Attachment 2

PROPOSED TECHNICAL SPECIFICATIONS

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION		PAGE
SECTION 1	INTRODUCTION	
1.0	DEFINITIONS	1.0-1
SECTION 2	SAFETY STANDARDS	
2.1	REACTOR CORE, Limiting Combination of Power, Pressure and Temperature	2.1-1
SECTION 3	LIMITING CONDITIONS FOR OPERATION	
3.0	LIMITING CONDITIONS FOR OPERATION (GENERAL)	3.0-1
3.1	REACTOR COOLANT SYSTEM	3.1-1
	3.1.1 Maximum Reactor Coolant Activity	3.1-1
	3.1.2 Operational Components	3.1-3
	3.1.3 Combined Heat-up, Cool-down and Pressure Limitations	3.1-10
	3.1.4 Leakage	3.1-18
	3.1.5 Pressurizer Relief Valves	3.1-21
	3.1.6 Pressurizer	3.1-22
	3.1.7 Reactor Coolant System Vents	3.1-23
3.2	CHEMICAL AND VOLUME CONTROL SYSTEM	3.2-1
3.3	SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS	3.3-1
	3.3.1 Operating Status	3.3-1
	3.3.2 Shutdown Status	3.3-6
	3.3.3 Minimum Water Volume and Boron Concentration In the Refueling Water Storage Tank	3.3-8
	3.3.4 Minimum Solution Volume Hydrazine Concentration in the Hydrazine Tank	3.3-9
	3.3.5 Primary Coolant System Pressure Isolation Valves	3.3-10

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION		PAGE
3.4	TURBINE CYCLE	3.4-1
	3.4.1 Operating Status	3.4-1
	3.4.2 Maximum Secondary Coolant Activity	3.4-3
	3.4.3 Auxiliary Feedwater System	3.4-4
	3.4.4 Auxiliary Feedwater Storage Tank	3.4-5
3.5	INSTRUMENTATION AND CONTROL	3.5-1
	3.5.1 Reactor Trip System Instrumentation	3.5-1
	3.5.2 Control Rod Insertion Limits	3.5-6
	3.5.3 Control and Shutdown Rod Misalignment	3.5-10
	3.5.4 Rod Position Indicating System	3.5-13
	3.5.5 Containment Isolation Instrumentation	3.5-15
	3.5.6 Accident Monitoring Instrumentation	3.5-19
	3.5.7 Auxiliary Feedwater Instrumentation	3.5-21
	3.5.8 Radioactive Liquid Effluent Instrumentation	3.5-24
	3.5.9 Radioactive Gaseous Process and Effluent Monitoring Instrumentation	3.5-27
	3.5.10 Radiation Monitoring Instrumentation	3.5-30
3.6	CONTAINMENT SYSTEMS	3.6-1
	3.6.1 Containment Sphere	3.6-1
	3.6.2 Containment Isolation Valves	3.6-3
	3.6.3 Hydrogen Monitors and Hydrogen Recombiners	3.6-6
3.7	AUXILIARY ELECTRICAL SUPPLY	3.7-1
3.8	FUEL LOADING AND REFUELING	3.8-1
3.9	MODERATOR TEMPERATURE COEFFICIENT	3.9-1
3.10	INCORE INSTRUMENTATION	3.10-1
3.11	CONTINUOUS POWER DISTRIBUTION MONITORING	3.11-1
3.12	CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM	3.12-1
3.13	SHOCK SUPPRESSORS (SNUBBERS) OPERABILITY	3.13-1
3.14	FIRE PROTECTION SYSTEMS OPERABILITY	3.14-1

