a Reactivity Control System Flow Paths - Operating

TECHNICAL REQUIREMENT

7.1.12 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.12-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.12 The capability of equipment listed in TRM Table 7.1.12-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.12-1.

Table 7.1.12-1
ECCS - Boric Acid Gravity Feed Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-CH-508 2-CH-509, Boric Acid Gravity Feed MOVs	Both valves must be capable of being cycled by local manual hand-wheel operation.	With one valve nonfunctional, establish minimum TECHNICAL REQUIREMENT (3.1.2.7) inventory in the Boric Acid Storage Tank (BAST) associated with the FUNCTIONAL valve, or perform ACTIONS b.1, b.2 for fire areas R-1, R-7, R-9, R-13, R-14, R-16, and R-17.

Table 7.1.12-2
ECCS - Boric Acid Gravity Feed Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-CH-508 2-CH-509, Boric Acid Gravity Feed MOVs	Verify that both valves can be manually cycled using their hand-wheels.	In accordance with the MOV surveillance frequency established by the PM Program.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.6.1.5 Containment Air Temperature

TECHNICAL REQUIREMENT

7.1.13 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.13-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.13 The capability of equipment listed in TRM Table 7.1.13-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.13-2.

Table 7.1.13-1
Containment Systems - Air Temperature Indication
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	TE-8108 Containment Ambient Temperature Indication	Instrument loop must be FUNCTIONAL from the Control Room.	With instrument loop nonfunctional from Control Room, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-4, R-5, R-6, R-8, R-10, R-11, and R-12.

Table 7.1.13-2
Containment Systems - Air Temperature Indication
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	TE-8108 Containment Ambient Temperature Indication	Perform CHANNEL CALIBRATION to verify the instrument loop is calibrated.	Once per REFUELING cycle

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.6.2.1 Containment Spray and Cooling Systems

TECHNICAL REQUIREMENT

7.1.14 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.14-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.14 The capability of equipment listed in TRM Table 7.1.14-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.14-2.

Table 7.1.14-1
Containment Systems - Containment Spray and Cooling Systems
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-CS-4A 2-CS-4B Containment Spray Stop Valves	Each valve must be capable of being closed by local manual hand-wheel operation.	With valve 2-CS-4A nonfunctional, perform ACTIONS b.1, b.2 for fire area R-2. With valve 2-CS-4B nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, and R-17.

Table 7.1.14-2
Containment Systems - Containment Spray and Cooling Systems
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-CS-4A 2-CS-4B Containment Spray Stop Valves	Verify that each valve can be manually cycled using its hand-wheel. Valves are manually cycled when entering shutdown cooling, and securing from shutdown cooling during plant heat-up. No additional surveillance testing needs to be done.	Once every 48 months.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.1.2 Auxiliary Feedwater Pumps

TECHNICAL REQUIREMENT

7.1.15 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.15-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.15 The capability of equipment listed in TRM Table 7.1.15-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.15-2.

Table 7.1.15-1
Plant System - Auxiliary Feedwater Pumps
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
A	SV-4188 AFW Pump Turbine Stm Supply Valve	Valve must be capable of being operated from Panel C10.	With valve nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire areas R-1 and R-13.	With valve nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-24), R-3 (FHA T-1A)
B	P4 & H-21 AFW Pump and Turbine Governor Control	1. Pump/Turbine must be FUNCTIONAL from the Control Room. 2. Pump/Turbine must be FUNCTIONAL from Panel C10.	With Pump/Turbine nonfunctional, from either location, perform ACTIONS b.1, b.2 for fire areas R-3, R-11, R-16, and R-17.	With Pump/Turbine nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-24), R-3 (FHA T-1A)
C	2-MS-202 AFW Steam Supply from SG 2, MOV	Valve must be open, incapable of remote operation with disconnect switch opened ("off" position).	With the valve closed and/or disconnect closed, perform ACTIONS b.1, b.2 for fire areas R-1, R-2, and R-13.	N/A
D	P-9A P-9B Motor Driven AFW pumps	Pumps must be FUNCTIONAL from the Control Room.	1. With P-9A nonfunctional: a) Perform ACTIONS b.1, b.2 for fire areas R-2 and R-10 b) Verify P4 FUNCTIONAL, or Perform ACTIONS b.1, b.2 for fire area R-8 c) Verify P9B FUNCTIONAL, or Perform ACTIONS b.1, b.2 for fire area R-12. 2. With P-9B nonfunctional, verify P4 FUNCTIONAL, or Perform ACTIONS b.1, b.2 for fire areas R-7 and R-14.	1. With P-9A nonfunctional Risk Mitigation Areas are: R-2 (FHA A-8D, T-8, T-10) 2. With P-9B nonfunctional Risk Mitigation Areas are: R-14 (FHA T-7, T-9)

Table 7.1.15-1
Plant System - Auxiliary Feedwater Pumps
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
E	2-FW-44 AFW Pump P4 Discharge Iso MOV	Valve must be capable of being opened by local manual handwheel operation.	With valve nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1 and R-13.	N/A
F	2-FW-43A 2-FW-43B AFW Flow Control (AOVs)	1. Each valve must be capable of local manual handwheel operation, and failing open when de-energized. 2. Valve 2-FW-43B must be capable of remote operation from Panel C10.	1. Verify that the applicable AFW regulating valve bypass valves 2-FW-56A and/or 2-FW-56B, and isolation valves 2-FW-10A/10B and/or 2-FW-11A/B can be manually cycled. 2. Verify that the AFW regulating valve bypass valve 2-FW-56B, and isolation valves 2-FW-10B or 2-FW-11B can be manually cycled.	N/A

Table 7.1.15-2
Plant System - Auxiliary Feedwater Pumps
Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	SV-4188 AFW Pump Turbine Stm Supply Valve	Verify that control of the steam supply valve SV-4188 can be isolated from the Control Room and operated from Panel C10.	Once per REFUELING cycle.
B	P4 & H-21 AFW Pump and Turbine Governor Control	1. In accordance with Technical Specification 4.7.1.2.a.1 and 4.7.1.2.a.2.b. 2. In accordance with Technical Specification 4.7.1.2.a.2.b.	1. In accordance with Technical Specification 4.7.1.2.a. 2. In accordance with Technical Specification 4.7.1.2.a.
C	2-MS-202 AFW Steam Supply from SG 2, MOV	Administrative control over the disconnect switch key found in procedure SP-2610C, and plant heatup procedure.	N/A
D	P-9A P-9B Motor Driven AFW pumps	In accordance with Technical Specification 4.7.1.2.a.1 and 4.7.1.2.a.2.a for each motor driven pump.	In accordance with Technical Specification 4.7.1.2.a.
E	2-FW-44 AFW Pump P4 Discharge Iso MOV.	Verify that the valve can be manually cycled using its handwheel.	In accordance with the MOV surveillance frequency established by the PM Program.
F	2-FW-43A 2-FW-43B AFW Flow Control (AOVs)	1.a. Verify that each valve can be manually cycled using its handwheel, and 1.b. That each valve will fail open on loss of control power. 2. Verify that 2-FW-43B can be isolated from the Control Room and operated from Panel C10.	1. Once every 48 months. 2. Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.1.3 Condensate Storage Tank

TECHNICAL REQUIREMENT

7.1.16 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.16-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.16 The capability of equipment listed in TRM Table 7.1.16-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.16-2.

Table 7.1.16-1
Condensate Storage Tank (CST) Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	LT-5282 CST Level Indication	1. Instrument loop must be FUNCTIONAL at Panel C10. 2. Instrument loop must be FUNCTIONAL from the Control Room.	1. With loop nonfunctional from Panel C10, perform ACTIONS b.1, b.2 for fire areas R-1 & R-13. 2. With loop nonfunctional from Control Room, verify that the local CST level indicator (LIS-5489) is FUNCTIONAL, or perform ACTIONS b.1, b.2 for all fire areas.
B	LIS-5489 CST local Level Indicator	Local CST Level Indication must be available.	With local indication nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, and R-10.

Table 7.1.16-2
Condensate Storage Tank Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	LT-5282 CST Level Indication	Perform CHANNEL CALIBRATION to verify the instrument loop is calibrated.	Once per REFUELING cycle.
B	LIS-5489 CST local Level Indication	Perform Instrument Calibration to verify the instrument is calibrated.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.1.5 Main Steam Isolation Valves

TECHNICAL REQUIREMENT

7.1.17 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.17-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.17 The capability of equipment listed in TRM Table 7.1.17-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.17-2.

Table 7.1.17-1
Main Steam Isolation Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-MS-64A 2-MS-64B Main Steam Iso Valves (AOVs)	2-MS-64A must be capable of closure from Bottle-up Panel C70A. 2-MS-64B must be capable of closure from Bottle-up Panel C70B.	With one valve nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-2, and R-17.
B	2-MS-65A SG MSIV Bypass Valve (MOV)	Valve must be closed with its disconnect switch locked open.	With 2-MS-65A open or disconnect closed, perform ACTIONS b.1, b.2 for fire areas R-1 and R-17.
C	2-MS-65B SG MSIV Bypass Valve (MOV)	Valve must be closed with its disconnect switch locked open.	With 2-MS-65B open or disconnect closed, perform ACTIONS b.1, b.2 for fire areas R-1 and R-2.

