AmerGen Energy Company, LLC
200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

AmerGen[SM]
An Exelon Company

TS 6.18.d

5928-06-20446
April 24, 2006

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

Three Mile Island, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Submittal of Changes to Technical Specifications Bases

In accordance with the requirement of Three Mile Island, Unit 1 Technical Specification 6.18.d, AmerGen Energy Company, LLC, hereby submits a complete updated copy of the Three Mile Island, Unit 1 Technical Specifications Bases, which includes changes through the date of this letter.

If you have any questions or require further information, please contact Tom Loomis at 610-765-5510.

Sincerely,

D. P. Helker

David P. Helker
Manager - Licensing

Enclosure: 1) TMI, Unit 1 TS and Bases

cc: S. J. Collins, Administrator, Region 1, USNRC (w/o Enclosure)
D. M. Kern, USNRC Senior Resident Inspector (w/o Enclosure)
F. E. Saba, Project Manager, USNRC (w/ Enclosure)
R. Guzman, Project Manager USNRC (w/o Enclosure)
File No. 06022

A001

TMI-1 TECHNICAL SPECIFICATIONS
APPENDIX A
Table of Effective Pages

PAGE NO.	AMENDMENT NO.
TMI Unit 1 Tech Specs	
Table of Contents	
i	173
ii	248
iii	246
iv	248
v	256
vi	216
vii	234
Section 1.0	
1-1	157
1-2	175
1-3	225
1-4	166
1-4a	Deleted
1-5	246
1-6	197
1-6a	Deleted
1-7	251
1-7a	Deleted
1-8	199
1-9	175
Section 2.0	
2-1	247
2-2	247
2-3	247
2-4	157
2-4a, Figure 2.1-1	247
2-4b, Figure 2.1-2	184
2-4c, Figure 2.1-3	247
2-5	184
2-6	247
2-7	247
2-8	247
2-9	157
2-10	247
2-11, Figure 2.3-1	247
2-12, Figure 2.3-2	184
Section 3.0	
3-1	98
3-1a	98
3-2	222
3-3	234
3-4	234
3-5	234
3-5a, Figure 3.1-1	234
3-5b, Figure 3.1-2	234
3-6	157
3-6a	78
3-7	ECR TM 04-00911

PAGE NO.	AMENDMENT NO.
3-8	204
3-9	204
3-9a	Deleted
3-9b, Figure 3.1-2a	204
3-10	157
3-11	157
3-12	246
3-13	141
3-13a	149
3-14	246
3-15	ECR TM 04-00601
3-15a	246
3-15b	141
3-16	157
3-17	157
3-18	157
3-18a	211
3-18b, Figure 3.1-3	Deleted
3-18c	167
3-18d	234
3-18e	234
3-18f	234
3-18g	186
3-18h	186
3-19	196
3-20	196
3-21	227
3-22	227
3-23	229
3-24	227
3-24a	227
3-25	242
3-26	242
3-26a	242
3-26b	242
3-26c	242
3-26d	242
3-27	189
3-27a	189
3-28	193
3-28a	157
3-29	247
3-30	189
3-30a	189
3-31	189
3-32	200
3-32a	193
3-32b	189
3-32c	189
3-33	246
3-34	211
3-34a	211
3-35	219
3-35a	211

TMI-1 TECHNICAL SPECIFICATIONS
APPENDIX A
Table of Effective Pages

PAGE NO.	AMENDMENT NO.	PAGE NO.	AMENDMENT NO.
3-35b	211	3-55	109
3-36	ECR TM 04-01026	3-56	157
3-36a	211	3-56a	157
3-36b	168	3-56b	109
Figure 3.5-2A	Deleted	3-57	115
Figure 3.5-2B	Deleted	3-58	115
Figure 3.5-2C	Deleted	3-59	182
Figure 3.5-2D	Deleted	3-60	157
Figure 3.5-2E	Deleted	3-61	226
Figure 3.5-2F	Deleted	3-62	226
Figure 3.5-2G	Deleted	3-62a	245
Figure 3.5-2H	Deleted	3-62b	245
Figure 3.5-2I	Deleted	3-62c	248
Figure 3.5-2J	Deleted	3-62d	248
Figure 3.5-2K	Deleted	3-62e	226
Figure 3.5-2L	Deleted	3-62f	226
3-37	159	3-63	106
3-37a	224	3-64	106
3-38	211	3-65	Deleted
3-39	Deleted	3-66	Deleted
3-39a, Figure 3.5-1	167	3-67	Deleted
3-39b, Figure 3.5-2	167	3-68	Deleted
3-39c, Figure 3.5-3	167	3-69	Deleted
3-40	211	3-70	Deleted
3-40a	100	3-71	Deleted
3-40b	242	3-72	Deleted
3-40c	242	3-73	Deleted
3-40d	240	3-74	Deleted
3-40e	166	3-75	Deleted
3-40f	182	3-76	Deleted
3-40g	216	3-77	Deleted
3-40h	216	3-78	Deleted
3-40i	216	3-79	Deleted
3-40j	216	3-80	157
3-41	246	3-81	41
3-41a	246	3-82	Deleted
3-41b	201	3-83	Deleted
3-41c	246	3-84	Deleted
3-41d	246	3-85	Deleted
3-41e	246	3-86	146
3-42	224	3-87	Deleted
3-43	258	3-87a	Deleted
3-43a	224	3-88	Deleted
3-44	257	3-89	Deleted
3-45	257	3-90	Deleted
3-45a	257	3-91	Deleted
3-46	129	3-92	Deleted
3-47	Deleted	3-93	Deleted
3-48	Deleted	3-94	Deleted
3-49	Deleted	3-95	187
3-50	Deleted	3-95a	139
3-51	Deleted	3-95b	Deleted
3-52	Deleted	3-95c	Deleted
3-53	Deleted	3-96	197
3-54	Deleted	3-97	Deleted

TMI-1 TECHNICAL SPECIFICATIONS
APPENDIX A
Table of Effective Pages

PAGE NO.	AMENDMENT NO.	PAGE NO.	AMENDMENT NO.
3-98	Deleted	4-10b	240
3-99	Deleted	4-10c	225
3-100	Deleted	4-11	172
3-101	Deleted	4-12	118
3-102	Deleted	4-13	198
3-103	Deleted	4-14	Deleted
3-104	Deleted	4-15	Deleted
3-105	Deleted	4-16	Deleted
3-105a	Deleted	4-17	Deleted
3-106	Deleted	4-18	Deleted
3-107	Deleted	4-19	Deleted
3-108	Deleted	4-20	Deleted
3-109	Deleted	4-21	Deleted
3-110	Deleted	4-22	Deleted
3-111	Deleted	4-23	Deleted
3-112	Deleted	4-24	Deleted
3-112a	Deleted	4-25	Deleted
3-113	Deleted	4-26	Deleted
3-114	Deleted	4-27	Deleted
3-115	Deleted	Figure 4.2-1	Deleted
3-116	Deleted	4-28	Deleted
3-117	Deleted	4-29	236
3-118	Deleted	4-30	201
3-119	Deleted	4-31	Deleted
3-119a	Deleted	4-32	Deleted
3-120	Deleted	4-33	Deleted
3-121	Deleted	4-34	Deleted
3-122	Deleted	4-34a	Deleted
3-123	Deleted	4-34b	Deleted
3-124	Deleted	4-34c	Deleted
3-125	Deleted	4-35	251
3-126	Deleted	4-35a	Deleted
3-127	Deleted	4-36	187
3-128	254	4-37	251
3-129	251	4-38	246
		4-38a	Deleted
Section 4		4-39	212
		4-40	157
4-1	256	4-41	234
4-1a	256	4-42	225
4-1b	256	4-43	225
4-2	227	4-44	167
4-2a	255	4-45	235
4-2b	255	4-46	243
4-3	255	4-47	175
4-4	255	4-48	211
4-5	225	4-49	157
4-5a	227	4-50	211
4-6	227	4-51	246
4-7	255	4-52	242
4-7a	191	4-52a	242
4-8	246	4-53	
4-9	211	4-54	97
4-10	225	4-55	223
4-10a	240	4-55a	226

TMI-1 TECHNICAL SPECIFICATIONS
APPENDIX A
Table of Effective Pages

PAGE NO.	AMENDMENT NO.
4-55b	245
4-55c	245
4-55d	248
4-55e	248
4-55f	122
4-55g	226
4-56	129
4-57	Deleted
4-58	167
4-59	149
4-60	219
4-61	149
4-62	106
4-63	106
4-64	149
4-65	106
4-66	110
4-67	167
4-68	167
4-69	Deleted
4-70	Deleted
4-71	Deleted
4-72	Deleted
4-73	Deleted
4-73a	Deleted
4-73b	Deleted
4-73c	Deleted
4-74	Deleted
4-75	Deleted
4-76	Deleted
4-76a	Deleted
4-77	47
4-78	237
4-79	209
4-80	237
4-80a	Deleted
4-81	237
4-82	237
4-83	237
4-83a	237
4-84	47
4-85	129
4-86	47
4-87	197
4-88	Deleted
4-89	Deleted
4-90	Deleted
4-91	Deleted
4-92	Deleted
4-93	Deleted
4-94	Deleted
4-95	Deleted
4-96	Deleted
4-97	Deleted
4-98	Deleted

PAGE NO.	AMENDMENT NO.
4-99	Deleted
4-100	Deleted
4-101	Deleted
4-102	Deleted
4-103	Deleted
4-104	Deleted
4-105	Deleted
4-106	Deleted
4-107	Deleted
4-108	Deleted
4-109	Deleted
4-110	Deleted
4-111	Deleted
4-112	Deleted
4-113	Deleted
4-114	Deleted
4-115	Deleted
4-116	Deleted
4-117	Deleted
4-118	Deleted
4-119	Deleted
4-120	Deleted
4-121	Deleted
4-122	Deleted

Section 5.0

5-1	218
5-2	157
5-3	157
5-4	233
5-5	183
5-6	170
5-7	231
5-7a Figure 5-4	170
5-7b Figure 5-5	170
5-8	212
5-9	212
5-10	246
5-11	Deleted
Figure 5-1	246
Figure 5-2	218
Figure 5-3	246

Section 6.0

6-1	219
6-1a	219
6-2	219
6-3	241
6-4	218
6-5	239
6-6	239
6-7	252

TMI-1 TECHNICAL SPECIFICATIONS
APPENDIX A
Table of Effective Pages

PAGE NO.	AMENDMENT NO.	PAGE NO.	AMENDMENT NO.
6-8	252		
6-9	252		
6-10	219		
6-11	218		
6-11a	197		
6-11b	197		
6-11c	244		
6-12	254		
6-12a	Deleted		
6-13	254		
6-13a	Deleted		
6-14	Deleted		
6-15	Deleted		
6-16	Deleted		
6-17	173		
6-18	185		
6-19	233		
6-19a	Deleted		
6-20	219		
6-21	252		
6-22	213		
6-23	218		
6-24	253		
6-25	256		
6-26	256		
Figure 6-1	Deleted		
Figure 6-2	Deleted		

THREE MILE ISLAND
NUCLEAR STATION
UNIT 1

LICENSE NO. DPR-50

APPENDIX A
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

Section		Page
TECHNICAL SPECIFICATIONS		
1	DEFINITIONS	1-1
1.1	RATED POWER	1-1
1.2	REACTOR OPERATING CONDITIONS	1-1
1.2.1	Cold Shutdown	1-1
1.2.2	Hot Shutdown	1-1
1.2.3	Reactor Critical	1-1
1.2.4	Hot Standby	1-1
1.2.5	Power Operation	1-1
1.2.6	Refueling Shutdown	1-1
1.2.7	Refueling Operation	1-2
1.2.8	Refueling Interval	1-2
1.2.9	Startup	1-2
1.2.10	T_{avg}	1-2
1.2.11	Heatup-Cooldown Mode	1-2
1.2.12	Station, Unit, Plant, and Facility	1-2
1.3	OPERABLE	1-2
1.4	PROTECTIVE INSTRUMENTATION LOGIC	1-2
1.4.1	Instrument Channel	1-2
1.4.2	Reactor Protection System	1-2
1.4.3	Protection Channel	1-3
1.4.4	Reactor Protection System Logic	1-3
1.4.5	Engineered Safety Features System	1-3
1.4.6	Degree of Redundancy	1-3
1.5	INSTRUMENTATION SURVEILLANCE	1-3
1.5.1	Trip Test	1-3
1.5.2	Channel Test	1-3
1.5.3	Channel Check	1-4
1.5.4	Channel Calibration	1-4
1.5.5	Heat Balance Check	1-4
1.5.6	Heat Balance Calibration	1-4
1.6	POWER DISTRIBUTION	1-5
1.6.1	Quadrant Power Tilt	1-5
1.6.2	Axial Power Imbalance	1-5
1.7	CONTAINMENT INTEGRITY	1-5
1.8	FIRE SUPPRESSION WATER SYSTEM	1-5
1.9	DELETED	1-6
1.10	DELETED	1-6
1.11	DELETED	1-6
1.12	DOSE EQUIVALENT I-131	1-6
1.13	SOURCE CHECK	1-6
1.14	SOLIDIFICATION - DELETED	1-6
1.15	OFFSITE DOSE CALCULATION MANUAL	1-6
1.16	PROCESS CONTROL PROGRAM	1-6
1.17	GASEOUS RADWASTE TREATMENT SYSTEM	1-6
1.18	VENTILATION EXHAUST TREATMENT SYSTEM	1-7
1.19	PURGE-PURGING	1-7
1.20	VENTING	1-7
1.21	REPORTABLE EVENT	1-7
1.22	MEMBER(S) OF THE PUBLIC	1-7
1.23	SUBSTANTIVE CHANGES	1-7
1.24	CORE OPERATING LIMITS REPORT	1-8
1.25	FREQUENCY NOTATION	1-8

Amendment No. ~~11~~, ~~72~~, ~~125~~, ~~137~~, ~~142~~, ~~150~~, ~~155~~,173

TABLE OF CONTENTS

Section		Page
2	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	2-1
2.1	Safety Limits, Reactor Core	2-1
2.2	Safety Limits, Reactor System Pressure	2-4
2.3	Limiting Safety System Settings, Protection Instrumentation	2-5
3	LIMITING CONDITIONS FOR OPERATION	3-1
3.0	General Action Requirements	3-1
3.1	Reactor Coolant System	3-1a
3.1.1	Operational Components	3-1a
3.1.2	Pressurization, Heatup and Cooldown Limitations	3-3
3.1.3	Minimum Conditions for Criticality	3-6
3.1.4	Reactor Coolant System Activity	3-8
3.1.5	Chemistry	3-10
3.1.6	Leakage	3-12
3.1.7	Moderator Temperature Coefficient of Reactivity	3-16
3.1.8	Single Loop Restrictions	3-17
3.1.9	Low Power Physics Testing Restrictions	3-18
3.1.10	Control Rod Operation (Deleted)	3-18a
3.1.11	Reactor Internal Vent Valves	3-18c
3.1.12	Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)	3-18d
3.1.13	Reactor Coolant System Vents	3-18f
3.2	Deleted	3-19
3.3	Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems	3-21
3.4	Decay Heat Removal (DHR) Capability	3-25
3.4.1	Reactor Coolant System (RCS) Temperature Greater than 250 Degrees F	3-25
3.4.2	RCS Temperature Less Than or Equal to 250 Degrees F	3-26a
3.5	Instrumentation Systems	3-27
3.5.1	Operational Safety Instrumentation	3-27
3.5.2	Control Rod Group and Power Distribution Limits	3-33
3.5.3	Engineered Safeguards Protection System Actuation Setpoints	3-37
3.5.4	Incore Instrumentation (Deleted)	3-38
3.5.5	Accident Monitoring Instrumentation	3-40a
3.5.6	Deleted	3-40f
3.5.7	Remote Shutdown System	3-40g
3.6	Reactor Building	3-41
3.7	Unit Electrical Power System	3-42
3.8	Fuel Loading and Refueling	3-44
3.9	Deleted	3-46
3.10	Miscellaneous Radioactive Materials Sources	3-46
3.11	Handling of Irradiated Fuel	3-55
3.12	Reactor Building Polar Crane	3-57
3.13	Secondary System Activity	3-58
3.14	Flood	3-59
3.14.1	Periodic Inspection of the Dikes Around TMI	3-59
3.14.2	Flood Condition for Placing the Unit in Hot Standby	3-60
3.15	Air Treatment Systems	3-61
3.15.1	Emergency Control Room Air Treatment System	3-61
3.15.2	Reactor Building Purge Air Treatment System (Deleted)	3-62a
3.15.3	Auxiliary and Fuel Handling Building Air Treatment System (Deleted)	3-62c
3.15.4	Fuel Handling Building ESF Air Treatment System	3-62e

TABLE OF CONTENTS

Section		Page
3.16	SHOCK SUPPRESSORS (SNUBBERS)	3-63
3.17	REACTOR BUILDING AIR TEMPERATURE	3-80
3.18	FIRE PROTECTION (DELETED)	3-86
3.19	CONTAINMENT SYSTEMS	3-95
3.20	SPECIAL TEST EXCEPTIONS (DELETED)	3-95a
3.21	RADIOACTIVE EFFLUENT INSTRUMENTATION (DELETED)	3-96
3.21.1	RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION (DELETED)	3-96
3.21.2	RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION (DELETED)	3-96
3.22	RADIOACTIVE EFFLUENTS (DELETED)	3-96
3.22.1	LIQUID EFFLUENTS (DELETED)	3-96
3.22.2	GASEOUS EFFLUENTS (DELETED)	3-96
3.22.3	SOLID RADIOACTIVE WASTE (DELETED)	3-96
3.22.4	TOTAL DOSE (DELETED)	3-96
3.23	RADIOLOGICAL ENVIRONMENTAL MONITORING (DELETED)	3-96
3.23.1	MONITORING PROGRAM (DELETED)	3-96
3.23.2	LAND USE CENSUS (DELETED)	3-96
3.23.3	INTERLABORATORY COMPARISON PROGRAM (DELETED)	3-96
3.24	REACTOR VESSEL WATER LEVEL	3-128
4	SURVEILLANCE STANDARDS	4-1
4.1	OPERATIONAL SAFETY REVIEW	4-1
4.2	REACTOR COOLANT SYSTEM INSERVICE INSPECTION	4-11
4.3	TESTING FOLLOWING OPENING OF SYSTEM (DELETED)	4-13
4.4	REACTOR BUILDING	4-29
4.4.1	CONTAINMENT LEAKAGE TESTS	4-29
4.4.2	STRUCTURAL INTEGRITY	4-35
4.4.3	DELETED	4-37
4.4.4	DELETED	4-38
4.5	EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING	4-39
4.5.1	EMERGENCY LOADING SEQUENCE	4-39
4.5.2	EMERGENCY CORE COOLING SYSTEM	4-41
4.5.3	REACTOR BUILDING COOLING AND ISOLATION SYSTEM	4-43
4.5.4	ENGINEERED SAFEGUARDS FEATURE (ESF) SYSTEMS LEAKAGE	4-45
4.6	EMERGENCY POWER SYSTEM PERIODIC TESTS	4-46
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4-48
4.7.1	CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS	4-48
4.7.2	CONTROL ROD PROGRAM VERIFICATION (DELETED)	4-50

Amendment ~~72~~, ~~81~~, ~~108~~, ~~129~~, ~~137~~, ~~146~~, ~~147~~, ~~158~~, ~~191~~, ~~197~~, ~~198~~, ~~211~~, ~~215~~, 246

TABLE OF CONTENTS

Section		Page
4.8	DELETED	4-51
4.9	DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING	4-52
4.9.1	REACTOR COOLANT SYSTEM (RCS) TEMPERATURE GREATER THAN 250 DEGREES F	4-52
4.9.2	RCS TEMPERATURE LESS THAN OR EQUAL TO 250 DEGREES F	4-52a
4.10	REACTIVITY ANOMALIES	4-53
4.11	REACTOR COOLANT SYSTEM VENTS	4-54
4.12	AIR TREATMENT SYSTEMS	4-55
4.12.1	EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM	4-55
4.12.2	REACTOR BUILDING PURGE AIR TREATMENT SYSTEM (DELETED)	4-55b
4.12.3	AUXILIARY AND FUEL HANDLING BUILDING AIR TREATMENT SYSTEM (DELETED)	4-55d
4.12.4	FUEL HANDLING BUILDING ESF AIR TREATMENT SYSTEM	4-55f
4.13	RADIOACTIVE MATERIALS SOURCES SURVEILLANCE	4-56
4.14	DELETED	4-56
4.15	MAIN STEAM SYSTEM INSERVICE INSPECTION	4-58
4.16	REACTOR INTERNALS VENT VALVES SURVEILLANCE	4-59
4.17	SHOCK SUPPRESSORS (SNUBBERS)	4-60
4.18	FIRE PROTECTION SYSTEMS (DELETED)	4-72
4.19	OTSG TUBE INSERVICE INSPECTION	4-77
4.19.1	STEAM GENERATOR SAMPLE SELECTION AND INSPECTION METHODS	4-77
4.19.2	STEAM GENERATOR TUBE SAMPLE SELECTION AND INSPECTION	4-77
4.19.3	INSPECTION FREQUENCIES	4-79
4.19.4	ACCEPTANCE CRITERIA	4-80
4.19.5	REPORTS	4-81
4.20	REACTOR BUILDING AIR TEMPERATURE	4-86
4.21	RADIOACTIVE EFFLUENT INSTRUMENTATION (DELETED)	4-87
4.21.1	RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION (DELETED)	4-87
4.21.2	RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION (DELETED)	4-87
4.22	RADIOACTIVE EFFLUENTS (DELETED)	4-87
4.22.1	LIQUID EFFLUENTS (DELETED)	4-87
4.22.2	GASEOUS EFFLUENTS (DELETED)	4-87
4.22.3	SOLID RADIOACTIVE WASTE (DELETED)	4-87
4.22.4	TOTAL DOSE (DELETED)	4-87
4.23.1	MONITORING PROGRAM (DELETED)	4-87
4.23.2	LAND USE CENSUS (DELETED)	4-87
4.23.3	INTERLABORATORY COMPARISON PROGRAM (DELETED)	4-87

Amendment No. ~~11, 22, 30, 41, 47, 55, 72, 78, 85, 87, 119, 122, 129,~~
~~137, 146, 197, 242, 245, 246~~, 248

TABLE OF CONTENTS

Section		Page
5	DESIGN FEATURES	5-1
5.1	SITE	5-1
5.2	CONTAINMENT	5-2
5.2.1	REACTOR BUILDING	5-2
5.2.2	REACTOR BUILDING ISOLATION SYSTEM	5-3
5.3	REACTOR	5-4
5.3.1	REACTOR CORE	5-4
5.3.2	REACTOR COOLANT SYSTEM	5-4
5.4	NEW AND SPENT FUEL STORAGE FACILITIES	5-6
5.4.1	NEW FUEL STORAGE	5-6
5.4.2	SPENT FUEL STORAGE	5-6
5.5	AIR INTAKE TUNNEL FIRE PROTECTION SYSTEMS	5-8
6	ADMINISTRATIVE CONTROLS	6-1
6.1	RESPONSIBILITY	6-1
6.2	ORGANIZATION	6-1
6.2.1	CORPORATE	6-1
6.2.2	UNIT STAFF	6-1
6.3	UNIT STAFF QUALIFICATIONS	6-3
6.4	TRAINING	6-3
6.5	REVIEW AND AUDIT	6-3
6.5.1	TECHNICAL REVIEW AND CONTROL	6-4
6.5.2	INDEPENDENT SAFETY REVIEW	6-5
6.5.3	AUDITS	6-7
6.5.4	DELETED	6-8
6.6	REPORTABLE EVENT ACTION	6-10
6.7	SAFETY LIMIT VIOLATION	6-10
6.8	PROCEDURES AND PROGRAMS	6-11
6.9	REPORTING REQUIREMENTS	6-12
6.9.1	ROUTINE REPORTS	6-12
6.9.2	DELETED	6-14
6.9.3	ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT	6-17
6.9.4	ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT	6-18
6.9.5	CORE OPERATING LIMITS REPORT	6-19
6.10	RECORD RETENTION	6-20
6.11	RADIATION PROTECTION PROGRAM	6-22
6.12	HIGH RADIATION AREA	6-22
6.13	PROCESS CONTROL PROGRAM	6-23
6.14	OFFSITE DOSE CALCULATION MANUAL (ODCM)	6-24
6.15	DELETED	6-24
6.16	DELETED	6-24
6.17	MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS	6-25
6.18	TECHNICAL SPECIFICATION (TS) BASES CONTROL PROGRAM	6-25

Amendment No. ~~11, 47, 72, 77, 129, 150, 173, 212, 252, 253~~, 256

LIST OF TABLES

TABLE	TITLE	PAGE
1.2	Frequency Notation	1-8
2.3-1	Reactor Protection System Trip Setting Limits	2-9
3.1.6.1	Pressure Isolation Check Valves Between the Primary Coolant System and LPIS	3-15a
3.5-1	Instruments Operating Conditions	3-29
3.5-1A	DELETED	
3.5-2	Accident Monitoring Instruments	3-40c
3.5-3	Post Accident Monitoring Instrumentation	3-40d
3.5-4	Remote Shutdown System Instrumentation and Control	3-40i
3.21-1	DELETED	
3.21-2	DELETED	
3.23-1	DELETED	
3.23-2	DELETED	
4.1-1	Instrument Surveillance Requirements	4-3
4.1-2	Minimum Equipment Test Frequency	4-8
4.1-3	Minimum Sampling Frequency	4-9
4.1-4	Post Accident Monitoring Instrumentation	4-10a
4.19-1	Minimum Number of Steam Generators to be Inspected During Inservice Inspection	4-84
4.19-2	Steam Generator Tube Inspection	4-85
4.21-1	DELETED	
4.21-2	DELETED	
4.22-1	DELETED	
4.22-2	DELETED	
4.23-1	DELETED	

Amendment No. ~~59, 72, 100, 106, 118, 137, 142, 147, 150, 173, 197,~~ 216

LIST OF FIGURES

FIGURE	TITLE	PAGE
2.1-1	Core Protection Safety Limit TMI-1	2-4a
2.1-2	DELETED	
2.1-3	Core Protection Safety Bases TMI-1	2-4c
2.3-1	TMI-1 Protection System Maximum Allowable Setpoints	2-11
2.3-2	DELETED	
3.1-1	Reactor Coolant System Heatup/Cooldown Limitations (Applicable thru 29 EFPY)	3-5a
3.1-2	Reactor Coolant Inservice Leak and Hydrostatic Test (Applicable thru 29 EFPY)	3-5b
3.1-2a	Dose equivalent I-131 Primary Coolant Specific Actual Limit vs. Percent of RATED THERMAL POWER	3-9b
3.1-3	DELETED	
3.3-1	Makeup Tank Pressure vs Level Limits	3-24a
3.5-2A thru 3.5-2M	DELETED	
3.5-1	Incore Instrumentation Specification Axial Imbalance Indication	3-39a
3.5-2	Incore Instrumentation Specification Radial Flux Tilt Indication	3-39b
3.5-3	Incore Instrumentation Specification	3-39c
3.11-1	Transfer Path to and from Cask Loading Pit	3-56b
4.17-1	Snubber Functional Test - Sample Plan 2	4-67
5-1	Extended Plot Plan TMI	N/A
5-2	Site Topography 5 Mile Radius	N/A
5-3	Gaseous Effluent Release Points and Liquid Effluent Outfall Locations	N/A
5-4	Minimum Burnup Requirements for Fuel in Region II of the Pool A Storage Racks	5-7a
5-5	Minimum Burnup Requirements for Fuel in the Pool "B" Storage Racks	5-7b

Amendment Nos. ~~11, 17, 29, 39, 45, 50, 59, 72, 106, 109, 120, 126, 134, 142, 150, 164, 167, 168, 184, 211, 227,~~ 234

SECTION 1.0

DEFINITIONS

1. DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is a steady state reactor core output of 2568 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 COLD SHUTDOWN

The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is no more than 200°F. Pressure is defined by Specification 3.1.2.

1.2.2 HOT SHUTDOWN

The reactor is in the hot shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is at or greater than 525°F.

1.2.3 REACTOR CRITICAL

The reactor is critical when the neutron chain reaction is self-sustaining and Keff = 1.0.

1.2.4 HOT STANDBY

The reactor is in the hot standby condition when all of the following conditions exist:

a. Tavg is greater than 525°F

b. The reactor is critical

c. Indicated neutron power on the power range channels is less than two percent of rated power

1.2.5 POWER OPERATION

The reactor is in a power operating condition when the indicated neutron power is above two percent of rated power as indicated on the power range channels.

1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 REFUELING OPERATION

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 REFUELING INTERVAL

Time between normal refuelings of the reactor. This is defined as once per 24 months.

1.2.9 STARTUP

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.2.10 T_{ave}

T_{ave} is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.

1.2.11 HEATUP - COOLDOWN MODE

The heatup-cooldown mode is the range of reactor coolant temperature greater than 200°F and less than 525°F.

1.2.12 STATION, UNIT, PLANT, AND FACILITY

Station, unit, plant, and facility as used in these technical specifications all refer to TMI Unit 1.

1.3 OPERABLE

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

1.4 PROTECTION INSTRUMENTATION LOGIC

1.4.1 INSTRUMENT CHANNEL

An instrument channel is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An instrument channel may be either analog or digital.

1.4.2 REACTOR PROTECTION SYSTEM

The reactor protection system is **described in Section 7.1** of the Updated FSAR. It is that combination of protection channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protection trip breakers, and activating relays or coils.

1.4.3 PROTECTION CHANNEL

A PROTECTION CHANNEL as **described in Section 7.1** of the updated FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers, and bistable modules provided for every reactor protection safety parameter) is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. It includes a shutdown bypass circuit, a protection channel bypass circuit and a reactor trip module.

1.4.4 REACTOR PROTECTION SYSTEM LOGIC

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as **described in Section 7.1** of the updated FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.

1.4.5 ENGINEERED SAFETY FEATURES SYSTEM

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7.1-4 of the updated FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant engineered safety features equipment on a two-of-three basis for any given parameter.

1.4.6 DEGREE OF REDUNDANCY

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.5 <u>INSTRUMENTATION SURVEILLANCE</u>

1.5.1 TRIP TEST

A TRIP TEST is a test of logic elements in a protection channel to verify their associated trip action.

1.5.2 CHANNEL TEST

A CHANNEL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practical to verify OPERABILITY, including alarm and/or trip functions.

1.5.3 CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.5.4 CHANNEL CALIBRATION

An instrument CHANNEL CALIBRATION is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

1.5.5 HEAT BALANCE CHECK

A HEAT BALANCE CHECK is a comparison of the indicated neutron power and core thermal power.

1.5.6 HEAT BALANCE CALIBRATION

A HEAT BALANCE CALIBRATION is an adjustment of the power range channel amplifiers output based on the core thermal power determination.

Amendment No. ~~137~~, ~~157~~,166

1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100\left[\frac{\textit{Power in Any Core Quadrant}}{\textit{Average Power of All Quadrants}} - 1\right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.

b. All passive Containment Isolation Valves (CIVs) and isolation devices, including manual valves and blind flanges, are closed as required by the "Containment Integrity Check List" attached to the operating procedure, "Containment Integrity and Access Limits." Normally closed passive CIVs may be unisolated intermittently under administrative control.

c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.

d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

1.9 DELETED

1.10 DELETED

1.11 DELETED

1.12 DOSE EQUIVALENT I-131

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (*microcurie/gram*) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites". [Or in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.]

1.13 SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.14 DELETED

1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluent, in the calculation of gaseous and liquid effluent monitoring *Alarm/Trip Setpoints*, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.3 and 6.9.4.

1.16 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.17 GASEOUS RADWASTE TREATMENT

The GASEOUS RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce radioactive gaseous effluent by collecting primary coolant system off gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.18 VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluent by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodine or particulates from the gaseous exhaust system prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEMS.

1.19 PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is required to purify the confinement.

1.20 VENTING

VENTING is the controlled process of discharging air as gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions in such a manner that replacement air or gas is not provided. Vent used in system name does not imply a VENTING process.

1.21 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

1.22 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include any individual except when that individual is receiving an occupational dose.

1.23 SUBSTANTIVE CHANGES

SUBSTANTIVE CHANGES are those which affect the activities associated with a document or the document's meaning or intent. Examples of non-substantive changes are: (1) correcting spelling; (2) adding (but not deleting) sign-off spaces; (3) blocking in notes, cautions, etc.; (4) changes in corporate and personnel titles which do not reassign responsibilities and which are not referenced in the Appendix A Technical Specifications; and (5) changes in nomenclature or editorial changes which clearly do not change function, meaning or intent.

1.24 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is a TMI-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these operating limits is addressed in individual specifications.

1.25 FREQUENCY NOTATION

The *FREQUENCY NOTATION* specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2. All Surveillance Requirements shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. The 25% extension applies to all frequency intervals with the exception of "F." No extension is allowed for intervals designated "F."

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shiftly (once per 12 hours)
D	Daily (once per 24 hours)
W	Weekly (once per 7 days)
M	Monthly (once per 31 days)
Q	Quarterly (once per 92 days)
S/A	Semi-Annually (once per 184 days)
R	Refueling Interval (once per 24 months)
P S/U	Prior to each reactor startup, if not done during the previous 7 days
P S/A	Within six (6) months prior to each reactor startup
P	Completed prior to each release
N/A (NA)	Not applicable
E	Once per 18 months
F	Not to exceed 24 months

Bases

Section 1.25 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each REFUELING INTERVAL. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each REFUELING INTERVAL. Likewise, it is not the intent that REFUELING INTERVAL surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Section 1.25 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, axial power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1. If the actual pressure/temperature point is below and to the right of the line, the safety limit is exceeded.

2.1.2 The combination of reactor thermal power and axial power imbalance (power in the top half of core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the protective limit as defined by the locus of points (solid line) for the specified flow set forth in the Axial Power Imbalance Protective Limits given in the Core Operating Limits Report (COLR). If the actual-reactor- thermal-power/axial-power-imbalance point is above the line for the specified flow, the protective limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed, departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in excessive cladding temperature and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Reference 1) and BWC (Reference 2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel with Inconel intermediate spacer grids and the BWC correlation applies to Mark-B fuel with zircaloy or M5 intermediate spacer grids (non-mixing vane). The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, accounting only for DNBR correlation uncertainty, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). This corresponds to a Statistical Design Limit (SDL) of 1.313 (BWC) which accounts for all uncertainties considered with the statistical core design methodology (Reference 4). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits.

The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based on the nuclear power peaking factors given in Reference 3 and the COLR which define the reference design peaking condition in the core for operation at the maximum power. Once the reference peaking condition and the associated thermal-hydraulic situation has been established for the hot channel, then all other combinations of axial flux shapes and their accompanying radials must result in a condition which will not violate the previously established design criteria on DNBR. The flux shapes examined include a wide range of positive and negative offset for steady state and transient conditions.

These design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion, and form the core DNBR design basis.

The Axial Power Imbalance Protective Limits curves in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing:

a. The DNBR limit produced by a total nuclear power peaking factor consisting of the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

b. The maximum allowable local linear heat rate that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the axial power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of the Axial Power Imbalance Protective Limits given in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3. The curves of Figure 2.1-3 represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 22 percent, (BAW-2), or 26 percent (BWC) whichever condition is more restrictive. The curves of Figures 2.1-1 and 2.1-3 were developed assuming a reactor coolant design flow rate of 102% of 352,000 gpm.

The maximum thermal power for each reactor coolant pump operating condition (four pump, three pump, and one pump in each loop) given in the COLR is due to a power level trip produced by the flux-flow ratio multiplied by the minimum flow rate for the given pump combination plus the maximum calibration and instrumentation error.

Using a local quality limit of 22 percent (BAW-2), or 26 percent (BWC) at the point of minimum DNBR as a basis for curves 2 and 3 of Figure 2.1-3 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or BWC correlation continually increases from the point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

For each curve of Figure 2.1-3, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than the Statistical Design Limit (SDL) of 1.313 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2), or 26 percent (BWC) for the particular reactor coolant pump situation. Curve 1 is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of this curve will be above and to the left of the other curves.

<u>REFERENCES</u>

(1) UFSAR, Section 3.2.3.1.1 - "Fuel Assembly Heat Transfer Design"

(2) BWC Correlation of Critical Heat Flux, <u>BAW-10143P-A</u>, Babcock & Wilcox, Lynchburg, Virginia, April 1985

(3) UFSAR, Section 3.2.3.1.1.3 - "Nuclear Power Factors"

(4) BAW-10187 P-A, "Statistical Core Design For B&W-Designed 177 FA Plants," B&W Fuel Company, Lynchburg, Virginia, March, 1994.