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION		PAGE
3.15	RADIOACTIVE LIQUID EFFLUENTS	3.15-1
	3.15.1 Liquid Effluents Concentration	3.15-1
	3.15.2 Liquid Waste Dose	3.15-2
	3.15.3 Liquid Effluent Treatment	3.15-3
3.16	RADIOACTIVE GASEOUS EFFLUENTS	3.16-1
	3.16.1 Dose Rate	3.16-1
	3.16.2 Dose, Noble Gases	3.16-2
	3.16.3 Dose, Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form	3.16-3
	3.16.4 Gaseous Radwaste Treatment	3.16-4
	3.16.5 Gas Storage Tank	3.16-6
	3.16.6 Explosive Gas Mixture	3.16-7
3.17	DOSE	3.17-1
3.18	RADIOLOGICAL ENVIRONMENTAL MONITORING	3.18-1
	3.18.1 Monitoring Program	3.18-1
	3.18.2 Land Use Census	3.18-7
	3.18.3 Interlaboratory Comparison Program	3.18-9
3.19	SOLID RADIOACTIVE WASTE	3.19-1
3.20	OVERPRESSURE PROTECTION SYSTEMS	3.20-1
SECTION 4	SURVEILLANCE REQUIREMENTS	
4.0	SURVEILLANCE REQUIREMENTS (GENERAL)	4.0-1
	4.1.1 Operational Safety Items	4.1-1
	4.1.2 Radioactive Liquid Effluent Instrumentation	4.1-13
	4.1.3 Radioactive Gaseous Process and Effluent Monitoring Instrumentation	4.1-16
	4.1.4 Containment Isolation Instrumentation	4.1-19
	4.1.5 Accident Monitoring Instrumentation	4.1-22
	4.1.6 Pressurizer Relief Valves	4.1-24
	4.1.7 Pressurizer	4.1-25
	4.1.8 Auxiliary Feedwater Instrumentation	4.1-26
	4.1.9 Auxiliary Feedwater System Surveillance	4.1-28
	4.1.10 Auxiliary Feedwater Storage Tank Surveillance	4.1-30
	4.1.11 Radiation Monitoring Instrumentation	4.1-31
	4.1.12 Reactor Coolant System Vents	4.1-33
	4.1.13 Leakage and Leakage Detection Systems	4.1-34

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION			PAGE
4.2	SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM		4.2-1
	4.2.1	Safety Injection and Containment Spray System Periodic Testing	4.2-1
	4.2.2	Primary Coolant System Pressure Isolation Valves	4.2-6
4.3	CONTAINMENT SYSTEMS		4.3-1
	4.3.1	Containment Testing	4.3-1
	4.3.2	Containment Isolation Valves	4.3-7
	4.3.3	Hydrogen Monitors and Hydrogen Recombiners	4.3-8
4.4	EMERGENCY POWER SYSTEM PERIODIC TESTING		4.4-1
4.5	RADIOACTIVE LIQUID EFFLUENTS		4.5-1
	4.5.1	Liquid Effluents Concentration	4.5-1
	4.5.2	Liquid Effluent Dose	4.5-5
	4.5.3	Liquid Waste Treatment	4.5-6
4.6	RADIOACTIVE GASEOUS EFFLUENTS		4.6-1
	4.6.1	Dose Rate	4.6-1
	4.6.2	Dose, Noble Gases	4.6-5
	4.6.3	Dose, Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form	4.6-6
	4.6.4	Gaseous Radwaste Treatment	4.6-7
	4.6.5	Gas Storage Tank	4.6-8
	4.6.6	Explosive Gas Mixture	4.6-9
4.7	INSERVICE INSPECTION REQUIREMENTS		4.7-1
4.8	REACTIVITY ANOMALIES		4.8-1
4.9	REACTOR VESSEL SURVEILLANCE PROGRAM		4.9-1
4.10	AUGMENTED INSERVICE INSPECTION OF HIGH ENERGY LINES OUTSIDE CONTAINMENT		4.10-1
4.11	CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM		4.11-1
4.12	MISCELLANEOUS RADIOACTIVE MATERIAL SOURCES		4.12-1
4.13	[Deleted]		4.13-1

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION		PAGE
4.14	SHOCK SUPPRESSOR (SNUBBERS) SURVEILLANCE	4.14-1
4.15	FIRE PROTECTION SYSTEM SURVEILLANCE	4.15-1
4.16	INSERVICE INSPECTION OF STEAM GENERATOR TUBING	4.16-1
4.17	DOSE	4.17-1
4.18	RADIOLOGICAL ENVIRONMENTAL MONITORING	4.18-1
	4.18.1 Monitoring Program	4.18-1
	4.18.2 Land Use Census	4.18-5
	4.18.3 Interlaboratory Comparison Program	4.18-6
4.19	SOLID RADIOACTIVE WASTE	4.19-1
4.20	OVERPRESSURE PROTECTION	4.20-1
SECTION 5	DESIGN FEATURES	
5.1	SITE DESCRIPTION	5.1-1
5.2	CONTAINMENT	5.2-1
5.3	REACTOR	5.3-1
5.4	AUXILIARY EQUIPMENT	5.4-1
SECTION 6	ADMINISTRATIVE CONTROLS	
6.1	RESPONSIBILITY	6.1-1
6.2	ORGANIZATION	6.2-1
	6.2.1 Offsite and Onsite Organization	6.2-1
	6.2.2 Unit Staff	6.2-1
6.3	UNIT STAFF QUALIFICATIONS	6.3-1
6.4	TRAINING	6.4-1
6.5	REVIEW AND AUDIT	6.5-1
	6.5.1 Onsite Review Committee (OSRC)	6.5-1
	6.5.2 Technical Review and Control	6.5-3
	6.5.3 Nuclear Safety Group (NSG)	6.5-5