Table 7.1.17-2
Main Steam Isolation Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-MS-64A 2-MS-64B Main Steam Iso Valves (AOVs)	1. Verify that 2-MS-64A can be closed from Bottle-up Panel C70A. 2. Verify that 2-MS-64B can be closed from Bottle-up Panel C70B.	Once per REFUELING cycle.
B	2-MS-65A SG MSIV Bypass Valve (MOV)	Administrative control for disconnect switch key provided in plant start-up procedure.	N/A
C	2-MS-65B SG MSIV Bypass Valve (MOV)	Administrative control for the opening coil provided in plant start-up procedure.	N/A

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.1.7 Atmospheric Dump Valves

TECHNICAL REQUIREMENT

7.1.18 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.18-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.18 The capability of equipment listed in TRM Table 7.1.18-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.18-2.

Table 7.1.18-1
Atmospheric Dump Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-MS-190A 2-MS-190B Atmospheric Steam Dump Valves (AOVs)	1. Each valve must be capable of failing closed when the valves' solenoid is deenergized. 2.a. Valve 2-MS-190A must be capable of closure from Panel C70A, and, 2.b. Valve 2-MS-190B must be capable of closure from Panel C70B. 3. Each valve must be capable of local manual handwheel operation. 4. Valve 2-MS-190B must be capable of remote operation from panel C10.	1. Failure to close with solenoid deenergized requires isolation of instrument air to the affected valve, for fire area R-11. 2.a. With 2-MS-190A nonfunctional from C70A perform ACTIONS b.1, b.2 for fire areas R-1, R-13 and R-17. 2.b. With 2-MS-190B nonfunctional from C70B, perform ACTIONS b.1, b.2 for fire areas R-1, and R-2. 3. With one valve nonfunctional, perform ACTIONS b.1, b.2 for **all** fire areas. 4. With loss of remote operation from Panel C10 for 2-MS-190B, ensure local manual operation is available.

Table 7.1.18-2
Atmospheric Dump Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-MS-190A 2-MS-190B Atmospheric Steam Dump Valves (AOVs)	1. From the Control room verify each valve will close when its quick open solenoid is de-energized and its valve positioning instrument loop is FUNCTIONAL and set to “close”. 2.a. Verify that valve 2-MS-190A can be closed at Panel C70A. 2.b. Verify that valve 2-MS-190B can be closed at Panel C70B 3. Addressed under Tech. Spec. 4.7.1.7. 4. Verify that 2-MS-190B can be operated from Panel C10.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.1.8 Steam Generator Blowdown Isolation Valves

TECHNICAL REQUIREMENT

7.1.19 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.19-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.19 The capability of equipment listed in TRM Table 7.1.19-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.19-2.

Table 7.1.19-1
Steam Generator Blowdown Isolation Valves
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	2-MS-220A 2-MS-220B SG Blowdown Isolation AOVs	1.b. Valve 2-MS-220A must be capable of closure from Panel C70A, and 1.c. Valve 2-MS-220B must be capable of closure from Panel C70B. 2. Each valve must be capable of failing closed when the valves' solenoid is deenergized.	1. With one valve nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, and R-17. 2. With one valve nonfunctional, perform ACTIONS b.1, b.2 for fire area R-11.

Table 7.1.19-2
Steam Generator Blowdown Isolation Valves
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-MS-220A 2-MS-220B (SG Blowdown Isolation AOVs)	1.b. Verify that valve 2-MS-220A can be closed at Panel C70A. 1.c. Verify that valve 2-MS-220B can be closed at Panel C70B. 2. Verify that each valve will fail closed on loss of control power.	Once per REFUELING cycle.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.3.1 RBCCW System

TECHNICAL REQUIREMENT

7.1.20 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.20-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.20 The capability of equipment listed in TRM Table 7.1.20-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.20-2.

Table 7.1.20-1
Reactor Building Closed Cooling Water (RBCCW) System
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
A	2-RB-13.1A 2-RB-13.1B Shutdown HX Disch Header AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	With valve 2-RB-13.1A nonfunctional, verify that 2-RB-14A is FUNCTIONAL, or perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-6, R-8, R-10, and R-11. With valve 2-RB-13.1B nonfunctional, verify that 2-RB-14B is FUNCTIONAL, or perform ACTIONS listed for fire areas R-1, R-5, R-7, R-9, R-13, R-14, R-16, and R-17. With valves 2-RB-13.1A and 2-RB-13.1B nonfunctional, verify that valves 2-RB-14A and 2-RB-14B are FUNCTIONAL, or perform ACTIONS listed for all fire areas.	N/A
B	2-RB-211A 2-RB-211B RBCCW Pump A inlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A

Table 7.1.20-1
Reactor Building Closed Cooling Water (RBCCW) System
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
C	2-RB-211C 2-RB-211D RBCCW Pump B inlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A
D	2-RB-211E 2-RB-211F RBCCW Pump C inlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A

Table 7.1.20-1
Reactor Building Closed Cooling Water (RBCCW) System
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
E	2-RB-4.1A 2-RB-4.1B RBCCW HX A Outlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A
F	2-RB-4.1C 2-RB-4.1D RBCCW HX B Outlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A

Table 7.1.20-1
Reactor Building Closed Cooling Water (RBCCW) System
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
G	2-RB-4.1E 2-RB-4.1F RBCCW HX C Outlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX 'A', or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX 'B', or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A
H	P11A P11B P11C RBCCW Pumps	Each RBCCW Pump must be FUNCTIONAL from the control room.	With P11A nonfunctional, perform ACTIONS listed for fire areas R-2 and R-3. With P11B nonfunctional, perform ACTIONS listed for fire area R-11. With P11C nonfunctional, perform ACTIONS listed for fire areas R-3 and R-14.	With P11B supplying A train, Risk Mitigation Areas are: R-2 (FHA T-8, T-10), R-3 (FHA T-1A, T-1C), R-16 (FHA I-1A, I-1C). With P11B supplying B train, Risk Mitigation Areas are: R-11 (FHA A-28), R-14 (FHA T-7).

Table 7.1.20-1
Reactor Building Closed Cooling Water (RBCCW) System
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
I	2-RB-251A 2-RB-251B RBCCW Pump Disch. Cross-connect AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A to SDC HX ‘A’, or perform ACTIONS listed for Fire Areas R-2, and R-3. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11A or P-11B to SDC HX ‘A’, or perform ACTIONS listed for Fire Areas R-8, R-10, and R-11. Verify that RBCCW System can be manually aligned to provide cooling flow from P-11B or P-11C to SDC HX ‘B’, or perform ACTIONS listed for Fire Areas R-7, R-9, R-13, R-14, R-16, and R-17.	N/A

Table 7.1.20-2
Reactor Building Closed Cooling Water (RBCCW) System Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-RB-13.1A 2-RB-13.1B Shutdown HX Disch Header AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
B	2-RB-211A 2-RB-211B RBCCW Pump A inlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
C	2-RB-211C 2-RB-211D RBCCW Pump B inlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
D	2-RB-211E 2-RB-211F RBCCW Pump C inlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
E	2-RB-4.1A 2-RB-4.1B RBCCW HX A Outlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
F	2-RB-4.1C 2-RB-4.1D RBCCW HX B Outlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
G	2-RB-4.1E 2-RB-4.1F RBCCW HX C Outlet AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.
H	P11A P11B P11C RBCCW Pumps	In accordance with Technical Specification Surveillance Requirement 4.0.5.	In accordance with Technical Specification Surveillance Requirement 4.0.5
I	2-RB-251A 2-RB-251B RBCCW Pump Disch. Cross-connect AOVs	Verify that each valve can be manually cycled using is handwheel.	Once every 48 months.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.7.4.1 Service Water System

TECHNICAL REQUIREMENT

7.1.21 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.21-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.21 The capability of equipment listed in TRM Table 7.1.21-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.21-2.

Table 7.1.21-1
Service Water System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
A	P5A & L1A P5C & L1C SW Pumps & Strainers	Each Service Water Pump and its associated Strainer must be FUNCTIONAL from the Control Room.	With P5A or its Strainer L1A nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-8, R-10, and R-11. With P5C or its Strainer L1C nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-7, R-9, R-13, R-14, R-16 and R-17.	With P5B supplying A train, Risk Mitigation Areas are: R-2 (FHA T-8,T-10), R-3 (FHA T-1A, T-1C) With P5B supplying B train, Risk Mitigation Areas are: R-11 (FHA A-28) R-14 (FHA T-7)
B	2-SW-97A 2-SW-97B SW Pump Discharge Hdr Crossconn. AOVs	1. Each Valve must be capable of being cycled by local manual handwheel operation. 2. Valve 2-SW-97A must be maintained in the closed position.	1. With one or both valve(s) nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-3 and R-16. 2. When 2-SW-97A is open remove the instrument air source from 2-SW-97B so it cannot spuriously open due to a fire or perform ACTIONS b.1, b.2 for fire area R-2.	N/A

Table 7.1.21-1
Service Water System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Compensatory Measures if Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
C	2-SW-2A 2-SW-2B 2-SW-2C SW Pump Disch Valve	Each Valve must be capable of being cycled by local manual handwheel operation.	With 2-SW-2A nonfunctional, align SW pump P5B to cool Train A, or perform ACTIONS b.1, b.2 for fire areas R-3 and R-11. With 2-SW-2B nonfunctional, align SW pump P5A to cool Train A and P5C to cool Train B, or perform ACTIONS b.1, b.2 for fire areas R-1, R-3, R-11, and R-16. With 2-SW-2C nonfunctional, align SW pump P5B to cool Train B, or perform ACTIONS b.1, b.2 for fire areas R-1 and R-16.	N/A
D	2-SW-3.1A 2-SW-3.1B RBCCW HX Inlet AOVs	Each Valve must be capable of being cycled by local manual handwheel operation.	With 2-SW-3.1A nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1 and R-14. With 2-SW-3.1B nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1 and R-2.	N/A