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

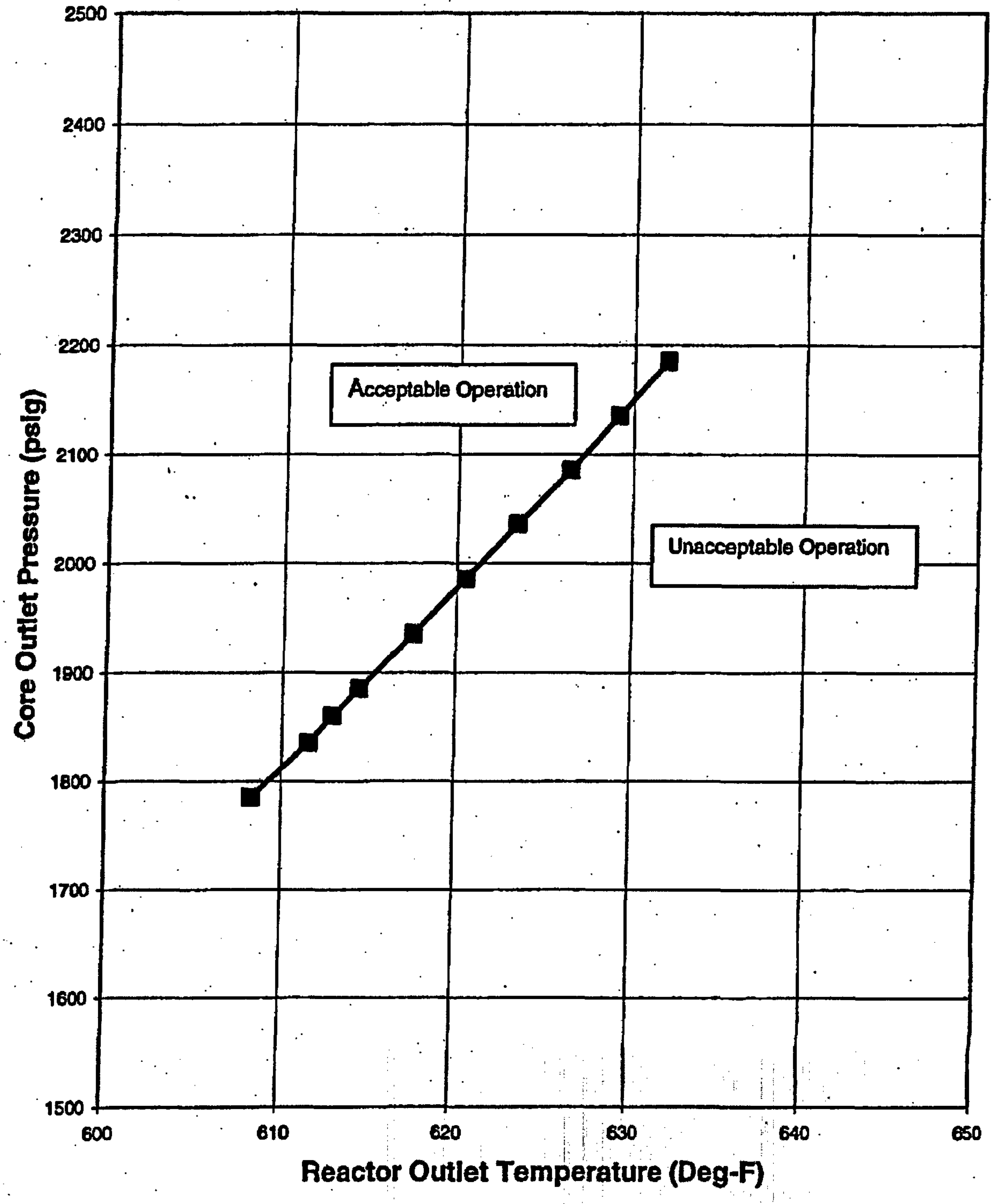
The reactor coolant system (Reference 1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure (Reference 2). The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established (Reference 2). The maximum settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by the presence of a pressurizer electromatic relief valve (Reference 3).

References

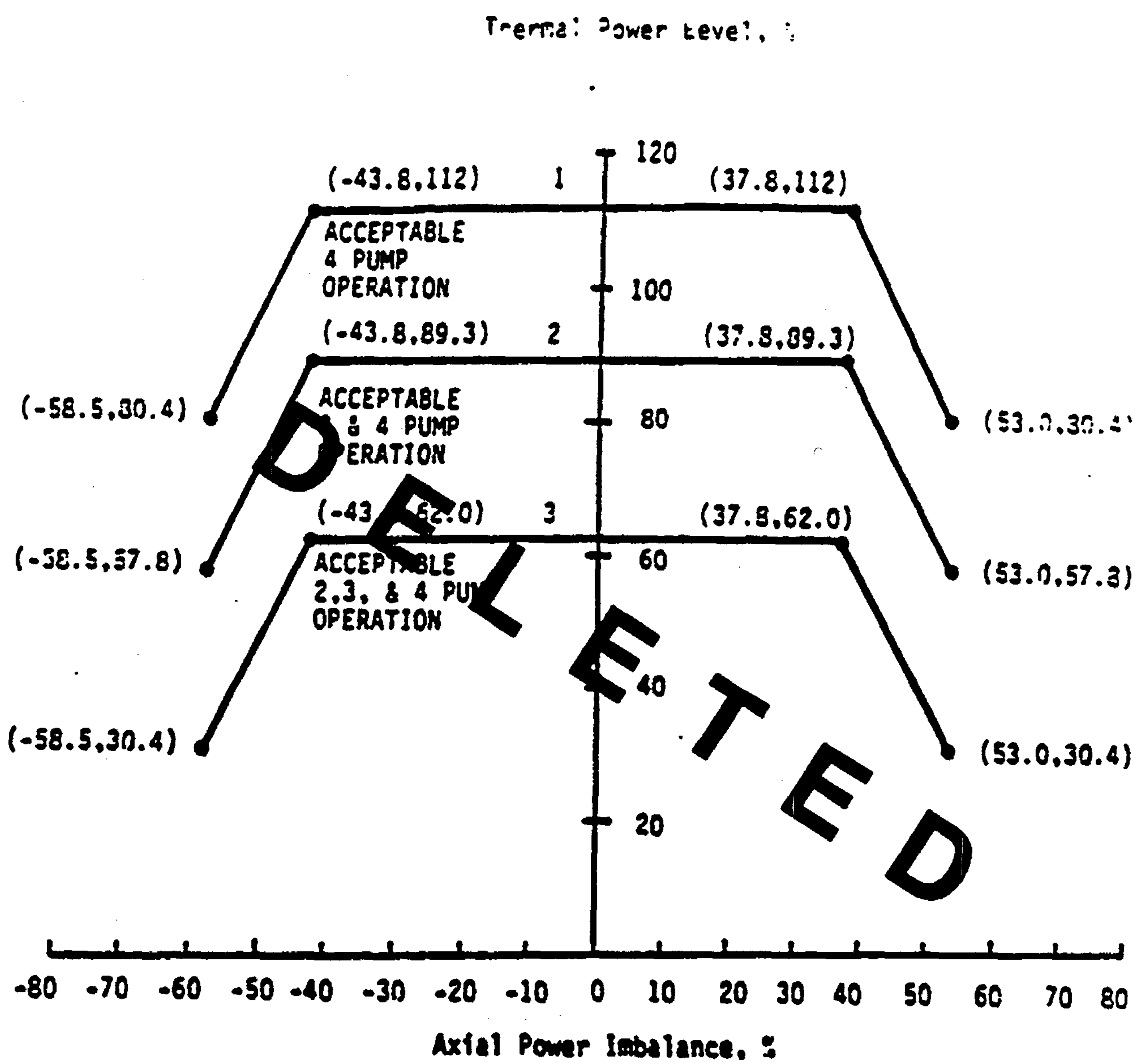
(1) UFSAR, Section 4.0 - "Reactor Coolant System"

(2) UFSAR, Section 4.3.10 - "Safety Limits and Conditions"

(3) UFSAR, Table 4.2-8 - "Reactor Coolant System Pressure Settings"



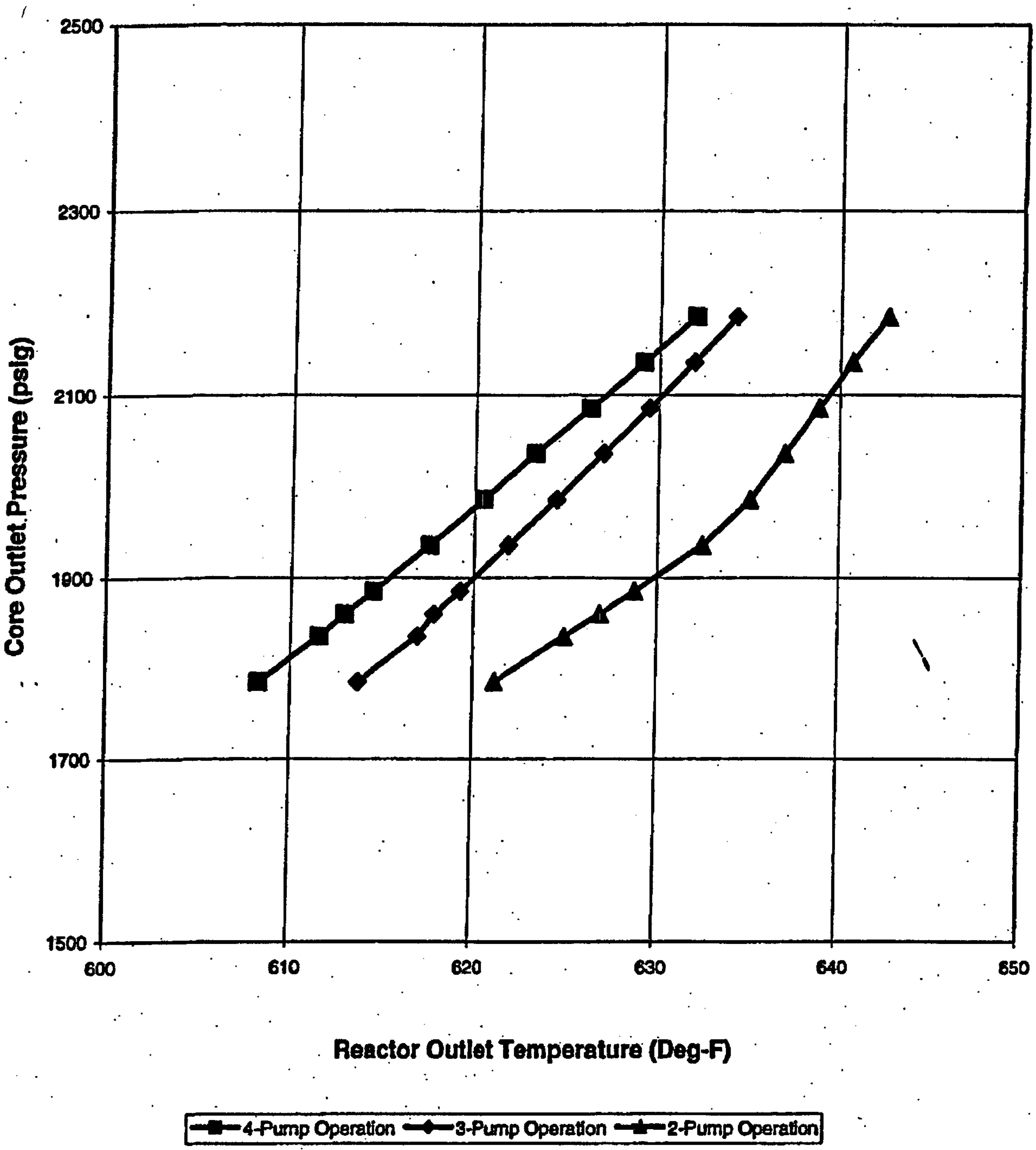
CORE PROTECTION SAFETY LIMIT
TMI-1
FIGURE 2.1-1



Curve	Reactor Coolant Flow (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

CORE PROTECTION SAFETY LIMITS
TMI-1

Figure 2.1-2



RC Pumps	Reactor Coolant Flow (lbs/hr)	Power	Pumps Operating (Type of Limit)
4	137.77X10[6]	112%	Four Pumps (DNBR Limit)
3	See COLR	See COLR	Three Pumps (DNBR Limit)
2	See COLR	See COLR	One Pump In Each Loop (DNBR Limit)

CORE PROTECTION SAFETY BASES
TMI-1
FIGURE 2.1-3

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, axial power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoints for Axial Power Imbalance as given in the COLR.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. These trip setpoints are setting limits on the setpoint side of the protection system bistable comparators. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operations with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.1% of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis (Reference 1).

a. Overpower trip based on flow and imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power to flow ratio is adequate to prevent a DNBR of less than the Statistical Design Limit of 1.313 (BWC) should a low flow condition exist due to any malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power to flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1 are given in the COLR.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage.

For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking Kw/ft limits or DNBR limits. The axial power imbalance (power in the top half of the core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below the Statistical Design Limit of 1.313 (BWC) by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (Reference 2). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated over-pressurization transients (Reference 3). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure and variable low pressure trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (References 4, 5, and 6). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. The revised low pressure trip of 1900 psig and the variable low pressure (16.25 T_{out} - 8113) trip setpoint prevent the minimum core DNBR from decreasing below the Statistical Design Limit of 1.313 (BWC). Figure 2.3-1 shows the high pressure, low pressure, high temperature and variable low pressure trip setpoints.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (618.8F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520° to 620°F. The trip setpoint of the channel is 618.8F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1.2°F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10% above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) UFSAR, Section 1.4.6 - "Criterion 6 - Reactor Core Design"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

(3) "Justification for Raising Setpoint for Reactor Trip on High Pressure," BAW-1890, Rev. 0, Babcock and Wilcox, September 1985.

(4) UFSAR, Section 14.1.2.7 - "Stuck-Out, Stuck-In, or Dropped Control Rod Accident"

(5) UFSAR, Section 14.1.2.9 - "Steam Line Break"

(6) UFSAR, Section 14.3, Reference 28 - "ECCS Analysis of B&W's 177-FA Lowered Loop NNS," BAW-10103-A, Rev. 3, Babcock and Wilcox, Lynchburg, Virginia, July 1977.

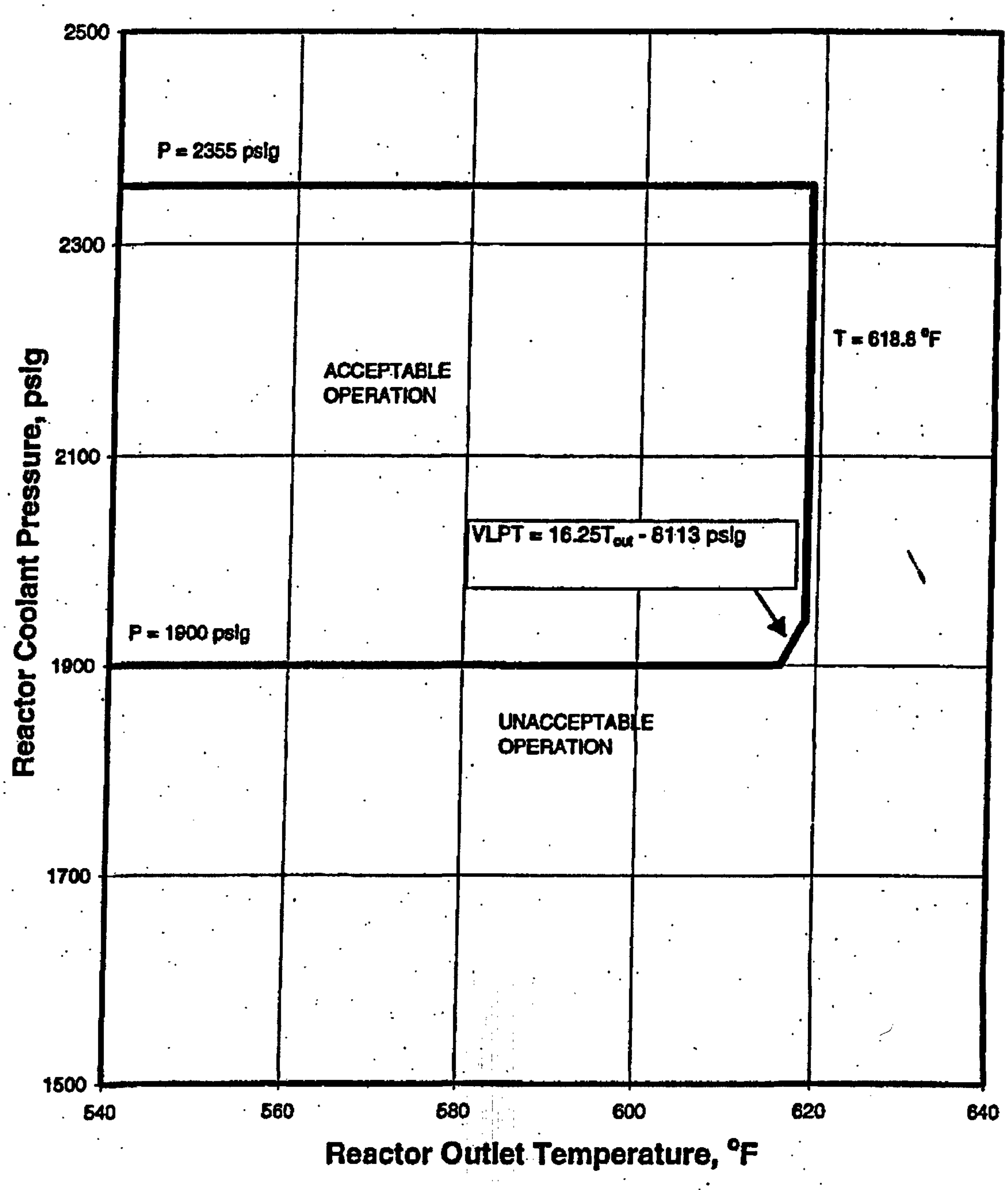
(7) UFSAR, Section 14.1.2.6 - "Loss of Coolant Flow"

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS (5)

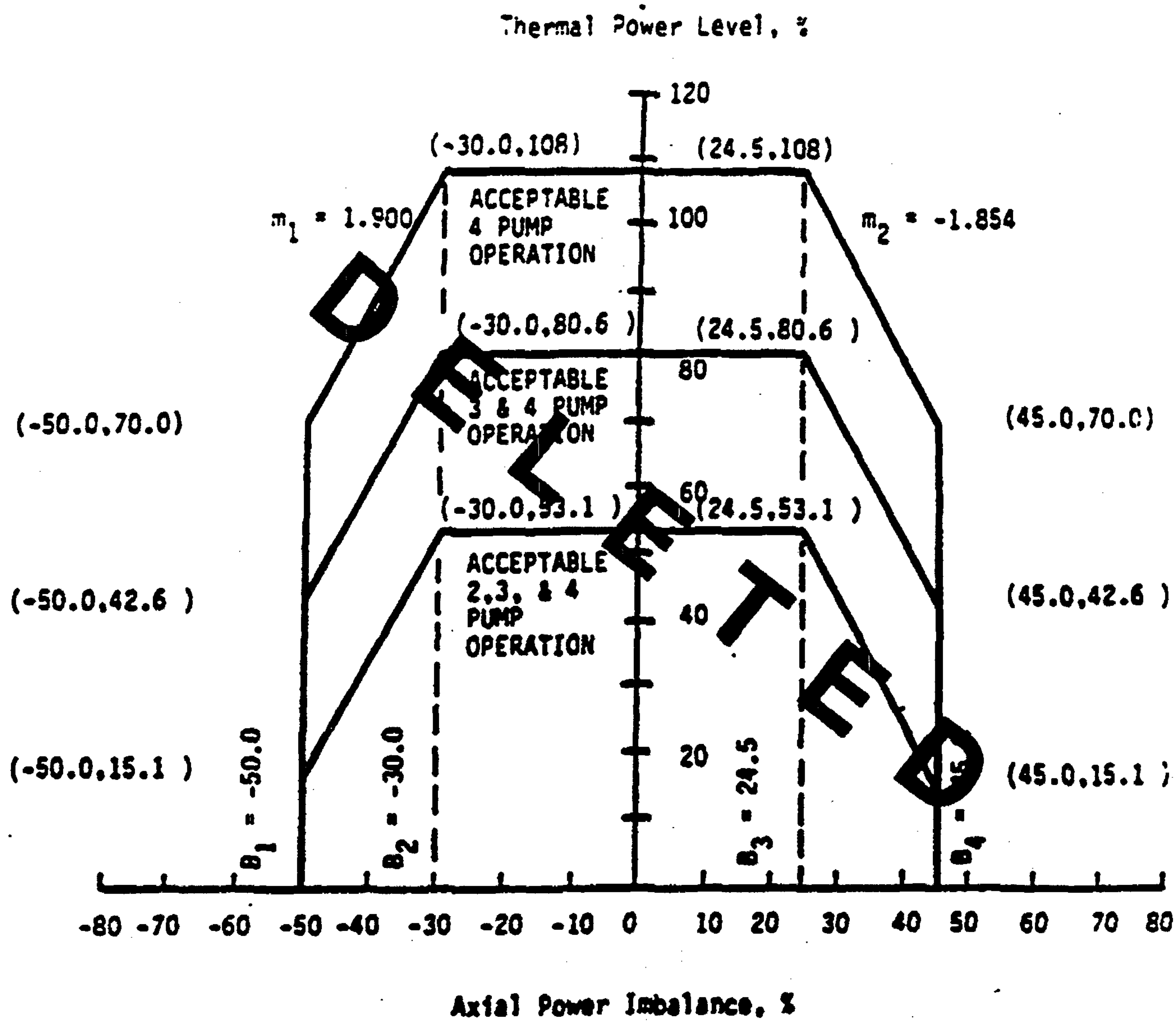
	Four Reactor Coolant Pumps Operating (Nominal Operating) <u>Power – 100%</u>	Three Reactor Coolant Pumps Operating (Nominal Operating) <u>Power - 75%</u>	One Reactor Coolant Pump Operating in Each Loop (Nominal <u>Operating Power - 49%)</u>	Shutdown <u>Bypass</u>
1. Nuclear power, max. % of rated power	105.1	105.1	105.1	5.0(2)
2. Nuclear power based on flow (1) and imbalance max. of rated power	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (4) on pump monitors max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Reactor coolant temp. F., max.	618.8	618.8	618.8	618.8
7. High Reactor Building pressure, psig max.	4	4	4	4
8. Variable low reactor coolant system pressure, psig min.	(16.25 T_{out} - 8113)(6)	(16.25 T_{out} - 8113)(6)	(16.25 T_{out} - 8113)(6)	Bypassed

(1) Reactor coolant system flow, %
(2) Administratively controlled reduction set during reactor shutdown.
(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(5) Trip settings limits are limits on the setpoint side of the protection system bistable connectors.
(6) T_{out} is in degrees Fahrenheit (F).



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
TMI-1
FIGURE 2.3-1

Amendment No. ~~13, 17, 28, 39, 45,~~
~~78, 126, 135, 142, 167,~~ 247



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS FOR
AXIAL POWER IMBALANCE
TMI-1

Figure 2.3-2

SECTION 3.0

LIMITING CONDITIONS FOR OPERATION

3. LIMITING CONDITIONS FOR OPERATION

3.0 GENERAL ACTION REQUIREMENTS

3.0.1 When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in :

1. At least HOT STANDBY within the next 6 hours.
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.

Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.

BASES

This specification delineates the action to be taken for circumstances not directly provided for in the action requirements of individual specifications and whose occurrence would violate the intent of the specification.

3.1 REACTOR COOLANT SYSTEM

3.1.1 OPERATIONAL COMPONENTS

Applicability

Applies to the operating status of reactor coolant system components.

Objective

To specify those limiting conditions for operation of reactor coolant system components which must be met to ensure safe reactor operations.

Specification

3.1.1.1 Reactor Coolant Pumps

a. Pump combinations permissible for given power levels shall be as shown in Specification Table 2.3.1.

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

c. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

a. Both steam generators shall be operable whenever the reactor coolant average temperature is above 250°F.

3.1.1.3 Pressurizer Safety Valves

a. The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ± 1%.

b. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature (References 1, 2, and 3).

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents (Reference 4). The pressurizer code safety valve lift set point shall be set at 2500 psig $\pm$1% allowance for error. **Surveillance requirements are specified in the Inservice Testing Program. Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with ASME Section XI (Reference 5) and the assumptions of TMI-1 safety analysis.**

References

(1) UFSAR, Tables 9.5-1 and 9.5-2

(2) UFSAR, Sections 4.2.5.1 and 9.5 – "Decay Heat Removal"

(3) UFSAR, Section 4.2.5.4 – "Secondary System"

(4) UFSAR, Section 4.3.10.4 – "System Minimum Operational Components"

(5) UFSAR, Section 4.3.7 – "Overpressure Protection"

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until 29 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding 29 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specification submitted in accordance with 10 CFR 50, Appendix G.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional temperature step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 240°F indicated RCS temperature.

The unirradiated reference nil ductility temperature (RT_{NDT}) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in Reference No. 5.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at 29 effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and for the limiting weld metals (SA-1526 & WF-25).

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes and corresponding fluence accumulations, based on predicted fuel cycle design conditions through 29 effective full power years of operation are described in Reference 6.

Based on the predicted RT_{NDT} after 29 effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established by FTI calculation, Reference No. 7, in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2 and ASME Code Section XI, Appendix G, as modified by ASME Code Case N-640 and N-588. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figure 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in RCS "A" loop hot leg. RCS "A" is most conservative and bounds use of "B".

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits.

(5) FTI Calculation No. 32-5011059-00, "TMI-1 Reactor Vessel Adjusted RTNDT Values for 23 and 29 EFPY."

(6) FTI Calculation No. 86-5010023-00, "TMI Cycle 5-11 Final Report."

(7) FTI Calculation No. 32-5011638-02, "TMI-1 29 EFPY P/T Limits."

Figure 3.1-1
Reactor Coolant System Heatup/Cooldown Limitations
[Applicable through 29 EFPY]

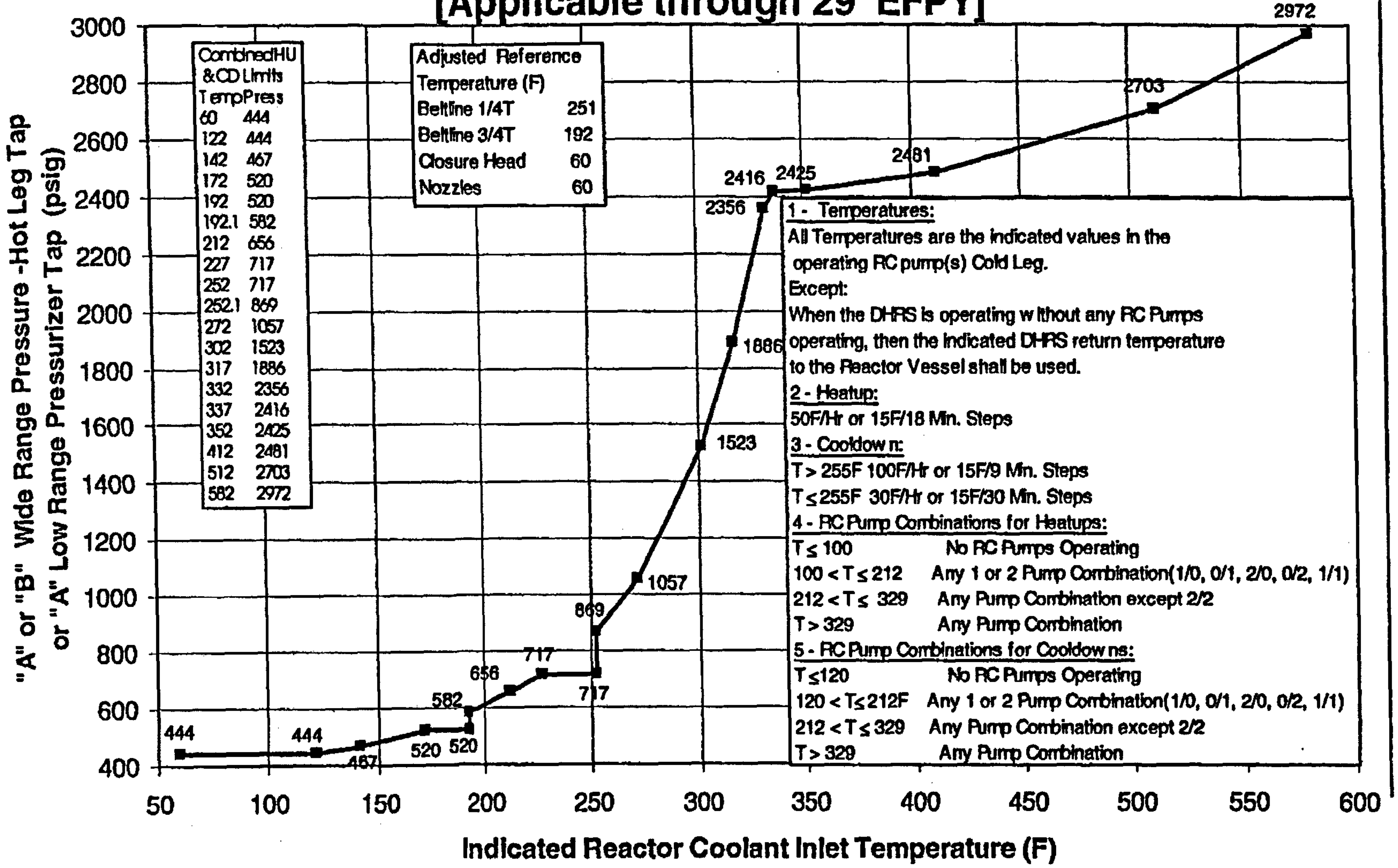
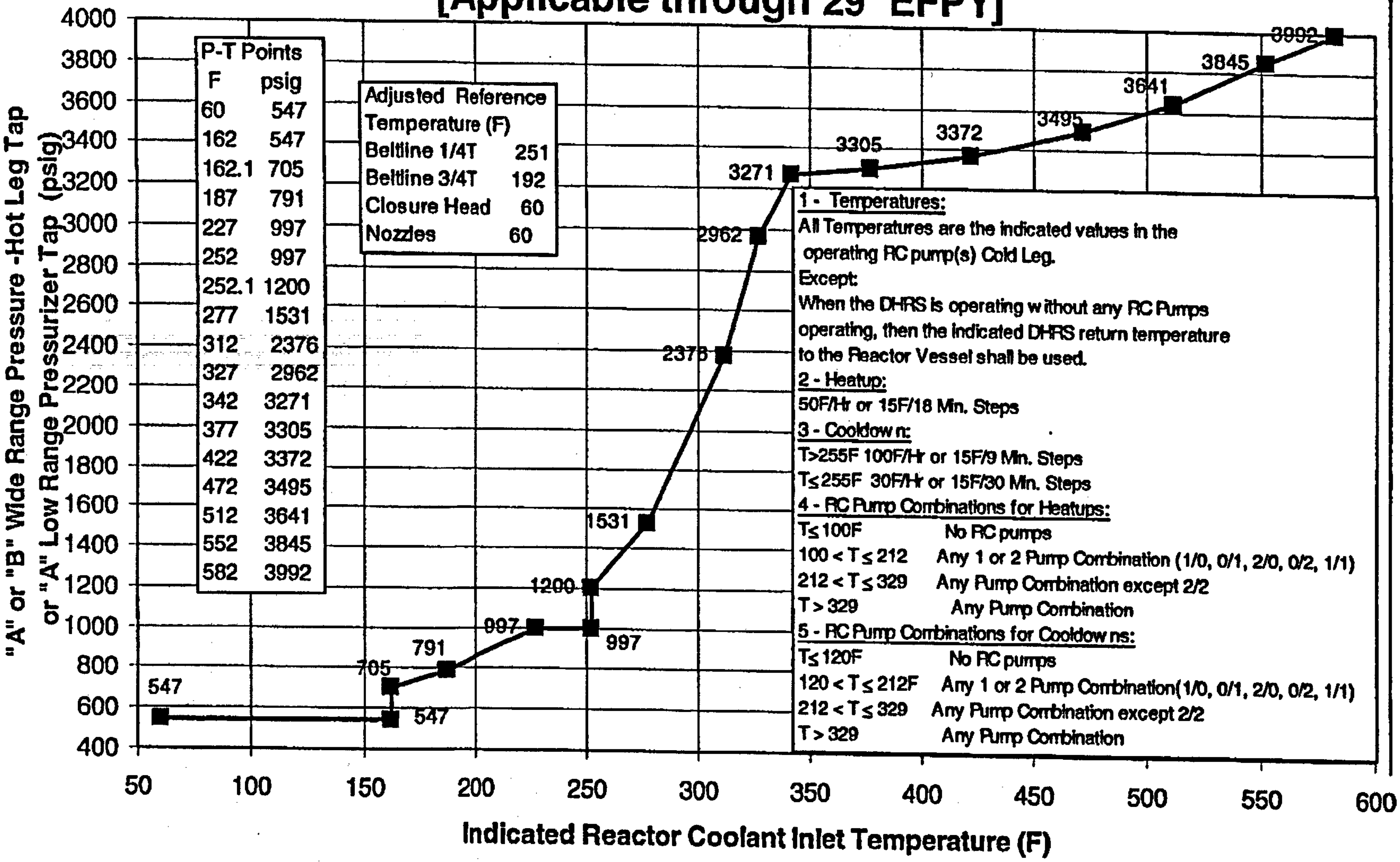


Figure 3.1-2
Reactor Coolant Inservice Leak Hydrostatic Test
[Applicable through 29 EFPY]



P-T Points	
F	psig
60	547
162	547
162.1	705
187	791
227	997
252	997
252.1	1200
277	1531
312	2376
327	2962
342	3271
377	3305
422	3372
472	3495
512	3641
552	3845
582	3992

Adjusted Reference Temperature (F)	
Beltline 1/4T	251
Beltline 3/4T	192
Closure Head	60
Nozzles	60

1 - Temperatures:
All Temperatures are the indicated values in the operating RC pump(s) Cold Leg.
Except:
When the DHRS is operating without any RC Pumps operating, then the indicated DHRS return temperature to the Reactor Vessel shall be used.

2 - Heatup:
50F/Hr or 15F/18 Min. Steps

3 - Cooldown:
T>255F 100F/Hr or 15F/9 Min. Steps
T≤255F 30F/Hr or 15F/30 Min. Steps

4 - RC Pump Combinations for Heatups:

T≤100F	No RC pumps
100 < T ≤ 212	Any 1 or 2 Pump Combination (1/0, 0/1, 2/0, 0/2, 1/1)
212 < T ≤ 329	Any Pump Combination except 2/2
T > 329	Any Pump Combination

5 - RC Pump Combinations for Cooldowns:

T≤120F	No RC pumps
120 < T ≤ 212F	Any 1 or 2 Pump Combination(1/0, 0/1, 2/0, 0/2, 1/1)
212 < T ≤ 329	Any Pump Combination except 2/2
T > 329	Any Pump Combination

Amendment ~~25, 134, 167, 176, 208~~, 234

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

c. To assure sufficient pressurizer heater capacity to maintain natural circulation conditions during a loss of offsite power.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 Pressurizer

3.1.3.4.1 The reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.

3.1.3.4.2 A minimum of 107 kw of pressurizer heaters, from each of two pressurizer heater groups shall be OPERABLE. Each OPERABLE 107 kw of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the established manual transfer scheme.

(a) With the pressurizer inoperable due to one (1) inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

(b) With the pressurizer inoperable due to two (2) inoperable emergency power supplies to the pressurizer heaters either restore the inoperable emergency power supplies within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

3.1.3.5 Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality with the following exceptions:

a. Inoperable rod per 3.5.2.2.

b. Physics Testing per 3.1.9.

c. Shutdown margin may not be reduced below 1% Δ k/k per 3.5.2.1.

d. Exercising rods per 4.1.2.

Following safety rod withdrawal, the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to deboration.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. Calculations show that above 525°F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent delta k/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient and the small integrated delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT+10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The availability of at least 107 kw in pressurizer heater capability is sufficient to maintain primary system pressure assuming normal system heat losses. Emergency power to heater groups 8 or 9, supplied via a manual transfer scheme, assures redundant capability upon loss of offsite power.

The requirements that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This requirement does not prohibit rod withdrawal when the reactor will remain more than 1% dk/k shutdown with the rod(s) withdrawn (e.g., rod latch verification).

The requirements for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

3.1.4.1 LIMITING CONDITION FOR OPERATION

The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.35 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram*

3.1.4.2 APPLICABILITY: at all times except refueling

3.1.4.3 ACTION:

MODES: Power Operation, Start-Up, Hot Standby

a. With the specific activity of the primary coolant greater than 0.35 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours** during one continuous time interval or exceeding the limit line shown on Figure 3.1-2a, be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when DOSE EQUIVALENT I-131 is below 0.35 microcurie/gram.

b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when primary coolant activity is less than $100/\bar{E}$ microcuries/gram.

MODES: At all times except refueling.

c. With the specific activity of the primary coolant greater than 0.35 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram perform the sampling and analysis requirements of Table 4.1-3 until the specific activity of the primary coolant is restored to within its limits.

Bases

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will be well within the Part 100 limit following a steam generator tube rupture accident or steam line break accident with postulated accident induced steam generator tube leakage in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based on a parametric evaluation by the NRC of typical site locations. These values are conservative, in that the specific site parameters of TMI-1, such as site boundary, location and meteorological conditions, were not considered in this evaluation.

* $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

** The time period begins from the time the sample is taken.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.35 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.1-2a, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Proceeding to HOT SHUTDOWN prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

The NRC staff has performed a generic analysis of airborne radiation released via the Reactor Building Purge Isolation Valves. The dose contribution due to the radiation contained in the air and steam released through the purge isolation valves prior to closure was found to be acceptable provided that the requirements of Specifications 3.1.4.1, 3.1.4.2 and 3.1.4.3 are met.