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

SECTION		PAGE
6.6	REPORTABLE EVENT ACTION	6.6-1
6.7	SAFETY LIMIT VIOLATION	6.7-1
6.8	PROCEDURES AND PROGRAMS	6.8-1
6.9	REPORTING REQUIREMENTS	6.9-1
	6.9.1 Routine Reports	6.9-1
	6.9.2 Special Reports	6.9-5
6.10	RECORD RETENTION	6.10-1
6.11	RADIATION PROTECTION PROGRAM	6.11-1
6.12	HIGH RADIATION AREA	6.12-1
6.13	PROCESS CONTROL PROGRAM (PCP)	6.13-1
6.14	OFFSITE DOSE CALCULATION MANUAL (ODCM)	6.14-1
6.15	MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS AND SOLID)	6.15-1
6.16	ENVIRONMENTAL PROTECTION	6.16-1

FIGURES

FIGURE	TITLE	PAGE
2.1.1	Safety Limits Pressure, Temperature, Power 100% Nominal Flow	2.1-5
3.1.3a	San Onofre Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY	3.1-12
3.1.3b	San Onofre Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY	3.1-13
3.1.3c	Effect of Fluence and Copper Content on Shift of RTNDT for Reactor Vessel Steels Exposed to 550°F Temperature	3.1-14
3.1.3d	Fast Neutron Fluence (E>1 MEV) as a Function of Full Power Service Life	3.1-15
3.5.2.1	Control Group Insertion Limits	3.5-9
5.1.1	Exclusion Area	5.1-2
6.2-1	[Figure Deleted]	
6.2-2	[Figure Deleted]	

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
TECHNICAL SPECIFICATIONS
TABLES

TABLE	TITLE	PAGE
1.1	Frequency Notation	1.0-7
1.2	Operational Modes	1.0-8
2.1	Maximum Safety Systems Settings	2.1-4
3.1.3.1	San Onofre Nuclear Generating Station Unit 1 Reactor Vessel Toughness Data (unirradiated)	3.1-16
3.3.5-1	Primary Coolant System Pressure Isolation Valves	3.3-11
3.5.1-1	Reactor Trip System Instrumentation	3.5-2
3.5.3-1	Accident Analyses Requiring Reevaluation in the Event of an Inoperable Rod	3.5-12
3.5.5-1	Containment Isolation Instrumentation	3.5-16
3.5.5-2	Containment Isolation Instrumentation Trip Set Points	3.5-18
3.5.6-1	Accident Monitoring Instrumentation	3.5-20
3.5.7-1	Auxiliary Feedwater Instrumentation	3.5-22
3.5.7-2	Auxiliary Feedwater Instrumentation Trip Set Points	3.5-23
3.5.8.1	Radioactive Liquid Effluent Monitoring Instrumentation	3.5-25
3.5.9.1	Radioactive Gaseous Effluent Monitoring Instrumentation	3.5-28
3.5.10-1	Radiation Monitoring Instrumentation	3.5-31
3.6.2-1	Power Operated or Automatic Containment Isolation Valve Summary	3.6-5
3.14.1	Fire Hose Stations	3.14-6
3.14.2	Fire Detection Instruments	3.14-7

TABLES

TABLE	TITLE	PAGE
3.18.1	Radiological Environmental Monitoring Program	3.18-3
3.18.2	Reporting Levels For Radioactivity Concentration in Environmental Samples	3.18-6
4.1.1	Reactor Trip System Instrumentation Surveillance Requirements	4.1-2
4.1.2	Minimum Equipment Check and Sampling Frequency	4.1-4
4.1.3	Minimum Frequencies for Testing, Calibrating, and/or Checking of Instrument Channels	4.1-10
4.1.2.1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	4.1-14
4.1.3.1	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	4.1-17
4.1.4-1	Containment Isolation Instrumentation Surveillance Requirements	4.1-20
4.1.5-1	Accident Monitoring Instrumentation Surveillance Requirements	4.1-23
4.1.8-1	Auxiliary Feedwater Instrumentation Surveillance Requirements	4.1-27
4.1.11-1	Radiation Monitoring Instrumentation Surveillance Requirements	4.1-32
4.1.13-1	Leakage Detection Systems	4.1-35
4.4-1	Battery Surveillance Requirements	4.4-5
4.5.1.1	Radioactive Liquid Waste Sampling and Analysis Program	4.5-2
4.6.1.1	Radioactive Gaseous Waste Sampling an Analysis Program	4.6-2
4.18.1	Maximum Values for the Lower Limits of Detection (LLD)	4.18-2
6.2-1	Minimum Shift Crew Composition	6.2-4