Table 7.1.21-1
Service Water System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see ACTION "d")
E	2-SW-9A 2-SW-9B 2-SW-9C RBCW HX Outlet Valves	Each Valve must be capable of being closed by local manual handwheel operation.	With valves 2-SW-9A and 2-SW-9B nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3, R-8, R-10, and R-11. With valves 2-SW-9B and 2-SW-9C nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-7, R-9, R-13, R-14, R-16, and R-17. With all valves nonfunctional, perform ACTIONS b.1, b.2 for all fire areas.	N/A
F	2-SW-5A 2-SW-5B 2-SW-5C TBCCW HX A, B & C Inlet Valves	Each Valve must be capable of being closed by local manual handwheel operation.	With a valve nonfunctional, perform ACTIONS b.1, b.2 for all fire areas except R-12.	N/A
G	2-SW-12B 2-SW-12C SW Isolation to EDG A and EDG B	Valve must be capable of being closed by local manual handwheel operation.	With 2-SW-12B nonfunctional, perform ACTIONS b.1, b.2 for fire area R-3 and R-14. With 2-SW-12C nonfunctional, perform ACTIONS b.1, b.2 for fire area R-2.	N/A

Table 7.1.21-1
Service Water System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures	
			Compensatory Measures if Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
H	2-SW-296 Fire Water Cross-Tie to “A” EDG Service Water Header	Valve must be capable of being cycled by local manual handwheel operation.	With 2-SW-296 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-3.	N/A

Table 7.1.21-2
Service Water System Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	P5A & L1A P5C & L1C S.W. Pumps & Strainers	In accordance with Technical Specification Surveillance Requirement 4.0.5.	In accordance with Technical Specification Surveillance Requirement 4.0.5.
B	2-SW-97A 2-SW-97B SW Pump Discharge Hdr Cross-conn. AOVs	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
C	2-SW-2A 2-SW-2B 2-SW-2C SW Pump Disch Valves.	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
D	2-SW-3.1A 2-SW-3.1B RBCCW HX Inlet AOVs	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
E	2-SW-9A 2-SW-9B 2-SW-9C RBCW HX Outlet Valves	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
F	2-SW-5A 2-SW-5B 2-SW-5C TBCCW HX A, B & C Inlet Valves	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
G	2-SW-12B 2-SW-12C SW Isolation to EDG A and EDG B	Verify that each valve can be manually cycled using its handwheel.	Once every 48 months.
H	2-SW-296 Drain on SW to EDG B	Verify valve can be manually cycled using its handwheel.	Once every 48 months.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.8.1.1 Electrical Power Systems AC Sources

TECHNICAL REQUIREMENT

7.1.22 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.22-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.22 The capability of equipment listed in TRM Table 7.1.22-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.22-2.

Table 7.1.22-1
Electrical Power, AC Sources
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	Station Blackout (SBO) Alternate AC Diesel Gen.	The Unit 3 Alternate AC (Station Blackout) Diesel Generator must be capable of powering Unit 2 Bus 24E.	With the Unit 3 Alternate AC (SBO) Diesel Generator nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-11, and R-16.

Table 7.1.22-2
Electrical Power, AC Sources
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	Station Blackout (SBO) Alternate AC Diesel Gen.	1. Verify availability of Unit 3 Alternate AC (SBO) Diesel Generator. 2. Verify that breakers required to align electric power from Unit 3 SBO Diesel Generator to Unit 2 can be manually cycled.	1. Once per REFUELING cycle. 2. In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.8.2.1 Onsite AC PWR Distribution

TECHNICAL REQUIREMENT

7.1.23 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.23-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.23 The capability of equipment listed in TRM Table 7.1.23-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.23-2.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
A	Bus 24C (A3) 4160 V Switchgear Facility Z1	All 4160 V breakers must be capable of being cycled using local manual controls.	1. With any local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-11; and in addition if: 2. With breaker A302 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1 and R-9; 3. With breaker A303 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-13; 4. With breaker A304 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1; 5. With breaker A305 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-2, R-3, and R-16; 6. With breaker A307 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-13; 7. With breaker A310 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-7, 9, 10, 13, 16 and R-17; 8. With breaker A312 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
B	Bus 24D (A4) 4160 V Switchgear Facility Z2	All 4160 V breakers must be capable of being cycled using local manual controls.	1. With any local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1, and R-16; and in addition if: 2. With breaker A401 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-14; 3. With breaker A408 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-14; 4. With breaker A410 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-14; 5. With breaker A405 nonfunctional, perform ACTIONS b.1, b. 2 for fire areas R-7, R-9, R-10, 13, 14, and R-17; 6. With breaker A406 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-13; 7. With breaker A409 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-11; 8. With breaker A411 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-10, and R-14.
C	Bus 24E (A5) 4160 V Switchgear Facility Z5	All 4160 V breakers must be capable of being cycled using local manual controls.	With any local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1, R-11, and R-16.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
D	Bus 24A (A1) 4160 V Switchgear Facility 1	4160 V feeder breakers A102, A103, and A104 must be capable of being cycled using local manual controls.	1. With breaker A102 local control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1. 2. With breaker A103 local control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-3 and R-13. 3. With breaker A104 local control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-11 and R-13.
E	Bus 24B (A2) 4160 V Switchgear Facility 2	4160 V feeder breakers A204, A205, and A206 must be capable of being cycled using local manual controls.	1. With breaker A204 local control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-13. 2. With breaker A205 local control nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-11 and R-13. 3. With breaker A206 local control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1.
F	Bus 22A (B01) 480 V load center	MCC feeder breaker B0105 must be capable of being tripped using local manual controls.	With breaker B0105 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3 and R-16.
G	Bus 22B (B02) 480 V load center	MCC feeder breaker B0205 must be capable of being tripped using local manual controls.	With breaker B0205 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3 and R-16.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
H	Bus 22E (B05) 480 V Switchgear Facility Z1	Breakers B0504 and B0505 must be capable of being cycled using local manual controls.	With any one of these breakers nonfunctional, perform ACTIONS b.1, b.2 for fire area R-11.
I	Bus 22F (B06) 480 V Switchgear Facility Z2	Breakers B0603, B0605, B0606, B0608, B0609, and B0611 must be capable of being cycled using local manual controls.	With any one of these breakers nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1 and R-16.
J	MCC B51 (MCC-22- 1E)	Breakers B5105, B5110, B5111, B5112, B5118, B5119, and B5145 must be capable of being cycled using local manual controls.	1. With breaker B5105 nonfunctional perform ACTIONS b.1, b.2 fire areas R-2 and R-4. 2. With breaker B5110, B5111, or B5112 nonfunctional perform ACTIONS b.1, b.2 fire area R-15. 3. With breaker B5118, or B5119 nonfunctional perform ACTIONS b.1, b.2 fire area R-2. 4. With breaker B5145 nonfunctional perform ACTIONS b.1, b.2 fire area R-4.
K	MCC B52 (MCC-22- 2E)	Breaker B5222 must be capable of being cycled using local manual controls.	With breaker B5222 nonfunctional perform ACTIONS b.1, b.2 fire areas R-2,R-3, and R-16.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
L	MCC B61 (MCC-22- 1F)	Breakers B6105, B6109, B6110, and B6119 must be capable of being cycled using local manual controls.	1. With breaker B6105 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-4. 2. With breaker B6109, B6110 or B6119 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-15 and R-17.
M	MCC B62 (MCC-22- 2F)	Breakers B6203 and B6226 must be capable of being cycled using local manual controls.	1. With breaker B6203 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-13. 2. With breaker B6226 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2, R-3 and R-16.
N	VA 20 120VAC Instrument Panel VIAC-2	All 120 VAC breakers must be capable of being cycled using local manual operation.	1. With any local breaker control nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1, and if 2. Breaker VA2004 is nonfunctional, perform LCO 3.7.13.23 ACTION Statement for fire area R-2.

Table 7.1.23-1
Onsite Power - AC Distribution System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
R	Emergency Lighting Units (ELUs)	All ELUs must be capable of at least 8 hours of illumination.	With any ELUs nonfunctional, replace defective ELUs with battery powered lanterns, and verify proper placement on a daily basis.

Table 7.1.23-2
Onsite Power - AC Distribution System Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	Bus 24C (A3) 4160 V Switchgear Facility Z1	Verify that each breaker can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
B	Bus 24D (A4) 4160 V Switchgear Facility Z2	Verify that each breaker can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
C	Bus 24E (A5) 4160 V Switchgear Facility Z5	Verify that each breaker can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
D	Bus 24A (A1) 4160 V Switchgear Facility 1	Verify that breakers A102, A103, and A104 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
E	Bus 24B (A2) 4160 V Switchgear Facility 2	Verify that breakers A204, A205, and A206 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
F	Bus 22A (B01) 480 V load center	Verify that breaker B0105 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
G	Bus 22B (B02) 480 V load center	Verify that breaker B0205 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
H	Bus 22E (B05) 480 V Switchgear Facility Z1	Verify that breakers B0504 and B0505 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).