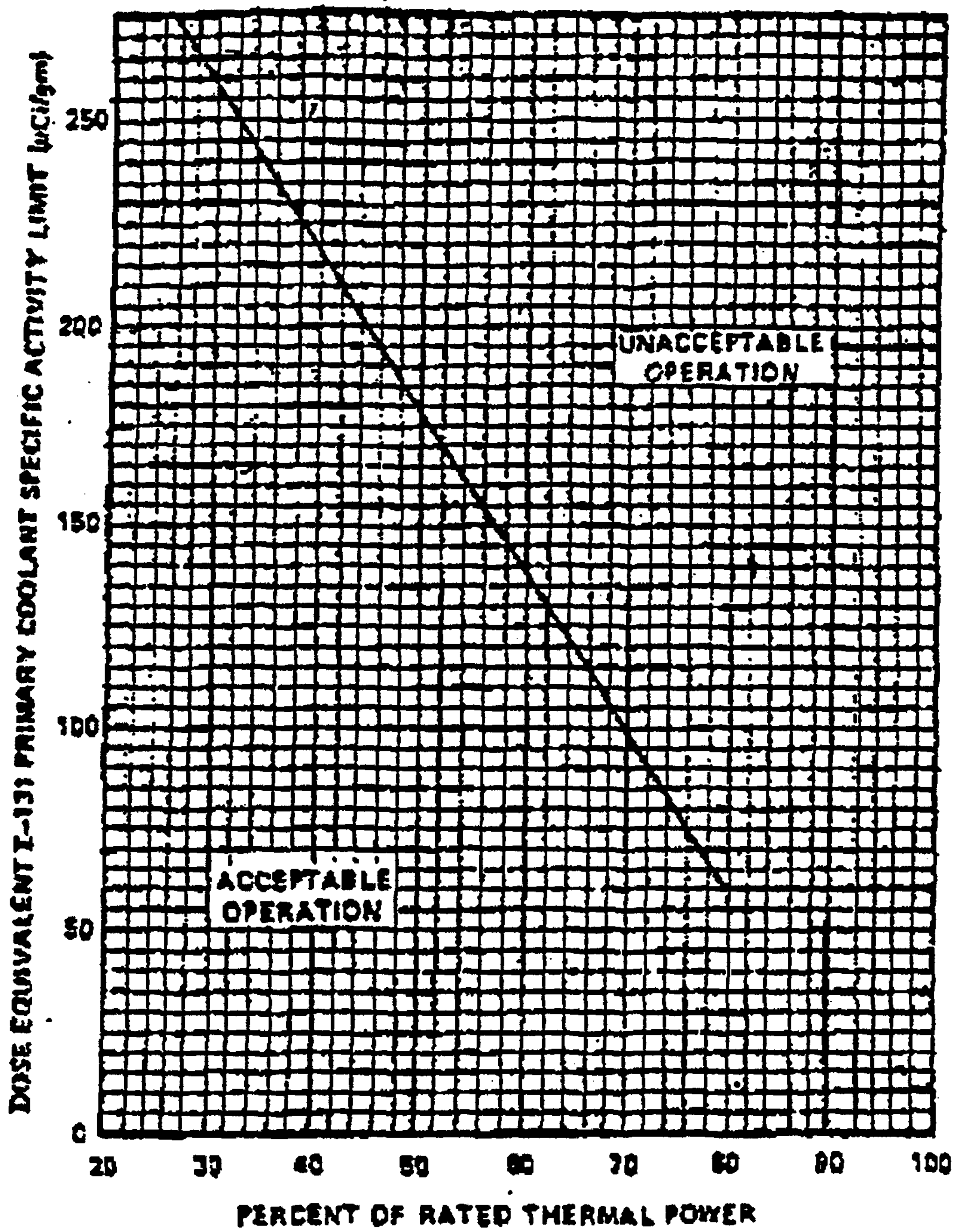


FIGURE 3.1-2a

Dose equivalent I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER (with the Primary Coolant Specific Activity >0.35 µCi/gram Dose Equivalent 1-131).

Amendment No. ~~108~~, ~~167~~, 204

3.1.5 CHEMISTRY

Applicability

Applies to acceptable concentrations of impurities for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities.

Specification

3.1.5.1 If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to $\leq$ 0.1 ppm.

3.1.5.2 If the concentration of chloride in the primary coolant exceeds 0.15 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to $\leq$ 0.15 ppm.

3.1.5.3 If the concentration of fluorides in the primary coolant exceeds 0.10 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to $\leq$ 0.10 ppm.

3.1.5.4 If the concentration limits for oxygen, chloride or fluoride given in 3.1.5.1, 3.1.5.2, and 3.1.5.3 above are not restored within 24 hours of detection, the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

3.1.5.5 If the oxygen, chloride, or fluoride concentration of the primary coolant system exceeds 1.0 ppm the reactor shall be brought to the hot shutdown condition using normal shutdown procedure and action is to be taken to return the system to within normal operation specifications. If normal operating specifications have not been reached in 12 hours, the reactor will then be brought to a cold shutdown condition.

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack (References 1 and 2).

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical. The requirement that the oxygen concentration not exceed 0.1 ppm during power operation is added assurance that stress corrosion cracks will not occur (Reference 3).

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, or increase the hydrogen concentration in the makeup tank).

Because of the time dependent nature of any adverse effects arising from chlorides, fluorides, or oxygen concentrations in excess of the limits, and because the condition can be corrected, it is unnecessary to shutdown immediately.

The oxygen, chloride, or fluoride limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time (Reference 3). Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analysis are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the hot shutdown condition within 12 hours thereafter and corrective action will continue. If the normal operational limits are not restored within an additional 24 hour period the reactor shall be placed in cold shutdown condition within 24 hours thereafter.

The maximum limit of 1 ppm for the oxygen, chloride, or fluoride concentration that will not be exceeded was selected because these values have been shown to be safe at 550°F (Reference 4). It is prudent to restrict operation to hot shutdown conditions, if these limits are reached.

REFERENCES

(1) UFSAR, Section 9.2 - "Chemical Addition and Sampling System"

(2) UFSAR, Table 9.2-3 - "Reactor Coolant Quality"

(3) Corrosion and Wear Handbook, D.J. DePaul, Editor

(4) Stress Corrosion of Metals, Logan

3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system.

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the dose rate limits of the ODCM.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7, except that such losses when added to leakage shall not exceed 30 gpm. If leakage plus losses exceeds 30 gpm, the reactor shall be placed in HOT SHUTDOWN within 24 hours of detection.

3.1.6.10 Operating conditions of POWER OPERATION, STARTUP and HOT SHUTDOWN apply to the operational status of the high pressure isolation valves between the primary coolant system and the low pressure injection system.

a. During all operating conditions in this specification, all pressure isolation valves listed in Table 3.1.6.1 that are located between the primary coolant system and the LPIS shall function as pressure isolation devices except as specified in 3.1.6.10.b. Valve leakage shall not exceed the amount indicated in Table 3.1.6.1.(a)

b. In the event that integrity of any high pressure isolation check valves specified in Table 3.1.6.1 cannot be demonstrated, reactor operation may continue provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition. (b)

c. If Specification 3.1.6.10.a or 3.1.6.10.b cannot be met, an orderly shutdown shall be accomplished by achieving HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within an additional 30 hours.

Bases

Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactive contamination and required cleanup or, in the case of reactor coolant, it could develop into a still more serious problem and, therefore, the first indications of such leakage will be followed up as soon as practical. The unit's makeup system has the capability to makeup considerably more than 30 gpm of reactor coolant leakage plus losses.

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks.

(a) For the purpose of this specification, integrity is considered to have been demonstrated by meeting Specification 4.2.7.

(b) Motor operated valves shall be placed in the closed position and power supplies deenergized.

Bases (Continued)

Although some leak rates on the order of gallons per minute may be tolerable from a dose point of view, it is recognized that leaks in the order of drops per minute through any of the barriers of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation, and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature and location of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation.

When reactor coolant leakage occurs to the Reactor Building, it is ultimately conducted to the Reactor Building sump. Although the reactor coolant is safely contained, the gaseous components in it escape to the Reactor Building atmosphere. There, the gaseous components become a potential hazard to plant personnel, during inspection tours within the Reactor Building, and to the general public whenever the Reactor Building atmosphere is periodically purged to the environment.

When reactor coolant leakage occurs to the Auxiliary Building, it is collected in the Auxiliary Building sump. The gases escaping from reactor coolant leakage within the Auxiliary Building will be collected in the Auxiliary and Fuel Handling Building exhaust ventilation system and discharged to the environment via the unit's Auxiliary and Fuel Handling Building vent. Since the majority of this leakage occurs within confined, separately ventilated cubicles within the Auxiliary Building, it incurs very little hazard to plant personnel.

In regard to the surveillance specification 4.2.7, the isolation valves may be tested at a reduced pressure in accordance with the Franklin Research Center Report titled "Primary Coolant System Pressure Isolation Valves for TMI-1" (FRC Task 212) dated October 24, 1980, Section 2.2.2.

When reactor coolant leakage occurs to the nuclear services closed cooling water system, the leakage, both gaseous and liquid, is contained because the nuclear services closed cooling water system surge tank is a closed tank that is maintained above atmospheric pressure. The leakage would be detected by the nuclear services closed cooling water system monitor and by purge tank liquid level, both of which alarm in the control room. Since the nuclear services closed cooling water system's only potential contact with reactor coolant is in the sample coolers, it is considered not to be a hazard. However, if reactor coolant leakage to this receptor occurred and the surge tank's relief valve discharged, radioactive gases could be discharged to the environment via the unit's auxiliary and fuel handling building vent.

Bases (Continued)

When reactor coolant leakage occurs to the intermediate cooling closed cooling water system, the leakage is indicated by both the intermediate cooling water monitor (RM-L9) and the intermediate cooling closed cooling water surge tank liquid level indicator, both of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank.

When reactor coolant leakage occurs to either of the decay heat closed cooling water systems, the leakage is indicated by the affected system's radiation monitor (RM-L2 or RM-L3 for system A and B, respectively) and surge tank liquid level indicator, all four of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank of the affected system.

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the dose rate limits of the ODCM would be exceeded.

When the reactor coolant leaks to the secondary sides of either steam generator, all the gaseous components and a very small fraction of the ionic components are carried by the steam to the main condenser. The gaseous components exit the main condenser via the unit's vacuum pump which discharges to the condenser vent past the condenser off-gas monitor. The condenser off-gas monitor will detect any radiation, above background, within the condenser vent.

However, buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

Bases (Continued)

If reactor coolant leakage is to the containment, it may be identified by one or more of the following methods:

a) The containment radiation monitor is a three channel monitor consisting of a particulate channel, an iodine channel, and a gaseous channel. All three channels read out in the Control Room and alarm to indicate an increase in containment activity.

   The containment particulate channel is sensitive to the presence of Rb-88, a daughter product of Kr-88, in the containment air sample. Since this activity originates predominantly in the Reactor Coolant System, an increase in monitor readings could be indicative of increasing RCS leakage. The sensitivity of the particulate monitor is such that a leakrate of less than 1 gpm will be detected within one (1) hour under normal plant operating conditions.

b) The mass balance technique is a method of determining leakage by stabilizing the Reactor Coolant System and observing the change in water inventory over a given time period. Level decreases in the Makeup Tank may also serve as an early indication of abnormal leakage.

c) The Reactor Building sump receives leakage from systems inside containment. Sump level readings are checked and recorded regularly for rate of water accumulation. High accumulation rates alert the operators to increase their surveillance of possible leak sources. Level is detected in one-half inch increments which correspond to a volume of ~56 gallons.

d) Deleted.

The leakage detection capability provided by the above methods can be used to determine potential pressure boundary faults. Such leakage, while tolerable from a dose point of view, could be indicative of material degradation which if not dealt with promptly, could develop into larger leaks.

This specification is concerned with leakage from the Reactor Coolant System (RCS) and Makeup and Purification System (MUPS). The methods discussed above provide a means of detecting, as early as possible, leakage which could be the result of a fault in the reactor coolant system pressure boundary. The primary method used at TMI-1 for quantifying RCS and MUPS leakage is the mass balance technique.

Bases (Continued)

The unidentified leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Total reactor coolant leakage is limited by this specification to 10 gpm. This limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage.

The primary to secondary leakage through the steam generator tubes is limited to 1 gpm total. This limit ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of a steam line break. Steam generator leakage is quantified by analysis of secondary plant activity.

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. Water inventories around the auxiliary building sump.

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by TMI-1 Plant Operations.

TABLE 3.1.6.1

PRESSURE ISOLATION CHECK VALVES BETWEEN THE PRIMARY COOLANT SYSTEM & LPIS

System	Valve No.	Maximum(a) Allowable Leakage
Low Pressure Injection		($\leq$5.0 GPM for all valves)
Train A	CF-V5A DH-V22A	($\leq$5.0 GPM for all valves)
Train B	CF-V5B DH-V22B	($\leq$5.0 GPM for all valves)

Footnote:

(a)

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

Applicability

Applies to maximum positive moderator temperature coefficient of reactivity at full power conditions.

Objective

To assure that the moderator temperature coefficient stays within the limits calculated for safe operation of the reactor.

Specification

3.1.7.1 The moderator temperature coefficient shall not be positive at power levels above 95% of rated power.

3.1.7.2 The moderator temperature coefficient shall be $\leq +0.9\times10^{-4}$ delta k/k/F at power levels $\leq$ 95% of rated power.

Bases

A non-positive moderator coefficient (Reference 1) at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ delta k/k/F. All other accident analyses as reported in the UFSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ delta k/k/F.

A non-positive moderator coefficient at power levels above 95% of rated power is also required to prevent overpressurization of the reactor coolant system in the event of a feedwater line break (see Specification 2.3.1, Basis C, Reactor Coolant System Pressure).

The Final Acceptance Criteria states that post-LOCA clad temperature will not exceed 2200°F (Reference 2.)

REFERENCES

(1) UFSAR, Section 3.2.2.1.5.4 - "Moderator Temperature Coefficient"
(2) UFSAR, Section 14 - Tables 14.2-1, 14.2-13, 14.2-14

3.1.8 Single Loop Restrictions

Applicability

Applies to single loop operation of the reactor coolant system

Specification

3.1.8.1 Single loop operation while the reactor is critical is prohibited.

Bases

The restriction prohibiting single loop operation with TMI-1 may be lifted, provided that: (1) analyses of TMI-1 support single loop operation, (2) testing on TMI-1 supports the analysis of single loop operation, and (3) any additional equipment necessary for single loop operation is installed at TMI-1.

3.1.9 LOW POWER PHYSICS TESTING RESTRICTIONS

Applicability

Applies to Reactor Protection System requirements for low power physics testing.

Objective

To assure an additional margin of safety during low power physics testing.

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protection System Requirements

a. Below 1720 psig Shutdown Bypass trip setting limits shall apply in accordance with Table 2.3-1.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.2 Startup Rate Rod Withdrawal Hold (Reference 1) Shall be operable At All Times.

3.1.9.3 Shutdown margin may not be reduced below 1% delta k/k per 3.5.2.1.

Bases

The above specification provides additional safety margins during low power physics testing, as is also provided for startup (Reference 2.)

REFERENCES

(1) UFSAR, Section 7.2.2.1.b - "Reactivity Rate Limits"

(2) UFSAR, Section 14.1.2.2 - "Startup Accident"

This page intentionally left blank

3.1.11 REACTOR INTERNALS VENT VALVES

Applicability

Applies to Reactor Internals Vent Valves

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specifications

3.1.11.1 The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.16.

Amendment No. ~~42~~,167
(8-16-78)

3.1.12 Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 LTOP Protection

If the reactor vessel head is installed and indicated RCS temperature is ≤ 329°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed, and

b. Pressurizer level is maintained ≤ 100 inches. If pressurizer level is > 100 inches, restore level to ≤ 100 inches within 1 hour.

3.1.12.2 The PORV settings shall be as follows:

a. Low Temperature Overpressure Protection Setpoint

1. When indicated RCS temperature is ≤ 329°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and

2. The PORV will have a maximum lift setpoint of 552 psig.

With the PORV setpoint above the maximum value, within 8 hours either:

1. restore the setpoint below the maximum value, or

2. verify pressurizer level is ≤ 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.3 allowing the PORV to be taken out of service.

b. Unless the Low Temperature Overpressure Protection Setpoint is in effect, the PORV lift setpoint will be a minimum of 2425 psig.

With the PORV setpoint below the minimum value, within 8 hours either:

1. restore the setpoint above the minimum value, or

2. close the associated block valve, or

3. close the PORV, and remove power from PORV

4. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.1.12.3 When the indicated RCS temperature is below 329°F the PORV shall not be taken out of service, nor shall it be isolated from the system unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out.
b. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed.
c. Head of the Reactor Vessel is removed.

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

a. With the PORV Block Valve inoperable, within 1 hour either:

1. restore the PORV Block Valve to OPERABLE status or

2. close the PORV (verify closed) and remove power from the PORV

3. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the PORV block valve inoperable, restore the inoperable valve to OPERABLE status prior to startup from the next COLD SHUTDOWN unless the COLD SHUTDOWN occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a COLD SHUTDOWN occurs within this 90 day period, restore the inoperable valve to OPERABLE status prior to startup for the next fuel cycle.

Bases

If the PORV is removed from service while the RCS is below 329°F, sufficient measures are incorporated to prevent severe overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2. Above 329°F, the PORV setpoint has been chosen to limit the potential for inadvertent discharge or cycling of the PORV. Other action such as removing the power to the PORV has the same effect as raising the setpoint which also satisfies this requirement. There is no upper limit on this setpoint as the Pressurizer Safety Valves (T.S. 3.1.1.3) provide the required overpressure relief.

Below 329°F, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

In both cases, the setting is specified to reflect the nominal value which allows for normal variations in the temperature setpoint while maintaining the tolerances assumed in the bases for T.S. 3.1.2. Either pressure actuation setpoint is acceptable within the temperature range between 313°F and 329°F.

With RCS temperatures less than 329°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 100 inches to allow time for action to prevent severe overpressurization in the event of any single failure.

The PORV block valve is required to be OPERABLE during the HOT STANDBY, STARTUP, and POWER OPERATION in order to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

For protection from severe overpressurization during HPI testing, refer to Section 4.5.2.1.c.

3.1.13 Reactor Coolant System Vents

Applicability

Provides the limiting conditions for operation of the Reactor Coolant System Vents. These limiting conditions for operation (LCO) are applicable only when Reactor is critical.

Objective

To ensure that sufficient vent flow paths are operable during the plant operating modes mentioned above.

Specification

3.1.13.1 At least one reactor coolant system vent path consisting of at least two power operated valves in series, powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head* (RC-V42 & RC-V43)
b. Pressurizer steam space (RC-V28 & RC-V44)
c. Reactor coolant system high point (either RC-V40A and 41A) or (RC-40B and 41B)

Action

3.1.13.2 a. With one of the above reactor coolant system vent paths inoperable, the inoperable vent path shall be maintained closed, with power removed from the valve actuators in the inoperable vent path. The inoperable vent path shall be restored to OPERABLE status within 30 days, or the plant shall be in HOT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above reactor coolant system vent paths inoperable, maintain the inoperable vent path closed, with power removed from the valve actuators in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT SHUTDOWN within an additional 6 hours and in COLD SHUTDOWN within the following 30 hours.

* This specification becomes binding after installation and initially being declared operable.

Bases

The safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plants ability to deal with large quantities of noncondensible gas which could interfere with natural circulation (i.e., core cooling).

The reactor vessel head vent (RC-V42 & RC-V43 in series) provides the capability of venting noncondensible gases from the majority of the reactor vessel head as well as the Reactor Coolant hot legs (to the elevation of the top of the outlet nozzles) and cold legs (through vessel internals leakage paths, to the elevation of the top of the inlet nozzles). This vent is routed to containment atmosphere.

Venting for the pressurizer steam space (RC-V28 and RC-V44 in series) has been provided to assure that the pressurizer is available for Reactor Coolant System pressure and volume control. This vent is routed to the Reactor Coolant Drain Tank.

Additional venting capability has been provided for the Reactor Coolant hot leg high points (RC-V40A, B, RC-41A, B), which normally cannot be vented through the Reactor vessel head vent or pressurizer steam-space vent. These vents relieve to containment atmosphere through a rupture disk (set at low pressure).

The above vent systems are seismically designed and environmentally qualified in accordance with the May 23, 1980 Commission Order and Memorandum per NUREG-0737, Item II.B.1. The high point vents do not fall within the scope of 10 CFR 50.49, since the vents are not relied upon during or following any design basis event (Reference 1). The power operated valves (2 in series in each flow path) which are powered from emergency buses fail closed on loss of power. All vent valves for the reactor vessel head vent, pressurizer vent and loop B high point vent are powered from the class 1E "B" bus. The vent valves for the loop A high point vent are powered from the class 1E "A" bus. The power operated valves are controlled in the Control Room. The individual vent path lines are sized so that an inadvertent valve opening will not constitute a LOCA as defined in 10 CFR 50.46(c)(1). These design features provide a high degree of assurance that these vent paths will be available when needed, and that inadvertent operation or failures will not significantly hamper the safe operation of the plant (Reference 2).

REFERENCES

(1) UFSAR, Section 4.2.3.9 - "Reactor Coolant System Venting"

(2) UFSAR, Section 7.3.2.2.c (16) - "Reactor Coolant System Venting"

3.2 MAKEUP AND PURIFICATION AND CHEMICAL ADDITION SYSTEMS

DELETED

THIS PAGE LEFT BLANK INTENTIONALLY

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability
Applies to the operating status of the emergency core cooling, reactor building emergency cooling, and reactor building spray systems.

Objective
To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The reactor shall not be made critical unless the following conditions are met:

3.3.1.1 Injection Systems

a. The borated water storage tank (BWST) shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

b. Two Makeup and Purification (MU)/High Pressure Injection (HPI) pumps are OPERABLE in the engineered safeguards mode powered from independent essential buses. Specification 3.0.1 applies.

c. Two decay heat removal pumps are OPERABLE. Specification 3.0.1 applies.

d. Two decay heat removal coolers and their cooling water supplies are OPERABLE. (See Specification 3.3.1.4) Specification 3.0.1 applies.

e. Two BWST level instrument channels are OPERABLE.

f. The two reactor building sump isolation valves (DH-V-6A/B) shall be remote-manually OPERABLE. Specification 3.0.1 applies

g. **MU Tank (MUT) pressure and level shall be maintained within the Unrestricted Operating Region of Figure 3.3-1.**

  1) **With MUT conditions outside of the Unrestricted Operating Region of Figure 3.3-1, restore MUT pressure and level to within the Unrestricted Operating Region within 72 hrs. Specification 3.0.1 applies.**

  2) **Operation with MUT conditions within the Prohibited Region of Figure 3.3-1 is prohibited. Specification 3.0.1 applies.**

3.3.1.2 Core Flooding System

a. Two core flooding tanks (CFTs) each containing 940 $\pm$30 ft^3 of borated water at 600 $\pm$25 psig shall be available. Specification 3.0.1 applies.

b. CFT boron concentration shall not be less than 2,270 ppm boron. Specification 3.3.2.1 applies.

c. The electrically operated discharge valves from the CFT will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

d. DELETED

e. CFT vent valves CF-V-3A and CF-V-3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. $\pm$6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 $\pm$.5 weight percent (%). Specification 3.3.2.1 applies.

c. All manual valves in the discharge lines of the NaOH tank shall be locked open. Specification 3.3.2.1 applies.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.*

3.3.2.1 If the CFT boron concentration is outside of limits, or NaOH tank is outside the limits of 3.3.1.3.b or any manual valve in the NaOH tank discharge lines are not locked open, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both CFTs shall be OPERABLE at all times.

b. Both the motor operated valves associated with the CFTs shall be fully open at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be verified to be OPERABLE.

* **In accordance with AmerGen License Change Application dated February 14, 2001, and any requirements in the associated NRC Safety Evaluation, a portion of the Nuclear Service Water System piping between valves NR-V-3 and NR-V-5 may be removed from service and Nuclear Services River Water flow realigned through a portion of the Secondary Services River Water System piping for up to 14 days. This note is applicable for one time use during TMI Unit 1 Operating Cycle 13.**

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location and sump recirculation switchover setpoint. The limits on contained water volume, NaOH concentration and boron concentration ensure a pH value of

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)

Bases (Contd.)

between 8.0 and 11.0 of the solution sprayed within containment after a design basis accident. The minimum pH of 8.0 assures that iodine will remain in solution while the maximum pH of 11.0 minimizes the potential for caustic damage to mechanical systems and components. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

Maintaining MUT pressure and level within the limits of Fig 3.3-1 ensures that MUT gas will not be drawn into the pumps for any design basis accident. Preventing gas entrainment of the pumps is not dependent upon operator actions after the event occurs. The plant operating limits (alarms and procedures) will include margins to account for instrument error.

The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

The iodine removal function of the reactor building spray system requires one spray pump and sodium hydroxide tank contents.

The spray system utilities common suction lines with the decay heat removal system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.2 and 3.3.3 provided requirements in Specification 3.3.4 are met which assure operability of the duplicate components. The specified maintenance times are a maximum. Operability of the specified components shall be based on the satisfactory completion of surveillance and inservice testing and inspection required by Technical Specification 4.2 and 4.5.

The allowable maintenance period of up to 72 hours may be utilized if the operability of equipment redundant to that removed from service is verified based on the results of surveillance and inservice testing and inspection required by Technical Specification 4.2 and 4.5.

In the event that the need for emergency core cooling should occur, operation of one makeup pump, one decay heat removal pump, and both core flood tanks will protect the core. In the event of a reactor coolant system rupture their operation will limit the peak clad temperature to less than 2,200 °F and the metal-water reaction to that representing less than 1 percent of the clad.

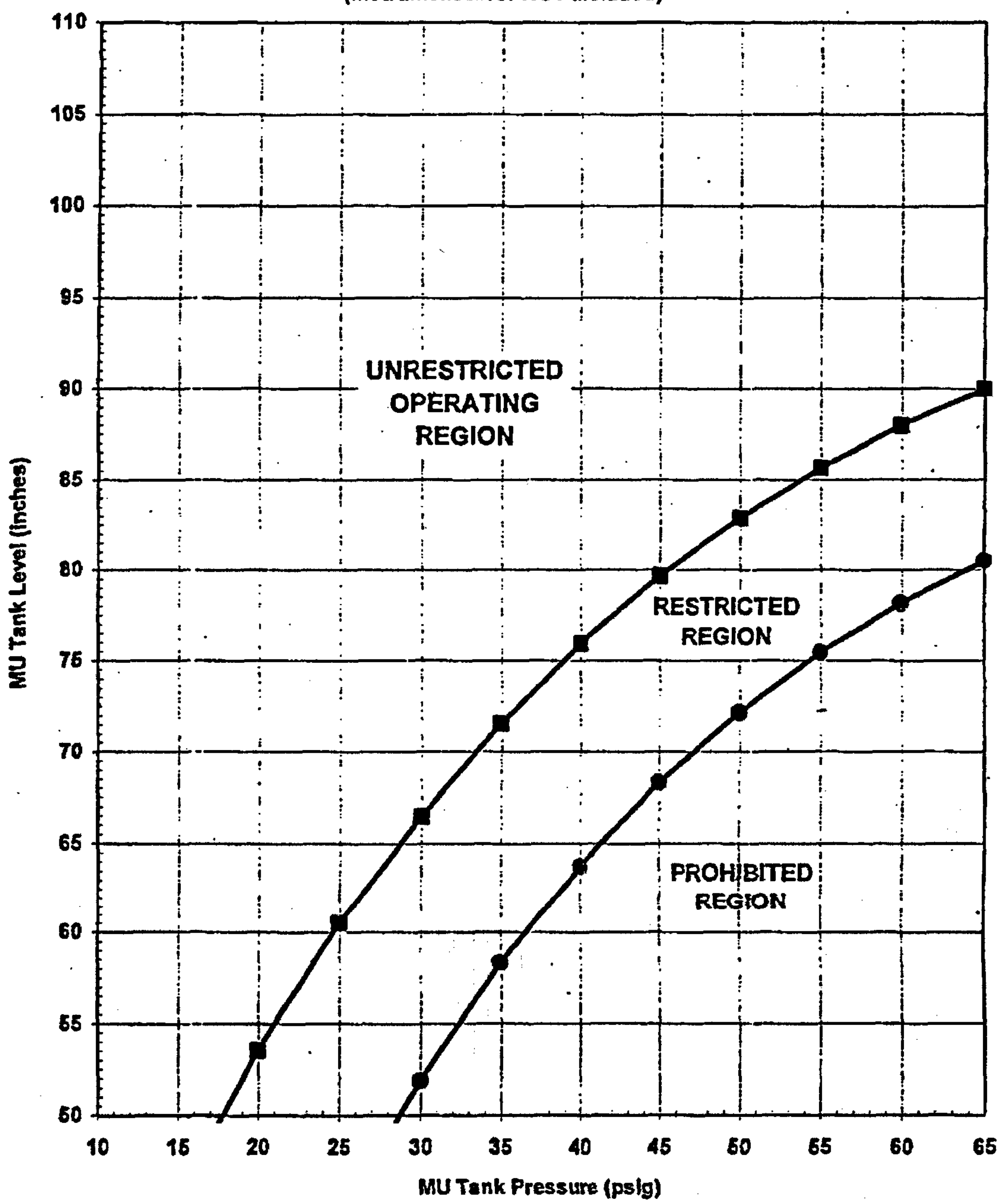
Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

REFERENCES

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

(2) UFSAR, Section 14.2.2.3 - "Large Break LOCA"

FIGURE 3.3-1
Makeup Tank Pressure vs Level Limits
(Instrument Error NOT Included)



3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

*To define the conditions necessary to assure continuous capability of DHR.**

Specification

3.4.1 Reactor Coolant System (RCS) temperature greater than 250 degrees F.

3.4.1.1 Three independent Emergency Feedwater (EFW) Pumps and two redundant flowpaths to each Once Through Steam Generator (OTSG) shall be OPERABLE ** with:

a. Two EFW Pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW Pump capable of being powered from two OPERABLE main steam supply paths.

(1) With one main steam supply path inoperable, restore the inoperable steam supply path to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 12 hours.

(2) With one EFW Pump or any EFW flowpath inoperable, restore the inoperable pump or flowpath to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours.

(3) With one main steam supply path to the turbine-driven EFW Pump and one motor-driven EFW Pump inoperable, restore the steam supply or the motor-driven EFW Pump to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 12 hours.

(4) With more than one EFW Pump or both flowpaths to either OTSG inoperable, initiate action immediately to restore at least two EFW Pumps and one flowpath to each OTSG:

---

* These requirements supplement the requirements of Specifications 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

** HSPS operability is specified in Specification 3.5.1. When HSPS is not required to be OPERABLE, EFW is OPERABLE by manual control of pumps and valves from the Control Room.

Notes:

1. Specification 3.0.1 and all other actions requiring shutdown or changes in REACTOR OPERATING CONDITIONS are suspended until at least two EFW Pumps and one EFW flowpath to each OTSG are restored to OPERABLE status.

2. While performing surveillance testing, more than one EFW Pump or both flowpaths to a single OTSG may be inoperable for up to 8 hours provided that:

   (a) At least one motor-driven EFW Pump shall remain OPERABLE, and

   (b) With the reactor in STARTUP, HOT STANDBY, or POWER OPERATION, a designated qualified individual who is in communication with the control room shall be continuously stationed in the immediate vicinity of the affected EFW local manual valves. On instruction from the Control Room, the individual shall realign the valves from the test mode to their operational alignment.

b. Four of six Turbine Bypass Valves (TBVs) OPERABLE. With more than two TBVs inoperable, restore operability of at least four TBVs within 72 hours.

c. The Condensate Storage Tanks (CSTs) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST.

   (1) With a CST inoperable, restore the CST to operability within 72 hours or be in HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours.

   (2) With more than one CST inoperable, restore at least one CST to OPERABLE status or be subcritical within 1 hour, in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

3.4.1.2.1 With the Reactor between 250 degrees F and HOT SHUTDOWN, and having been subcritical for at least one (1) hour, two (2) Main Steam Safety Valves (MSSVs) per OTSG shall be OPERABLE. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3.4.1.2.2 With the Reactor between HOT SHUTDOWN and 5% power, and having been subcritical for at least one (1) hour, two (2) MSSVs per OTSG shall be OPERABLE provided the overpower trip setpoint in the RPS is set to less than 5% full power. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3.4.1.2.3 Except as provided in Specification 3.4.1.2.2 above, when the Reactor is above HOT SHUTDOWN, all eighteen (18) MSSVs shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of MSSVs Disabled on Any OTSG	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than three (3) MSSVs inoperable, restore at least fifteen (15) MSSVs to OPERABLE status within 4 hours or be in HOT SHUTDOWN within the next 6 hours.

3.4.2 RCS temperature less than or equal to 250 degrees F.

3.4.2.1 At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").

b. DHR String (Loop "B").

c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.

d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

3.4.2.2 Operation of the means for DHR may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

3.4.2.4 Specification 3.4.2.1 does not apply when either of the following conditions exist:

a. Decay heat generation is less than 188 KW with the RCS full.

b. Decay heat generation is less than 100 KW with the RCS drained down for maintenance.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal DHR is by the OTSGs with the steam dump to the condenser when RCS temperature is above 250 degrees F and by the DHR System below 250 degrees F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the OTSG is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the OTSGs is provided by the main feedwater system.

The Emergency Feedwater (EFW) System supplies adequate feedwater to the OTSGs at accident pressures, removing heat from the Reactor Coolant System (RCS) to support safe shutdown of the reactor when the normal feedwater supply is unavailable. EFW is not required for normal plant startup and shutdown.

The turbine-driven EFW Pump and two motor-driven EFW Pumps take suction from the Condensate Storage Tanks (CSTs) and deliver flow to a common discharge header. Flowpath redundancy is provided for those portions of the EFW flowpath containing active components between the pumps and each of the OTSGs. Each EFW line to an OTSG includes two redundant flowpaths, each equipped with an automatic control valve (EF-V-30A/B/C/D) and a manual isolation valve (EF-V-52A/B/C/D). Each redundant flowpath is capable of providing adequate flow to the associated OTSG. Heat removed from the OTSGs returns to the Main Condenser through the Turbine Bypass Valves (TBVs) or discharges to the atmosphere through the Main Steam Safety Valves (MSSVs) and/or the Atmospheric Dump Valves (ADVs). An unlimited supply of river water to the EFW Pumps is available using either of the two Reactor Building Emergency Cooling Water (Reactor River Water) Pumps (RR-P-1A/B).

Redundant main steam supply paths are provided to the turbine-driven EFW Pump for certain events involving loss of one steam supply (e.g., main steam and feedwater line breaks). An operable Main Steam supply path delivers steam to the turbine-driven EFW Pump upon HSPS actuation or by operator action from the control room when HSPS is not required. During low pressure conditions, additional steam supply paths from Main Steam (MS-V-10A/B) or Auxiliary Steam can be made available to the turbine-driven EFW Pump as necessary.

During design basis events the EFW System can withstand any single active failure and still perform its function. The limiting design basis accident for the EFW System is a loss of feedwater event with off-site power available. In the event of a loss of all AC power, which assumes multiple single failures, the turbine-driven EFW Pump alone delivers the necessary EFW flow. Consideration of additional failures in the EFW System or Heat Sink Protection System (HSPS) is not required for this event. Additionally, the EFW System capabilities are sufficient to deliver the required flow in licensing basis events (e.g., ATWS failure to trip events, Generic Letter 81-14 seismic events, and the Station Blackout event).

The most limiting EFW flow requirement is met when at least two EFW Pumps are operable and at least one EFW flowpath to each OTSG is operable. When three pumps and two flowpaths to each OTSG are operable, the EFW System can withstand any single active failure. Examples of single active failures include: failure of any one EFW Pump to actuate, failure of one HSPS train to actuate, or failure of one redundant flowpath to either OTSG. Initially after a shutdown, any two EFW Pumps are required to remove RCS heat with one pump eventually sufficing as the decay heat production rate diminishes.

Bases (Continued)

If EFW were required during surveillance testing, minor operator action (e.g., opening a local isolation valve or manipulating a control switch from the control room) may be needed to restore operability of the required pumps or flowpaths. An exception to permit more than one EFW Pump or both EFW flowpaths to a single OTSG to be inoperable for up to 8 hours during surveillance testing requires 1) at least one motor-driven EFW Pump operable, and 2) an individual involved in the task of testing the EFW System must be in communication with the control room and stationed in the immediate vicinity of the affected EFW flowpath valves. Thus the individual is permitted to be involved in the test activities by taking test data and his movement is restricted to the area of the EFW Pump and valve rooms where the testing is being conducted.

The allowed action times are reasonable, based on operating experience, to reach the required plant operating conditions from full power in an orderly manner and without challenging plant systems. Without at least two EFW Pumps and one EFW flowpath to each OTSG operable, the required action is to immediately restore EFW components to operable status, and all actions requiring shutdown or changes in Reactor Operating Condition are suspended. With less than two EFW pumps or no flowpath to either OTSG operable, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, which might result in a trip. The seriousness of this condition requires that action be started immediately to restore EFW components to operable status. TS 3.0.1 is not applicable, as it could force the unit into a less safe condition.

The EFW system actuates on: 1) loss of all four Reactor Coolant Pumps, 2) loss of both Main Feedwater Pumps, 3) low OTSG water level, or 4) high Reactor Building pressure. A single active failure in the HSPS will neither inadvertently initiate the EFW system nor isolate the Main Feedwater system. OTSG water level is controlled automatically by the HSPS system or can be controlled manually, if necessary.