APPENDIX A

As required by Section 3, Paragraph B, of Provisional Operating License DPR-13, any changes made to the San Onofre Nuclear Generating Station, Unit 1 Technical Specifications will be authorized by the Nuclear Regulatory Commission. All changes to these Technical Specifications are indicated by a vertical Line in the right-hand margin. The length of the line shows the extent of the changed material.

The amendment numbers assigned by the NRC appear in the lower right-hand corner of the page. In general, the highest amendment number is associated with the change bars on each page.

APPENDIX A
IDENTIFICATION OF CHANGES*

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 1	July 21, 1970	Revises Technical Specification 5.2
No. 2 (Amendment No. 3)	November 13, 1970	Adds new Technical Specification 3.9, and revises Technical Specification 5.3
No. 3	November 16, 1971	Revises Technical Specification 3.9
No. 4	January 18, 1972	Revises Technical Specification 3.5.2
No. 5	October 20, 1971	Revises Technical Specification 4.1
No. 6	August 24, 1971	Revises Technical Specification 6.1 and 6.2
No. 7	January 13, 1972	Revises Technical Specifications 3.1.4 and 3.6
No. 8	April 13, 1973	Revises Technical Specification 3.9
No. 9 (Amendment No. 4)	June 5, 1973	Revises Technical Specification 1.0, and adds new Technical Specification 6.5
No. 10	June 26, 1973	Revises Technical Specification 4.5
No. 11	July 17, 1973	Revises Technical Specification 5.3
No. 12	September 17, 1973	Revises Technical Specifications 4.1, 4.2 and 4.4
No. 13	November 13, 1973	Adds new Technical Specification 3.10

*The above numbered designated changes relate to Appendix A Technical Specifications.

APPENDIX A
IDENTIFICATION OF CHANGES* (Continued)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 14	April 12, 1974	Revises Technical Specifications 3.1.3 and 4.7, and adds new Technical Specification 4.9
No. 15 (Amendment No. 5)	October 15, 1974	Revises Technical Specification 4.3
No. 17 (Amendment No. 7)	December 20, 1974	Revises Technical Specifications 1.0, 3.5.2, 3.10 and Table 4.1.1 and adds Technical Specification 3.11
No. 18 (Amendment No. 8)	January 18, 1975	Revises Technical Specifications 3.10 and 3.11
No. 20 (Amendment No. 10)	February 18, 1975	Revises Technical Specification 3.9
No. 21 (Amendment No. 11)	May 13, 1975	Revises Technical Specifications 3.5 and 3.11
No. 22 (Amendment No. 12)	April 18, 1975	Revises Technical Specification 6.0
No. 23 (Amendment No. 13)	July 15, 1975	Adds new Technical Specification 4.10
No. 24 (Amendment No. 14)	September 16, 1975	Adds new Technical Specifications 3.12 and 4.11
No. 25 (Amendment No. 15)	September 29, 1975	Adds new Technical Specification 4.12 and revises Technical Specifications 6.9 and 6.10
No. 26 (Amendment No. 16)	January 17, 1976	Revises Figure 6.2.1.1, deletes Technical Specification 6.6, and revises Technical Specifications 6.5.2.2, 6.5.2.7.g, and 6.9

*The above numbered designated changes relate to Appendix A Technical Specifications.