Table 7.1.23-2
Onsite Power - AC Distribution System Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
I	Bus 22F (B06) 480 V Switchgear Facility Z2	Verify that breakers B0603, B0605, B0606, B0608, B0609, and B0611 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
J	MCC B51 (MCC-22-1E)	Verify that Breakers B5105, B5110, B5111, B5112, B5118, B5119, and B5145 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
K	MCC B52 (MCC-22-2E)	Verify that breakers that supply Appendix R loads are capable of being manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
L	MCC B61 (MCC-22-1F)	Verify that breakers B6105, B6109, B6110, and B6119 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
M	MCC B62 (MCC-22-2F)	Verify that breakers that supply Appendix R loads are capable of being manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
N	VA 20 120VAC Instrument Panel VIAC-2	Verify that breakers are capable of being manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
R	Emergency Lighting Units (ELUs)	Verify capability of ELUs to provide at least 8 hours of operation.	In accordance with the Emergency Lighting Unit (ELU) surveillance frequency established by the Preventive Maintenance Program.

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.8.2.3 DC Power Distribution

TECHNICAL REQUIREMENT

7.1.24 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.24-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.24 The capability of equipment listed in TRM Table 7.1.24-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.24-2.

Table 7.1.24-1
Electric Power - D.C. Distribution Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
B	D01 BUS 201A 125 VDC Switchgear	Breakers D0103, D0104, D0105, D0108, and D0111 must be capable of being cycled using local manual operation.	1. With breaker D0103 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1. 2. With breaker D0104 or D0111 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R- 3, R-8 and R-11. 3. With breaker D0105 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-3, R-13, R-14 and R-17. 4. With breaker D0108 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-15.
C	D02 BUS 201B 125 VDC Switchgear	All breakers on this panel must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1. 2. With breaker D0204 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-3, R-8 and R-11. 3. With breaker D0205 nonfunctional, perform ACTIONS b.1, b.2 for fire area R-15. 4. With breaker D0208 nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2 and R-8.

Table 7.1.24-1
Electric Power - D.C. Distribution Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
D	DV10 Panel 201A-1V 125 VDC Vital Instrument Panel.	All 125 VDC breakers on Panel DV10 must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional perform ACTIONS b.1, b.2 for fire area R-15, and in addition if: 2. Breakers DV1011, DV1015, DV1017, or DV1020 are nonfunctional perform ACTIONS b.1, b.2 for fire area R-17; 3. Breaker DV1008 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-3; 4. Breaker DV1013 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-4 and R-17; 5. Breaker DV1014 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-2, 4, 5, 6, 8, 10, 11 and R-17; 6. Breaker DV1018 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-2, 3, 11, 13 and R-14; 7. Breaker DV1019 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-13; 8. Breaker DV1008 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-3, 13 and R-14.

Table 7.1.24-1
Electric Power - D.C. Distribution Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
E	DV20 Panel 201B-1V 125 VDC Vital Instrument Panel.	All 125 VDC breakers on Panel DV20 must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional perform ACTIONS b.1, b.2 for fire areas R-1 and R-15, and in addition if: 2. Breaker DV2014 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-4, 5, 6, 7, 9, 13, 14, 16 and R-17; 3. Breaker DV2015 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-8; 4. Breaker DV2018 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-3, 11 and R- 14; 5. Breaker DV2020 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-8; 6. Breaker DV2021 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-13 and R- 14.
F	D11 Panel 201A-1 125 VDC Distribution	All 125 VDC breakers on Panel D11must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional perform ACTIONS b.1, b.2 for fire area R-15, and in addition if: 2. Breaker D1104 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-11.
G	D12 Panel 201A-2 125 VDC Distribution	All 125 VDC breakers on Panel D12 must be capable of being cycled using local manual operation.	With any local breaker control nonfunctional perform ACTIONS b.1, b.2 for fire area R-15.

Table 7.1.24-1
Electric Power - D.C. Distribution Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")
H	D21 Panel 201B-1 125 VDC Distribution	All 125 VDC breakers on Panel D21 must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional perform ACTIONS b.1, b.2 for fire area R-15, and in addition if: 2. Breaker D2104 is nonfunctional perform ACTIONS b.1, b.2 for fire areas R-3 and R-11.
I	D22 Panel 201B-2 125 VDC Distribution	All 125 VDC breakers on Panel D22 must be capable of being cycled using local manual operation.	1. With any breaker nonfunctional perform ACTIONS b.1, b.2 for fire area R-15, and in addition if: 2. Breaker D2216 is nonfunctional perform ACTIONS b.1, b.2 for fire area R-2.

Table 7.1.24-2
Electric Power - D.C. Distribution Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
B	D01 BUS 201A 125 VDC Switchgear	Verify that breakers D0103, D0104, D0105, D0108, and D0111 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
C	D02 BUS 201B 125 VDC Switchgear	Verify that breakers D0204, D0205, and D0208 and D0211 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
D	DV10 Panel 201A-1V 125 VDC Vital Instrument Panel.	Verify that all breakers on Panel DV10 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
E	DV20 Panel 201B-1V 125 VDC Vital Instrument Panel.	Verify that all breakers on Panel DV20 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
F	D11 Panel 201A-1 125 VDC Distribution	Verify that all breakers on Panel D11 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
G	D12 Panel 201A-2 125 VDC Distribution	Verify that all breakers on Panel D12 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
H	D21 Panel 201B-1 125 VDC Distribution	Verify that all breakers on Panel D21 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).
I	D22 Panel 201B-2 125 VDC Distribution	Verify that all breakers on Panel D22 can be manually cycled.	In accordance with the breaker surveillance frequency established by the Preventive Maintenance Program (Refer to TRM Appendix R Bases, item No. 9).

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: None

TECHNICAL REQUIREMENT

7.1.25 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.25-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.25 The capability of equipment listed in TRM Table 7.1.25-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.25-2.

Table 7.1.25-1
Fire Protection Miscellaneous System Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Risk Significant Component Not Restored in 72 Hours (see Action "d")
A	2-FIRE-258 2-FIRE-259 2-FIRE-559 2-FIRE-560 Fire Prot. Cross-tie to EDG SW Cooling	1. Valve 2-FIRE-258 must be capable of being opened, and; 2. Valves 2-FIRE-259, 2-FIRE-559, and 2-FIRE-560 must be capable of being closed, by local manual handwheel operation.	With any valve nonfunctional, perform ACTIONS b.1, b.2 for fire area R-3. Evaluation for continued operation.	N/A
B	Fire Suppression Systems	Refer to the following TRs: 3.7.9.1 Fire Suppression Water 3.7.9.2 Spray/Sprinkler Systems 3.7.9.3 Fire Hose Stations 3.7.9.4 Halon System	As specified in the ACTION Statements for the following: TR 3.7.9.1 TR 3.7.9.2 TR 3.7.9.3 TR 3.7.9.4	N/A
C	2-FIRE-34 Cross-tie to AFW suction	Valve 2-FIRE-34 must be capable of being opened.	N/A	With 2-FIRE-34 nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-1A, A-1B, A-1G, A-12A and A-24), R-3 (FHA T-1A), R-11 (FHA A-28 and R-16 (FHA I-1A and I-1B)

Table 7.1.25-2
Fire Protection Miscellaneous System Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	2-FIRE-258 2-FIRE-259 2-FIRE-559 2-FIRE-560 Fire Prot. Crosstie to EDG SW Cooling	Verify that each valve can be manually cycled by local manual handwheel operation.	Once every 48 months.
B	Fire Suppression Systems	Refer to the following TSRs: 4.7.9.1 Fire Suppression Water 4.7.9.2 Spray/Sprinkler Systems 4.7.9.3 Fire Hose Stations 4.7.9.4 Halon System	As specified in the TSRs for the following: 4.7.9.1 Fire Suppression Water 4.7.9.2 Spray/Sprinkler Systems 4.7.9.3 Fire Hose Stations 4.7.9.4 Halon System
C	2-FIRE-34 Cross-tie to AFW suction	N/A	N/A

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: None

TECHNICAL REQUIREMENT

7.1.26 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.26-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.26 The capability of equipment listed in TRM Table 7.1.26-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements shall be verified per the TECHNICAL SURVEILLANCE REQUIREMENTS of TRM Table 7.1.26-2.

Table 7.1.26-1
Support Equipment, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Compensatory Measures for Components Requiring Prompt Action (see ACTION "f")
A	Hand-held Radios	1. At least eight (8) hand-held radios must be FUNCTIONAL. Three (3) in the Control Room, and five (5) in the Fire Shutdown Storage Box. 2. Communications from manual action locations to Control Room base station (using Remote Desktop Controller and hand-held radios) must be FUNCTIONAL. Note: If Main Radio Control Station located in MPS1 Control Room is nonfunctional than Remote Desktop Controller in MPS2 Control Room will also be nonfunctional. 3. Communications from manual action locations to dedicated Control Room portable radio unit (with hardwired external antenna) must be FUNCTIONAL. 4. Primary or Secondary (800 MHz) Repeaters must be FUNCTIONAL.	1. With less than 8 hand-held radios FUNCTIONAL, replace or restore radios to FUNCTIONALITY within 7 days or perform ACTIONS b.1, b.2 for all fire areas EXCEPT R-12. 2. With Remote Desktop Controller nonfunctional, perform ACTIONS b.1, b.2 for all fire areas EXCEPT R-3, R-11, R-12, and R-16. 3. With dedicated portable radio nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-3, R-11, and R-16. 4. If the primary repeater is nonfunctional, verify that the Secondary Repeater is FUNCTIONAL, or perform ACTIONS b.1, b.2 for all fire areas. 5. If the Secondary repeater is nonfunctional, verify that the Primary Repeater is FUNCTIONAL, and perform ACTIONS b.1, b.2 for fire area R-3. 6. If neither repeater is FUNCTIONAL, immediately attempt to restore one repeater to operation, and perform ACTIONS b.1, b.2 for all fire areas.	N/A