The MSSVs will be able to relieve to atmosphere the total steam flow if necessary. Below 5% power, only a minimum number of MSSVs need to be operable as stated in Specifications 3.4.1.2.1 and 3.4.1.2.2. This is to provide OTSG overpressure protection during hot functional testing and low power physics testing. Additionally, when the Reactor is between hot shutdown and 5% full power operation, the overpower trip setpoint in the RPS shall be set to less than 5% as is specified in Specification 3.4.1.2.2. The minimum number of MSSVs required to be operable allows margin for testing without jeopardizing plant safety. Plant specific analysis shows that one MSSV is sufficient to relieve reactor coolant pump heat and stored energy when the reactor has been subcritical by 1% delta K/K for at least one hour. Other plant analyses show that two (2) MSSVs on either OTSG are more than sufficient to relieve reactor coolant pump heat and stored energy when the reactor is below 5% full power operation but had been subcritical by 1% delta K/K for at least one hour subsequent to power operation above 5% full power. According to Specification 3.1.1.2a, both OTSGs shall be operable whenever the reactor coolant average temperature is above 250 degrees F. This assures that all four (4) MSSVs are available for redundancy. During power operations at 5% full power or above, if MSSVs are inoperable, the power level must be reduced, as stated in Specification 3.4.1.2.3 such that the remaining MSSVs can prevent overpressure on a turbine trip.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY (Continued)

Bases (Continued)

The minimum amount of water in the CSTs required by Specification 3.4.1.1.c, provides at least 12 hours of DHR with steam being discharged to the atmosphere. This provides adequate time to align alternate water sources for RCS cooldown. After cooling to 250 degrees F, the DHR System is used to achieve further cooling.

When the RCS temperature is below 250 degrees F, a single DHR String (Loop), or single OTSG with an EFW Pump and a flowpath capable of supporting natural circulation is sufficient to provide removal of decay heat at all times following the cooldown to 250 degrees F. The DHR String (Loop) redundancy required by Specification 3.4.2.1 is achieved with independent active components capable of maintaining the RCS subcooled. A single DHR flowpath with redundant active components is sufficient to meet the requirements of Specifications 3.4.2.1.a and 3.4.2.1.b. The requirement to maintain two operable means of DHR ensures that a single active failure does not result in a complete loss of DHR capability. The requirement to keep a DHR Loop in operation as necessary to maintain the RCS subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

With the Reactor Vessel head removed and 23 feet of water above the Reactor Vessel flange, a large heat sink is available for core cooling. In this condition, only one DHR Loop is required to be operable because the volume of water above the Reactor Vessel flange provides a large heat sink which would allow sufficient time to recover active DHR means.

Following extensive outages or major core off-loading, the decay heat generation being removed from the Reactor Vessel is so low that ambient losses are sufficient to maintain core cooling and no other means of heat removal is required. The system is passive and requires no redundant or diverse backup system. Decay heat generation is calculated in accordance with ANSI 5.1-1979 to determine when this situation exists (Reference 4).

REFERENCES

(1) UFSAR, Table 6.1-4 - ECCS "Single Failure Analysis"

(2) UFSAR, Section 9.5 - "Decay Heat Removal System"

(3) UFSAR, Section 10.6 - "Emergency Feedwater System"

(4) TMI Unit 1 Calculation C-3320-85-001, "RCS Decay Heat Removal-Ambient Losses," Revision 0, February 28, 1985

3.5 INSTRUMENTATION SYSTEMS

3.5.1 OPERATIONAL SAFETY INSTRUMENTATION

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5-1, Column "A" and "B" are met, except as provided in Table 3.5-1, Column "C". Specification 3.0.1 applies.

3.5.1.2 The key operated channel bypass switch associated with each reactor protection channel may be used to lock the reactor trip module in the untripped state as indicated by a light. Only one channel shall be locked in this untripped state at any one time. Unit operation at rated power shall be permitted to continue with Table 3.5-1, Column "A". Only one channel bypass key shall be kept in the control room.

3.5.1.3 In the event the number of protection channels operable falls below the limit given under Table 3.5-1, Column "A", operation shall be limited as specified in Column "C". Specification 3.0.1 applies.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation (except for required maintenance or testing).

3.5.1.5 During START-UP when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade.

3.5.1.6 During START-UP, HOT STANDBY or POWER OPERATION, in the event that a control rod drive trip breaker is inoperable, within one hour place the breaker in trip. Specification 3.0.1 applies.

3.5.1.7 During START-UP, HOT STANDBY or POWER OPERATION, in the event that one of the control rod drive trip breaker diverse trip features (shunt trip or undervoltage trip attachment) is inoperable:

a. Restore to OPERABLE status within 48 hours or

b. Within one additional hour place the breaker in trip.

Specification 3.0.1 applies.

3.5.1.7.1 Power may be restored through the breaker with the failed trip feature for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.8 During STARTUP, HOT STANDBY or POWER OPERATION, in the event that one of the two regulating control rod power SCR electronic trips is inoperable, within one hour:

a. Place the inoperable SCR electronic trip in the tripped condition or

b. Remove the power supplied to the associated SCRs. Specification 3.0.1 applies.

3.5.1.8.1 Power may be restored through the SCRs with the failed electronic trip for up to two hours for surveillance testing per T.S. 4.1.1.

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup.

The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 460 volt sources. Redundant trip devices are employed in each of these sources. The AC Trip Breaker is one means to trip a source. The redundant means is a parallel configuration consisting of two DC Trip Breakers and five SCR power supplies. The SCRs are turned off by the "electronic trip relays."

Diverse trip features are provided on each breaker. These are the undervoltage relay and shunt trip attachment. Each trip feature is tested separately. Failure of one breaker trip feature does not result in loss of redundancy and a reasonable time limit is provided for corrective action.

Failure in the untripped state of a breaker or SCR electronic trip results in loss of redundancy and prompt action is required. Failure of both trip features on one breaker is considered failure of the breaker.

Power may be restored through the failed breaker (SCRs) for a limited time to perform required testing.

The 4.16kv ES Bus Undervoltage Relays detect a degraded voltage or Loss of Voltage on the associated ES Bus. Detection of low voltage will separate the ES bus from the offsite power, initiate load shedding and start the associated diesel generator. The relays do not function during design basis events where acceptable offsite voltage is available. If the voltage relays on either train are not operable, the time permitted for repair is consistent with other safety related equipment. If both trains are affected then shutdown is initiated in accordance with Specification 3.0.1 since automatic response of the diesel generator is required to assure completion of the safety function if offsite power is degraded or lost.

Automatic initiation of EFW is provided on loss of all reactor coolant pumps, loss of both main feedwater pumps, low OTSG level, and high reactor building pressure. High reactor building pressure would be indicative of a loss of coolant accident, main steam line or feedwater line break inside the reactor building. Operability of these instruments is required in order to assure that the EFW system will actuate and control at the appropriate OTSG level without operator action for those events where timely initiation of EFW is required.

Automatic isolation of main feedwater is provided on low OTSG pressure in order to maintain appropriate RCS cooling (minimize overcooling) following a loss of OTSG integrity and minimize the energy released to the Reactor Building atmosphere.

HSPS instrument operability specified meets the single failure criterion for the EFW system. Four instrument channels are provided for automatic EFW initiation on OTSG low level and high reactor building pressure, and for automatic main feedwater isolation on low OTSG pressure. Normal trip logic is two out of four. With one of the 4 channels in bypass, a second channel may be taken out of service (placed in the tripped position) and no single active failure will prevent actuation of the associated HSPS train actuation logic. No single active failure of either HSPS train will prevent the other HSPS train from operating to supply EFW to both OTSGs.

REFERENCE

(1) B&W Report No. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," Rev. 0, October 1985

TABLE 3.5-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. <u>Reactor Protection System</u>			
1. Manual pushbutton	1	0	(a)
2. Power range instrument channel	2	1	(a)
3. Intermediate range instrument channels	1	0	(a) (b)
4. Source range instrument channels	1	0	(a) (c)
5. Reactor coolant temperature instrument channels	2	1	(a)
6. Reactor Coolant Pressure-Temperature Instrument channels	2	1	(a)
7. Flux / imbalance / flow	2	1	(a)
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	(a)
b. Low reactor coolant pressure instrument channels	2	1	(a)

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
A. Reactor Protection System (cont'd)			
9. Power/number of pumps instrument channels	2	1	(a)
10. High reactor building pressure channels	2	1	(a)

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, source range instrumentation is not required.

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
B. <u>Other Reactor Trips</u>			
1. Loss of Feedwater (c)	2	1	(a)
2. Turbine Trip (c)	2	1	(b)

(a) Restore the conditions of Column (A) and Column (B) within one hour or reduce indicated reactor power to less than 7% within an additional 6 hours.

(b) Restore the conditions of Column (A) and Column (B) within one hour or reduce indicated reactor power to less than 45% within an additional 6 hours.

(c) Trip may be defeated during low power physics tests.

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
C. Engineered Safety Features			
1. Makeup and Purification System (high pressure injection mode)			
a. Reactor Coolant Pressure Instrument Channels	2	1(b)	(a)
b. Reactor Building 4 psig Instrument Channels	2	1(b)	(a)
c. Manual Pushbutton (also actuates Low Pressure Injection)	2	N/A	(g)
2. Decay Heat System (low pressure injection mode)			
a. Reactor Coolant Pressure Instrument Channels	2	1(b)	(a)
b. Reactor Building 4 psig Instrument Channels	2	1(b)	(a)
c. Reactor Coolant Pressure D.H. Valve Interlock Bistable	1	0	Open circuit breaker at MCC for DH-V1 or DH-V2 with the affected valve in the closed position within 4 hours or maintain R.C. pressure less than 350 psig.

AMENDMENT NO 70, 78, 124, 189, 2003-32

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
C. Engineered Safety Features (cont'd)			
3. Reactor Building Isolation and Cooling System			
a. Reactor Bldg. 4 psig Instrument Channel	2	1(b)	(a)
b. Manual Pushbuttons			
i. 4 psig feature	2	N/A	(g)
ii. 30 psig feature	2	N/A	(g)
c. Deleted			
d. Reactor Building 30 psig pressure switches	2	1	(c)
e. RCS Pressure less than 1600 psig	2	1(b)	(a)
f. Reactor Building Purge Line Isolation (AH-V1A and AH-V1D) High Radiation	1	0	(f)
4. Reactor Building Spray System			
a. Reactor Building 30 psig pressure switches	2	1	(d)
b. Spray Pump Manual Switches	2	N/A	(g)
5. 4.16KV ES Bus Undervoltage Relays			
a. Degraded Grid Voltage Relays	2	1	(e)
b. Loss of Voltage Relay	2	1	(e)

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

C. <u>Engineered Safety Features (cont'd)</u>

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the reactor in HOT SHUTDOWN within an additional 6 hours and COLD SHUTDOWN within the following 24 hours.

(b) The minimum degree of redundancy may be reduced to 0 up to 8 hours for surveillance testing.

(c) The Operability requirement is two out of three pressure switches in each train, with a minimum degree of redundancy of one, in each train.

1. If the minimum conditions are not met on one train, restore the function to OPERABLE within 48 hours, or place the reactor in HOT SHUTDOWN within 6 hours.

2. If the minimum conditions are not met on either train, then place the reactor in HOT SHUTDOWN in 6 hours and in COLD SHUTDOWN within the following 30 hours.

(d) The Operability requirement is two out of three pressure switches in each train, with a minimum degree of redundancy of one, in each train.

1. If the minimum conditions are not met on one train, restore the function to OPERABLE within 72 hours, or place the reactor in HOT SHUTDOWN within 6 hours.

2. If the minimum conditions are not met on either train, then place the reactor in HOT SHUTDOWN in 6 hours and COLD SHUTDOWN within the following 24 hours.

(e) The operability requirement for the undervoltage relay, its associated auxiliary relays, and the timer

1. If one 4.16 kv ES Bus does not meet the minimum conditions, restore the function to operable status within 72 hours or be in hot shutdown within an additional 6 hours.

2. If both 4.16 kv Buses do not meet the minimum conditions, then restore at least one 4.16 kv ES Bus to meet the minimum conditions within 1 hour or be in hot shutdown within an additional 6 hours.

(f) Discontinue Reactor Building purging and close AHV-1A, 1B, 1C, and 1D within 8 hours.

INSTRUMENTS OPERATING CONDITIONS

C. Engineered Safety Features (cont'd)

(g) The Operability requirement is for the manual actuation switch for the specified feature on each train to be OPERABLE.

1. If the manual actuation switch on one train is inoperable, restore the switch to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 6 hours.

2. If both manual actuation switches for that feature are inoperable, then place the reactor in HOT SHUTDOWN in 6 hours and COLD SHUTDOWN within the following 24 hours.

Amendment No. 189

TABLE 3.5-1 (Cont'd)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot Be Met
D. Heat Sink Protection System			
1. EFW Auto Initiation			
a. Loss of both Feedwater pumps	N/A(b)	N/A(b)	(a)
b. Loss of all RC Pumps	N/A(b)	N/A(b)	(a)
c. OTSG A Low Level	2	1	(a)
d. OTSG B Low Level	2	1	(a)
e. High Reactor Building Pressure	2	1	(a)
2. MFW Isolation			
a. OTSG A Low Pressure	2	1	(a)
b. OTSG B Low Pressure	2	1	(a)
3. EFW Level Control			
a. OTSG A Level Control	N/A(b)	N/A(b)	(a)
b. OTSG B Level Control	N/A(b)	N/A(b)	(a)

(a) Restore the conditions of Column (A) and Column (B) within 72 hours, or place the unit in HOT SHUTDOWN within the next 12 hours.

(b) Operability requirements are specified in Section 3.5.1.9.

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall not be less than one percent $\Delta K/K$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 in the safety or regulating rod banks shall not be permitted. Verify SDM $\geq$ 1% $\Delta k/k$ or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent $\Delta k/k$ hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, and cannot be aligned per 3.5.2.2.f, power shall be reduced to $\leq$ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to $\leq$ 70% of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F^N_{\Delta H}$) limits per the COLR have not been exceeded within 72 hours.

Amendment No. ~~17~~, ~~150~~, ~~211~~, 246
(5-18-76)

f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation may continue provided that within 1 hour the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2.

g. If the inoperable rod in Paragraph "e" above is in groups 5, 6, 7, or 8, the other rods in the group may be trimmed to the same position. Normal operation of 100 percent of the thermal power allowable for the reactor coolant pump combination may then continue provided that within 1 hour the rod that was declared inoperable is maintained within allowable group average position limits in 3.5.2.5.

3.5.2.3 The worth of single inserted control rods during criticality is limited by the restriction of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.

3.5.2.4 Quadrant Tilt:

a. Except for physics tests, the quadrant tilt, as determined using the full incore system (FIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

The FIS is OPERABLE for monitoring quadrant tilt provided the number of valid symmetric string individual SPND signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore system is not OPERABLE and except for physics tests quadrant tilt as determined using the power range channels for each quadrant (out of core detector system)(OCD), shall not exceed the values in CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests, quadrant tilt as determined using the minimum incore system (MIS), shall not exceed the values in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if quadrant tilt exceeds the tilt limit, allowable power shall be reduced 2 percent for each 1 percent tilt in excess of the tilt limit. For less than four pump operation, thermal power shall be reduced 2 percent below the thermal power allowable for the reactor coolant pump combination for each 1 percent tilt in excess of the tilt limit.

e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

1. Verify $F_Q(Z)$ and $F^N_{\Delta H}$ are within limits of the COLR once per 2 hours and restore QPT to $\leq$ steady state limit within 24 hours, or perform steps 2, 3, & 4 below.

2. The protection system reactor power/imbalance envelope trip setpoints shall be reduced 2 percent in power for each 1 percent tilt, in excess of the tilt limit, or when thermal power is equal to or less than 50% full power with four reactor coolant pumps running, set the nuclear overpower trip setpoint equal to or less than 60% full power.

3. The control rod group withdrawal limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

4. The operational imbalance limits in the CORE OPERATING LIMITS REPORT shall be reduced 2 percent in power for each 1 percent tilt in excess of the tilt limit.

f. Except for physics or diagnostic testing, if quadrant tilt is in excess of the maximum tilt limit defined in the CORE OPERATING LIMITS REPORT and using the applicable detector system defined in 3.5.2.4.a, b, and c above, reduce thermal power to ≤15% FP within 2 hours. Diagnostic testing during power operation with a quadrant tilt is permitted provided that the thermal power allowable is restricted as stated in 3.5.2.4.d above.

g. Quadrant tilt shall be monitored on a minimum frequency of once every 12 hours when the QPT alarm is inoperable and every 7 days when the alarm is operable during power operation above 15 percent of rated power. When QPT has been restored to ≤ steady state limit, verify hourly for 12 consecutive hours, or until verified acceptable at ≥95% FP.

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ±5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F_{\Delta H}^{N}$ shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

2. If regulating rods are .inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within 2 hours or reduce power to ≤ power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Plant Manager.

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F_{\Delta H}^{N}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

e. If an acceptable axial power imbalance is not achieved within 24 hours, reactor power shall be reduced to ≤40% FP within 2 hours.

f. Axial power imbalance shall be monitored on a minimum frequency of once every 12 hours when axial power imbalance alarm is OPERABLE, and every 1 hour when imbalance alarm is inoperable during power operation above 40 percent of rated power.

3.5.2.8 A power map shall be taken at intervals not to exceed 31 effective full power days using the incore instrumentation detection system to verify the power distribution is within the limits shown in the CORE OPERATING LIMITS REPORT.

Bases

The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. The effects of the APSRs are included in the limit development. Additional conservatism included in the limit development is introduced by application of:

a. Nuclear uncertainty factors

b. Thermal calibration uncertainty

c. Fuel densification effects

d. Hot rod manufacturing tolerance factors

e. Postulated fuel rod bow effects

f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

The incore instrumentation system uncertainties used to develop the axial power imbalance and quadrant tilt limits accounted for various combinations of invalid Self Powered Neutron Detector (SPND) signals. If the number of valid SPND signals falls below that used in the uncertainty analysis, then another system shall be used for monitoring axial power imbalance and/or quadrant tilt.

For axial power imbalance and quadrant power tilt measurements using the incore detector system, the minimum incore detector system consists of OPERABLE detectors configured as follows:

Axial Power Imbalance

a. Three detectors in each of three strings shall lie in the same axial plane with one plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-planes.

c. The detectors shall not have radial symmetry.

Quadrant Power Tilt

a. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

A system of 52 incore flux detector assemblies with seven detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core nuclear instrumentation calibration and for core power distribution verification.

a. The out-of-core instrumentation calibration includes:

   1. Calibrations of the split detectors at initial reactor startup, during the power escalation program, and periodically thereafter.

   2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

   3. Confirmation that the out-of-core axial power splits are as expected.

b. Core power distribution verification includes:

   1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

   2. Subsequent checks during operation to ensure that power distribution is consistent with calculations.

   3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

c. The minimum requirement for 23 individual incore detectors is based on the following:

   1. An adequate axial imbalance indication can be obtained with nine individual detectors. Figure 3.5-1 shows a typical set of three detector strings with three detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

   2. Figure 3.5-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from two detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

   3. Figure 3.5-3 combines Figures 3.5-1 and 3.5-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions.

The 25 ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod bank)

Control rod groups are withdrawn in sequence beginning with group 1. Groups 5,6 and 7 are overlapped 25 percent. The normal position at power is for group 7 to be partially inserted. Group 8 position is maintained consistent with the core design which may include withdrawal of APSRs and long-term operation with APSRs fully withdrawn. When APSR withdrawal is specified in the core design, appropriate limits for time of the withdrawal and restrictions for subsequent APSR insertion are included in the COLR.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (Reference 1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than: 0.65% delta k/k at rated power. These values have been shown to be safe by the safety analysis of the hypothetical rod ejection accident (Reference 2). A maximum single inserted control rod worth of 1.0% delta k/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth 1.0% delta k/k at beginning of life, hot, zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than 0.65% delta k/k ejected rod worth at rated power.

The plant computer will scan for tilt and imbalance and will satisfy the technical specification requirements. If the computer is out of service, then manual calculation for tilt above 15 percent power and imbalance above 40 percent power must be performed as specified until the computer is returned to service.

Reduction of the nuclear overpower trip setpoint to 60% full power when thermal power is equal to or less than 50% full power maintains both core protection and an operability margin at reduced power similar to that at full power.

During the physics testing program, the high flux trip setpoints are administratively set as follows to assure an additional safety margin is provided:

Test Power	Test Setpoint
0	<5%
≤80	90%
>80	105.1%

REFERENCES

(1) *UFSAR, Section 3.2.2.1.2* - "Reactivity Control Distribution"

(2) *UFSAR, Section 14.2.2.2* - "Rod Ejection Accident"

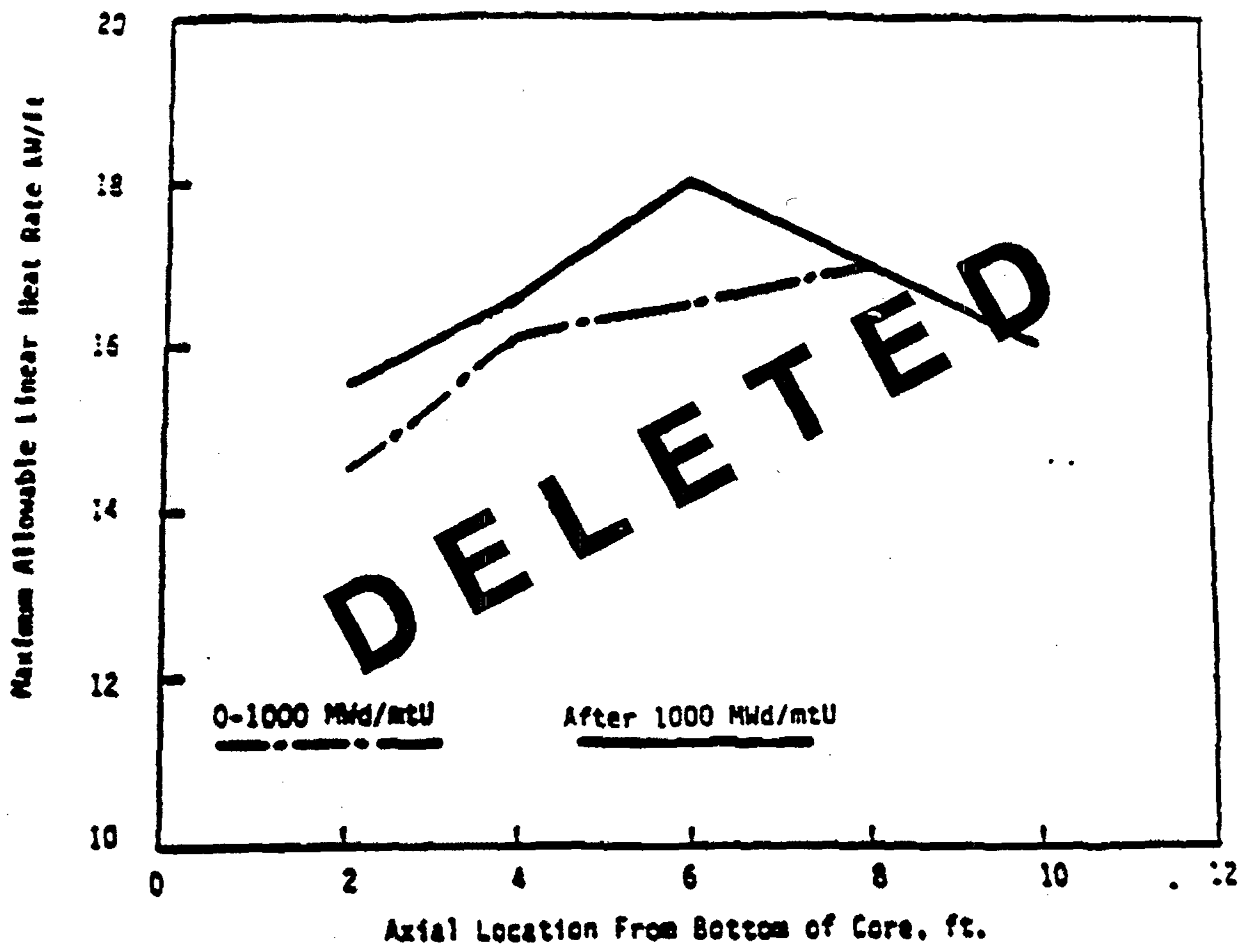


FIGURE DELETED AND INCORPORATED

INTO THE CORE OPERATING LIMITS REPORT.

LOCA LIMITED MAXIMUM
ALLOWABLE LINEAR HEAT RATE

TMI-1

Figure 3.5-2M

Amendment No. ~~142~~, ~~157~~, ~~167~~, 168

3.5.3 ENGINEERED SAFEGUARDS PROTECTION SYSTEM ACTUATION SETPOINTS

Applicability:

This specification applies to the engineered safeguards protection system actuation setpoints.

Objective:

To provide for automatic initiation of the engineered safeguards protection system in the event of a breach of Reactor Coolant System integrity.

Specification:

3.5.3.1 The engineered safeguards protection system actuation setpoints and permissible bypasses shall be as follows:

Initiating Signal	Function	Setpoint
High Reactor Building Pressure (1)	Reactor Building Spray	$\leq$ 30 psig
	Reactor Building Isolation	$\leq$ 30 psig
	High-Pressure Injection	$\leq$ 4 psig
	Low-Pressure Injection	$\leq$ 4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	$\leq$ 4 psig
Low Reactor Coolant System Pressure	High Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Low Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Reactor Building Isolation	$\geq$ 1600 psig(2)
4.16 kv E.S. Buses Undervoltage Relays		
Degraded Voltage	Switch to Onsite Power Source and load shedding	3760 volts (4)
Degraded voltage timer		10 sec (5)
Loss of voltage	Switch to Onsite Power Source and load shedding	2400 Volts (6)
Loss of voltage timer		1.5 sec (7)

(1) May be bypassed for reactor building leak rate test.

(2) May be bypassed below 1775 psig on decreasing pressure and is automatically reinstated before 1800 psig on increasing pressure.

(3) May be bypassed below 925 psig on decreasing pressure and is automatically reinstated before exceeding 950 psig on increasing pressure.

(4) Minimum allowed setting is 3740 v. Maximum allowed setting is 3773 v.

(5) Minimum allowed time is 8 sec. maximum allowed time is 12 sec.

(6) Minimum allowed setting is 2200 volts, maximum allowed setting is 2860 volts

(7) Minimum allowed time is 1.0 second, maximum allowed time is 2.0 seconds.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressures to prevent spurious initiation (Reference 1).

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.. Bypass of HPI below 1775 psig and LPI below 925 psig, prevents ECCS actuation during normal system cooldown (References 1 and 2).

4.16 KV ES Bus Undervoltage Relays

The basis for the degraded grid voltage relay setpoint is to protect the safety related electrical equipment from loss of function in the event of a sustained degraded voltage condition on the offsite power system. The timer setting prevents spurious transfer to the onsite source for transient conditions.

The loss of voltage relay and timers detect loss of offsite power condition and initiate transfer to the onsite source with minimal time delay.

The minimum and maximum degraded voltage setpoint are "as found" readings.

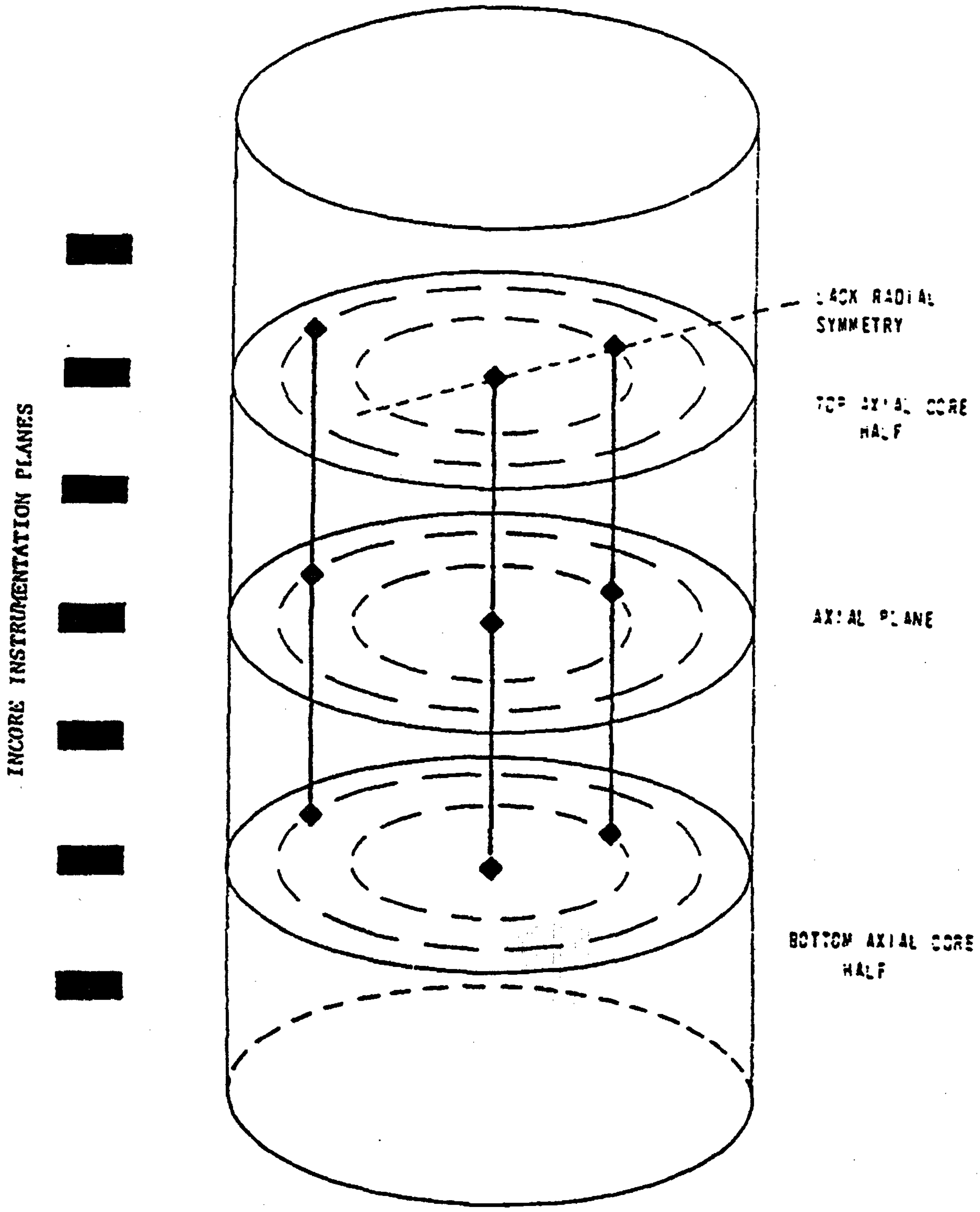
References

(1) UFSAR, Table 7.1-3

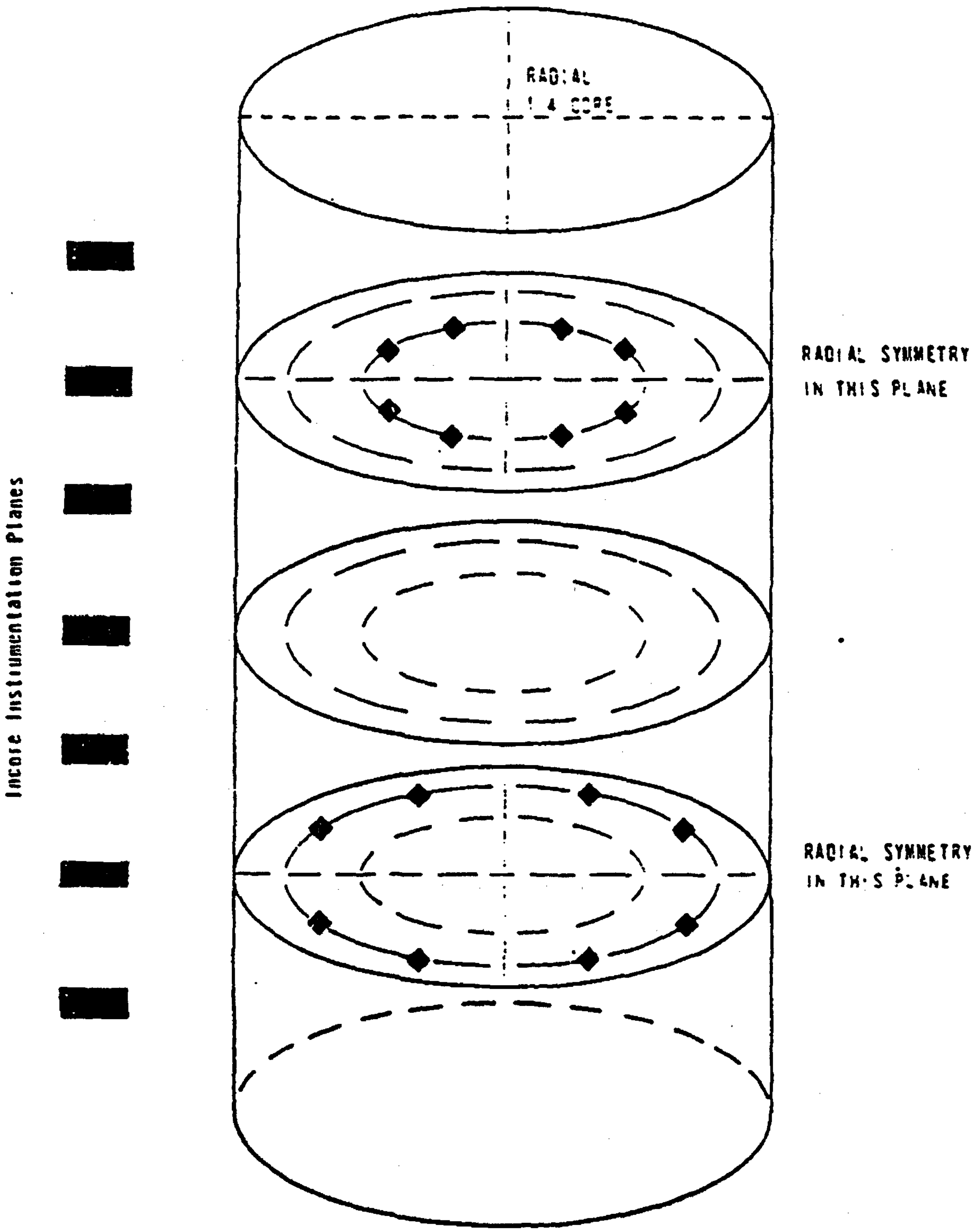
(2) UFSAR, Section 14.1.2.10 - "Steam Generator Tube Failure"

This page intentionally Left Blank

(Page 3-39 deleted)

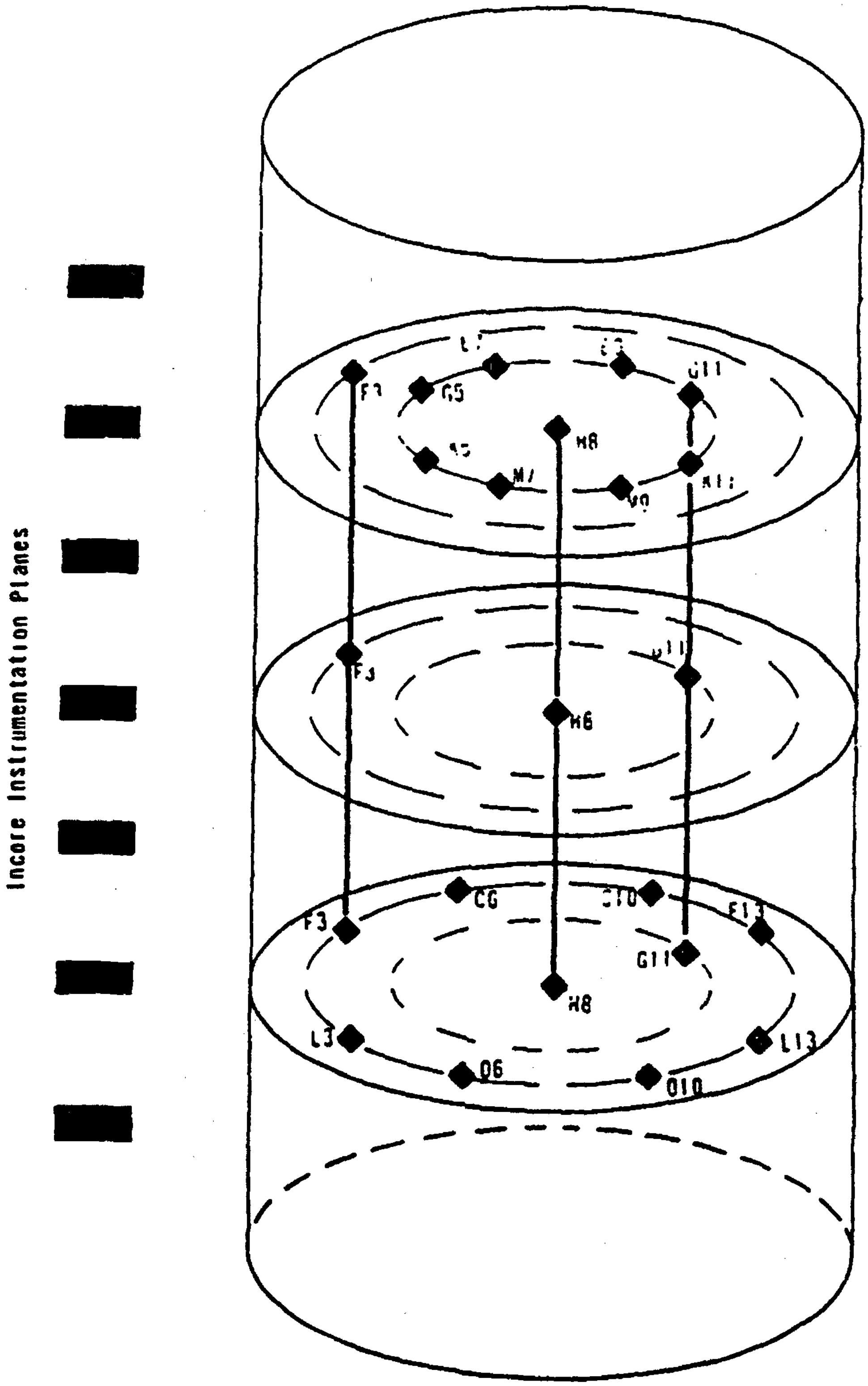


INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION
THREE MILE ISLAND NUCLEAR STATION UNIT 1



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX INDICATION
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-2



INCORE INSTRUMENTATION SPECIFICATION

THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 3.5-3

This page intentionally Left Blank

3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Applicability

Applies to the operability requirements for the instruments identified in Table 3.5-2 and Table 3.5-3 during STARTUP, POWER OPERATION and HOT STANDBY.