APPENDIX A
IDENTIFICATION OF CHANGES* (Continued)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 28 (Amendment No. 18)	January 15, 1976	Adds new Technical Specification 4.13
No. 30 (Amendment No. 21)	July 21, 1976	Adds new Technical Specifications 3.13 and 4.14
No. 31 (Amendment No. 22)	October 26, 1976	Revised Technical Specification 4.1
No. 32 (Amendment No. 23)	December 7, 1976	Adds new Technical Specification 6.12
No. 33 (Amendment No. 24)	February 18, 1977	Revises Technical Specification 4.3
No. 34 (Amendment No. 25)	April 1, 1977	Revises Technical Specifications Sections 3, 4, and 5
No. 36 (Amendment No. 27)	September 22, 1977	Revises Technical Specification 6.3
No. 38 (Amendment No. 29)	December 20, 1977	Revises Technical Specifications 3.1, 3.4 and 4.1
No. 39 (Amendment No. 30)	February 6, 1978	Revises Technical Specification 6.9
No. 40 (Amendment No. 31)	March 8, 1978	Adds new Technical Specifications 3.14 and 4.15
No. 41 (Amendment No. 33)	April 7, 1978	Revises Technical Specification 4.14
No. 42 (Amendment No. 35)	August 4, 1978	Revise Technical Specifications 3.5.2 and 3.11
No. 43 (Amendment No. 36)	September 25, 1978	Revises Table 4.1.2 and Technical Specification 3.8

*The above numbered designated changes relate to Appendix A Technical Specifications.

APPENDIX A
IDENTIFICATION OF CHANGES* (Continued)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 44 (Amendment No. 37)	October 31, 1978	Revises Technical Specifications Sections 3, 4, and 5
No. 45 (Amendment No. 38)	November 17, 1978	Revises Technical Specifications 3.1.3 and deletes Technical Specification 6.12
No. 46 (Amendment No. 39)	November 30, 1978	Revises Technical Specifications Section 5
No. 48 (Amendment No. 42)	July 6, 1979	Revises Technical Specifications 6.2 and 6.5
No. 49 (Amendment No. 43)	July 19, 1979	Revises Technical Specifications 2.1 and 3.1.2, Deletes Figure 2.1.2, Revises Table 3.5.1
No. 50 (Amendment No. 44)	July 19, 1979	Revises Technical Specification 3.14, 4.15 and 6.2.2
No. 52 (Amendment No. 46)	September 26, 1979	Revises Technical Specification 4.7
No. 54 (Amendment No. 49)	May 29, 1980	Revises Technical Specification 3.5
No. 55 (Amendment No. 50)	June 25, 1980	Revises Technical Specifications 6.2 and 6.5
No. 56 (Order for Modification of License)	October 24, 1980	Revises Technical Specifications 6.10.2 and 6.12
No. 57 (Amendment No. 52)	February 6, 1981	Revises Technical Specification 3.7
No. 59 (Amendment No. 54)	May 7, 1981	Revises Technical Specification 6

*The above numbered designated changes relate to Appendix A Technical Specifications.

APPENDIX A
IDENTIFICATION OF CHANGES* (Continued)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 60 (Amendment No. 55)	June 8, 1981	Revises Technical Specification 2.1, 3.1, 3.5, 3.11 and 4.16
No. 61 (Amendment No. 56)	June 11, 1981	Revises Technical Specification 1.0, 3.5, 3.6, 4.1, 4.4 and adds 3.0
No. 62 (Order for Modification of License)	April 20, 1981	Revises Technical Specification 3.3 and 4.2
No. 63 (Amendment No. 57)	November 5, 1981	Revises Technical Specification 4.2
No. 64 (Amendment No. 58)	December 16, 1981	Revises Technical Specification 1.0, 3.1, 3.5, 3.6, 4.1, 4.3, 6.3, 6.4, 6.10, 6.13, 6.14 and 6.15
No. 65 (Amendment No. 59)	December 17, 1981	Revises Technical Specification 1.0 and 3.1
No. 66 (Amendment No. 61)	September 10, 1982	Revises Technical Specification 6.11
No. 67 (Amendment No. 62)	September 24, 1982	Revises Technical Specification 4.2.3
No. 68 (Amendment No. 63)	October 8, 1982	Revises Technical Specification 3.13, 4.14 and 6.10
No. 69 (Amendment No. 64)	November 17, 1982	Revised Technical Specification 3.0 and 6.9
No. 70 (Amendment No. 65)	December 1, 1982	Revises Technical Specification 4.1
No. 71 (Amendment No. 66)	February 2, 1983	Revises Technical Specification 6.1, 6.2, 6.3, 6.5 and 6.8

*The above numbered designated changes relate to Appendix A Technical Specifications.