Table 7.1.26-1
Support Equipment, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Compensatory Measures for Components Requiring Prompt Action (see ACTION "f")
B	HOT SHUTDOWN (HSD) Storage Box Inventory	All items on inventory must be available to support COLD SHUTDOWN for fire area R-2. Refer to applicable station procedure for inventory.	With any inventory item missing, perform ACTIONS b.1, b.2 for fire area R- 2.	N/A
C	Fire Shutdown (FSD) Storage Box Inventory	All items on inventory must be available to support COLD SHUTDOWN for fire area(s) where required. Refer to applicable station procedure for inventory.	With any inventory item missing, perform ACTIONS b.1, b.2 for fire area(s) where required.	N/A
D	COLD SHUTDOWN Repair Material	All materials required to support COLD SHUTDOWN repairs must be available in the Unit 2 Appendix R storage area. Repair Material shall not be utilized as plant spare material. Refer to applicable station procedure for materials required.	With any inventory item missing or non-functional perform ACTION b.1 for fire areas where item is required. This ACTION is in addition to the Prompt ACTION b.2 taken. See applicable station procedure for fire area identification. Note: Some items are shared with Unit 3. Notify the Shift Manager of any impacts.	Within 12 hours of any inventory item identified as missing or non-functional, perform ACTION b.2 for fire area where item is required. See applicable station procedure for fire area identification. Note: Some items are shared with Unit 3. Notify the Shift Manager of any impacts.

Table 7.1.26-1
Support Equipment, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Compensatory Measures for Components Requiring Prompt Action (see ACTION "f")
E	F156A, F156B, F156C, F156D Portable Fans	1. A minimum of two fans shall be FUNCTIONAL and stored in designated cabinets in the corridor, el. 14'-6" adjacent to the Turbine Bldg entrance. 2. Power shall be available at the dedicated 480V outlets in both DC Switchgear Rooms.	1. With less than 2 fans FUNCTIONAL, or the dedicated power supply (REC-1,2 and 3) in the East (Z1) DC Switchgear Room nonfunctional, perform ACTIONS b.1, b.2 for fire area R-3. 2. With less than 2 fans FUNCTIONAL, or the dedicated power supply (REC-7, 8, and 9) in the West (Z2) DC Switchgear Room nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-1 and R-16.	N/A
F	F157A, F157B, F157C, F157D Portable Fans	1. A minimum of two fans shall be FUNCTIONAL, and stored in designated cabinet on the platform (El.46'-10") in the West 480V Switchgear Room. 2. Power shall be available at the dedicated 480V outlets in both 480V Switchgear Rooms.	1. With less than 2 fans FUNCTIONAL, or the dedicated power supply (REC-4, 5, and 6) in the West (Z1) 480V Switchgear Room nonfunctional, perform ACTIONS b.1, b.2 for fire areas R-2 and R-3. 2. With less than 2 fans FUNCTIONAL, or the dedicated power supply (REC-10, 11, and 12) in the East (Z2) 480V Switchgear Room nonfunctional, perform ACTIONS b.1, b.2 for fire area R-1.	N/A

Table 7.1.26-1
Support Equipment, Appendix R Components
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Compensatory Measures if Component Not Restored in 14 Days (see ACTION "a")	Compensatory Measures for Components Requiring Prompt Action (see ACTION "f")
G	Compressed air bottle 3A and valves 2-IA-601, 598, 600, 891, 910, 566, and 2- IA-599 (regulator). Compressed air bottle 3B and valves 2-IA-603, 590, 595, 566 and 2-IA-591 (regulator). Compressed air bottle 3C and valves 2-IA-885, 887, 889, 910, 566, and 2-IA-886 (regulator).	A minimum of one compressed air bottle shall be FUNCTIONAL (1000 psig min.) and capable of alignment to 2-CH-517, 2-CH-518, and 2-CH-519 using indicated valving.	With all three compressed air bottles nonfunctional, 1. Perform ACTIONS b.1, b.2 for fire area R-1, AND 2. Verify that 2-CH-517, 2-CH-518, and 2-CH-519 can be cycled by manual operation or perform ACTIONS b.1, b.2 for ALL remaining fire areas.	N/A

Table 7.1.26-2
Support Equipment, Appendix R Components
TECHNICAL SURVEILLANCE REQUIREMENTS

TSR Item	Comp ID	TECHNICAL SURVEILLANCE REQUIREMENT	Frequency
A	Hand-held Radios	1. Verify FUNCTIONALITY of the portable radios from all remote shutdown areas. 2. Verify FUNCTIONALITY of both Control Room base stations and repeaters.	Once per REFUELING cycle
B	HOT SHUTDOWN (HSD) Storage Box Inventory	Verify all items are available to support COLD SHUTDOWN.	Once per REFUELING cycle
C	Fire Shutdown (FSD) Storage Box Inventory	Verify all items are available to support COLD SHUTDOWN.	Once per REFUELING cycle
D	COLD SHUTDOWN Repair Material	Verify all materials available in the Unit 2 Appendix R Storage Area to support COLD SHUTDOWN.	Once per REFUELING cycle
E	F156A, F156B, F156C, F156D Portable Appendix R fans	Verify each fan can be powered from the outlets in both DC Switchgear Rooms, and fans are stored in their designated location.	Once per REFUELING cycle
F	F157A, F157B, F157C, F157D Portable Appendix R fans	Verify each fan can be powered from the outlets in both 480 V Switchgear Rooms, and fans are stored in their designated location.	Once per REFUELING cycle
G	Compressed air bottle 3A and valves 2-IA-601, 598, 600, 891, 910, 566, and 2-IA-599 (regulator). Compressed air bottle 3B and valves 2-IA-603, 590, 595, 566, and 2-IA-591 (regulator). Compressed air bottle 3C and valves 2-IA-885, 887, 889, 910, 566, and 2-IA-886 (regulator).	1. Verify each required instrument air valve and regulator can be manually opened. 2. Verify each compressed air bottle is FUNCTIONAL.	Once per REFUELING cycle

7.1 APPENDIX R SAFE SHUTDOWN REQUIREMENTS

REFERENCE TECHNICAL SPECIFICATION: 3.8.1.1 - A. C. Sources Operating

TECHNICAL REQUIREMENT

7.1.27 The Appendix R Safe Shutdown Related (ARSR) equipment listed in the TRM Table 7.1.27-1 shall be FUNCTIONAL.

APPLICABILITY:

MODES 1, 2, 3, and 4.

ACTION:

With an ARSR component listed in the above referenced component table nonfunctional (unable to meet its intended Appendix R shutdown function), take the ACTION as specified in the above table under Compensatory Measures.

TECHNICAL SURVEILLANCE REQUIREMENTS

7.1.27 The capability of equipment listed in TRM Table 7.1.27-1 to meet its Appendix R shutdown related FUNCTIONALITY requirements are verified per the SURVEILLANCE REQUIREMENTS of TS 4.8.1.1.2.a & b.

Table 7.1.27-1
Plant System - Emergency Diesel Generators
FUNCTIONALITY Requirements and Compensatory Measures

TR Item	Comp ID	FUNCTIONALITY Description	Component Not Restored in 14 Days (see ACTION “a”)	Risk Significant Component Not Restored in 72 Hours (see ACTION “d”)
A	H7A (includes F38A, A312, 2-SW-89A, 2-SW-231A)	The ‘A’ Emergency Diesel Generator must be capable of powering Bus 24C	(Tech Spec Limiting)	With ‘A’ EDG nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-12A, A-14B, A-14D, A-17, A-24) R-2 (FHA A-13, T-8, T-10) R-3 (FHA T-1A, T-1C, T-1F) R-10 (FHA A-21) R16 (FHA I-1A)
B	H7B (includes F38B, A401, 2-SW-89B, 2-SW-231B)	The ‘B’ Emergency Diesel Generator must be capable of powering Bus 24D	(Tech Spec Limiting)	With ‘B’ EDG nonfunctional, Risk Mitigation Areas are: R-1 (FHA A-12A, A-14B, A-14D, A-24) R-3 (FHA T-1A) R-7 (FHA A-15, A-31) R-9 (FHA A-20) R-14 (FHA T-7, T-9) R16 (FHA I-1A) R-17 (FHA A-15)

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

1. CORE OPERATING LIMITS REPORT

This CORE OPERATING LIMITS REPORT for Millstone 2 has been prepared in accordance with the requirements of Technical Specification 6.9.1.8 a through 6.9.1.8 d. The Technical Specifications affected by this report are listed below:

Section	Specification	
2.1	3/4.1.1.1	SHUTDOWN MARGIN – (SDM)
2.2		Deleted (Ref. License Amendment 280)
2.3	3/4.1.1.4	Moderator Temperature Coefficient (MTC)
2.4	3/4.1.3.6	Regulating CEA Insertion Limits
2.5	3/4.2.1	Linear Heat Rate
2.6	3/4.2.3	TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR -- F_r^T
2.7	3/4.2.6	DNB Margin

Terms appearing in capitalized type are DEFINED TERMS as defined in Section 1.0 of the Technical Specifications.

2. OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC approved methodologies specified in Section 3.

2.1 SHUTDOWN MARGIN – (SDM) (Specification 3/4.1.1.1)

The SHUTDOWN MARGIN shall be $\geq$ 3.6% ΔK/K

2.2 Deleted

2.3 Moderator Temperature Coefficient (Specification 3/4.1.1.4)

The moderator temperature coefficient shall be:

a. Less positive than 0.7×10^{-4} ΔK/K/°F whenever THERMAL POWER is $\leq$70% of RATED THERMAL POWER,

b. Less positive than 0.4×10^{-4} ΔK/K/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER,

c. Less negative than -3.0×10^{-4} ΔK/K/°F at RATED THERMAL POWER.