Objectives

To assure operability of key instrumentation useful in diagnosing situations which could represent or lead to inadequate core cooling or evaluate and predict the course of accidents beyond the design basis.

Specification

3.5.5.1 The minimum number of channels identified for the instruments in Table 3.5-2, shall be OPERABLE. With the number of instrumentation channels less than the minimum required, restore the inoperable channel(s) to OPERABLE status within seven (7) days (48 hours for pressurizer level) or be in at least HOT SHUTDOWN within the next six (6) hours and in COLD SHUTDOWN within an additional 30 hours. Prior to startup following a COLD SHUTDOWN, the minimum number of channels shown in Table 3.5-2 shall be operable.

3.5.5.2 The channels identified for the instruments specified in Table 3.5-3 shall be OPERABLE. With the number of instrumentation channels less than required, restore the inoperable channel(s) to OPERABLE in accordance with the action specified in Table 3.5-3.

Bases

The Saturation Margin Monitor provides a quick and reliable means for determination of saturation temperature margins. Hand calculation of saturation pressure and saturation temperature margins can be easily and quickly performed as an alternate indication for the Saturation Margin Monitors.

Discharge flow from the two (2) pressurizer code safety valves and the PORV is measured by differential pressure transmitters connected across elbow taps downstream of each valve. A delta-pressure indication from each pressure transmitter is available in the control room to indicate code safety or relief valve line flow. An alarm is also provided in the control room to indicate that discharge from a pressurizer code safety or relief valve is occurring. In addition, an acoustic monitor is provided to detect flow in the PORV discharge line. An alarm is provided in the control room for the acoustic monitor.

3.5.5 ACCIDENT MONITORING INSTRUMENTATION (Continued)

The Emergency Feedwater System (EFW) is provided with two channels of flow instrumentation on each of the two discharge lines. Local flow indication is also available for the EFW System.

Although the pressurizer has multiple level indications, the separate indications are selectable via a switch for display on a single display. Pressurizer level, however, can also be determined via the patch panel and the computer log. In addition, a second channel of pressurizer level indication is available independent of the NNI.

Although the instruments identified in Table 3.5-2 are significant in diagnosing situations which could lead to inadequate core cooling, loss of any one of the instruments in Table 3.5-2 would not prevent continued, safe, reactor operation. Therefore, operation is justified for up to 7 days (48 hours for pressurizer level). Alternate indications are available for Saturation Margin Monitors using hand calculations, the PORV/Safety Valve position monitors using discharge line thermocouple and Reactor Coolant Drain Tank indications, and for EFW flow using Steam Generator level and EFW Pump discharge pressure. Pressurizer level has two channels, one channel from NNI (2 D/P instrument strings through a single indicator) and one channel independent of the NNI. Operation with the above pressurizer level channels out of service is permitted for up to 48 hours. Alternate indication would be available through the plant computer.

The operability of design basis accident monitoring instrumentation as identified in Table 3.5-3, ensures that sufficient information is available on selected plant parameters to monitor and assess the variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Rev. 3, May 1983.) These instruments will be maintained for that purpose.

TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

FUNCTION	INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS
1	Saturation Margin Monitor	2	1
2	Safety Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line
3	PORV Position Monitor	2	1*
4	Emergency Feedwater Flow	2 per OTSG	1 per OTSG
5	Pressurizer Level	2	1
6	Backup Incore Thermocouple Display Channel	4 thermocouples/core quadrant	2 thermocouples/core quadrant

* With the PORV Block Valve closed in accordance with Specification 3.1.12.4.a, the minimum number of channels is zero.

TABLE 3.5-3

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	REQUIRED NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS	ACTION
1.	High Range Noble Gas Effluent			
	a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)	1	1	A
	b. Condenser Vacuum Pump Exhaust (RM-G25)	1	1	A
	c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)	1	1	A
	d. Reactor Building Purge Exhaust (RM-A9-Hi)	1	1	A
	e. Reactor Building Purge Exhaust (RM-G24)	1	1	A
	f. Main Steam Lines Radiation (RM-G26/RM-G27)	1 each OTSG	1 each OTSG	A
2.	Containment High Range Radiation (RM-G22/G-23)	2	2	A
3.	Containment Pressure	2	1	B
4.	Containment Water Level			
	a. Containment Flood (LT-806/807)	2	1	B
	b. Containment Sump (LT-804/805)	1	0	C
5.	DELETED			
6.	Wide Range Neutron Flux	2	1	A
7.	Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)	2	1	A
8.	Reactor Coolant System Hot Leg Water Temperature (TE-958, 960; TI-958A, 960A)	2	1	A
9.	Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)	2	1	A
10.	Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)	2/OTSG	1/OTSG	A
11.	Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)	2/Tank	1/Tank	A

TABLE 3.5-3 (Continued)

ACTIONS

A. With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirements:

1. either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2. prepare and submit a Special Report within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

B. 1. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

C. Restore the inoperable sump level instrument to OPERABLE status prior to startup following the COLD SHUTDOWN subsequent to its inoperability declaration.

3.5.6 DELETED

3.5.7 REMOTE SHUTDOWN SYSTEM

Applicability

Applies to the operability requirements for the Remote Shutdown System Panel "B" Functions in Table 3.5-4 during STARTUP, POWER OPERATION AND HOT STANDBY.

Objectives

To assure operability of the instrumentation and controls necessary to place and maintain the unit in HOT SHUTDOWN from a location other than the control room.

Specification

The minimum number of functions identified in Table 3.5-4 shall be OPERABLE. With the number of functions less than the minimum required, restore the required function to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within an additional 12 hours.

Bases

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from locations other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as HOT SHUTDOWN.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in HOT SHUTDOWN. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches HOT SHUTDOWN following a unit shutdown and can be maintained safely in HOT SHUTDOWN for an extended period of time.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected unit parameters to place and maintain the unit in HOT SHUTDOWN should the control room become inaccessible.

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in HOT SHUTDOWN.

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19.

The controls, instrumentation, and transfer switches are those required for: Reactor Coolant Inventory Control, Reactor Coolant System Pressure and Temperature Control, Decay Heat Removal, Reactivity Monitoring, OTSG Level and Pressure Control, Reactor Coolant Flow Control, and Electrical Power.

The Remote Shutdown System instruments and control circuits covered by this specification do not need to be energized to be considered OPERABLE. This specification is intended to ensure the Remote Shutdown System instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation. The operability of components and equipment are determined by their respective Technical Specification requirements. If a component required for safe shutdown is placed in its fail-safe condition, as permitted by Technical Specifications, then the safety function has been assured and the remote shutdown panel function is considered operable.

Entry into an applicable REACTOR OPERATING CONDITION while relying on the specification actions is allowed even though the specification actions may eventually require a unit shutdown. This is acceptable due to the low probability of an event requiring these instruments.

The conditions of the specification may be entered independently for each Function listed on Table 3.5-4 and completion times of inoperable Functions will be tracked separately for each Function.

TABLE 3.5-4 (Sheet 1 of 2)

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

Function/Instrument or Control Parameter	Required Number of Functions
1. Reactor Coolant	
Coolant Temperature	1
Inlet Temperature	1
Coolant Pressure	1
Pressurizer Level	1
RC-V-2	1
RC-V-3	1
2. Emergency Feedwater Controls	
EFW A Flow Indicator	1
EFW B Flow Indicator	1
OTSG A Level	1
OTSG B Level	1
EF-V-30B	1
EF-V-30D	1
3. OTSG "B" Pressure Control	
Outlet Pressure	1
MS-V-4B	1
MS-V-8B	1
MS-V-8A	1

TABLE 3.5-4 (Sheet 2 of 2)

Function/Instrument or Control Parameter	Required Number of Functions
4. Decay Heat Removal	
Cooler Outlet Temperature	1
Pump Inlet Temperature	1
Flow	1
5. Reactor Neutron Power	
Source Range Flux	1
6. Makeup Control and Status	
MU-P-1B	1
MU-P-1C	1
MU-P-3B	1
MU-P-3C	1
MU-V-2A	1
MU-V-2B	1
MU-V-8	1
MU-V-14B	1
MU-V-16C	1
MU-V-16D	1
MU-V-18	1
MU-V-20	1
MU-V-32 Indicator	1
MU-V-37	1
DH-T-1 BWST Level	1
Makeup Tank Level	1
7. Decay Heat Closed Cycle Cooling Water	
DC-P-1B (Auxiliary "B" Panel)	1
8. Diesel Generator	
EG-Y-1B	1

3.6 REACTOR BUILDING

Applicability

Applies to the CONTAINMENT INTEGRITY of the reactor building as specified below.

Objective

To assure CONTAINMENT INTEGRITY.

Specification

3.6.1 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200 degrees F or greater.

c. Nuclear fuel is in the core.

3.6.2 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% $\Delta k/k$ shall not be made by control rod motion or boron dilution unless CONTAINMENT INTEGRITY is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual Containment Isolation Valves (CIVs) which should be closed are closed and are conspicuously marked.

3.6.6 When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV* in the line will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

b. For lines isolable by one CIV, where the other barrier is a closed system, the line shall be isolated by at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 72 hours or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

* All CIVs required to isolate the penetration.

3.6 <u>REACTOR BUILDING (Continued)</u>

3.6.7 DELETED

3.6.8 While CONTAINMENT INTEGRITY is required (see Specification 3.6.1), if a 48" reactor building purge valve is found to be inoperable perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (see Specification 6.8.5), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below.

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 6.8.5 and perform either (1) or (2) below:

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test in accordance with the Reactor Building Leakage Rate Testing Program. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31 degrees and (AH-V-1B&C) shall be limited to less than 33 degrees open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons.

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

(3) Performance of Technical Specification required surveillances.

(4) Radiation Surveys.

(5) Engineering support of safety-related modifications for pre-outage planning.

(6) Purging prior to shutdown to prevent delaying of outage commencement (24 hours prior to shutdown).

c. Containment purging is permitted for Reactor Building pressure control.

d. To the extent practicable the above containment entries shall be scheduled to coincide, in order to minimize instances of purging.

3.6.11 When the reactor is in COLD SHUTDOWN or REFUELING SHUTDOWN, continuous purging is permitted with the Reactor Building purge isolation valves opened fully.

3.6.12 Personnel or emergency air locks:

a. At least one door in each of the personnel or emergency air locks shall be closed and sealed during personnel passage through these air locks.

b. One door of the personnel or emergency air lock may be open for maintenance, repair or modification provided the other door of the air lock is verified closed within 1 hour, locked within 24 hours, and verified to be locked closed monthly. Air lock doors in high radiation areas may be verified locked closed by administrative means.

c. Entry and exit is permissible to perform repairs on the affected personnel or emergency air lock components. With both air locks inoperable due to inoperability of only one door in each airlock, entry and exit is permissible for 7 days under administrative controls. With the personnel or emergency air lock door interlock mechanism inoperable, entry and exit is permissible under the control of a dedicated individual.

d. With one or more air locks inoperable for reasons other than "b" or "c" above, initiate action immediately to evaluate the overall containment leakage rate with respect to the requirements of Specification 6.8.5, verify a door is closed in the affected air lock within 1 hour, and restore the affected air lock(s) to operable status within 24 hours or the reactor shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

BASES

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

The primary Containment Isolation Valves (CIVs) are identified in UFSAR Table 5.3-2. Additional vent, drain, test and other manually operated valves which complete the containment boundary are identified in the containment integrity checklist. For the purpose of this specification, check valves and relief valves identified in the containment integrity checklist are defined to be active valves.

The loss of redundant capability for containment isolation is limited for all penetrations after which the containment penetration must be isolated. Isolation of certain penetrations may require the closure of multiple CIVs due to piping branches.

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.

2. For those CIVs where the second barrier is a closed system within the Reactor Building, there is no other CIV to isolate the penetration. If operability cannot be regained, the valve must be closed within 72 hours or the plant must commence shut down. An action time of 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a containment isolation boundary and the relative importance of supporting containment integrity.

The definition of Containment Integrity permits normally closed CIVs, except for the 48 inch purge valves, to be unisolated intermittently or manual control to be substituted for automatic control under administrative control. Administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment (Reference 1). The dedicated individual can be responsible for closing more than one valve provided that the valves are in close vicinity and can be closed in a timely manner. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the containment penetrations containing these valves may not be opened under administrative control.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable

BASES (Continued)

of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30 degree open position.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to COLD SHUTDOWN from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Reference 2), and the Reactor Building Leakage Rate Testing Program. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

Entry and exit is allowed to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock. With both air locks inoperable due to inoperability of one door in each of the two air locks, entry and exit is allowed for use of the air locks for 7 days under administrative controls. Containment entry may be required to perform Technical Specifications (TS) Surveillance and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

With one or more air locks inoperable for reasons other than those described in 3.6.12 "b" or "c," Section 3.6.12.d requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour would otherwise be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

BASES (Continued)

Section 3.6.12.d requires that one door in the affected containment air locks(s) must be verified to be closed within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. 24 hours is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

References

(1) NRC Generic Letter 91-08
(2) 10 CFR 50, Appendix J.

3.7 UNIT ELECTRIC POWER SYSTEM

Applicability
Applies to the availability of electrical power for operation of the unit auxiliaries.

Objective
To define those conditions of electrical power availability necessary to ensure:

a. Safe unit operation

b. Continuous availability of engineered safeguards

Specification

3.7.1 The reactor shall not be made critical unless all of the following requirements are satisfied:

a. All engineered safeguards buses, engineered safeguards switchgear, and engineered safeguards load shedding systems are operable.

b. One 7200 volt bus is energized.

c. Two 230 kV lines are in service.

d. One 230 kV bus is in service.

e. Engineered safeguards diesel generators are operable and at least 25,000 gallons of fuel oil are available in the storage tank.

f. Station batteries are charged and in service. Two battery chargers per battery are in service.

3.7.2 The reactor shall not remain critical unless all of the following requirements are satisfied:

a. Offsite Sources:

(i.) Two 230 kV lines are in service to provide auxiliary power to Unit 1, except as specified in Specification 3.7.2e below.

(ii.) The voltage on the 230 kV grid is sufficient to power the safety related ES loads, except as specified in Specification 3.7.2.h below.

b. Both 230/4.16 kV unit auxiliary transformers shall be in operation except that within a period not to exceed eight hours in duration from and after the time one Unit 1 auxiliary transformer is made or found inoperable, two diesel generators shall be operable, and one of the operable diesel generator will be started and run continuously until both unit auxiliary transformers are in operation. This mode of operation may continue for a period not exceeding 30 days.

c. Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days* provided that the redundant diesel generator is:

1. verified to be operable immediately;

2. within 24 hours, either:

   a. determine the redundant diesel generator is not inoperable due to a common mode failure; or,

   b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

With one diesel generator inoperable, in addition to the above, verify that: All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

d. If one Unit Auxiliary Transformer is inoperable and a diesel generator becomes inoperable, the unit will be placed in HOT SHUTDOWN within 12 hours. If one of the above sources of power is not made operable within an additional 24 hours the unit shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

e. If Unit 1 is separated from the system while carrying its own auxiliaries, or if only one 230 kV line is in service, continued reactor operation is permissible provided one emergency diesel generator shall be started and run continuously until two transmission lines are restored.

f. The engineered safeguards electrical bus, switchgear, load shedding, and automatic diesel start systems shall be operable except as provided in Specification 3.7.2c above and as required for testing.

g. One station battery may be removed from service for not more than eight hours.

h. If it is determined that a trip of the Unit 1 generator, in conjunction with LOCA loading, will result in a loss of offsite power to Engineered Safeguards buses, the plant shall begin a power reduction within 24 hours and be in HOT SHUTDOWN in an additional 6 hours, except as provided in Specification 3.7.2.e above.

* The 7-day allowed outage time of Technical Specification 3.7.2.c, which was entered on April 2, 2006 at 2100 hours, may be extended one time by an additional 3 days to complete repair and testing of EG-Y-1A.

Bases

The Unit Electric Power System is designed to provide a reliable source of power for balance of plant auxiliaries and a continuously available power supply for the engineered safeguards equipment. The availability of the various components of the Unit Electric Power System dictates the operating mode for the station.

Verification of emergency diesel generator and station battery operability normally consists of verifying that the surveillance is current, and that other available information does not indicate inoperability.

It is recognized that while testing the redundant emergency diesel generator (EDG) in accordance with surveillance requirement 4.6.1.a, the EDG will not respond to an automatic initiation signal. In this situation, the 12 hour time clock will not be entered per the provisions of section 3.7.2.f. due to the low probability of an event occurring while the EDG is being tested.

Trip of TMI-1 could result in a change in the 230 kV system (Grid) voltage at the TMI substation. The predicted voltage following a loss of the unit is referred to as the Post-Contingency voltage for trip of TMI-1. The transmission system operator monitors 230 kV system conditions for Post Contingency voltages. If the Post-Contingency voltage is less than the value required to support safety related ES loads, the transmission system operator will notify the TMI Unit 1 control room. The required voltage setpoint values for dual or single auxiliary transformer operation are specified by degraded grid calculations. The appropriate setpoint for the current plant condition(s) is provided to the Grid operator. The required voltage setpoint is based on the Large Break LOCA loading which results in the greatest ES loads.

Upon receipt of a valid Post-Contingency voltage Alarm for Loss of TMI-1, TMI will implement the Low System (Grid) Voltage Procedure. An allowed action time of 24 hours provides the transmission system operator time to take actions to reconfigure the 230 kV system for improved voltage support. The time allowed has been evaluated for the level of risk associated with the increased reliance on use of the onsite sources.

3.8 FUEL LOADING AND REFUELING

Applicability: Applies to fuel loading and refueling operations.

Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by RM-G6 and RM-G7. Radiation levels in the spent fuel storage area shall be monitored by RM-G9. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door in each of the personnel and emergency air locks shall be capable of being closed.* The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

----------------------------- NOTE -----------------------------

The equipment hatch may be open if all of the following conditions are met:

1) The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes,
2) A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and
3) Reactor Building Purge Exhaust System is in service.

3.8.7 During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or capable of being closed,* or

2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

* Administrative controls shall ensure that the Reactor Building Purge Exhaust System is in service, appropriate personnel are aware that air lock doors and/or other penetrations are open, a specific individual(s) is designated and available to close the air lock doors and other penetrations as part of a required evacuation of containment. Any obstruction(s) (e.g., cable and hoses) that could prevent closure of an air lock door or other penetration will be capable of being quickly removed.

3.8.8 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.9 The reactor building purge isolation valves, and associated radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than 7 days prior to initial fuel movement in the reactor building.

3.8.10 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.11 During the handling of irradiated fuel in the Reactor Building at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange, as determined by a shiftly check and a daily verification. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the UFSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core (Reference 1). The boron concentration will be sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, however only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

Per Specification 3.8.6 and 3.8.7, the personnel and emergency air lock doors, and penetrations may be open during movement of irradiated fuel in the containment provided a minimum of one door in each of the air locks, and penetrations are capable of being closed in the event of a fuel handling accident, and the plant is in REFUELING SHUTDOWN or REFUELING OPERATION with at least 23 feet of water above the fuel seated within the reactor pressure vessel. The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel. Should a fuel handling accident occur inside containment, a minimum of one door in each personnel and emergency air lock, and the open penetrations will be closed following an evacuation of containment. Administrative controls will be in place to assure closure of at least one door in each air lock, as well as other open containment penetrations, following a containment evacuation.

Specification 3.8.6 is modified by a NOTE:

-------------------------------------- NOTE ------------------------------

The equipment hatch may be open if all of the following conditions are met:

1) The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes,
2) A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and
3) Reactor Building Purge Exhaust System is in service.

These restrictions include administrative controls to allow the opening of the reactor building equipment hatch during the handling of irradiated fuel in the Reactor Building provided that 1) The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes, 2) A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and 3) Reactor Building Purge Exhaust System is in service. The Reactor Building Equipment Hatch Missile Shield Barrier includes steel plating on the bottom of the shield structure, which acts to restrict a release of post-accident fission products. The capability to close the reactor building missile shield barrier includes requirements that the barrier is capable of being closed and that any cables or hoses across the opening have quick disconnects to ensure the barrier is capable of being closed within 45 minutes. The 45-minute closure time for the reactor building missile shield barrier starts when the control room communicates the need to shut the Reactor Building Equipment Hatch Missile Shield Barrier. This 45-minute requirement is significantly less than the fuel handling accident analysis assumption that the reactor building remains open to the outside environment for a two-hour period subsequent to the accident. Placing reactor building purge exhaust in service will ensure any release from the reactor building will be monitored, and ensure continued air flow into the Reactor Building in the event of a fuel handling accident. The Reactor Building purge valve high radiation interlock will be bypassed to ensure continued air flow into the Reactor Building in the event of a Fuel Handling Accident.

The administrative controls will also include the responsibility to be able to communicate with the control room, and the responsibility to ensure that the reactor building missile shield barrier is capable of being closed in the event of a fuel handling accident. These administrative controls will ensure reactor building closure would be established in the event of a fuel handling accident inside containment.

Provisions for equivalent isolation methods in Technical Specification 3.8.7 include use of a material (e.g. temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for other containment penetrations during fuel movements.

Specification 3.8.9 requires testing of the reactor building purge isolation system. This system consists of the four reactor building purge valves and the associated reactor building purge radiation monitor(s). The test verifies that the purge valves will automatically close when they receive initiation signals from the radiation detectors that monitor reactor building purge exhaust, and the valves remain open when the isolation system is bypassed. The test is performed no more than 7 days prior to the start of fuel movement in the reactor building to ensure that the monitors, purge valves, and associated interlocks are functioning prior to operations that could result in a fuel handling accident within the reactor building. The Fuel Handling Accident analysis assumes that the four purge valves remain open.

Specification 3.8.10 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours (Reference 2).

<u>REFERENCES</u>

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

(2) UFSAR, Section 14.2.2.1(2) - "FHA Inside Containment"

3.9 DELETED

3.10 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.10.1 The source leakage test performed pursuant to Specification 4.13 shall be capable of detecting the presence of 0.005 µCi of radioactive material on the test sample. If the test reveals the presence of 0.005 µCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations; and a Special Report of the test results that show the presence of $\geq$ .005 µCi of removable contamination shall be prepared and submitted to the NRC Region I Administrator within 90 days after completion of the test. Sealed sources are exempt from such leak tests when the source contains 100 µCi or less of beta and/or gamma emitting material or 5 µCi or less of alpha emitting material.

3.10.2 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

Bases

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3.11 Handling of Irradiated Fuel

Applicability

Applies to the operation of the fuel handling building crane when within the confines of Unit 1 and there is any spent fuel in storage in the Unit 1 fuel handling building.

Objective

To define the lift conditions and allowable areas of travel when loads to be lifted and transported with the fuel handling building crane are in excess of 15 tons or between 1.5 tons and 15 tons or consist of irradiated fuel elements.

Specification

3.11.1 Spent fuel elements having less than 120 days for decay of their irradiated fuel shall not be loaded into a spent fuel transfer cask in the shipping cask area.

3.11.2 The key operated travel interlock system for automatically limiting the travel area of the fuel handling building crane shall be imposed whenever loads in excess of 15 tons are to be lifted and transported with the exception of fuel handling bridge maintenance.

3.11.3 The lowest surface of all loads in excess of 15 tons shall be administratively limited to an elevation one foot or less above the concrete surface at the nominal 348 ft-0 in. elevation in the fuel handling building.

3.11.4 Loads in excess of hook capacity shall not be lifted, except for load testing.

3.11.5 Following modifications or repairs to any of the load bearing members, the crane shall be subjected to a test lift of 125 percent of its rated load.

3.11.6 Administrative controls shall require the use of an approved procedure with an identified safe load path for loads in excess of 3,000 lbs. handled above the Spent Fuel Pool Operating Floor (348' elevation).

3.11.7 During transfer of the cask to and from the cask loading pit, the cask will be restricted to the transfer path shown in Figure 3.11-1. Administrative controls will be used to ensure that all lateral movements of the cask are performed at slow bridge and trolley speeds. During this transfer the cask lifting yoke shall be oriented in the East-West direction.

Bases

This specification will limit activity releases to unrestricted areas resulting from damage to spent fuel stored in the spent fuel storage pools in the postulated event of the dropping of a heavy load from the fuel handling building crane. A Fuel Handling accident analysis was performed assuming that the cask and its entire contents of ten fuel assemblies are sufficiently damaged as a result of dropping the cask, to allow the escape of all noble gases and iodine in the gap (Reference 1). This release was assumed to be directly to the atmosphere and to occur instantaneously. The site boundary doses resulting from this accident are 5.25 R whole body and 1.02 R to thyroid, and are within the limits specified in 10 CFR 100.

Specification 3.11.1 requires that spent fuel, having less than 120 days decay post-irradiation, not be loaded in a spent fuel transfer cask in order to ensure that the doses resulting from a highly improbable spent fuel transfer cask drop would be within those calculated above.

Specification 3.11.2 requires the key operated interlock system, which automatically limits the travel area of the fuel handling crane while it is lifting and transporting the spent fuel shipping cask, to be imposed whenever loads in excess of 15 tons are to be lifted and transported while there is any spent fuel in storage in the spent fuel storage pools in Unit 1. This automatically ensures that these heavy loads travel in areas where, in the unlikely event of a load drop accident, there would be no possibility of this event resulting in any damage to the spent fuel stored in the pools, any unacceptable structural damage to the spent fuel pool structure, or damage to redundant trains of safety related components. The shipping cask area is designed to withstand the drop of the spent fuel shipping cask from the 349 ft-0 in. elevation without unacceptable damage to the spent fuel pool structure (Reference 2).

Specification 3.11.3 ensures that the lowest surface of any heavy load never gets higher than one foot above the concrete surface of the 348 ft-0 in. elevation in the fuel handling building (nominal elevation 349 ft-0 in.) thereby keeping any impact force from an unlikely load drop accident within acceptable limits.

Specification 3.11.4 ensures that the proper capacity crane hook is used for lifting and transporting loads thus reducing the probability of a load drop accident.

Following modification or repairs, specification 3.11.5 confirms the load rating of the crane.

References

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"

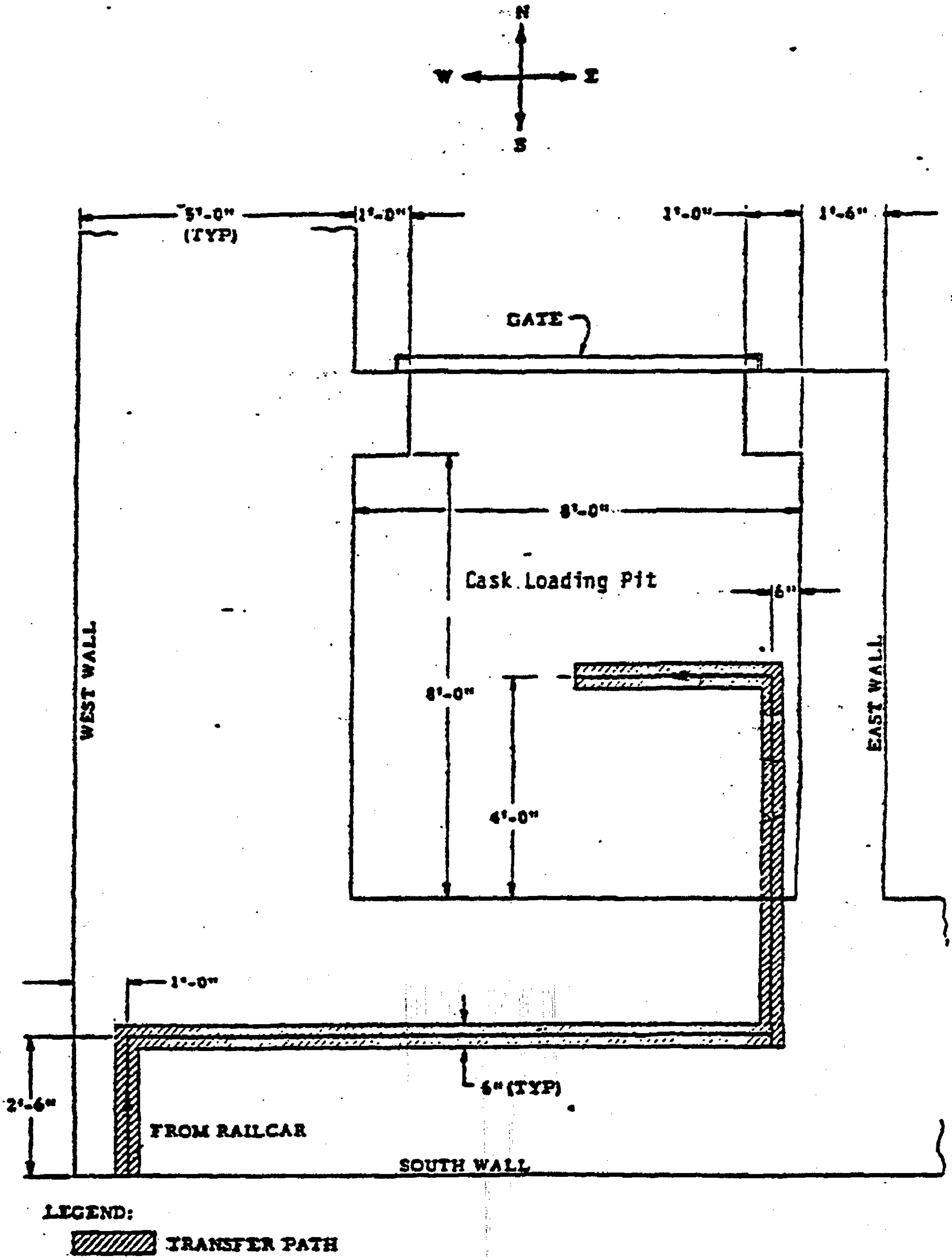
(2) UFSAR, Section 14.2.2.8 - "Fuel Cask Drop Accident"

Specification 3.11.6 imposes administrative limits on handling loads weighing in excess of 3000 lbs. to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the spent fuel pool, or to impact redundant safe shutdown equipment. The safe load path shall follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. Handling loads of less than 3000 lbs. without these restrictions is acceptable because the consequences of dropping loads in this weight range are comparable to those produced by the fuel handling accident considered in the FSAR and found acceptable.

Specification 3.11.7 in combination with 3.11.3 ensures the spent fuel cask is handled in a manner consistent with the load drop analysis (Reference 3).

<u>Reference</u>

(3) GPU Evaluation of Heavy Load Handling Operations at TMI-1 February 21, 1984, as transmitted to the NRC in GPUN Letter No. 5211-84-2013.



TRANSFER PATH TO AND FROM CASK LOADING PIT
(EL. 348'-0")

FIGURE 3.11-1

3.12 REACTOR BUILDING POLAR CRANE

Applicability

Applies to the use of the reactor building polar crane hoists over the steam generator compartments and the fuel transfer canal.

Objective

To identify those conditions for which the operation of the reactor building polar crane hoists are restricted.

Specification

3.12.1 The reactor building polar crane hoists shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.12.2 During the period when the reactor vessel head is removed and irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use.

3.12.3 During the period when the reactor coolant system is pressurized above 300 psig, and is above 200 F, and fuel is in the core, the reactor building polar crane hoists shall not be operated over the steam generator compartments.

Bases

Restriction of use of the reactor building polar crane hoists over the fuel transfer canal when the reactor vessel head is removed to permit those operations necessary for the fuel handling and core internals operations is to preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that any escape of fission products would result.

Restriction of use of the reactor building polar crane hoists over the steam generator compartments during the time when steam could be formed from dropping a load on the steam generator or reactor coolant piping resulting in rupture of the system is required to protect against a loss-of-coolant accident.

3.13 SECONDARY COOLANT SYSTEM ACTIVITY

Applicability

Applies to the limiting conditions for operation when reactor coolant system pressure is greater than 300 psig or T_{avg} is greater than 200°F.

Objective

To limit the inventory of activity in the secondary system.

Specification

3.13.1 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

3.13.2 With the specific activity of the secondary coolant system > 0.10μ Ci/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

Bases

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary-to-secondary tube leak in the steam generator of the affected steam line.

3.14 FLOOD

3.14.1 PERIODIC INSPECTION OF THE DIKES AROUND TMI

Applicability

Applies to inspection of the dikes surrounding the site.

Objective

To specify the minimum frequency for inspection of the dikes and to define the flood stage after which the dikes will be inspected.

Specification

3.14.1.1 The dikes shall be inspected at least once every six months and after the river has returned to normal, following the condition defined below:

a. The level of the Susquehanna River exceeds flood stage; flood stage is defined as elevation 307 feet at the Susquehanna River Gage at Harrisburg.

Bases

The earth dikes are compacted to provide a stable impervious embankment that protects the site from inundation during the design flood of 1,100,000 cfs. The rip-rap, provided to protect the dikes from wave action and the flow of the river, continues downward into natural ground for a minimum depth of two feet to prevent undermining of the dike (References 1 and 2).

Periodic inspection, and inspection of the dikes and rip-rap after the river has returned to normal from flood stage, will assure proper maintenance of the dikes, thus assuring protection of the site during the design flood.

References

(1) UFSAR, Section 2.6.5 - "Design of Hydraulic Facilities"

(2) UFSAR, Figure 2.6-17 - "Typical Dike Section"

3.14.2 FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY

Applicability

Applies to the river stage for placing the unit in hot standby.

Objective

To define the action taken in the event river elevation reaches 302 feet at the intake structure.

Specification

3.14.2.1 If the river stage reaches elevation 302 feet at the River Water Intake Structure, corresponding to 1,000,000 cfs river flow, the unit will be brought to the hot standby condition.

Bases

The dikes provided protect the plant site during the design flood of 1,100,000 cfs. The design flood corresponds to an elevation of approximately 303 feet at the River Water Intake Structure (Reference 1). The dike elevation at the intake structure is 305 feet. The minimum freeboard is at the downstream end of the plant site where the dike elevation is 304 feet providing a freeboard of approximately one foot. Adequate freeboard is provided to protect the plant site from flooding due to wave action during the design flood (Reference 2).

Placing the unit in hot standby when the river stage reaches 302 feet elevation provides an additional margin of conservatism by assuring that adequate freeboard exists during operation of the unit.

References

(1) UFSAR, Figure 2.6-15 - "Dike Freeboard - Design Flood"

(2) UFSAR, Section 2.6.4 - "Flood Studies"

3.15 AIR TREATMENT SYSTEMS

3.15.1 EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the emergency control room air treatment system and its associated filters.

Objective

To specify minimum availability and efficiency for the emergency control room air treatment system and its associated filters.

Specifications

3.15.1.1 Except as specified in Specification 3.15.1.3 below, both emergency treatment systems, AH-E18A fan and associated filter AH-F3A and AH-E18B fan and associated filter AH-F3B shall be operable at all times, per the requirements of Specification 3.15.1.2 below; when containment integrity is required and when irradiated fuel handling operations are in progress.

3.15.1.2 a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show < 0.05% DOP penetration and < 0.05% halogenated hydrocarbon penetration, except that the DOP test will be conducted with prefilters installed.

b. The results of laboratory carbon sample analysis shall show ≥ 95% radioactive methyl iodide decontamination efficiency when tested in accordance with ASTM D3803-1989 at 30°C, 95% R.H.

c. The fans AH-E18A and B shall each be shown to operate within ± 4000 CFM of design flow (40,000 CFM).

3.15.1.3 From and after the date that one control room air treatment system is made or found to be inoperable for any reason, reactor operation or irradiated fuel handling operations are permissible only during the succeeding 7 days provided the redundant system is verified to be OPERABLE.

3.15.1.4 From the date that both control room air treatment systems are made or found to be inoperable or if the inoperable system of 3.15.1.3 cannot be made operable in 7 days, irradiated fuel handling operations shall be terminated in 2 hours and reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 48 hours.

Bases

The emergency control room air treatment systems AH-E18A and 18B and their associated filters are two independent systems designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control building is designed to be automatically placed in the recirculation mode upon an RM-A1 high radiation alarm, air tunnel device actuation, ESAS actuation or station blackout condition. The emergency control room air treatment fan and filter AH-E18A or B and AH-F3A or B is designed to be manually started by the operator if a high radiation alarm from RM-A1 is indicated.

Prefilters and high efficiency particulate absolute (HEPA) filters are installed before the charcoal absorbers to prevent clogging of the iodine adsorbers and remove particulate activity. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If one system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 7 days, the reactor is shut down and brought to cold shutdown within 48 hours and irradiated fuel handling operations are terminated within 2 hours.