APPENDIX A
IDENTIFICATION OF CHANGES* (Continued)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 72 (Amendment No. 68)	May 3, 1983	Revises Technical Specification 3.7
No. 73 (Amendment No. 69)	May 3, 1983	Revises Technical Specification 6.5
No. 74 (Amendment No. 70)	December 6, 1983	Revises Technical Specifications 3.1.1 Table 4.1.2
No. 75 (Amendment No. 71)	February 17, 1984	Revises Technical Specification 3.6.2 and Table 3.6.2-1
No. 76** (Amendment No. 72)	February 17, 1984	Revises Technical Specifications, Table 3.5.5-2 and Sections 5.1, 5.2 and 5.3
No. 77 (Amendment No. 73)	April 23, 1984	Revises Technical Specification 3.6.1 and 3.8.
No. 78 (Amendment No. 75)	June 4, 1984	Revises Technical Specification 4.3.1
No. 80*** (Amendment No. 77)	October 4, 1984	Revises Technical Specifications 1.0, 3.1.2, 3.8 and Table 4.1.2
No. 81 (Amendment No. 81)	October 16, 1984	Revised Technical Specifications 3.13, 4.14, and 6.10.2.
No. 82 (Amendment No. 82)	November 7, 1984	Revised Technical Specifications 3.4.1, 3.4.3, 3.4.4, 3.5.7, 4.1.8, 4.1.9, 4.1.10, 4.4

*The above numbered designated changes relate to Appendix A Technical Specifications.

**As corrected by NRC.s letter dated June 5, 1984, "Correction to Amendment No. 72."

***As corrected by NRC.s letter dated September 24, 1984, "Correction to Amendment No. 77."

APPENDIX A
IDENTIFICATION OF CHANGES* (CONTINUED)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 83 (Amendment No. 83)	November 2, 1984	Revised Technical Specifications 1.0, 3.0, 3.1.1, 3.1.5, 3.1.7, 3.5.1, 3.5.5, 3.5.6, 3.5.10, 3.6.3, 4.0, 4.1, 4.1.5, 4.1.11., 4.1.12, 4.3.3, 6.9.3
No. 84 (Amendment No. 84)	November 14, 1984	Revises Technical Specifications 3.7 and 4.4
No. 85 (Amendment No. 86)	November 26, 1984	Revises Technical Specification 3.3.1
No. 86 (Amendment No. 79)	January 1, 1985	Add Technical Specifications 3.5.8, 3.5.9, 3.15, 3.16, 3.17, 3.18, 3.19, 4.1.2, 4.17, 4.18, 4.19, 6.16, 6.17 and 6.18 and revises Technical Specifications 1.0, 4.1.3, 4.5, 4.6, 5.1, 6.5, 6.8, 6.9.1, 6.9.2
No. 87 (Amendment No. 87)	February 8, 1985	Revises Technical Specification 4.3.1
No. 88**	November 23, 1984	Revises Basis to Technical Specification 3.5.2
No. 89 (Amendment No. 88)	March 6, 1985	Revises Technical Specification 6.2.2
No. 90 (Amendment No. 90)	August 5, 1985	Revises Technical Specification 3.5.8, 3.5.9, 3.15.2, 3.15.3, 3.16.2, 3.16.3, 3.16.4, 3.16.5, 3.16.6, 3.17, 3.18.1, 3.18.2, 3.18.3, 3.19, 4.1.2, 4.1.3, 4.5.1, 4.6.1, 6.5.2, 6.9.3 and adds 6.19.

*The above numbered designated changes relate to Appendix A Technical Specifications.

**Change incorporated per NRC letter dated November 23, 1984, "Change to Basis of Technical Specification 3.5.2."

APPENDIX A
IDENTIFICATION OF CHANGES* (CONTINUED)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 91 (Amendment No. 91)	November 14, 1985	Revises Technical Specifications 3.1.1, 3.1.4, 3.5.10, 3.15.2, 3.15.3, 3.16.2, 3.16.3, 3.16.4, 3.17, 3.18.1, 3.18.2, 4.2.3, 4.8, 4.12, 4.16 and 6.1 through 6.19
No. 92 (Amendment No. 92)	May 21, 1986	Revised Figures 3.1.3a and 3.1.3b
No. 93 (Amendment No. 93)	May 30, 1986	Revised Technical Specification 3.14
No. 94 (Amendment No. 94)	May 30, 1986	Revised Technical Specification 4.10
No. 95 (Amendment No. 95)	July 3, 1986	Revised Technical Specification 4.4
No. 96 (Amendment No. 96)	March 5, 1987	Revised Technical Specification 1.0, 3.1.1, 4.1.2, 6.9.1.5
No. 97 (Amendment No. 97)	April 7, 1987	Revised Technical Specification 2.1
No. 99 (Amendment No. 99)	May 18, 1987	Revised Technical Specification 3.6.2
No. 100***	December 2, 1987	Revises Basis to Technical Specification 4.12
No. 101 (Amendment No. 100)	April 26, 1988	Revises Table 4.1.2
No. 102 (Amendment No. 101)	May 6, 1988	Revises Technical Specification 4.16

*The above numbered designated changes relate to Appendix A Technical Specifications.