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

2.4 Regulating CEA Insertion Limits (Technical Specification 3/4.1.3.6)

The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown in Figure 2.4-1. CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits is restricted to:

a. ≤ 4 hours per 24 hour interval,

b. ≤ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. ≤ 14 Effective Full Power Days per 365 Effective Full Power Day interval.

2.5 Linear Heat Rate (Technical Specification 3/4.2.1)

The linear heat rate, including heat generated in the fuel, clad and moderator, shall not exceed:

a. 15.1 kw/ft whenever the reactor coolant flow rate (determined per Technical Specification 3/4.2.1) is ≥ 360,000 gpm.

b. 15.0 kw/ft whenever the reactor coolant flow rate (determined per Technical Specification 3/4.2.1) is < 360,000 gpm and ≥ 354,600 gpm.

c. 14.9 kw/ft whenever the reactor coolant flow rate (determined per Technical Specification 3/4.2.1) is < 354,600 gpm and ≥ 349,200 gpm.

During operation with the linear heat rate being monitored by the Excore Detector Monitoring System, the AXIAL SHAPE INDEX shall remain within the limits of Figure 2.5-1.

During operation with the linear heat rate being monitored by the Incore Detector Monitor System, the alarm setpoints shall be adjusted to less than or equal to the limit when the following factors are appropriately included in the setting of the alarms:

1. A measurement-calculational uncertainty factor of 1.07,

2. An engineering uncertainty factor of 1.03, and

3. A THERMAL POWER measurement uncertainty factor of 1.02.

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

2.6 TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR -- F_r^T (Technical Specification 3/4.2.3)

The calculated value of F_r^T at RATED THERMAL POWER shall be:

$\leq$ 1.690 whenever the reactor coolant flow rate is $\geq$ 360,000 gpm.

$\leq$ 1.664 whenever the reactor coolant flow rate is < 360,000 gpm and $\geq$ 354,600 gpm.

$\leq$ 1.639 whenever the reactor coolant flow rate is < 354,600 gpm and $\geq$ 349,200 gpm.

2.6.1 The Power Dependent F_r^T limits, whenever the reactor coolant flow rate is $\geq$ 360,000 gpm, are shown in Figure 2.6-1.

2.6.2 The Power Dependent F_r^T limits, whenever the reactor coolant flow rate is < 360,000 gpm and $\geq$ 354,600 gpm, are shown in Figure 2.6-2.

2.6.3 The Power Dependent F_r^T limits, whenever the reactor coolant flow rate is < 354,600 gpm and $\geq$ 349,200 gpm, are shown in Figure 2.6-3.

2.7 DNB Margin (Technical Specification 3/4.2.6)

The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the following limits:

Parameter	Limits
	Four Reactor Coolant Pumps Operations
a. Cold Leg Temperature	$\leq$ 549°F
b. Pressurizer Pressure	> 2225 psia*
c. Reactor Coolant Flow Rate	$\geq$ 360,000 gpm with Linear Heat Rate and F_r^T limits as specified in Sections 2.5 and 2.6. or $\geq$ 354,600 gpm with Linear Heat Rate and F_r^T limit reductions as specified in Sections 2.5 and 2.6. or $\geq$ 349,200 gpm with Linear Heat Rate and F_r^T limit reductions as specified in Sections 2.5 and 2.6.
d. AXIAL SHAPE INDEX	Figure 2.7-1

* Limit not applicable during either the THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

3. ANALYTICAL METHODS

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

3.1 EMF-96-029(P)(A) Volumes 1 and 2, “Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,” Siemens Power Corporation, January 1997.

Methodology for:

TS 3.1.1.1 – Shutdown Margin – (SDM)
TS 3.1.1.4 – Moderator Temperature Coefficient (MTC)
TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.2 ANF-84-73 Revision 5 Appendix B(P)(A), “Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events,” Advanced Nuclear Fuels, July 1990.

Methodology for:

TS 3.1.1.1 – Shutdown Margin – (SDM)
TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.3 XN-NF-82-21(P)(A) Revision 1, “Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations,” Exxon Nuclear Company, September 1983.

Methodology for:

TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.4 XN-75-32(P)(A) Supplements 1 through 4, “Computational Procedure for Evaluating Fuel Rod Bowing,” Exxon Nuclear Company, October 1983.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

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3.5 EMF-2328(P)(A), Revision 0 and Supplement 1(P)(A), “PWR Small Break LOCA Evaluation Model S-RELAP5 Based,” Framatome ANP, September 2015.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.6 Deleted

3.7 XN-NF-78-44(NP)(A), “A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors,” Exxon Nuclear Company, October 1983.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.8 XN-NF-621(P)(A) Revision 1, “Exxon Nuclear DNB Correlation for PWR Fuel Designs,” Exxon Nuclear Company, September 1983.

Methodology for:

TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.9 XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” Exxon Nuclear Company, October 1986.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

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3.10 ANF-88-133(P)(A) and Supplement 1, “Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU,” Advanced Nuclear Fuels Corporation, December 1991.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.11 XN-NF-85-92(P)(A) “Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results,” Exxon Nuclear Company, November 1986.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.12 ANF-89-151(P)(A), “ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events,” Advanced Nuclear Fuels Corporation, May 1992.

Methodology for:

TS 3.1.1.4 – Moderator Temperature Coefficient (MTC)
TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.13 EMF-1961(P)(A) Revision 0, “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors,” Siemens Power Corporation, July 2000.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r
TS 3.2.6 – DNB Margin

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3.14 EMF-2310(P)(A), Revision 1 and Supplement 2P-A, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” Framatome ANP, March 2023.

Methodology for:

TS 3.1.1.4 – Moderator Temperature Coefficient (MTC)
TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.15 EMF-92-153(P)(A), Revision 1, “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, January 2005.

Methodology for:

TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.16 EMF-92-116(P)(A) Revision 0 and Supplement 1(P)(A), “Generic Mechanical Design Criteria for PWR Fuel Designs,” Siemens Power Corporation, February 2015.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.17 BAW-10240(P)(A), Revision 0, “Incorporation of M5™ Properties in Framatome ANP Approved Methods,” Framatome ANP, Inc., May 2004.

Methodology for:

TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

3.18 EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” AREVA Inc., June 2016.

Methodology for:

TS 3.1.3.6 – Regulating CEA Insertion Limits
TS 3.2.1 – Linear Heat Rate
TS 3.2.3 – Total Unrodded Integrated Radial Peaking Factor – F^T_r

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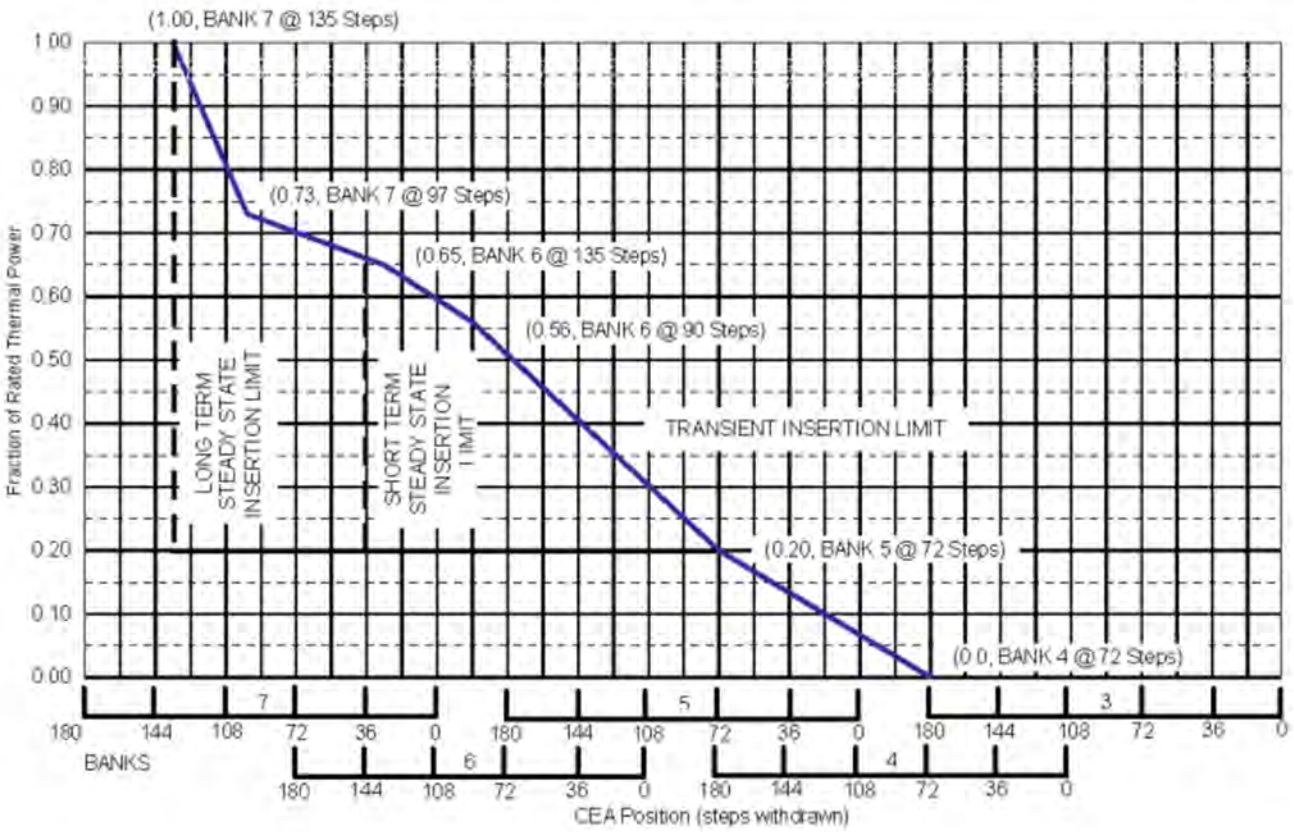
CORE OPERATING LIMITS REPORT, CYCLE 30



Note: Regulating CEAs that are ≥ 176 steps withdrawn are considered fully withdrawn and are acceptable per this figure.