If both systems are found to be inoperable, reactor shutdown shall be initiated and the reactor will be brought to cold shutdown in 48 hours and irradiated fuel handling operations will be stopped within 2 hours.

In-place testing for penetration and system bypass shall be performed in accordance with ANSI N510-1980. Charcoal samples shall be obtained in accordance with ANSI N509-1980. Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Guide 1.52, Revision 2. Any lot of charcoal adsorber which fails the laboratory test criteria shall be replaced with new adsorbent qualified according toASTM D3803-1989.

Laboratory testing of charcoal samples will be performed in accordance with the methods prescribed by ASTM D3803-1989. Design basis accident analyses assume the carbon adsorber is 90% efficient in its total radioiodine removal. Therefore, using a Safety Factor of 2 (Ref. 3), the acceptance criteria for the laboratory test of carbon adsorber is set at greater than or equal to 95% [(100 – 90) / 2 = 5% penetration].

References

(1) FSAR Section 9.8
(2) FSAR Figure 9-21
(3) **NRC Generic Letter 99-02, dated June 3, 1999.**

3.15.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Deleted

THIS PAGE LEFT BLANK INTENTIONALLY

3.15.3 AUXILIARY AND FUEL HANDLING BUILDING AIR TREATMENT SYSTEM

Deleted

THIS PAGE LEFT BLANK INTENTIONALLY

3.15.4 Fuel Handling Building ESF Air Treatment System

Applicability

Applies to the Fuel Handling Building (FHB) ESF Air Treatment System and its associated filters.

Objective

To specify minimum availability and efficiency for the FHB ESF Air Treatment System and its associated filters for irradiated fuel handling operations.

Specifications

3.15.4.1 Prior to fuel movement each refueling outage, two trains shall be operable. One train shall be operating continuously whenever TMI-1 irradiated fuel handling operations in the FHB are in progress.

a. With one train inoperable, irradiated fuel handling operations in the Fuel Handling Building may continue provided the redundant train is operating.

b. With both trains inoperable, handling of irradiated fuel in the Fuel Handling Building shall be suspended until such time that at least one train is operable and operating. Any fuel assembly movement in progress may be completed.

3.15.4.2 A FHB ESF Air Treatment System train is operable when its surveillance requirements are met and:

a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and carbon adsorber banks shall show < 0.05% DOP penetration and < 0.05% halogenated hydrocarbon penetration.

b. The results of laboratory carbon sample analysis shall show ≥ 95% radioactive methyl iodide decontamination efficiency when tested in accordance with ASTM D3803-1989 at 30°C, 95% R.H.

c. The fans AH-E-137A and B shall each be shown to operate within ± 10% of design flow (6,000 SCFM).

Bases

Compliance with these specifications satisfies the condition of operation imposed by the Licensing Board as described in NRC's letter dated October 2, 1985, item 1.c.

The FHB ESF Air Treatment System contains, controls, mitigates, monitors and records radiation release resulting from a TMI-1 postulated spent fuel accident in the Fuel Handling Building as described in the FSAR. Offsite doses will be less than the 10 CFR 100 guidelines for accidents analyzed in Chapter 14 (Reference 1).

Bases (Continued)

Normal operation of the FHB ESF Air Treatment System will be during TMI-1 irradiated fuel movements in the Fuel Handling Building. The system includes air filtration and exhaust capacity to ensure that any radioactive release to atmosphere will be filtered and monitored. Effluent radiation monitoring and sampling capability are provided.

The in-plant testing for penetration and system bypass shall be performed in accordance with ANSI N510-1980. Charcoal samples shall be obtained in accordance with ANSI N509-1980. Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Guide 1.52, Revision 2. Any lot of charcoal adsorber which fails the laboratory test criteria shall be replaced with new adsorbent qualified in accordance with ASTM D3803-1989.

Laboratory testing of charcoal samples will be performed in accordance with the test methods prescribed by ASTM D3803-1989. Testing of charcoal at 95% relative humidity will be required until such time that a surveillance to demonstrate operability of the heaters is incorporated by amendment into the specification. The accident analysis in FSAR Chapter 14 (Reference 1) assumes the charcoal adsorber is 90% efficient in its total radioiodine removal. Therefore, using a Safety Factor of 2 (Ref. 2), the acceptance criteria for the laboratory test of charcoal adsorber is set at greater than or equal to 95% [(100 – 90) / 2 = 5% penetration].

References

(1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"
(2) NRC Generic Letter 99-02, dated June 3, 1999.

3.16 SHOCK SUPPRESSORS (SNUBBERS)

LIMITING CONDITION FOR OPERATION

3.16.1 Each safety related snubber shall be OPERABLE.

APPLICABILITY:

Whenever the system protected by the snubber is required to be OPERABLE.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.17.1.g.2 on the attached component or declare the attached system inoperable and follow the appropriate action statement for that system.

BASES

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber due to failure to activate (lockup) is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. The consequence of snubber inoperability due to failure to extend or retract is an increase in the probability of structural damage to piping as a result of thermal motion. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component which is required to be operable must also be operable. During plant conditions other than operating, snubbers on those systems that are required to be operable during that plant condition are also required to be operable.

INTENTIONALLY BLANK

PAGES 3-64 through 3-79

3.17 REACTOR BUILDING AIR TEMPERATURE

Applicability

This specification applies to the average air temperature of the primary containment during power operations.

Objective

To assure that the temperatures assumed in the structural analysis of the Reactor Building are not exceeded.

Specification

3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.

3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.

3.17.3 The primary containment average air temperature shall be calculated as follows:

a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.

b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320 A list of locations is given below.

Location	Location
SE Wall Elev. 352'	NE Wall Elev 314'*
NW Sec Shield Elev 352'	S Wall Elev 314'*
NE Sec Shield Elev 352'	NW Wall Elev 314'*
E Wall Elev 382'	E Sec Shield Elev 352'
NE Sec Shield Elev 352'	S Rx Wall Elev 321'
NW Sec Shield Elev 352'	NE Wall Elev 287'*
NE Sec Shield Elev 352'	S Wall Elev 287'*
NW Sec Shield Elev 352'	NW Wall Elev 287'*
NW Wall Elev 352'	E Sec Shield Elev 352'
E Wall Elev 400'	NW Sec Shield Elev 287'*
S Sec Shield Elev 352'	NE Sec Shield Elev 364'
NW Sec Shield Elev 352'	N Sec Shield Elev 364'

NOTE: (1) * Detectors located below elev 320'.

Bases

The specified temperature limits assure that the containment design temperature and pressure will not be exceeded in the event of a design basis loss of coolant accident. The limits also assure the maintenance of acceptable ambient environmental conditions for safety-related components located inside the containment.

PAGES 3-82 THROUGH 3-85

INTENTIONALLY BLANK

THIS PAGE LEFT BLANK INTENTIONALLY

(Pages 3-87 through 3-94 deleted)

3.19 CONTAINMENT SYSTEMS

3.19.1 CONTAINMENT STRUCTURAL INTEGRITY

Applicability:

Applies to the structural integrity of the reactor building.

OBJECTIVE:

To verify containment structural integrity in accordance with the inservice tendon surveillance program for the reactor building prestressing system.

Specification

3.19.1.1 With the structural integrity of the containment not conforming to the inservice tendon surveillance program requirements of 4.4.2.1 for the tendon lift off forces, perform an engineering evaluation of the structural integrity of the containment to determine if COLD SHUTDOWN is required. The margins available in the containment design may be considered during the investigation. If the acceptability of the containment tendons cannot be established within 48 hours, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.19.1.2 DELETED

3.20 (DELETED)

3.21 <u>RADIOACTIVE EFFLUENT INSTRUMENTATION</u>

Deleted

3.21.1 Radioactive Liquid Effluent Instrumentation

Deleted

3.21.2 Radioactive Gaseous Process and Effluent Monitoring Instrumentation

Deleted

3.22 <u>RADIOACTIVE EFFLUENTS</u>

Deleted

3.22.1 Liquid Effluents

Deleted

3.22.2 Gaseous Effluents

Deleted

3.22.3 Solid Radioactive Waste

Deleted

3.22.4 Total Dose

Deleted

3.23 <u>RADIOLOGICAL ENVIRONMENTAL MONITORING</u>

Deleted

3.23.1 Monitoring Program

Deleted

3.23.2 Land Use Census

Deleted

3.23.3 Interlaboratory Comparison Program

Deleted

3.24 Reactor Vessel Water Level Indication

Applicability

Applies to the operability requirements for the Reactor Vessel Water Level Indication when the reactor is critical.

Objectives

To assure operability of the Reactor Vessel Water Level instrumentation which may be useful in diagnosing situations which could represent or lead to inadequate core cooling.

Specification

Two channels of the Reactor Vessel Water Level Instrumentation System shall be OPERABLE.

If one channel becomes INOPERABLE that channel shall be returned to OPERABLE within 30 days. If the channel is not restored within 30 days, within 14 days, submit a special report to the NRC providing the details of the inoperability, to include cause, action being taken and projected date for return to OPERABLE status.

With no channels OPERABLE, one channel shall be restored to OPERABLE status within 7 days. If at least one channel is not restored within 7 days, within 14 days, submit a special report to the NRC providing the details of the inoperability, to include cause, action being taken and projected date for return to OPERABLE status.

Bases

The Reactor Vessel Water Level Indication (Reference 1) provides indication of the trend in water inventory in the hot legs and reactor vessel during the approach to inadequate core cooling (ICC). In this manner additional information may be available to the operator to diagnose the approach of ICC and to assess the adequacy of responses taken to restore core cooling.

Each Reactor Vessel Water Level channel is comprised of a hot leg level indication and a reactor vessel level indication.

The system is required to be operable (as defined previously) when the plant is critical.

The system is an information system to aid the operator during the approach to inadequate core cooling. There is not regulatory limit for this system.

Inoperability of the system removes the availability of an information system. Other useful instrumentation for inadequate core cooling will be available. The Subcooling Margin Indication System is relied upon to determine subcooling margin when the reactor coolant pumps are operating or when natural circulation can be verified. When natural or forced circulation cannot be verified, the margin to saturation is determined by manual calculation, based on reactor coolant temperature (incore thermocouples) and pressure indications available in the control room and steam tables. See Tech. Spec. 3.5.5.

The system is not a required system to mitigate evaluated accidents. It may be useful to have the system operable but there will be no adverse impact if it is not operable.

The LCO action statement provides the level of emphasis required for an information system.

The Reactor Vessel Water Level is a Regulatory Guide 1.97 Category 1 variable.

<u>Reference</u>

(1) UFSAR, Update Section 7.3.2.2(c)10(d) - "Reactor Coolant Inventory Trending System".

(2) USNRC Regulatory Guide 1.97.

SECTION 4.0

SURVEILLANCE STANDARDS

4. SURVEILLANCE STANDARDS

4.0.1 During Reactor Operational Conditions for which a Limiting Condition for Operation (LCO) does not require a system/component to be operable, the associated surveillance requirements do not have to be performed. Prior to declaring a system/ component operable, the associated surveillance requirement must be current. Failure to perform a surveillance within the specified Frequency shall be failure to meet the LCO except as provided in 4.0.2.

4.0.2 If it is discovered that a surveillance was not performed within its specified frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

When the surveillance is performed within the delay period and the surveillance is not met, the LCO must immediately be declared not met, and the applicable condition(s) must be entered.

Bases

SR 4.0.1 establishes the requirement that SRs must be met during the REACTOR OPERATING CONDITIONS or other specified conditions in the SRs for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified frequency, in accordance with definition 1.25, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified frequency.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The system or components are known to be inoperable, although still meeting the SRs or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a REACTOR OPERATING CONDITION or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given REACTOR OPERATING CONDITION or other specified condition.

Surveillances, including surveillances invoked by LCO required actions, do not have to be performed on inoperable equipment because the actions define the remedial measures that apply. Surveillances have to be met and performed in accordance with the specified frequency, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with the specified frequency. Post maintenance testing may not be possible in the current REACTOR OPERATING CONDITION or other specified conditions in the SRs due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE 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 REACTOR OPERATING CONDITION or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency feedwater (EFW) pump maintenance during refueling that requires testing at steam pressures greater than 750 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.

b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the required surveillance has not been performed in accordance with Surveillance Standard 4.0.2 and not at the time that the specified frequency was not met.

The delay period provides an adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with required actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

Bases (Contd.)

When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering power operation 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, Surveillance Standard 4.0.2 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

Surveillance Standard 4.0.2 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of operating condition changes imposed by required LCO actions.

Failure to comply with specified surveillance frequencies is expected to be an infrequent occurrence. Use of the delay period established by Surveillance Standard 4.0.2 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance 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 surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. 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 surveillance should be treated as an emergent condition as discussed in 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. Missed surveillances for important components should be analyzed quantitatively. 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 surveillances will be placed in the licensee's Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the completion times of the required actions for the applicable LCO conditions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the completion time of the actions, restores compliance.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protection system, engineered safety feature protection system, and heat sink protection system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2, 4.1-3, and 4.1-5.

4.1.3 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the check, test and calibration at the frequencies shown in Table 4.1-4.

4.1.4 Each remote shutdown system function shown in Table 3.5-4 shall be demonstrated OPERABLE by the performance of the following check, test, and calibration:

a) Perform CHANNEL CHECK for each required instrumentation channel that is normally energized every 31 days.

b) Verify each required control circuit and transfer switch is capable of performing the intended function every refueling interval.

c) Perform CHANNEL CALIBRATION for each required instrumentation channel every refueling interval (excludes source range flux).

Bases

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. **The acceptance criteria for the daily check of the Makeup Tank pressure instrument will be maintained within the error used to develop the plant operating limit.** Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Bases (Cont'd)

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required semi-annually on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel (Reference 1).

The rotation schedule for the reactor protection channels is as follows:

a) Deleted
b) Semi-annually with one channel being tested every 46 days on a continuous sequential rotation.

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested every 46 days. The frequency of every 46 days on a continuous sequential rotation is consistent with the calculations of Reference 2 that indicate the RPS retains a high level of reliability for this interval.

Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested quarterly with one channel being tested every 23 days on a continuous sequential rotation. Calculations have shown that the frequency of every 23 days maintains a high level of reliability of the Reactor Trip System in Reference 4. The trip test checks all logic combinations and is to be performed on a rotational basis.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

Bases (Cont'd)

The equipment testing and system sampling frequencies specified in Tables 4.1-2, 4.1-3, and 4.1-5 are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"
(2) NRC SER for BAW-10167A, Supplement 1, December 5, 1988.
(3) BAW-10167, May 1986.
(4) BAW-10167A, Supplement 3, February 1998.

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
1. Protection Channel Coincidence Logic	NA	Q	NA	
2. Control Rod Drive Trip Breaker and Regulating Rod Power SCRs	NA	Q	NA	(1) Includes independent testing of shunt trip and undervoltage trip features.
3. Power Range Amplifier	D(1)	NA	(2)	(1) When reactor power is greater than 15%.
				(2) When above 15% reactor power run a heat balance check once per shift. Heat balance calibration shall be performed whenever heat balance exceeds indicated neutron power by more than two percent.
4. Power Range Channel	S	S/A	M(1)(2)	(1) When reactor power is greater than 60% verify imbalance using incore instrumentation.
				(2) When above 15% reactor power calculate axial offset upper and lower chambers after each startup if not done within the previous seven days.
5. Intermediate Range Channel	S(1)	P S/U	NA	(1) When in service.
6. Source Range Channel	S(1)	P S/A	NA	(1) When in service.
7. Reactor Coolant Temperature Channel	S	S/A	F	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	S/A	R	
9. Low Reactor Coolant Pressure Channel	S	S/A	R	
10. Flux-Reactor Coolant Flow Comparator	S	S/A	F	
11. Reactor Coolant Pressure-Temperature Comparator	S	S/A	R	
12. Pump Flux Comparator	S	S/A	R	
13. High Reactor Building Pressure Channel	S	S/A	F	
14. High Pressure Injection Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Low Pressure Injection Analog Channels			0	
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	S(1)	M(1)	F	(1) When CONTAINMENT INTEGRITY is required.
b. RCS Pressure 1600 psig	S(1)	M(1)	NA	(1) When RCS Pressure > 1800 psig.
c. Deleted				
d. Reactor Bldg. 30 psi pressure switches	S(1)	M(1	F	(1) When CONTAINMENT INTEGRITY is required.
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	W(1)	M(1)(2)	F	(1) When CONTAINMENT INTEGRITY is required.
f. Line Break Isolation Signal (ICCW & NSCCW)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required.
20. Reactor Building Spray System Logic Channel	NA	Q	NA	
21. Reactor Building Spray 30 psig pressure switches	NA	M	F	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Check with Relative Position Indicator
24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator.
25. Core Flooding Tanks				
a. Pressure Channels Coolant	NA	NA	F	
b. Level Channels	NA	NA	F	
26. Pressurizer Level Channels	S	NA	R	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
27. Makeup Tank **Instrument Channels:**				
a **Level**	D(1)	NA	R	(1) When Makeup and Purification System is in operation.
b. **Pressure**	D(1)	NA	R	
28. Radiation Monitoring Systems*				
a. RM-G6 (FH Bridge #1 Aux)	W(1)(2)	M(2)	Q(2)	(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning.
b. RM-G7 (FH Bridge #2 Main)	W(1)(2)	M(2)	Q(2)	
c. RM-G9 (FH Bridge-FH Bldg)	W(1)(3)	M(3)	E(3)	
d. RM-A2P (RB Atmosphere particulate)	W(1)(4)	M(4)	E(4)	
e. RM-A2I (RB Atmosphere iodine)	W(1)(4)	M(4)	Q(4)	(2) RM-G6 and RM-G7 operability requirements are given in T.S. 3.8.1. Surveillances are required to be current only when handling irradiated fuel.
f. RM-A2G (RB Atmosphere gas)	W(1)(4)	M(4)	E(4)	
				(3) RM-G9 operability requirements are given in T.S. 3.8.1.
				(4) RM-A2 operability requirements are given in T.S. 3.1.6.8
29. High and Low Pressure Injection Systems: Flow Channels	N/A	N/A	R	

* Includes only monitors indicated under this item. Other T.S. required radiation monitors are included in specifications 3.5.5.2, 4.1.3, Table 3.5-1 item C.3.f, and Table 4.1-1 item 19e.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
30. Borated Water Storage Tank Level Indicator	W	NA	R	
31. DELETED				
32. DELETED				
33. Containment Temperature	NA	NA	F	
34. Incore Neutron Detectors	M(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout when reactor power is greater than 15%.
35. Emergency Plant Radiation Instruments	M(1)	NA	F	(1) Battery Check.
36. (DELETED)				
37. Reactor Building Sump Level	NA	NA	R	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION		CHECK	TEST	CALIBRATE	REMARKS
38.	OTSG Full Range Level	W	NA	R	
39.	Turbine Overspeed Trip	NA	R	NA	
40.	BWST/NaOH Differential Pressure Indicator	NA	NA	F	
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	F	
42.	Diesel Generator Protective Relaying	NA	NA	R	
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)				
	a. Degraded Grid	NA	M(1)	A	(1) Relay operation will be checked by local test pushbuttons.
	b. Loss of Voltage	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F.
45.	Loss of Feedwater Reactor Trip	S(1)	S/A(1)	R	(1) When reactor power exceeds 7% power.
46.	Turbine Trip/Reactor Trip	S(1)	S/A(1)	F	(1) When reactor power exceeds 45% power.
47.	a. Pressurizer Code Safety Valve and PORV Tailpipe Flow Monitors	S(1)	NA	F	(1) When T_{ave} is greater than 525°F.
	b. PORV – Acoustic/Flow	NA	M(1)	R	(1) When T_{ave} is greater than 525°F.
48.	PORV Setpoints	NA	M(1)	R	(1) Per Specification 3.1.12 excluding valve operation.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
49. Saturation Margin Monitor	S(1)	M(1)	R	(1)When T_{ave} is greater than 525°F.
50. Emergency Feedwater Flow Instrumentation	NA	M(1)	F	(1)When T_{ave} is greater than 250°F.
51. Heat Sink Protection System				
a. EFW Auto Initiation Instrument Channels				(1)Includes logic test only.
1. Loss of Both Feedwater Pumps	NA	Q(1)	F	
2. Loss of All RC Pumps	NA	Q(1)	R	
3. Reactor Building Pressure	NA	Q	F	
4. OTSG Low Level	W	Q	R	
b. MFW Isolation OTSG Low Pressure	NA	Q	R	
c. EFW Control Valve Control System				
1. OTSG Level Loops	W	Q	R	
2. Controllers	W	NA	R	
d. HSPS Train Actuation Logic	NA	Q(1)	R	
52. Backup Incore Thermocouple Display	M(1)	NA	R	(1)When T_{ave} is greater than 250°F.
53. Deleted				
54. Reactor Vessel Water Level	NA	NA	R	

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2. Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3. Pressurizer Safety Valves	Setpoint	In accordance with the Inservice Testing Program
4. Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5. Refueling System Interlocks	Functional	Start of each refueling period
6. (Deleted)	--	--
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525 degrees F
8. (Deleted)	--	--
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visual inspection of Intake Pump House Floor	Not to exceed 24 months
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer Block Valve (RC-V2)	Functional*	Quarterly

* Function shall be demonstrated by operating the valve through one complete cycle of full travel.

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

	Item		Check	Frequency
1.	Reactor Coolant	a.	Specific Activity Determination to compare to the $100/\bar{E}$ µCi/gm limit	At least once each 7 days during POWER OPERATION, HOT STANDBY, START-UP, and HOT SHUTDOWN.
		b.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	i) 1 per 14 days during power operations.
				ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during power operation, start-up and hot standby.
				iii) # Once per 4 hours, whenever the specific activity exceeds 0.35 µCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$µCi/gram during all modes but refueling.
		c.	Radiochemical for $\bar{E}$ Determination	1 per 6 months* during power operation.
		d.	Chemistry (Cl, F and O2)	5 times/week when Tavg IS GREATER THAN 200°F.
		e.	Boron concentration	2 times/week
		f.	Tritium Radioactivity	Monthly
2.	Borated Water Storage Tank Water Sample		Boron concentration	Weekly and after each makeup when reactor coolant system pressure is greater than 300 psig or Tavg is greater than 200°F.
3.	Core Flooding Tank Water Sample		Boron concentration	Monthly and after each makeup when RCS pressure is greater than 700 psig.

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron Concentration greater than or equal to 600 ppmb	Weekly
5.	Secondary Coolant	Isotopic analysis for DOSE EQUIVALENT I-131 concentration	At least once per 72 hours when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.
6.	Deleted		
7.	Deleted		
8.	Deleted		
9.	Deleted		
10.	Sodium Hydroxide Tank	Concentration	Semi-Annually and after each makeup.
11.	Deleted		
12.	Deleted		

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

** Deleted

*** Deleted

TABLE 4.1-4

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK	TEST	CALIBRATE	REMARKS
1.	Noble Gas Effluent				
	a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)	W	M	F	(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that this monitor is functioning.
	b. Condenser Vacuum Pump Exhaust (RM-G25)	W(1)	M	F	
	c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)	W	M	F	
	d. Reactor Building Purge Exhaust (RM-A9-Hi)	W	M	F	
	e. Reactor Building Purge Exhaust (RM-G24)	W(1)	M	F	
	f. Main Steam Lines Radiation (RM-G26/RM-G27)	W(1)	M	F	
2.	Containment High Range Radiation (RM-G22/G23)	W	M	R	
3.	Containment Pressure	W	N/A	F	
4.	Containment Water Level	W	N/A	R	
5.	DELETED				
6.	Wide Range Neutron Flux	W	N/A	F	

TABLE 4.1-4 (Continued)

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	INSTRUMENTS	CHECK	TEST	CALIBRATE	REMARKS
7.	Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)	W	N/A	R	
8.	Reactor Coolant System Hot Leg (TE-958, 960; TI-958A, 960A)	W	N/A	R	
9.	Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)	W	N/A	R	
10.	Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)	W	N/A	R	
11.	Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)	W	N/A	F	

TABLE 4.1-5
SYSTEM SURVEILLANCE REQUIREMENTS

Item	Test	Frequency
1. Core Flood Tank	a. Verify two core flood tanks each contain 940 $\pm$ 30 ft^3 borated water.	S
	b. Verify that two core flood tanks each contain 600 $\pm$ 25 psig.	S
	c. Verify CF-V-1A&B are fully open.	S
	d. Verify power is removed from CF-V-1A&B and CF-V-3A&B valve operators	M

4.2 REACTOR COOLANT SYSTEM INSERVICE INSPECTION AND TESTING

Applicability

This technical specification applies to the inservice inspection (ISI) and inservice testing (IST) of the reactor coolant system pressure boundary and portions of other safety oriented system pressure boundaries.

Objective

The objective of the ISI and IST programs is to provide assurance of the continuing integrity of the reactor coolant system while at the same time minimizing radiation exposure to personnel in the performance of inservice inspections and tests.

Specification

4.2.1 ISI of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

4.2.2 IST of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the NRC.

4.2.3 (Deleted)

4.2.4 The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within the first ISI period, two reactor coolant pump motor flywheel assemblies within the first two ISI periods and all four by the end of the 10 year inspection interval. However, the U.T. procedure is developmental and will be used only to the extent that it is shown to be meaningful. The extent of coverage will be limited to those areas of the flywheel which are accessible without motor disassembly, i.e., can be reached through the access ports. Also, if radiation levels at the lower access ports are prohibitive, only the upper access ports will be used.

4.2.5 (Deleted)

4.2.6 (Deleted)

4.2.7 A surveillance program for the pressure isolation valves between the primary coolant system and the low pressure injection system shall be as follows:

1. Periodic leakage testing[a] at test differential pressure greater than 150 psid shall be accomplished for the valves listed in Table 3.1.6.1 for the following conditions:

   (a) prior to achieving hot shutdown after returning the valve to service following maintenance repair or replacement work, and

   (b) prior to achieving hot shutdown following a cold shutdown of greater than 72 hours duration unless testing has been performed within the previous 9 months.

2. Whenever integrity of a pressure isolation valve listed in Table 3.1.6.1 cannot be demonstrated, the integrity of the other remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Bases

Specifications 4.2.1 and 2 ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, paragraphs 55a(f) and 55a(g). Relief from any of the above requirements has been provided in writing by the NRC and is not a part of these technical specifications.

4.3 DELETED

(Pages 4-14 through 4-28 deleted)

4.4 REACTOR BUILDING

4.4.1 CONTAINMENT LEAKAGE TESTS

Applicability

Applies to containment leakage.

Objective.

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Testing (ILRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program at test frequencies established in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2 Local Leakage Rate Testing (LLRT) shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program. LLRT shall be performed at a pressure not less than peak accident pressure P_{ac} with the exception that the airlock door seal tests shall normally be performed at 10 psig and the periodic containment airlock tests shall be performed at a pressure not less than P_{ac}. LLRT frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Operability of the personnel and emergency air lock door interlocks and the associated control room annunciator circuits shall be determined at least once per six months. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable, except as provided in Technical Specification Section 3.8.6.

Bases (1)

The Reactor Building is designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions. The peak calculated Reactor Building pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig. The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}. Containment Isolation Valves are addressed in the UFSAR (Reference 2).

4.4 REACTOR BUILDING (Continued)

The Reactor Building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program (See Section 6.8.5). This program is contained in the surveillance procedures for Reactor Building inspection, Integrated Leak Rate Testing, and Local Leak Rate Testing. These periodic testing requirements verify that Reactor Building leakage rate does not exceed the assumptions used in the safety analysis. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

Periodic surveillance of the airlock interlock systems (Reference 4) is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an airlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

Reference

(1) UFSAR, Chapter 5.7.4 - "Post Operational Leakage Rate Tests"
(2) UFSAR, Tables 5.7-1 and 5.7-3
(3) DELETED.
(4) UFSAR, Table 5.7-2

(Pages 4-31 through 4-34, 4-34a, and 4-34b deleted)

4.4.2 Structural Integrity

Specification

4.4.2.1 Inservice Tendon Surveillance Requirements

The surveillance program for structural integrity and corrosion protection conforms to the requirements of Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a. The detailed surveillance program for the prestressing system tendons shall be based on periodic inspection and mechanical tests to be performed on selected tendons.

4.4.2.1.1 DELETED

4.4.2.1.2 DELETED

4.4.2.1.3 DELETED

4.4.2.1.4 Tendon Surveillance Previous Inspections

The tendon surveillance shall include the reexamination of all abnormalities (i.e., concrete scaling, cracking, grease leakage, etc.) discovered in the previous inspection to determine whether conditions have stabilized. The inspection program shall be modified accordingly if obvious deteriorating conditions are observed.

4.4.2.1.5 Inspection for Crack Growth at Dome Tendons in the Ring Girder Anchorage Areas

Concrete around the dome tendon anchorage areas shall be inspected for crack growth during ten and 15 year inspections by monitoring cracks greater than 0.005 inch in width. Select as a minimum nine dome tendon anchoring areas having concrete cracks with crack widths 0.005 inch. In the selection of dome tendon anchoring areas to be monitored, preference shall be given to those areas having cracks greater than 0.005 inch in width. The width, depth (if depths can be measured with simple existing plant instruments, (i.e., feeler gauges, wires) and length of the selected cracks shall be measured and mapped by charting. This inspection may be discontinued, if the concrete cracks show no sign of growth. If, however, these inspections indicate crack growth, an investigation of the causes and safety impact should be performed.

4.4.2.1.6 Reports

a. Within 3 months after the completion of each tendon surveillance a special report shall be submitted to the NRC Region I Administrator. This Report will include a section dealing with trends for the rate of prestress loss as compared to the predicted rate for the duration of the plant life (after an adequate number of surveillances have been completed).

b. Reports submitted in accordance with 10 CFR 50.73 shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and any corrective actions taken.

4.4.3 DELETED

BASES

For ungrouted, post-tensioned tendons, this surveillance requirement ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the TMI-1 Reactor Building Structural Integrity Tendon Surveillance Program. Testing and frequency are consistent with the requirements of Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a, and as described in the FSAR.

The modified visual inspection requirements pertaining to the dome tendons in the ring girder were implemented as a result of: 1) discovery of ring girder voids in 1977 and the potential that more undetected voids in the ring girder could exist, and 2) the number of dome tendon bearing areas having cracks appeared to be growing with time (Reference Amendment No. 59).

REFERENCES

(1) UFSAR, Section 5.7.5 - Tendon Stress Surveillances

4.4.4 DELETED

4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM & REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 *Emergency Loading Sequence*

Applicability: Applies to periodic testing requirements for safety actuation systems.

Objective: To verify that the emergency loading sequence and automatic power transfer is operable.

Specifications:

4.5.1.1 Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power.

-M. U. Pump
-D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
-R. B. Cooling Pump
-R. B. Ventilators
-D. H. Closed Cycle Cooling Pump
-N. S. Closed Cycle Cooling Pump
-D. H. River Cooling Pump
-N. S. River Cooling Pump
-D. H. and N. S. Pump Area Cooling Fan
-Screen House Area Cooling Fan
-Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psig Pressure Test Signal.)
-Motor Driven Emergency Feedwater Pump

c. Following successful transfer to the emergency diesel, the diesel generator breaker will be opened to simulate trip of the generator then re-closed to verify block load on the reclosure.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel.

Bases

The Emergency loading sequence and automatic power transfer controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system.

Automatic start and loading of the emergency diesel generator to meet the requirements of 4.5.1.1b/c above is described in Technical Specification 4.6.1.b.

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves (MU-V-14A/B & 16A/B/C/D) have completed their travel and the make-up pumps are running as evidenced by system flow. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

1) Indicated RCS temperature shall be greater than 329°F.
2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. During each refueling period and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance, which affects the position indicator.

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. Verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At intervals not to exceed 3 months, the components required for emergency core cooling will be tested.

b. The test will be considered satisfactory if the pumps and fans have been successfully started and the valves have completed their travel as evidenced by the control board component operating lights, and a second means of verification, such as: the station computer, verification of pressure/flow, or control board indicating lights initiated by separate limit switch contacts.

Bases

The emergency core cooling systems (Reference 1) are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Reference

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

4.5.3 REACTOR BUILDING COOLING AND ISOLATION SYSTEM

Applicability

Applies to testing of the reactor building cooling and isolation systems.

Objective

To verify that the reactor building cooling systems are operable Specification

4.5.3.1 System Tests

a. Reactor Building Spray System

1. At each refueling interval and simultaneously with the test of the emergency loading sequence, a Reactor Building 30 psi high pressure test signal will start the spray pump. Except for the spray pump suction valves, all engineered safeguards spray valves will be closed.

   Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

   The operation of the spray valves will be verified during the component test of the R. B. Cooling and Isolation System.

   The test will be considered satisfactory if the spray pumps have been successfully started.

2. Compressed air will be introduced into the spray headers to verify each spray nozzle is unobstructed at least every ten years.

b. Reactor Building Cooling and Isolation Systems

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system.

2. The test will be considered satisfactory if **measured system flow is greater than accident design flow rate.**

4.5.3.2 Component Tests

a. At intervals not to exceed three months, the components required for Reactor Building Cooling and Isolation will be tested.

b. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights and a second means of verification, such as: the station computer, local verification, verification of pressure/flow, or control board component operating lights initiated by separate limit switch contacts.

Bases

The Reactor Building Cooling and Isolation Systems and Reactor Building Spray System are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure (References 1 and 2).

The delivery capability of one Reactor Building Spray Pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump.

With the pumps shut down and the Borated Water Storage Tank outlet valve closed, the Reactor Building spray injection valves can each be opened and closed by operator action. With the Reactor Building spray inlet valves closed, low pressure air can be blown through the test connections of the Reactor Building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment.

The Reactor Building fans are normally operating periodically, constituting the test that these fans are operable.

Reference

(1) UFSAR, Section 6.2 - "Reactor Building Spray System"
(2) UFSAR, Section 6.3 - "Reactor Building Emergency Cooling System"

4.5.4 ENGINEERED SAFEGUARDS FEATURE (ESF) SYSTEMS LEAKAGE

Applicability

Applies to those portions of the Decay Heat, Building Spray, and Make-Up Systems, which are required to contain post accident sump recirculation fluid, when these systems are required to be operable in accordance with Technical Specification 3.3.

Objective

To maintain a low leakage rate from the ESF systems in order to prevent significant off-site exposures and dose consequences.

Specification

4.5.4.1 The total maximum allowable leakage into the Auxiliary Building from the applicable portions of the Decay Heat, Building Spray and Make-Up System components as measured during refueling interval tests in Specification 4.5.4.2 shall not exceed 15 gallons per hour.

4.5.4.2 Once each refueling interval the following tests of the applicable portions of the Decay Heat Removal, Building Spray and Make-Up Systems shall be conducted to determine leakage:

a. The applicable portion of the Decay Heat Removal System that is outside containment shall be leak tested with the Decay Heat pump operating, except as specified in "b".

b. Piping from the Reactor Building Sump to the Building Spray pump and Decay Heat Removal System pump suction isolation valves shall be pressure tested at no less than 55 psig.

c. The applicable portion of the Building Spray system that is outside containment shall be leak tested with the Building Spray pumps operating and BS-V-1A/B closed, except as specified in "b" above.

d. The applicable portion of the Make-Up system on the suction side of the Make-Up pumps shall be leak tested with a Decay Heat pump operating and DH-V-7A/B open.

e. The applicable portion of the Make-Up system from the Make-Up pumps to the containment boundary valves (MU-V-16A/D, 18, and 20) shall be leak tested with a Make-Up pump operating.

f. Visual inspection shall be made for leakage from components of these systems. Leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit of 15 gph (measured in standard room temperature gallons) for the accident recirculation portions of the Decay Heat Removal (DHR), Building Spray (BS), and Make-Up (MU) systems is based on ensuring that potential leakage after a loss-of-coolant accident will not result in off-site dose consequences in excess of those calculated to comply with the 10 CFR 50.67 limits (Reference 1 and 2). The test methods prescribed in 4.5.4.2 above for the applicable portions of the DH, BS and MU systems ensure that the testing results account for the highest pressure within that system during the sump recirculation phase of a design basis accident.

References

(1) UFSAR, Section 6.4.4 - "Design Basis Leakage"

(2) UFSAR, Section 14.2.2.5(d) - "Effects of Engineered Safeguards Leakage During Maximum Hypothetical Accident"

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirement of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiate start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be effected.

b. Automatically start and loading the emergency diesel generator in accordance with Specification 4.5.1.1.b/c including the following. This test will be conducted every refueling interval on each diesel generator.

(1) Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in ≤ 10 seconds.

(2) Verify that the diesel block loads upon simulated loss of offsite power in ≤ 30 seconds.

(3) The diesel operates with the permanently connected and auto connected load for ≥ 5 minutes.

(4) The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

(5) The diesel generator block loads and operates for ≥ 5 minutes upon reclosure of the diesel generator breaker.

c. Deleted.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded:

(1) every 92 days
(2) once within 24 hours after a battery discharge <105 V
(3) once within 24 hours after a battery overcharge >150 V
(4) If any cell parameters are not met, measure and record the parameters on each connected cell every 7 days thereafter until all battery parameters are met.

b. The voltage and specific gravity of a pilot cell will be measured and recorded weekly. If any pilot cell parameters are not met, perform surveillance 4.6.2.a on each connected cell within 24 hours and every 7 days thereafter until all battery parameters are met.

c. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

d. The battery will be subjected to a load test on a refueling interval basis.