***Change incorporated per NRC letter dated December 2, 1987, "Change to Basis of Technical Specification 4.12."

APPENDIX A
IDENTIFICATION OF CHANGES* (CONTINUED)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 103 (Amendment No. 102)	May 23, 1988	Revises Technical Specifications 3.1.2, 3.1.3, 3.2, 3.3.2, 3.20 and 4.20
No. 104** (Amendment No. 103)	June 9, 1988	Revises Technical Specifications 3.1.2 and 4.1
No. 105 (Amendment No. 104)	July 22, 1988	Revises Technical Specifications 1.0 and 4.4
No. 106 (Amendment No. 105)	August 3, 1988	Revises Technical Specifications 3.5.8, 3.5.9, 3.15.2, 3.15.3, 3.16.2, 3.16.3, 3.16.4, 3.17, 3.18.1, 3.18.2, 3.18.3, 3.19, 4.2.3, 4.14, 4.16, 4.19, 6.2.2, 6.5.3.5, 6.8.2, 6.9.1.10, 6.9.2 and 6.16.2
No. 107 (Amendment No. 106)	August 17, 1988	Revises Technical Specification 3.7
No. 109 (Amendment No. 109)	October 26, 1988	Revises Technical Specifications 4.1.6 and 4.11
No. 110 (Amendment No. 110)	October 18, 1988	Revises Technical Specifications 6.2 and 6.5
No. 111 (Amendment No. 111)	October 21, 1988	Revises Technical Specification 3.5.2
No. 112 (Amendment No. 112)	October 28, 1988	Revises Technical Specification 3.9
No. 113 (Amendment No. 113)	November 8, 1988	Deletes Technical Specification 4.13

*The above numbered designated changes relate to Appendix A Technical Specifications.
**As corrected by NRC letter dated August 17, 1988, "Errata for Amendment No. 103."

APPENDIX A
IDENTIFICATION OF CHANGES* (CONTINUED)

DESIGNATED CHANGE	EFFECTIVE DATE	DESCRIPTION
No. 114 (Amendment No. 114)	November 18, 1988	Revises Technical Specification 4.2.1
No. 115 (Amendment No. 115)	November 28, 1988	Revises Technical Specification 4.16
No. 117 (Amendment No. 117)	December 13, 1988	Revises Technical Specifications 1.0, 2.1, 3.5.1, 3.5.6, 3.11, 4.1.1, 4.1.5 and 4.4
No. 118 (Amendment No. 118)	January 25, 1989	Revises Technical Specification 4.3
No. 119 (Amendment No. 119)	February 6, 1989	Revises Technical Specification 3.1.4 and adds 4.1.13
No. 120 (Amendment No. 120)	March 20, 1989	Revises Technical Specification 3.3.2 and Bases for 3.7

APPENDIX A TO LICENSE DPR-13

SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page No.	Amendment No.
i	90
ii	90
iii	90, 102, 120
iv	90
v	90, 91, 102
vi	90, 91
vii	55, 92, 102, 110, 111, 120
viii	56, 58, 71, 79, 83, 104, 117
ix	31, 56, 58, 79, 83, 84, 91, 117
1.0-1	31, 56, 59 83, 117
1.0-2	31, 56, 59, 83, 117
1.0-3	31, 56, 59, 79, 83, 104
1.0-4	31, 56, 59, 79, 83, 117
1.0-5	77, 79, 83, 117
1.0-6	79, 83, 96 117
1.0-7	58, 83, 117
1.0-8	56, 83, 117
2.1-1	43, 55, 97, 117
2.1-2	43, 97, 117
2.1-3	43, 117
2.1-4	43, 97, 117
2.1-5	55
3.0-1	43, 56, 64, 83
3.0-2	56, 64, 83
3.1-1	29, 38, 70, 83, 91, 96