Figure 2.4-1
CEA Insertion Limit vs. THERMAL POWER With
Four Reactor Coolant Pumps Operating

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CORE OPERATING LIMITS REPORT, CYCLE 30

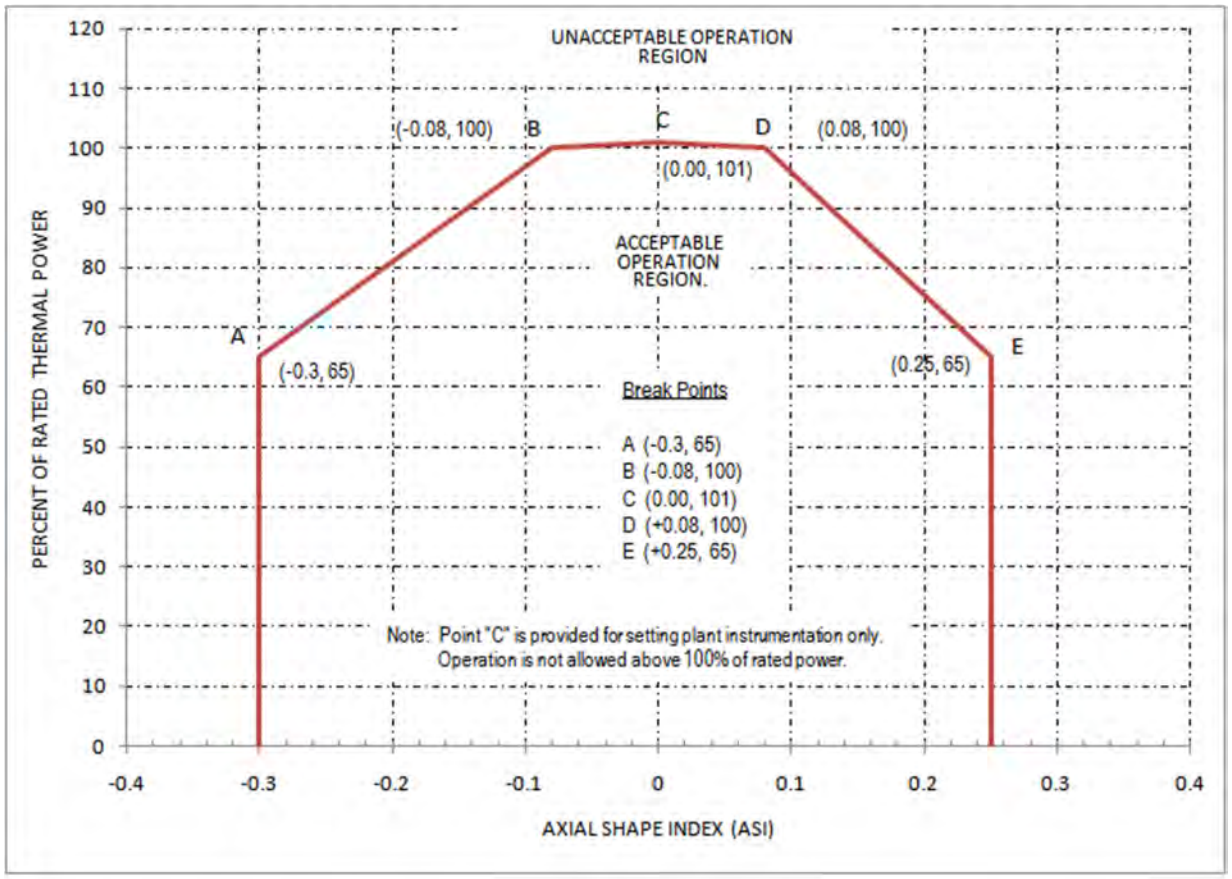
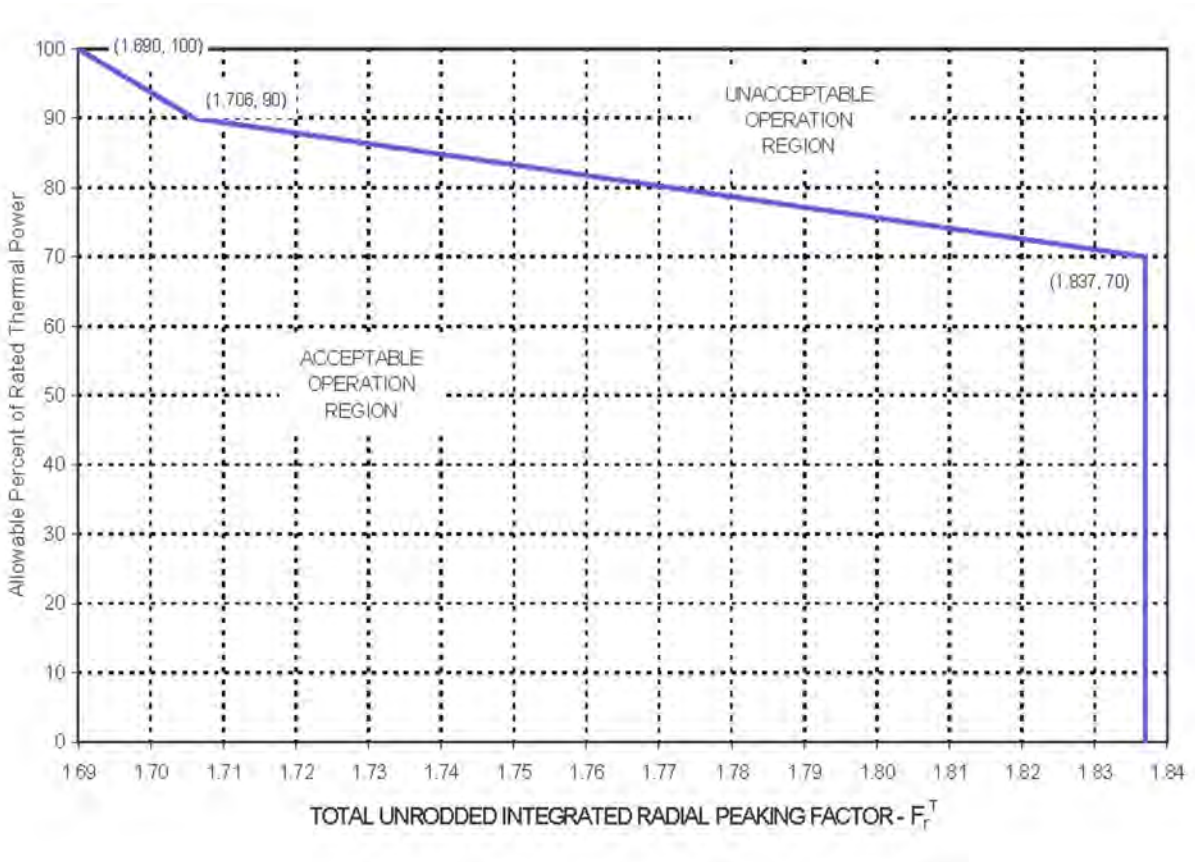


Figure 2.5-1
AXIAL SHAPE INDEX vs.
PERCENT OF ALLOWABLE POWER LEVEL

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30



Note: The F_r^T limit should be reduced for reactor coolant flow rates ≥ 349,200 gpm and < 360,000 gpm (see Section 2.6).