(1) Verify battery capacity exceeds that required to meet design loads.

(2) Any battery which is demonstrated to have less than 85% of manufacturers ratings during a capacity discharge test shall be replaced during the subsequent refueling outage.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted at least once each refueling:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the emergency power bus and energized. Upon completion of this test, the heaters shall be returned to their normal power bus.

(2) *Demonstrate that the pressurizer heaters breaker on the emergency* bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verify that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential overload condition. The testing frequency specified is intended to identify and *permit correction of any mechanical or electrical deficiency before it can* result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition in these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valve is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an *emergency bus provides assurance that these heaters can be energized during a* loss of offsite power condition to maintain natural circulation.

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 CONTROL ROD DRIVE SYSTEM FUNCTIONAL TESTS

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the axial power shaping rods (APSRs), from the fully withdrawn position to ¾ insertion (104 inches travel) shall not exceed 1.66 seconds at hot reactor coolant full flow conditions or 1.40 seconds for the hot no flow conditions (Reference 1). For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has actuated the 25% withdrawn reference switch during insertion from the fully withdrawn position. The specified trip time is based upon the safety analysis in UFSAR, Chapter 14 and the Accident Parameters as specified therein.

Each control rod drive mechanism shall be exercised by a movement of a minimum of 3% of travel every 92 days.This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod deviates from its group average position by more than nine inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCE

(1) UFSAR, Section 3.1.2.4.3 - "Control Rod Drive Mechanism"

The page intentionally left blank

4.8 DELETED

4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for DHR are capable of performing their design function.

Specification

4.9.1 Reactor Coolant System (RCS) Temperature greater than 250 degrees F.

4.9.1.1 Verify each Emergency Feedwater (EFW) Pump is tested in accordance with the requirements and acceptance criteria of the ASME Section XI Inservice Test Program.

Note: This surveillance is not required to be performed for the turbine-driven EFW Pump (EF-P-1) until 24 hours after exceeding 750 psig.

4.9.1.2 DELETED

4.9.1.3 At least once per 31 days, each EFW System flowpath valve from both Condensate Storage Tanks (CSTs) to the OTSGs via the motor-driven pumps and the turbine-driven pump shall be verified to be in the required status.

4.9.1.4 On a refueling interval basis:

a) Verify that each EFW Pump starts automatically upon receipt of an EFW test signal.

b) Verify that each EFW control valve responds upon receipt of an EFW test signal.

c) Verify that each EFW control valve responds in manual control from the control room and remote shutdown panel.

4.9.1.5 Prior to STARTUP, following a REFUELING SHUTDOWN or a COLD SHUTDOWN greater than 30 days, conduct a test to demonstrate that the motor driven EFW Pumps can pump water from the CSTs to the Steam Generators.

4.9.1.6 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 RCS Temperature less than or equal to 250 degrees F.*

4.9.2.1 On a daily basis, verify operability of the means for DHR required by Specification 3.4.2 by observation of console status indication.

* These requirements supplement the requirements of Specifications 4.5.2.2 and 4.5.4.

Bases

ASME Section XI specifies requirements and acceptance standards for the testing of nuclear safety related pumps. The quarterly EFW Pump test frequency specified by the ASME Section XI Code will be sufficient to verify that the turbine-driven and both motor-driven EFW Pumps are operable. Compliance with the normal acceptance criteria assures that the EFW Pumps are operating as expected. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Deferral of the requirement to perform IST on the turbine-driven EFW Pump is necessary to assure sufficient OTSG pressure to perform the test using Main Steam.

Daily verification of the operability of the required means for DHR ensures that sufficient DHR capability will be maintained.

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

Specification

4.10.1 Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation will be made to determine the cause of the discrepancy.

Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than one percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of one percent is considered a safe limit since a shutdown margin of at least one percent with the most reactive rod in the fully withdrawn position is always maintained.

4.11 REACTOR COOLANT SYSTEM VENTS

Applicability

Applies to Reactor Coolant System Vents.

Objective

To ensure that Reactor Coolant System vents are able to perform their design function.

Specification

4.11.1 Each reactor coolant system vent path shall be demonstrated OPERABLE once per refueling interval by cycling each power operated valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

BASES

Frequency of tests specified above are necessary to ensure that the individual Reactor Coolant System Vents will perform their functions. It is not advisable to perform these tests during Plant Power Operation, or when there is significant pressure in the Reactor Coolant System. Tests are, therefore, to be performed during either Cold Shutdown or Refueling.

4.12 AIR TREATMENT SYSTEM

4.12.1 EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the emergency control room air treatment system and associated components.

Objective

To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.1.1 At least every refueling interval, the pressure drop across the combined HEPA filters and charcoal adsorber banks of AH-F3A and 3B shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

4.12.1.2

a. The tests and sample analysis required by Specification 3.15.1.2 shall be performed initially and at least once per year for standby service or after every 720 hours of system operation and following significant painting, steam, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing which could affect the HEPA filter bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing which could effect the charcoal adsorber bank bypass leakage.

d. Each AH-E18A and B (AH-F3A and B) fan/filter circuit shall be operating at least 10 hours every month.

4.12.1.3 At least once per refueling interval, automatic initiation of the required Control Building dampers for isolation and recirculation shall be demonstrated as operable.

4.12.1.4 An air distribution test shall be performed on the HEPA filter bank initially, and after any maintenance or testing that could affect the air distribution within the system. The air distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 40,000 cfm (±10%) flow rate.

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per refueling cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon shall be performed in accordance with approved test procedures. Replacement adsorbent should be qualified according to ASTM D3803-1989. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable all adsorbent in the system shall be replaced. Tests of the HEPA filters with DOP aerosol shall also be performed in accordance with approved test procedures. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52 March 1978.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

If significant painting, steam, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the Vice President-TMI Unit 1.

Demonstration of the automatic initiation of the recirculation mode of operation is necessary to assure system performance capability. Dampers required for control building isolation and recirculation are specified in UFSAR Sections 7.4.5 and 9.8.1.

4.12.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Deleted

THIS PAGE LEFT BLANK INTENTIONALLY

4.12.3 AUXILIARY AND FUEL HANDLING BUILDING AIR TREATMENT SYSTEM

DELETED

THIS PAGE LEFT BLANK INTENTIONALLY

4.12.4 FUEL HANDLING BUILDING ESF AIR TREATMENT SYSTEM

Applicability

Applies to Fuel Handling Building (FHB) ESF Air Treatment System and associated components.

Objective

To verify that this sytem and associated components will be able to perform its design functions.

Specification

4.12.4.1 Each refueling interval prior to movement of irradiated fuel:

a. The pressure drop across the entire filtration unit shall be demonstrated to be less than 7.0 inches of water at 6,000 cfm flow rate (±10%).

b. The tests and sample analysis required by Specification 3.15.4.2 shall be performed.

4.12.4.2 Testing necessary to demonstrate operability shall be performed as follows:

a. The tests and sample analysis required by Specification 3.15.4.2 shall be performed following significant painting, steam, fire, or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank, and after any structural maintenance on the system housing that could affect the HEPA filter bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank, and after any structural maintenance on the system housing that could affect charcoal adsorber bank bypass leakage.

4.12.4.3 Each filter train shall be operated at least 10 hours every month.

4.12.4.4 An air flow distribution test shall be performed on the HEPA filter bank initially and after any maintenance or testing that could affect the air flow distribution within the system. The distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 6,000 cfm ± 10% flow rate.

Bases

The FHB ESF Air Treatment System is a system which is normally kept in a "standby" operating status. Tests and sample analysis assure that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedure should allow for the removal of a sample from one adsorber test canister. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The in-place test criteria for activated charcoal will meet the guidelines of ANSI-N510-1980. The laboratory test of charcoal will be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified in accordance with ASTM D3803-1989. Any HEPA filters found defective will be replaced with filters qualified in accordance with ANSI-N509-1980.

Pressure drop across the entire filtration unit of less than 7.0 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture buildup on the adsorbers and HEPA filters.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational movement of irradiated fuel. The determination of what is significant shall be made by the Vice President-TMI Unit 1.

4.13 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources previously subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.14 DELETED

4.15 MAIN STEAM SYSTEM INSERVICE INSPECTION

Applicability

This technical specification applies to the inservice inspection of four welds in the Main Steam System identified as MS-0001, MS-0002, MS-0003, and MS-0004L of the TMI-1 Inservice Inspection Program.

Objective

The objective of the Inservice Inspection Program is to provide assurance of the continuing integrity of that portion of the Main Steam System in which a postulated failure would produce pressures in excess of the compartment wall and/or slab capacities.

Specification

4.15.1. The four weld joints identified above shall be 100 percent inspected in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant components, defined in the TMI-1 Inservice Inspection Program. Inspections are to be performed at a frequency of once every 3-1/2 years (or during the nearest refueling outage).

Prior to initial plant operation, a preoperational inspection of the identified weld joints will be performed and any data acquired will be recorded to form a baseline on which to compare results of subsequent inspections.

Bases

Calculations (Reference 1) postulated that breaks in the main steam lines at the containment penetrations in small compartments No. 2 and No. 5 could produce pressures in excess of wall and/or slab capacities.

Inspections are conducted at an inspection frequency of 3 1/2 year intervals following initial plant startup. These inspections have revealed that no degradation of the welds has occurred during the inspection cycles up to and including the 9R outage inspection. Consequently, as further degradation is not expected to occur, justification to extend the inspection frequency to once every ten (10) years is being developed. The conclusions of the technical benefit review will be submitted to the NRC for evaluation in a Technical Specification change request.

Reference

(1) UFSAR, Appendix 14A, Section 7.2.1

4.16 REACTOR INTERNALS VENT VALVES SURVEILLANCE

Applicability

Applies to Reactor Internals Vent Valves.

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specification

	Item	Test	Frequency
4.16.1	Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs. (applied vertically upward).	Each Refueling Shutdown

Bases

Verifying vent valve freedom of movement insures that coolant flow does not bypass the core through reactor internals vent valves during operation and therefore insures the conservatism of Core Protection Safety limits as delineated in Figures 2.1-1 and 2.1-3, and the flux/flow trip setpoint.

4.17 SHOCK SUPPRESSORS (SNUBBERS)

SURVEILLANCE REQUIREMENTS

4.17.1 Each snubber shall be demonstrated OPERABLE by performance of the following inspection program.

a. Snubber Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation and may be treated independently. The Director-Radiological Health and Safety, will ensure that a review is performed for ALARA considerations on all snubbers which are located in radiation areas for the determination of their accessibility. This review shall be in accordance with the recommendations of Regulatory Guides 8.8 and 8.10. The determination shall be based upon the known or projected radiation levels at each snubber location which would render the area inaccessible during reactor operation and based upon the expected time to perform the visual inspection. Snubbers may also be determined to be inaccessible because of their physical location due to an existing industrial safety hazard at the specific snubber location. This determination shall be reviewed and approved by the management position responsible for occupational safety.

Snubbers accessible during reactor operation shall be inspected in accordance with the schedule stated below. Snubbers scheduled for inspection that are inaccessible during reactor operation because of physical location or radiation levels shall be inspected during the next reactor shutdown greater than 48 hours where access is restored* unless previously inspected in accordance with the schedule stated below.

Visual inspections shall include all safety related snubbers and shall be performed in accordance with the following schedule:

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period**#
0	24 months ± 25%
1	16 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

* Snubbers may continue to be inaccessible during reactor shutdown greater than 48 hours (e.g. if purging of the reactor building is not permitted).

** The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

The provisions of Table 1.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

c. Refueling Outage Inspections

At least once each refueling cycle during shutdown, a visual inspection shall be performed of all safety related snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired operability and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.17-1f. When the reservoir outlet port of a snubber is found to be uncovered by fluid, the snubber shall only be declared operable if functional testing in both extension and retraction directions is satisfactory and an engineering evaluation concludes that this snubber is operable.

e. Functional Tests*

At least once each refueling interval during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period, or the sample plan used in the prior test period shall be used:

1) At least 10% of the total each type of snubber in use in the plant shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.17.1f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

* The four 550,000 lb reactor coolant pump snubbers are not included. The functional test program for reactor coolant pump snubbers is implemented in accordance with the schedule and other requirements of the snubber testing program.

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.17-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.17.1f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.17-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region testing of that type of snubber may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, provided all snubbers tested with the failed equipment during the day of equipment failure are retested.

The representative sample selected for functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practicable that they are representative of the various configurations, operating environments, and the range of size and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If, during the functional test, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Snubber activation (restraining action or lockup) is achieved within the specified velocity range in both tension and compression.

2) Snubber release rate (bleed) is achieved in both tension and compression, within the specified range.

3) Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly, or parameters other than those specified, if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

1. Cause of Failure Evaluation

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers, irrespective of type, which may be subject to the same failure mode.

2. Damage Evaluation

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to activate (lockup) or fails to extend or retract, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type which are subject to the same defect shall be evaluated in a manner to ensure operability. This testing requirement shall be independent of the requirements stated in Specification 4.17.1e for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall have been tested to meet the functional test criteria before installation in the unit.

SURVEILLANCE REQUIREMENTS (Continued)

i. Snubber Seal Service Program

A snubber seal service life program shall be developed whereby the seal service life of hydraulic snubbers is monitored to ensure that the service life is not exceeded between surveillance inspections. The designated service life for the various seals shall be established based on engineering information. The seals shall be replaced so that the indicated service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.m.

Bases

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection includes verification of proper orientation, adequate hydraulic fluid level, and proper attachment of snubber to piping and structures.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule. Those snubbers which are inaccessible during reactor operation are not required to be inspected in accordance with the indicated inspection interval but must be inspected during the next shutdown when access is restored.

When the cause of the rejection of a snubber by visual inspection is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, that snubber may be exempted from being counted as inoperable if it is determined operable by functional testing. Generically susceptible snubbers are those snubbers which are of a specific make or model and have the same design features directly related to rejection of the snubbers by visual inspection, or are similiarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, one of the two sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.17-1.

Figure 4.17-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Snubber seal service life is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records. The requirement to monitor the snubber seal service life is included to ensure that the seals periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber seal service life. The requirements for the maintenance of records and the snubber seal service life are not intended to affect plant operation.

A technique and method for functional testing of the 550,000 lb. reactor coolant pump snubbers is currently under development. The functional test program shall be developed by Cycle 6 refueling or July 1, 1985, whichever is earlier. The functional test program shall be implemented in accordance with the schedule and other requirements of the program.

A list of individual snubbers with appropriate detailed information is maintained at the plant site. As a basis for permanent deletion of a snubber from the list of safety related snubbers, an engineering analysis must be performed to verify that the original safety analysis design criteria are either met or exceeded. Snubber additions and deletions are reported to the NRC in accordance with 10 CFR 50.59 requirements.

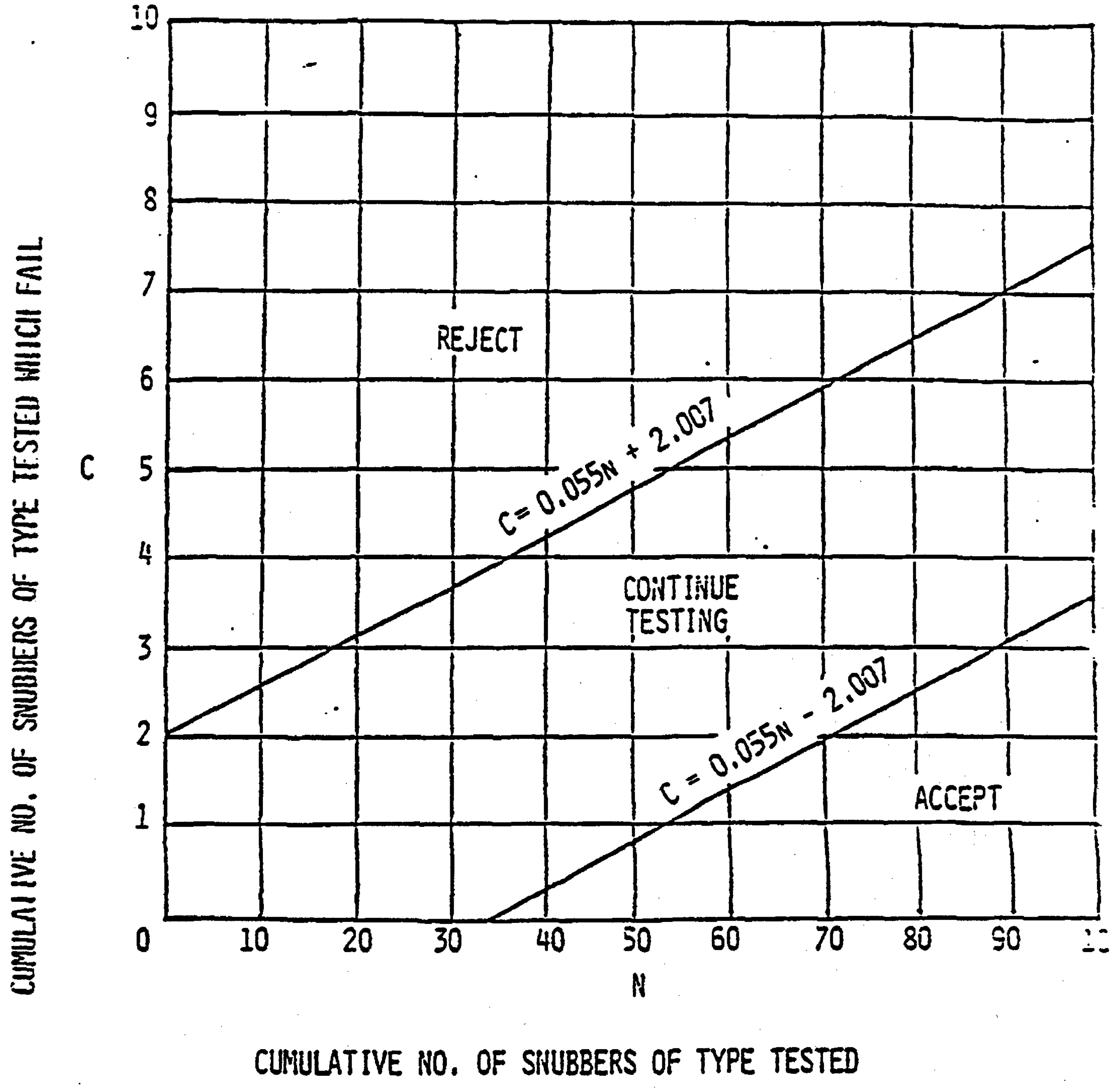


FIGURE 4.17-1

SNUBBER FUNCTIONAL TEST - SAMPLE PLAN 2

THIS PAGE INTENTIONALLY LEFT BLANK

(Pages 4-69 through 4-76A deleted)

4.19 OTSG TUBE INSERVICE INSPECTION

Applicability

This Technical Specification applies to the inservice inspection of the OTSG tube portion of the reactor coolant pressure boundary.

Objective

The objective of this inservice inspection program is to provide assurance of continued integrity of the tube portion of the Once-Through Steam Generators, while at the same time minimizing radiation exposure to personnel in the performance of the inspection.

Specification

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.1.6.3.

4.19.1 Steam Generator Sample Selection and Inspection Methods

a. Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.19.1 at the frequency specified in 4.19.3.

b. Inservice inspection of steam generator tubing shall include nondestructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall be calibrated to provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.19.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.19.2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.19.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.19.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except·

a. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (>20%).

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 4.19.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

   4. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

      (1) Group A-1: Tubes in rows 73 through 79 adjacent to the open inspection lane, and tubes between and on lines drawn from tube 66-1 to tube 75-15 and from 86-1 to 77-15.

      (2) Group A-2: Tubes having a drilled opening in the 15th support plate.

b. The tubes selected as the second and third samples (if required by Table 4.19.2) during each inservice inspection may be subjected to a partial tube inspection provided:

   1. The tubes selected for these second and third samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   2. The inspection includes those portions of the tubes where imperfections were previously found.

c. Implementation of the repair criteria for Inside Diameter (ID) Inter-Granular Attack (IGA) requires 100% bobbin coil inspection of all non-plugged tubes in accordance with AmerGen Engineering Report, ECR No. TM 01-00328, during all subsequent steam generator inspection intervals pursuant to Section 4.19.3. ID IGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes, as defined in that report.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected in a steam generator are degraded tubes and none of the inspected tubes are defective.

4.19.2 Specification (Continued)

C-2 One or more tubes, but not more than 1% of the total tubes inspected in a steam generator are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected in a steam generator are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES: (1) *In all inspections, previously degraded tubes whose degradation has not* been spanned by a sleeve must exhibit significant increase in the applicable degradation size measurement (> 0.24 volt bobbin coil amplitude increase for inside diameter IGA indications or > 10% further wall penetration for all other degradation) to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.19.2.a.4, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.19.3 Inspection Frequencies

The required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first (baseline) inspection was performed after 6 effective full power months but within 24 calendar months of initial criticality. The subsequent inservice inspections shall be performed not more than 24 calendar months after the previous inspection. If the results of *two consecutive inspections for a given group of tubes** *encompassing not less than 18* calendar months all fall into the C-1 category or demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval for that group may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.19-2 at 40 month intervals for a given group of tubes* fall into Category C-3 the inspection frequency for that group shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.19.3.a; the interval may then be extended to a maximum of once per 40 months.

---

* A group of tubes means:
(a) All tubes inspected pursuant to 4.19.2.a.4, or
(b) All tubes in a steam generator less those inspected pursuant to 4.19.2.a.4

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.19-2 during the shutdown subsequent to any of the following conditions:

1. A seismic occurrence greater than the Operating Basis Earthquake.

2. A loss of coolant accident requiring actuation of engineering safeguards, or

3. A major main steam line or feedwater line break.

d. After primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.6.3, an inspection of the affected steam generator will be performed in accordance with the following criteria:

1. If the leak is above the 14th tube support plate in a Group as defined in Section 4.19.2.a.4(1) all of the tubes in this Group in the affected steam generator will be inspected above the 14th tube support plate. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2. If the leaking tube is not as defined in Section 4.19.3.d.1, then an inspection will be performed on the affected steam generator(s) in accordance with Table 4.19-2.

4.19.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawing or specifications. Eddy current testing indications less than degraded tube criteria specified in a.3 below may be considered imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing:

(a) an inside diameter (I.D.) IGA indication with a bobbin coil indication ≥ 0.2 volt or ≥ 0.13 inches axial extent or ≥ 0.26 inches circumferential extent, or |

(b) imperfections $\geq 20\%$ of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

6. Repair Limit means the extent of degradation at or beyond which the tube shall be repaired or removed from service because it may become unserviceable prior to the next inspection.

   This limit is equal to 40% of the nominal tube wall thickness. Inside diameter IGA indications shall be repaired or removed from service if they exceed an axial extent of 0.25 inches, or a circumferential extent of 0.52 inches, or a through wall degradation dimensions of ≥ 40% if assigned.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c., above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

9. Inside Diameter Inter-Granular Attack (IGA) Indication means an indication initiating on the inside diameter surface and confirmed by diagnostic ECT to have a volumetric morphology characteristic of IGA.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, of all tubes exceeding the repair limit and all tubes containing throughwall cracks) required by Table 4.19-2.

4.19.5 Reports

a. DELETED

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 90 days following completion of the inspection and repairs (main generator breaker closure). The report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Location, bobbin coil depth estimate (if determined), bobbin coil amplitude (if determined), and axial and circumferential extent for each inside diameter IGA indication, and

4. Identification of tubes repaired or removed from service.

5. The number of tubes repaired or removed from service in each steam generator,

6. An assessment of growth of inside diameter IGA degradation in accordance with the volumetric ID IGA management program contained in AmerGen Engineering Report, ECR No. TM 01-00328, and

7. Results of in-situ pressure testing, if performed.

c. Results of steam generator tube inspections which fall into Category C-3 require notification in accordance with 10 CFR 50.72 prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence in accordance with 10 CFR 50.73.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

The program for inservice inspection of steam generator tubes is based on modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The Unit is expected to be operated in a manner such that the primary and secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the primary or secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result.

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant activity, Specification 3.1.6.3.

The extent of cracking during plant operation would be limited by the limitation of total steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired or removed from service.

Wastage-type defects are unlikely with proper chemistry treatment of the primary or the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled inservice steam generator tube examinations. For tubes with ID IGA indications, additional conservatism is being applied to evaluate circumferential and axial dimensions for determining final disposition of the tube. For ID IGA indications through wall dimension will continue to be assigned to those indications where amplitude response permits measuring through wall dimension. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness. Tubes with I.D. initiated intergranular degradation may remain in service without % T.W. sizing if the degradation morphology has been characterized as not crack-like by diagnostic eddy current inspection and the degradation is of limited circumferential and axial length to ensure tube structural integrity. Additionally, serviceability for accident leakage under the limiting postulated Main Steam Line Break (MSLB) accident will be evaluated by determining that this I.D. initiated degradation mechanism is inactive (e.g. comparison of the outage examination results with the results from past outages meets the requirements of AmerGen Engineering Report, ECR No. TM 01-00328) and by successful in-situ pressure testing of a sample of these degraded tubes to evaluate their accident leakage potential when in-situ pressure tests are performed.

Where experience in similar plants with similar water chemistry, as documented by USNRC Bulletins/Notices, indicate critical areas to be inspected, at least 50% of the tubes inspected should be from these critical areas. First sample inspections sample size may be modified subject to NRC review and approval.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 on the first sample inspection (See Table 4.19.2), these results will be reported to NRC pursuant to the requirements of Specification 4.19.5.c. Such cases will be considered by the NRC on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

NOTE: The eddy current examination voltages referred to in this section (section 4.19) are based on a normalization procedure that sets the bobbin coil prime frequency peak-to-peak response from the four 20% through-wall holes of an ASME calibration standard to 4 volts.

TABLE 4.19-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	None
No. of Steam Generators per Unit	Two
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION:

1. The Inservice Inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in that steam generator if the results of the first and subsequent inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.19-2
STEAM GENERATION TUBE INSPECTION(2)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.(1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes.
			C-3	Perform action for C-3 result of first sample.	C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in other S.G. Provide notification to NRC pursuant to 10CFR50.72.b.2.i and submit a report pursuant to 10CFR50.73.-a.2.ii.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Provide notification to NRC pursuant to 10CFR50.72 b.2.i and submit a report pursuant to 10CFR50.73.a.2.ii.	N/A	N/A

Notes: (1) $S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) For tubes inspected pursuant to 4.19.2.a.4: No action is required for C-1 results. For C-2 results in one or both steam generators plug or repair defective tubes. For C-3 results in one or both steam generators, plug or repair defective tubes and provide notification to NRC pursuant to 10 CFR 50.72.b.2.i followed by a written report pursuant to 10 CFR 50.73.a.2.ii.

4.20 REACTOR BUILDING AIR TEMPERATURE

Applicability

This specification applies to the average air temperature of the primary containment during power operations.

Objective

To assure that the temperatures used in the safety analysis of the reactor building are not exceeded.

Specification

4.20.1 When the reactor is critical, the reactor building temperature will be checked once each twenty-four (24) hours. If any detector exceeds 130°F (120°F below elevation 320) the arithmetic average will be computed to assure compliance with Specification 3.17.1.

4.21 RADIOACTIVE EFFLUENT INSTRUMENTATION

Deleted

4.21.1 Radioactive Liquid Effluent Instrumentation

Deleted

4.21.2 Radioactive Gaseous Process & Effluent Monitoring Instrumentation

Deleted

4.22 RADIOACTIVE EFFLUENTS

Deleted

4.22.1 Liquid Effluents

Deleted

4.22.2 Gaseous Effluents

Deleted

4.22.3 Solid Radioactive Waste

Deleted

4.22.4 Total Dose

Deleted

4.23 RADIOLOGICAL ENVIRONMENTAL MONITORING

Deleted

4.23.1 Monitoring Program

Deleted

4.23.2 Land Use Census

Deleted

4.23.3 Interlaboratory Comparison Program

Deleted

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

Applicability

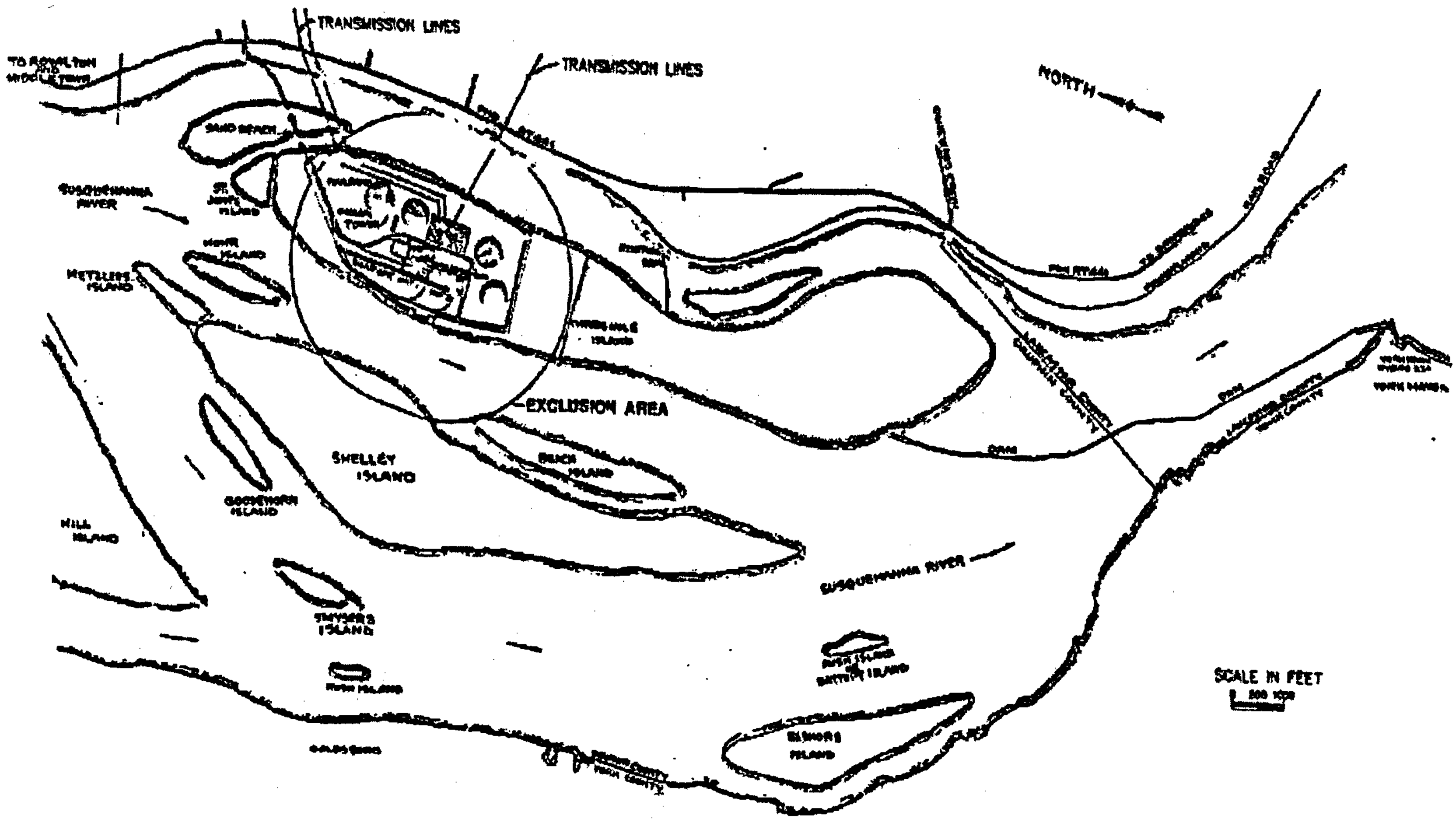
Applies to the location and extent of the exclusion boundary, restricted area, and low population zone.

Objective

To define the above by location and distance description.

Specification

5.1.1 The Three Mile Island Nuclear Station Unit 1 is located in an area of low population density about ten miles southeast of Harrisburg, PA. It is in Londonderry Township of Dauphin County, Pennsylvania, about two and one-half miles north of the southern tip of Dauphin County, where Dauphin is coterminal with York and Lancaster Counties. The station is located on an island approximately three miles in length situated in the Susquehanna River upstream from York Haven Dam. Figure 5.1 is an extended plot plan of the site showing the plant orientation and immediate surroundings. The Exclusion Area as defined in 10 CFR 100.3, is a 2,000 ft. radius, including portions of Three Mile Island, the river surface around it, and a portion of Shelley Island, which is owned by AmerGen Energy Company, LLC. The minimum distance of 2,000 ft. occurs on the shore of the mainland in a due easterly direction from the plant as shown on Figure 5.1 for the Exclusion Area. Figure 5-3 showing the physical location of the fence defines the "Restricted Area" surrounding the plant. The minimum distance of the "Restricted Area" is approximately 560 feet and is from the centerline of the TMI Unit 2 Reactor Building to a point on the westerly shoreline of Three Mile Island. The minimum distance to the outer boundary of the low population zone is two miles as shown on T.S. Figure 5-2, which also depicts the site topography for a radius of five miles. T.S. Figure 5-3 depicts the locations of gaseous effluent release points and liquid effluent outfalls (as tabularized on page 5-10), and the meteorological tower location (designated as 'weather tower' on the figure).

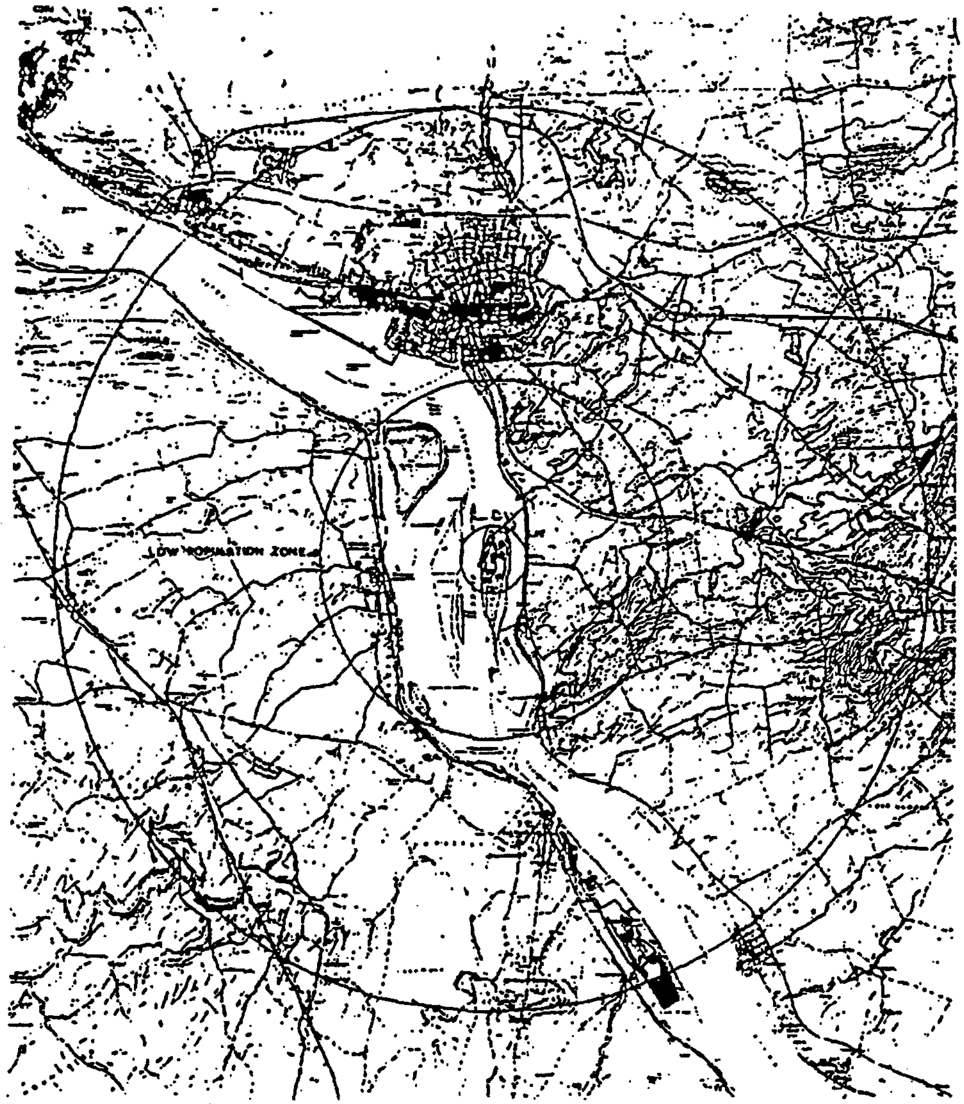


Amendment No. ~~140, 216~~, 246

AmerGen
Three Mile Island Nuclear Station

EXTENDED PLOT PLAN

CAD FILE: 6717R1,DWG FIG 5-1



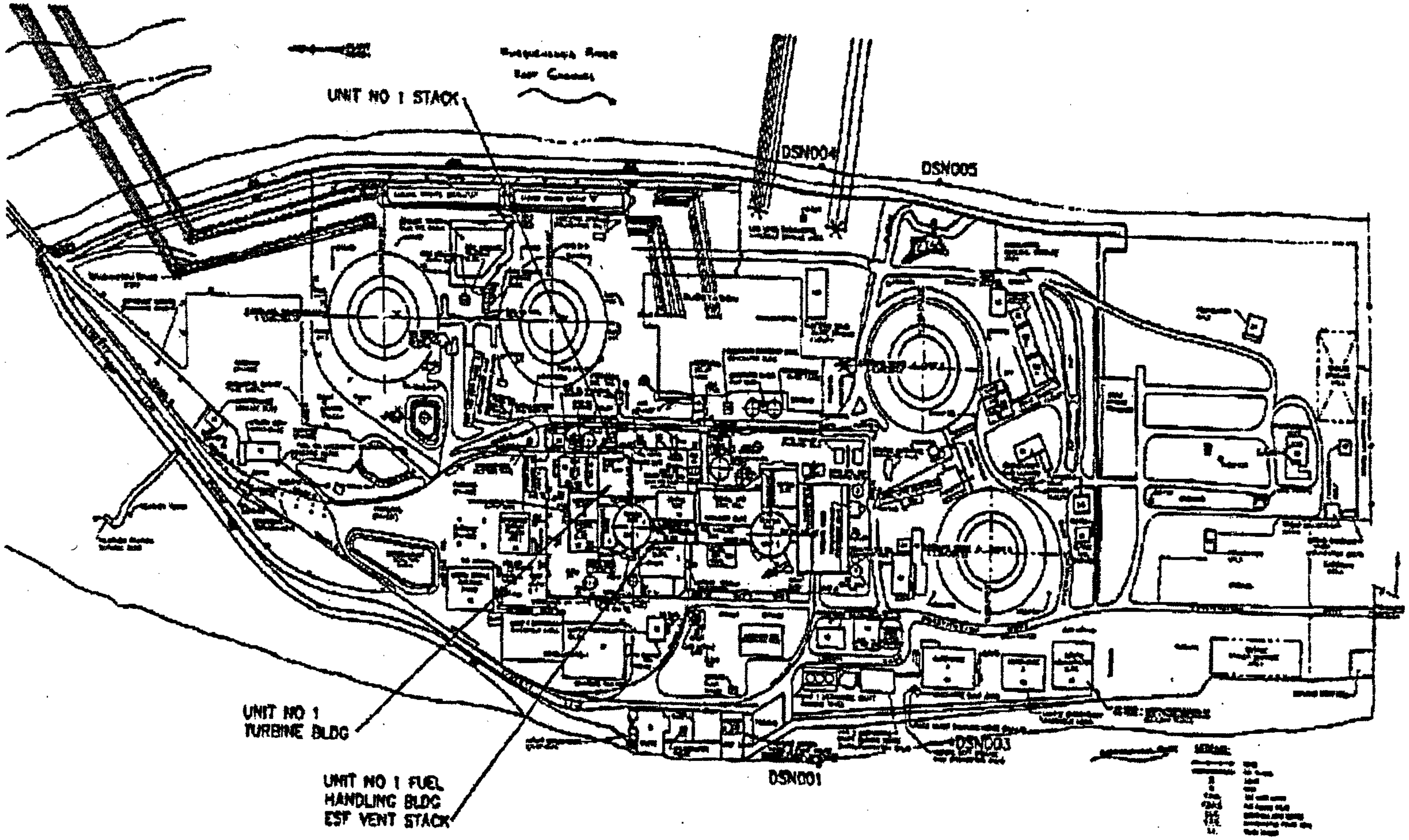
CONTOUR INTERVAL 20 FEET
DATUM IS MEAN SEA LEVEL

AmerGen

Site Topography
5 Mile Radius

Three Mile Island Nuclear Station

Fig. 5-2



1e-120-01-001 Rev. 55

Amendment No. ~~140, 216~~, 246

AmerGen
Three Mile Island Nuclear Station

Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

CAD FILE: 6716R15,DWG FIG 5-3

5.2 CONTAINMENT

Applicability

Applies to those design features of the containment system relating to operational and public safety.

Objective

To define the significant design features of the reactor containment.

Specification

Containment consists of two systems which are the reactor building and reactor building isolation system.

5.2.1 REACTOR BUILDING

The reactor building completely encloses the reactor and the associated reactor coolant systems. The reactor building is a reinforced concrete structure composed of cylindrical walls with a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The cylindrical walls are prestressed with a post-tensioning tendon system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three-way post-tensioning tendon system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness for containment.

The internal free volume of the reactor building is in excess of 2.0×10^6 cubic feet. The foundation mat is 9 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylindrical portion has an inside diameter of 130 ft, wall thickness of 3 ft 6 in., and a height of 157 ft from top of foundation slab to the spring line. The shallow dome roof has a large radius of 110 ft, a transition radius of 20 ft 6 in., a thickness of 3 ft, and an overall height of 32 ft 4 1/8 in.

The concrete containment building provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 55 psig and 281°F, respectively. The reactor building is designed for an external atmospheric pressure of 2.5 psi greater than the internal pressure.

Penetration assemblies are welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces (Reference 1).

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in Section 14, with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of energy sources and heat sinks.

5.2.2 REACTOR BUILDING ISOLATION SYSTEM

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves (Reference 2).

REFERENCES

(1) UFSAR Section 5.2.2.4.8 - "Penetrations and Openings"

(2) UFSAR Section 5.3.1 - "Isolation System - Design Bases"

5.3 REACTOR

Applicability

Applies to the design features of the reactor core and reactor coolant system.

Objective

To define the significant design features of the reactor core and reactor coolant system.

Specification

5.3.1 REACTOR CORE

5.3.1.1 A fuel assembly normally contains 208 fuel rods arranged in a 15 by 15 lattice. The reactor shall contain 177 fuel assemblies. Fuel rods shall be clad with zircaloy, ZIRLO, or zirconium-based M5 alloy materials and contain an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions. The details of the fuel assembly design are described in TMI-1 UFSAR Chapter 3.

5.3.1.2 The reactor core shall approximate a right circular cylinder with an equivalent diameter of 128.9 inches. The active fuel height is defined in TMI-1 UFSAR Chapter 3.

5.3.1.3 The core average and individual batch enrichments for the present cycle are described in TMI-1 UFSAR Chapter 3.

5.3.1.4 The control rod assemblies (CRA) and axial power shaping rod assemblies (APSRA) are distributed in the reactor core as shown in TMI-1 FSAR Chapter 3. The CRA and APSRA design data are also described in the UFSAR.

5.3.1.5 The TMI-1 core may contain burnable poison rod assemblies (BPRA) and gadolinia-urania integral burnable poison fuel pellets as described in TMI-1 UFSAR Chapter 3.

5.3.1.6 Reload fuel assemblies and rods shall conform to design and evaluation data described in the UFSAR. Enrichment shall not exceed a nominal 5.0 weight percent of U_{235}.

5.3.2 REACTOR COOLANT SYSTEM

5.3.2.1 The reactor coolant system shall be designed and constructed in accordance with code requirements. (Refer to UFSAR Chapter 4 for details of design and operation.)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F.

5.3.2.3 The reactor coolant system volume shall be less than 12,200 cubic feet.

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 NEW FUEL STORAGE

a. New fuel will normally be stored in the new fuel storage vault or spent fuel pools.

For the new fuel storage vault, the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21-1/8 inches in both directions. The spacing in the new fuel storage vault is sufficient to maintain K_{eff} less than 0.95 based on storage of fuel assemblies in clean unborated water or less than 0.98 based on storage in an optimum hypothetical low density moderator (fog or foam) for fuel assemblies with a nominal enrichment of 5.0 weight percent U^{235}. When fuel is being stored in the new fuel storage vault, twelve (12) storage locations (aligned in two rows of six locations each; transverse row numbers four and eight) must be left vacant of fissile or moderating material to provide sufficient neutron leakage to satisfy the NRC maximum allowable reactivity value under the optimum low moderator density condition.

For Spent Fuel Pool "A", the fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 11.1 inches in both directions for the Region I racks and 9.2 inches in both directions for the Region II racks. The spacing in the Spent Fuel Pool "A" storage locations for both Region I and II is adequate to maintain K_{eff} less than 0.95. Region I will store fuel with a maximum 5.0 percent initial enrichment. Region II will store new fuel with low enrichment. When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition.

For Spent Fuel Pool "B", the fuel assemblies are stored in racks in parallel rows, having nominal center to center distance of 13-5/8 inches in both directions. This spacing is sufficient to maintain a K_{eff} less than 0.95 based on fuel assemblies with a maximum enrichment of 4.37 weight percent U^{235}. When fuel is being moved in or over the Spent Fuel Storage Pool "B" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition.

b. Deleted.

c. New fuel may also be stored in shipping containers.

5.4.2 SPENT FUEL STORAGE (Reference 1)

a. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pools, which are located in the fuel handling building.

b. Whenever there is fuel in the pool except for initial fuel loading, the spent fuel pool is filled with water borated to the concentration used in the reactor cavity and fuel transfer canal.

c. Deleted.

d. The fuel assembly storage racks provided and the number of fuel elements each will store are listed by location below:

	Spent Fuel Pool A North End of Fuel Handling Building	Spent Fuel Pool B South End of Fuel Handling Building	Dry New Fuel Storage Area Fuel Handling Building
Fuel Assys.	846 *	496	54
Cores	4.78	2.8	0.37

NOTE: * Includes three spaces for accommodating failed fuel containers. An additional 648 storage locations can be installed to provide a total of 1494 locations or 8.44 cores.

e. All of the fuel assembly storage racks provided are designed to Seismic Class 1 criteria to the accelerations indicated below:

	Fuel Handling Building Dry New Fuel Storage Area And Spent Fuel Pool A	Fuel Handling Building Spent Fuel Pool B
Horiz.	0.38 g	**
Vertical	0.25 g	**

NOTE: ** The "B" pool fuel storage racks are designed using the floor response spectra of the Fuel Handling Building.

f. **DELETED**

g. When spent fuel assemblies are stored in the Spent Fuel Pool "A", Region II storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-4.

h. When spent fuel assemblies are stored in the Spent Fuel Pool "B", storage locations, the combination of initial enrichment and cumulative burnup for spent fuel assemblies shall be within the acceptable area of Figure 5-5.

<u>REFERENCES</u>

(1) UFSAR, Section 9.7 - "Fuel Handling System"

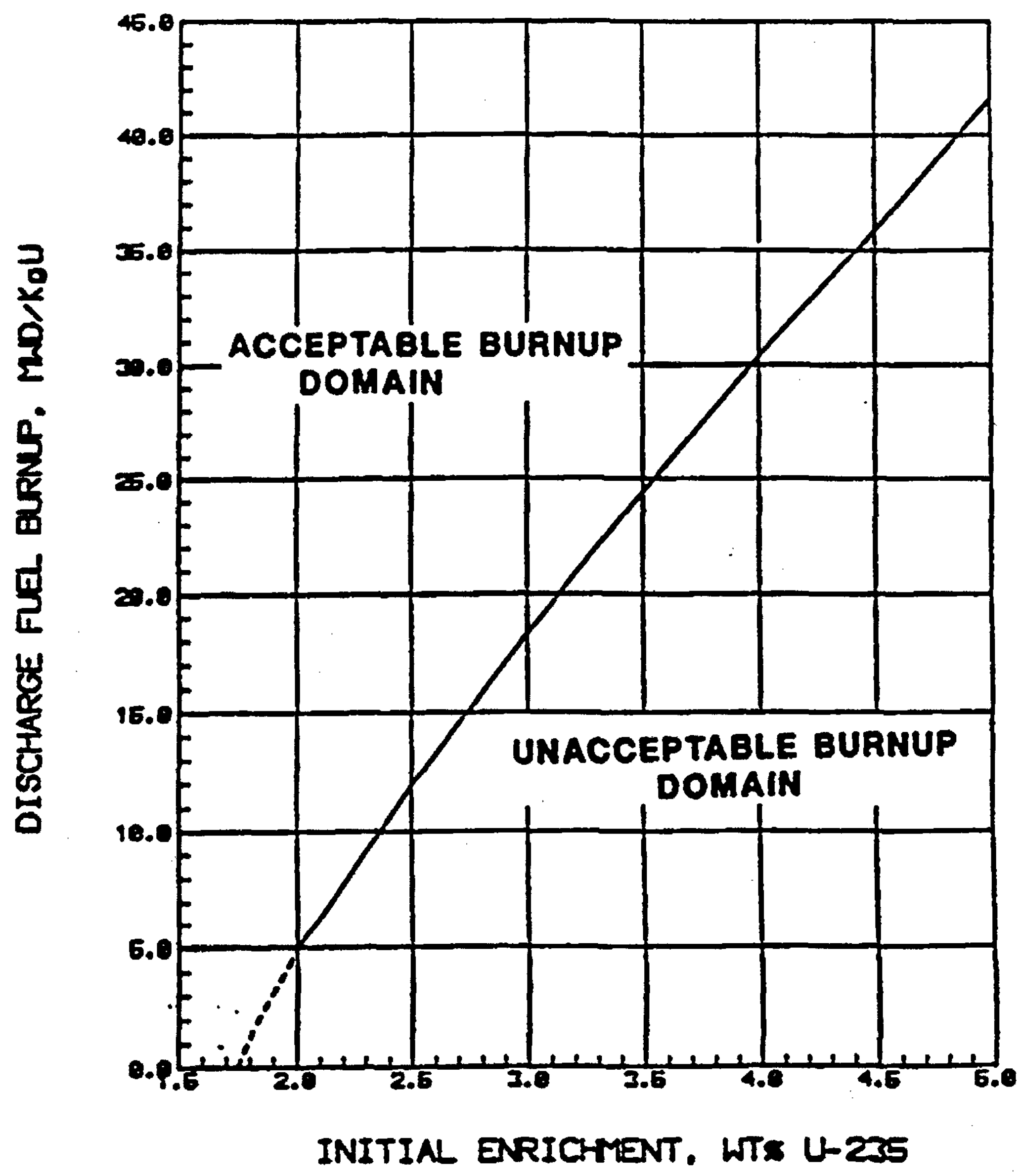


Figure 5-4 MINIMUM BURNUP REQUIREMENTS FOR FUEL IN REGION II OF THE POOL A STORAGE RACKS

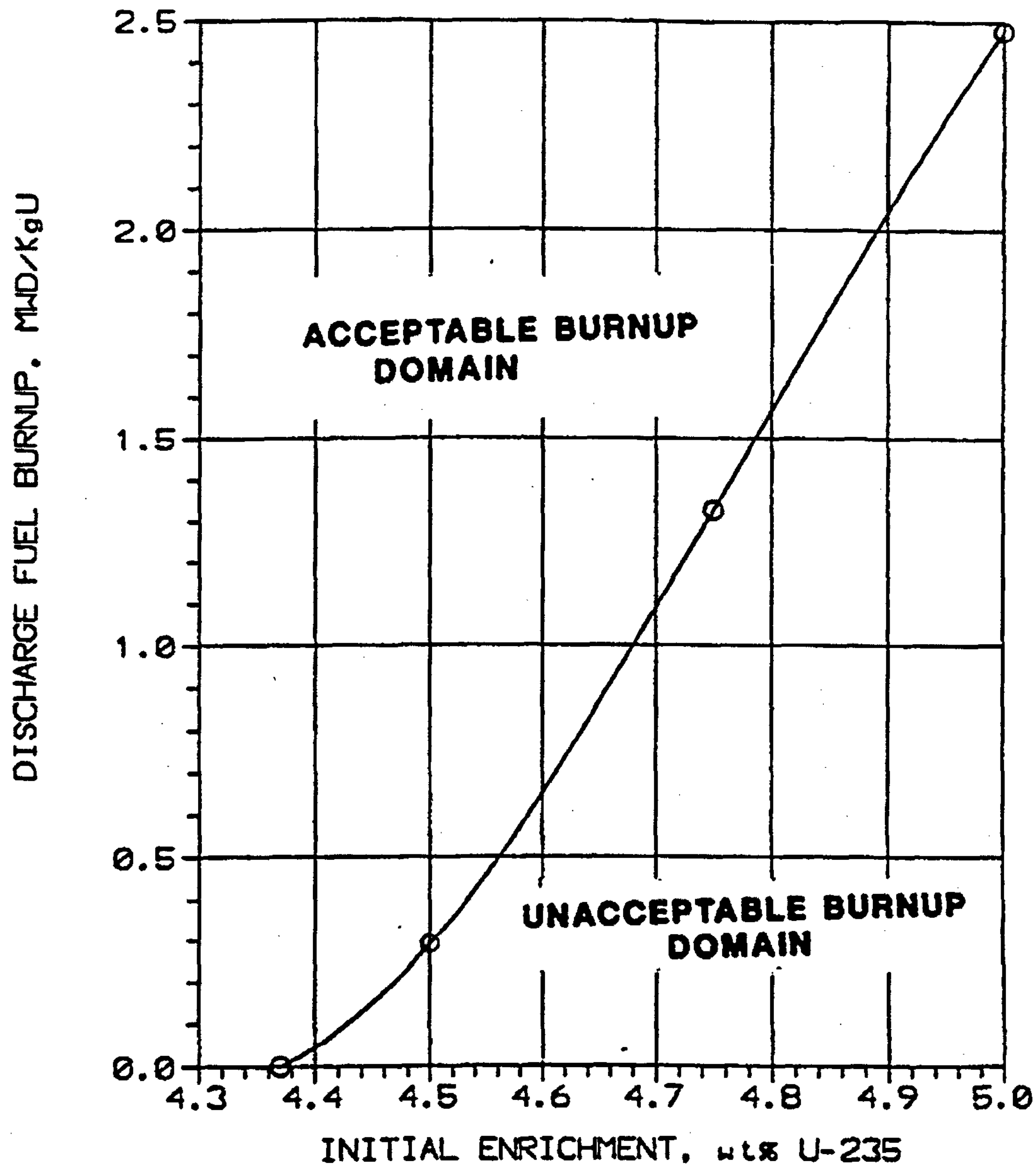


Figure 5-5 MINIMUM BURNUP REQUIREMENTS FOR FUEL IN THE POOL "B" STORAGE RACKS

5.5 AIR INTAKE TUNNEL FIRE PROTECTION SYSTEMS

Deleted

Intentionally left blank

ELEVATIONS FOR GASEOUS EFFLUENT RELEASE POINTS
(See Figure 5-3)

Unit 1 Stack	483' 7"
Unit 1 Turbine Building	425' 4"
Unit 1 Fuel Handling Building ESF Vent Stack	348'

LOCATIONS OF LIQUID EFFLUENT OUTFALLS PURSUANT TO NPDES
(See Figure 5-3)

Outfall No.	Description
DSN 001	Main Station Discharge
DSN 002	(Deleted)
DSN 003	Emergency Discharge from Unit 1 (if DSN 001 is blocked)
DSN 004	Emergency Discharge from Unit 1 (if Unit 1 MDCT blocked)
DSN 005	Stormwater and yard drainage and dewatering of natural draft cooling towers, maintenance dredging desiltation and basin dewatering, fire brigade training facility runoff, fire service water runoff.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President-TMI Unit 1 shall be responsible for TMI-1 operations and may, at any time, delegate his responsibilities in writing to the Plant Manager. He shall delegate the succession of his responsibilities in writing during his absence.

6.1.2 The Shift Manager (or during his absence from the Control Room, a designated individual), shall be responsible for the Control Room command function. A management directive to this effect signed by the Chief Nuclear Officer shall be reissued to all unit personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 CORPORATE

6.2.1.1 An onsite and offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

6.2.1.2 Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including operating organization positions. These relationships shall be documented and updated as appropriate, in the form of organizational charts. These organizational charts will be documented in the Updated FSAR and updated in accordance with 10 CFR 50.71e.

6.2.1.3 The Chief Nuclear Officer shall have corporate responsibility for overall plant nuclear safety and shall take measures to ensure acceptable performance of the staff in operating, maintaining, and providing technical support so that continued nuclear safety is assured.

6.2.2 UNIT STAFF

6.2.2.1 The Vice President-TMI Unit 1 shall be responsible for overall site safe operation and shall have control over those on site activities necessary for safe operation and maintenance of the site.

6.2.2.2 The unit staff organization shall meet the following:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be present in the control room when fuel is in the reactor.

c. At least two licensed Reactor Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. The Shift Manager or Control Room Supervisor # shall be in the control room at all times other than cold shutdown conditions (T_{ave} < 200°F) when he shall be onsite.

e. An individual ## qualified pursuant to 6.3.2 in radiation protection procedures shall be on site when fuel is in the reactor.

f. All REFUELING OPERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

g. A Site Fire Brigade ## of at least 5 members shall be maintained onsite at all times. The Site Fire Brigade shall not include 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.

h. The Shift Technical Advisor shall serve in an advisory capacity to the Shift Manager on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.2.2.3 Individuals who train the operating staff and those who carry out the health physics and quality assurance function shall have sufficient organizational freedom to be independent from operating pressures, however they may report to the appropriate manager on site.

If not SRO licensed, he shall have completed the SRO Training program.

The individual of item 6.2.2.2e and 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.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION(iii)

LICENSE CATEGORY QUALIFICATIONS	$T_{ave} > 200°$	$T_{ave} \leq 200°$
SRO(iv)	2	1(i)
RO(iv)	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1(ii)	None Required

(i) Does not include the Licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising (a) irradiated fuel handling and transfer activities onsite, and (b) all unirradiated fuel handling and transfer activities to and from the Reactor Vessel.

(ii) May be on a different shift rotation than licensed personnel.

(iii) Except for the Shift Manager, shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crewman being late or absent.

(iv) Pursuant to the requirements of 10 CFR 50.54(m).

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 of 1978 for comparable positions unless otherwise noted in the Technical Specifications, with the following exceptions: 1) the education and experience eligibility requirements for operator license applicants (described in Exelon letter RS-02-100, dated June 19, 2002), and changes thereto, shall be approved by the NRC and described in an applicable station training procedure, and 2) individuals who do not meet ANSI/ANS 3.1 of 1978, Section 4.5, are not considered technicians or maintenance personnel for purposes of determining qualifications but are permitted to perform work for which qualification has been demonstrated.

6.3.2 The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 of 1977. Each radiological controls technician/supervisor shall meet or exceed the qualifications of ANSI-N 18.1-1971, paragraph 4.5.2/4.3.2, or be formally qualified through an NRC approved TMI-1 Radiation Controls training program. All radiological controls technicians will be qualified through training and examination in each area or specific task related to their radiological controls functions prior to their performance of those tasks.

6.3.3 The Shift Technical Advisors shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in unit design, response and analysis of transients and accidents.

6.4 TRAINING

6.4.1 A training program for the Fire Brigade shall be maintained and shall meet or exceed the requirements of Section 600 of the NFPA Code.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL

The director of each department shall be responsible for ensuring the preparation, review, and approval of documents required by the activities described in 6.5.1.1 through 6.5.1.5 within his functional area of responsibility as assigned in the Review and Approval Matrix. Implementing approvals shall be performed at the cognizant manager level or above.

ACTIVITIES

6.5.1.1 Each procedure required by Technical Specification 6.8 and other procedures which affect nuclear safety, and substantive changes thereto, shall be prepared by a designated individual(s)/group knowledgeable in the area affected by the procedure. Each such procedure, and substantive changes thereto, shall be reviewed for adequacy by an individual(s)/group other than the preparer, but who may be from the same organization as the individual who prepared the procedure or change.

6.5.1.2 Proposed changes to the Appendix "A" Technical Specifications shall be reviewed by a knowledgeable individual(s)/group other than the individual(s) group who prepared the change.

6.5.1.3 Proposed modifications that affect nuclear safety to unit structures, systems and components shall be designed by an individual/organization knowledgeable in the areas affected by the proposed modification. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification but may be from the same division as the individual who designed the modification.

6.5.1.4 Proposed tests and experiments that affect nuclear safety shall be reviewed by a knowledgeable individual(s)/group other than the preparer but who may be from the same division as the individual who prepared the tests and experiments.

6.5.1.5 Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, shall be reviewed by a knowledgeable individual(s)/group other than the individual/group which performed the investigation.

6.5.1.6 All REPORTABLE EVENTS shall be reviewed by an individual/group other than the individual/group which prepared the report.

6.5.1.7 Special reviews, investigations or analyses and reports thereon as requested by the Vice President-TMI Unit 1shall be performed by a knowledgeable individual(s)/group.

6.5.1.8 The Security Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.5.1.9 The Emergency Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.5.1.10 A knowledgeable individual(s)/group shall review every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports to the Vice President-TMI Unit 1 covering evaluations, recommendations and disposition of the corrective action to prevent recurrence.

6.5.1.11 Major changes to radwaste systems shall be reviewed by a knowledgeable individual(s)/group other than the individuals(s)/group which prepared them.

6.5.1.12 Individuals responsible for reviews performed in accordance with 6.5.1.1 through 6.5.1.4 shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate personnel. Individuals responsible for reviews considered under 6.5.1.1, 6.5.1.3, and 6.5.1.4 shall render determinations in writing with regard to whether or not NRC approval is required pursuant to 10CFR50.59.

RECORDS

6.5.1.13 Written records of activities performed under Specifications 6.5.1.1 through 6.5.1.11 shall be maintained.

QUALIFICATIONS

6.5.1.14 Responsible Technical Reviewers shall meet or exceed the qualifications of ANSI/ANS 3.1 of 1978 Section 4.6, or 4.4 for applicable disciplines, or have 7 years of appropriate experience in the field of his specialty. Credit toward experience will be given for advanced degrees on a one-to-one basis up to a maximum of two years. Responsible Technical Reviewers shall be designated in writing.

6.5.2 INDEPENDENT SAFETY REVIEW FUNCTION

6.5.2.1 The director of each department shall be responsible for ensuring the independent safety review of the subjects described in 6.5.2.5 within his assigned area of safety review responsibility, as assigned in the Review and Approval Matrix.

6.5.2.2 Independent safety review shall be completed by an individual/group not having direct responsibility for the performance of the activities under review, but who may be from the same functionally cognizant organization as the individual/group performing the original work.

6.5.2.3 The licensee shall collectively have or have access to the experience and competence required to independently review subjects in the following areas:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Nondestructive testing
f. Instrumentation and control
g. Radiological safety
h. Mechanical engineering
i. Electrical engineering
j. Administrative controls and quality assurance practices
k. Emergency plans and related organization, procedures and equipment
l. Other appropriate fields associated with the unique characteristics of TMI-1.

6.5.2.4 Consultants may be utilized as determined by the cognizant department director to provide expert advice.

RESPONSIBILITIES

6.5.2.5 The following subjects shall be independently reviewed by the functionally assigned divisions:

a. Written evaluations of changes in the facility as described in the Updated Final Safety Analysis Report (UFSAR), of changes in procedures as described in the UFSAR, and of tests or experiments not described in the UFSAR, which are completed without prior NRC approval under the provisions of 10CFR50.59(c)(1). This review is to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or require NRC approval pursuant to 10CFR50.59. Such reviews need not be performed prior to implementation.

b. Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or requires NRC approval pursuant to 10CFR50.59. Matters of this kind shall be reviewed prior to submittal to the NRC.

c. Proposed changes to Technical Specifications or license amendments related to nuclear safety shall be reviewed prior to submittal to the NRC for approval.

d. Violations, deviations, and reportable events which require reporting to the NRC in writing. Such reviews are performed after the fact. Review of events covered under this subsection shall include results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Written summaries of audit reports in the areas specified in Section 6.5.3 and involving safety related functions.

f. Any other matters involving safe operation of the nuclear power plant which a reviewer deems appropriate for consideration, or which is referred to the independent reviewers.

6.5.2.6 QUALIFICATIONS

The independent reviewer(s) shall either have a Bachelor's Degree in Engineering or the Physical Sciences and five (5) years of professional level experience in the area being reviewed or have 9 years of appropriate experience in the field of his specialty. An individual performing reviews may possess competence in more than one specialty area. Credit toward experience will be given for advanced degrees on a one-for-one basis up to a maximum of two years.

RECORDS

6.5.2.7 Reports of reviews encompassed in Section 6.5.2.5 shall be prepared, maintained and transmitted to the cognizant department director and the Vice President-TMI Unit 1.

6.5.3 AUDITS

6.5.3.1 Audits of unit activities shall be performed in accordance with the Quality Assurance Topical Report (QATR). These audits shall encompass:

a. The conformance of unit operations to provisions contained within the Technical Specifications and applicable license conditions.

b. The performance, training and qualifications of the entire unit staff.

c. The verification of the non-conformances and corrective actions program to be properly implemented and documented as related to action taken to correct deficiencies occurring in unit equipment, structures, systems or methods of operation that affect nuclear safety.

d. The performance of activities required by the QATR to meet the criteria of Appendix "B" 10 CFR 50.

e. The Emergency Plan and implementing procedures.

f. The Security Plan and implementing procedures.

g. The Fire Protection Program and implementing procedures.

h. The Offsite Dose Calculation Manual (ODCM) and implementing procedures.

i. The Process Control Program and implementing procedures for solidification of radioactive wastes.

j. The performance of activities required by the Quality Assurance Program to meet criteria of Regulatory Guide 4.15, December, 1977.

k. Any other area of unit operation considered appropriate by the Chief Nuclear Officer.

6.5.3.2 Audits of the following shall be performed under the cognizance of the department director responsible for technical support.

a. An independent fire protection and loss prevention program inspection and audit shall be performed utilizing either qualified licensee personnel or an outside fire protection firm.

b. An inspection and audit of the fire protection and loss prevention program, by an outside qualified fire consultant.

RECORDS

6.5.3.3 Audit reports encompassed by sections 6.5.3.1 and 6.5.3.2 shall be forwarded for action to the management positions responsible for the areas audited within 60 days after completion of the audit. Upper management shall be informed per the QATR.

6.5.4 DELETED

DELETED

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR 50, and

b. Each REPORTABLE EVENT shall undergo an independent safety review pursuant to Specification 6.5.2.5.d.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The reactor shall be shutdown and operation shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. An immediate report shall be made to the Plant Manager, and Vice President-TMI Unit 1, and the event shall be reported to NRC in accordance with 10 CFR 50.72.

c. A complete analysis of the circumstances leading up to and resulting from the occurrence shall be prepared by the unit staff. This report shall include analysis of the effects of the occurrence and recommendations concerning operation of the unit and prevention of recurrence. This report shall be submitted to the Plant Manager and the Vice President-TMI Unit 1. The safety limit violation report shall be submitted to NRC in accordance with 10 CFR 50.73.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the items referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Surveillance and test activities of equipment that affects nuclear safety and radioactive waste management equipment.

c. Refueling Operations.

d. Security Plan Implementation.

e. Fire Protection Program Implementation.

f. Emergency Plan Implementation.

g. Process Control Program Implementation.

h. Offsite Dose Calculation Manual Implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, Revision 1.

j. Plant Staff Overtime, to limit the amount worked by staff performing safety-related functions in accordance with NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.8.2 Further, each procedure required by 6.8.1 above, and substantive changes thereto, shall be reviewed and approved as described in 6.5.1 prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the licensee's management staff qualified in accordance with 6.5.1.14 and knowledgeable in the area affected by the procedure. For changes which may affect the operational status of unit systems or equipment, at least one of these individuals shall be a member of unit management or supervision holding a Senior Reactor Operator's License on the unit.

c. The change is documented, reviewed and approved as described in 6.5.1 within 14 days of implementation.

6.8.4 a. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

(1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

(2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

(3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

b. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

(1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

(2) Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 times the concentrations specified in 10 CFR Part 20.1001 - 20.2402, Appendix B, Table 2, Column 2,

(3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

b. Radioactive Effluent Controls Program (continued)

(4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the unit to the site boundary conforming to Appendix I to 10 CFR Part 50,

(5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

(6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

(7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at, or beyond, the site boundary. The limits are as follows:

(a) For noble gases: less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

(b) For I-131, I-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ.

(8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

(9) Limitations on the annual quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50, and

(10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the NRC Region 1 Office unless otherwise noted.

6.9.1 Routine Reports

A. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the UFSAR, Chapter 13 and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described.

Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

B. Annual Reports. Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year. (A single submittal maybe made for the station. The submittal should combine those sections that are common to both units at the station.)

1. DELETED

2. The following information on aircraft movements at the Harrisburg International Airport:

   a. The total number of aircraft's movements (takeoffs and landings) at the Harrisburg International Airport for the previous twelve-month period.

   b. The total number of movements of aircraft larger than 200,000 pounds at the Harrisburg International Airport for the previous twelve-month period, broken down into scheduled and non-scheduled (including military) takeoffs and landings, based on a current estimate provided by the airport manager or his designee.

3. The following information from the periodic Leak Reduction Program tests shall be reported:

   a. Results of leakage measurements,
   b. Results of visual inspections, and
   c. Maintenance undertaken as a result of Leakage Reduction Program tests or inspections.

4. The following information regarding pressurizer power operated relief valve and pressurizer safety valve challenges shall be reported:

   a. Date and time of incident,
   b. Description of occurrence, and
   c. Corrective measures taken if incident resulted from an equipment failure.

5. The following information regarding the results of specific activity analysis in which the primary coolant exceeded limits of Technical Specifications 3.1.4.1 shall be reported:

   a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

   b. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations;

   c. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

   d. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and

   e. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

C. DELETED

6.9.2 DELETED

(Pages 6-14, 6-15, and 6-16 deleted)

6.9.3 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.3.1 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in: (1) the ODCM; and, (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

Note: A single submittal may be made for the station.

6.9.4 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.4.1 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Report shall include a summary of the quantities of radioactive liquid and gaseous effluent and solid waste released from the unit. The material provided shall be: (1) consistent with the objectives outlined in the ODCM and PCP; and, (2) in conformance with 10 CFR 50.36(a) and Section IV.B.1 of Appendix I to 10 CFR Part 50.

Note: A single submittal may be made for the station. The submittal should combine those sections that are common to both units at the station.

6.9.5 CORE OPERATING LIMITS REPORT

6.9.5.1 The core operating limits addressed by the individual Technical Specifications shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle.

6.9.5.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be those previously reviewed and approved by the NRC for use at TMI-1, specifically:

(1) BAW-10179 P-A, "Safety and Methodology for Acceptable Cycle Reload Analyses." The current revision level *shall* be specified in the COLR.

(2) TR-078-A, "TMI-1 Transient Analyses Using the RETRAN Computer Code", Revision 0. NRC SER dated 2/10/97.

(3) TR-087-A, "TMI-1 Core Thermal-Hydraulic Methodology Using the VIPRE-01 Computer Code", Revision 0. NRC SER dated 12/19/96.

(4) TR-091-A, "Steady State Reactor Physics Methodology for TMI-1", Revision 0. NRC SER dated 2/21/96.

(5) TR-092P-A, "TMI-1 Reload Design and Setpoint Methodology", Revision 0. NRC SER dated 4/22/97.

(6) BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel", NRC SER dated February 4, 2000.

6.9.5.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient/accident analysis limits) of the safety analysis are met.

6.9.5.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of normal station operation including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of periodic checks, tests and calibrations.

e. Records of reactor physics tests and other special tests related to nuclear safety.

f. Changes to procedures required by Specification 6.8.1.

g. Deleted

h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

i. Results of annual physical inventory verifying accountability of licensed sources on record.

j. Control Room Log Book.

k. Control Room Supervisor Log Book.

6.10.2 The following records shall be retained for the duration of Operating License DPR-50 unless otherwise specified in 6.10.1 above.

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Routine unit radiation surveys and monitoring records.

d. Records of doses received by all individuals for whom monitoring was required.

e. Records of radioactive liquid and gaseous wastes released to the environment, and records of environmental monitoring surveys.

f. Records of transient or operational cycles for those facility components which affect nuclear safety for a limited number of transients or cycles as defined in the Final Safety Analysis Report.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QATR.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Deleted.

l. Records of analyses required by the radiological environmental monitoring program.

m. Records of the service lives of all safety related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of solid radioactive shipments.

o. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation at 30 cm (11.8 in.) is greater than 100 mrem/hr. deep dose but less than 1000 mrem/hr. shall be barricaded and conspicuously posted as a High Radiation Area, and personnel desiring entrance shall obtain a Radiation Work Permit (RWP). Any individual or group of individuals entering a High Radiation Area shall (a) use a continuously indicating dose rate monitoring device or (b) use a radiation dose rate integrating device which alarms at a pre-set dose level (entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them), or (c) assure that a radiological control technician provides positive control over activities within the area and periodic radiation surveillance with a dose rate monitoring instrument.

b. In addition to the requirements of specification 6.12.1.a:

1. Any area accessible to personnel where an individual could receive in any one hour a deep dose in excess of 1000 mrem at 30 cm (11.8 in.) but less than 500 rads at one meter (3.28 ft.) from sources of radioactivity shall be locked or guarded to prevent unauthorized entry. The keys to these locked barricades shall be maintained under the administrative control of the respective Radiological Controls Supervisor.

2. For individual high radiation areas where an individual could receive in any one hour a deep dose in excess of 1000 mrem at 30 cm (11.8 in.) but less than 500 rads at one meter (3.28 ft.), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

The Radiation Work Permit is not required by Radiological Controls personnel during the performance of their assigned radiation protection duties provided they are following radiological control procedures for entry into High Radiation Areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

   a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

   b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and,

   c. documentation that the changes have been reviewed and approved pursuant to 6.8.2.

2. Shall become effective upon review and approval by licensee management.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Licensee initiated changes to the ODCM:

1. Shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

   a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

   b. a determination that the changes did not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation that the changes have been reviewed and approved pursuant to 6.8.2.

2. Shall become effective upon review and approval by licensee management.

6.15 DELETED

6.16 DELETED

6.17 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.17.1 Licensee initiated safety related changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Annual Report (Specification 6.9.1B) for the period in which the evaluation was reviewed. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individuals in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and approved.

2. Shall become effective upon review and approval in accordance with Section 6.5.1.

6.18 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license or

   2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b.1 or 6.18.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71 (e).