Figure 2.6-1
TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR
vs. Allowable RATED THERMAL POWER
For Reactor Coolant Flow Rates ≥ 360,000 gpm

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CORE OPERATING LIMITS REPORT, CYCLE 30

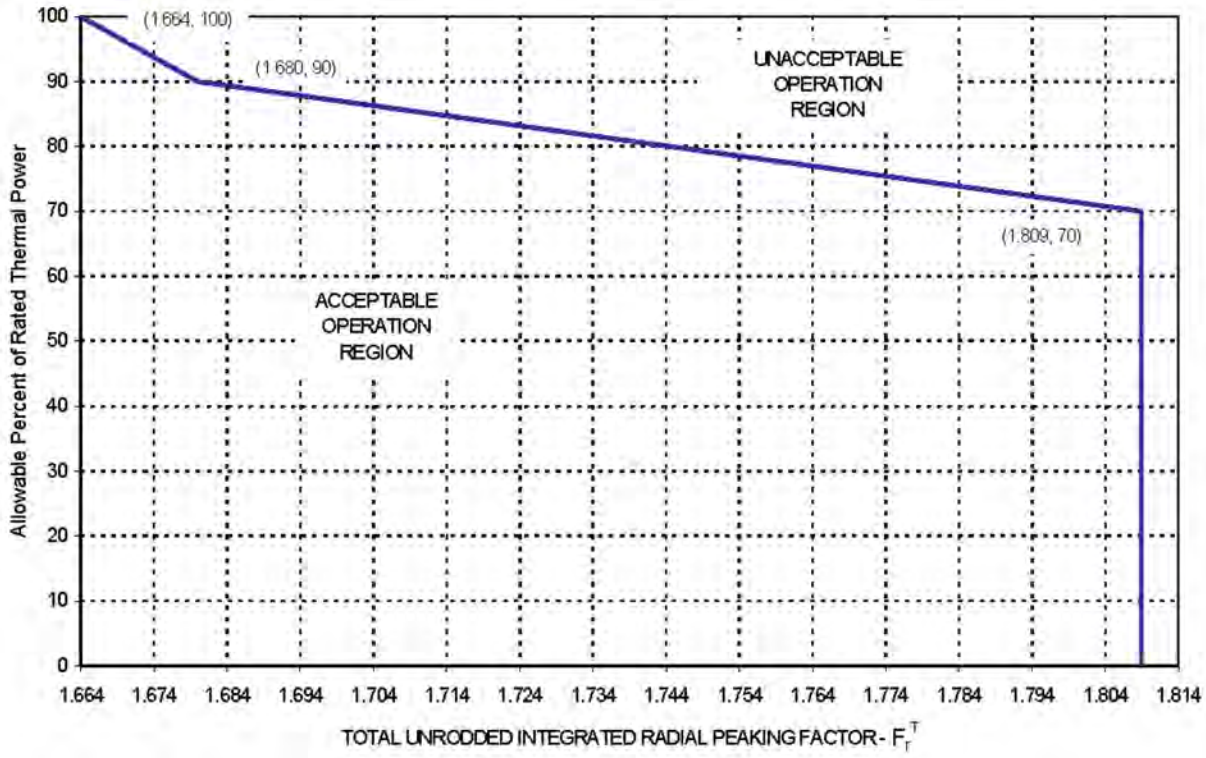

Figure 2.6-2
TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR
vs. Allowable RATED THERMAL POWER
For Reactor Coolant Flow Rates < 360,000 gpm and ≥ 354,600 gpm

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

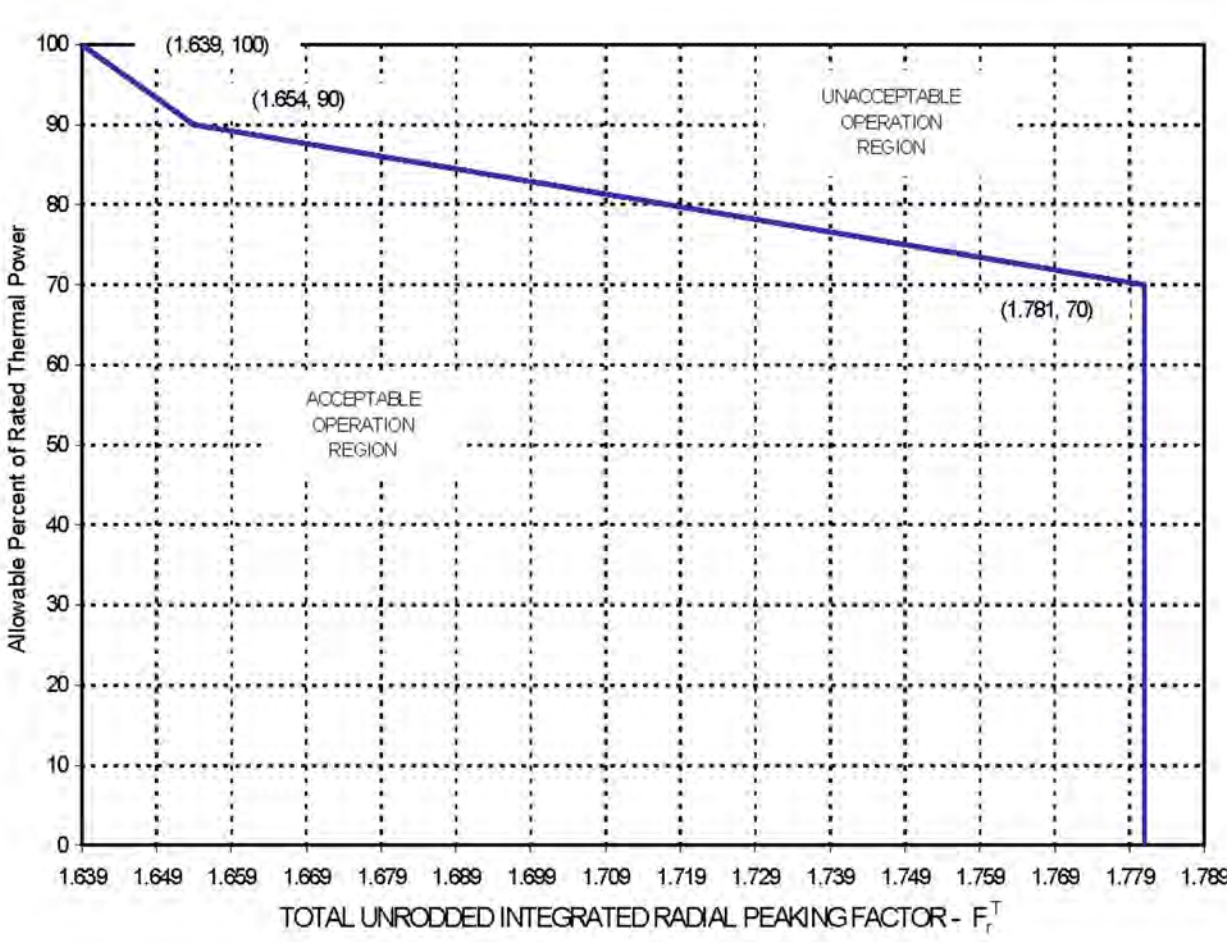

Figure 2.6-3
TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR
vs. Allowable RATED THERMAL POWER
For Reactor Coolant Flow Rates < 354,600 gpm and ≥ 349,200 gpm

APPENDIX 8.1
CORE OPERATING LIMITS REPORT, CYCLE 30

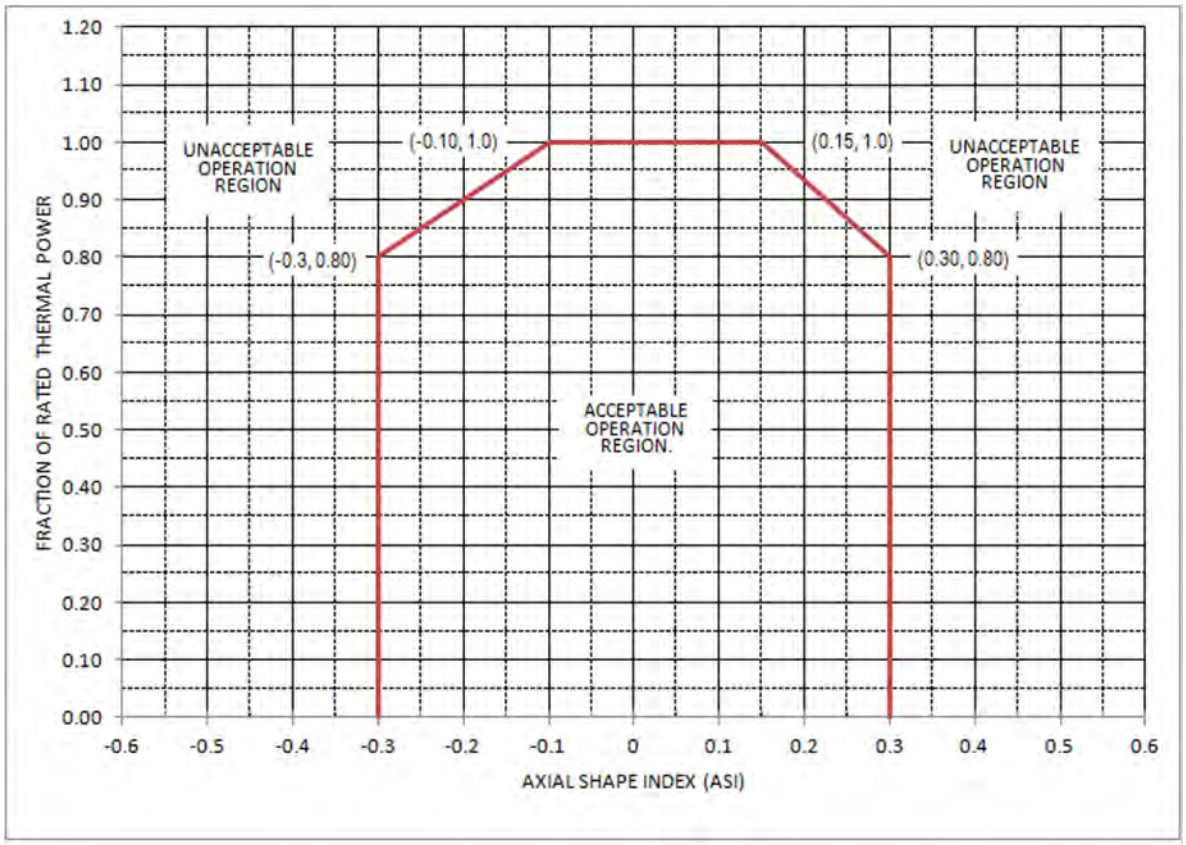

Figure 2.7-1
AXIAL SHAPE INDEX Operating Limits With
Four Reactor Coolant Pumps Operating