Page No.	Amendment No.
3.1-2	29, 83, 96
3.1-3	43, 77, 103
3.1-4	77
3.1-5	77
3.1-6	77, 102
3.1-7	77, 102, 103
3.1-8	43, 102, 103
3.1-9	77, 102
3.1-10	Change No. 14 38, 102
3.1-11	Change No. 14 38, 102
3.1-12	Change No. 14 92
3.1-13	Change No. 14 92
3.1-14	Change No. 14 102
3.1-15	Change No. 14 102
3.1-16	Change No. 14 102
3.1-17	Change No. 14 102
3.1-18	Change No. 7 37, 55, 91, 119
3.1-19	Change No. 7 55, 119
3.1-20	37, 55, 119
3.1-21	58, 59, 83
3.1-22	58
3.1-23	83
3.2-1	102
3.2-2	25, 102
3.3-1	25, 37, 86
3.3-2	25
3.3-3	25, 38, 86
3.3-4	25, 37
3.3-5	25
3.3-6	25, 102, 120
3.3-7	25, 102
3.3-8	25, 38

Page No.	Amendment No.
3.3-9	25
3.3-10	NRC order 4/20/81
3.3-11	NRC order 4/20/81
3.4-1	29, 82
3.4-2	31, 82
3.4-3	29
3.4-4	82
3.4-5	82
3.5-1	83, 117
3.5-2	43, 56, 58, 83, 117
3.5-3	43, 56, 58, 83, 117
3.5-4	55, 58, 83, 117, 118
3.5-5	83, 117
3.5-6	7, 11, 25, 35, 55, 56, 111
3.5-7	7, 11, 25, 35, 49, 55, 56, 111
3.5-8	11, 49, 111
3.5-9	11, 25, 56, 111
3.5-10	56
3.5-11	56
3.5-12	56
3.5-13	56
3.5-14	56
3.5-15	58, 83
3.5-16	58, 83
3.5-17	58, 83
3.5-18	58, 72
3.5-19	58, 83
3.5-20	58, 83, 117
3.5-21	58, 82
3.5-22	58, 82
3.5-23	58, 82
3.5-24	79, 90, 105

APPENDIX A TO LICENSE DPR-13

SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page No.	Amendment No.
3.5-25	79, 90
3.5-26	79
3.5-27	79, 90, 105
3.5-28	79, 90
3.5-29	79, 90
3.5-30	83
3.5-31	83
3.5-32	83, 91
3.6-1	25, 56, 58, 73
3.6-2	Change No. 7 56
3.6-3	58, 99
3.6-4	58, 71, 99
3.6-5	58, 59, 71, 99
3.6-6	58, 59, 83
3.7-1	25, 52, 68, 84
3.7-2	25, 84, 106
3.7-3	25, 84
3.7-4	25, 52, 68, 84, 106
3.7-5	25, 52, 68, 84, 106, 120
3.8-1	25, 36, 37, 73, 77
3.8-2	36, 77
3.8-3	25, 36, 73, 77
3.8-4	36, 73, 77
3.8-5	36, 73, 77
3.9-1	3, 10, 112
3.9-2	3, 10, 112
3.10-1	7, 8, 112
3.10-2	7, 112
3.11-1	7, 8, 11, 35, 55, 112, 117
3.11-2	7, 11, 112, 117
3.12-1	14
3.13-1	21, 63, 81
3.13-2	21, 63, 81
3.14-1	31, 44
3.14-2	31, 44, 93
3.14-3	31, 93
3.14-4	31, 93
3.14-5	31
3.14-6	31
3.14-7	31
3.15-1	79
3.15-2	79, 90, 91, 105
3.15-3	79, 90, 91, 105
3.16-1	79
3.16-2	79, 90, 91, 105
3.16-3	79, 90, 91, 105
3.16-4	79, 90, 91, 105
3.16-5	79
3.16-6	79, 90
3.16-7	79, 90
3.17-1	79, 90, 91, 105
3.17-2	79, 90
3.18-1	79, 91, 105
3.18-2	79, 90
3.18-3	79
3.18-4	79
3.18-5	79
3.18-6	79
3.18-7	79, 91, 105
3.18-8	79, 90
3.18-9	79, 90, 105
3.19-1	79, 90, 105
3.20-1	102
3.20-2	102
4.0-1	83
4.0-2	83
4.1-1	29, 56, 83, 117
4.1-2	7, 22, 83, 117
4.1-3	117
4.1-4	25, 29, 70, 96, 117
4.1-5	25, 29, 70, 96
4.1-6	25, 29, 56, 70, 100
4.1-7	25, 29, 36, 77, 103
4.1-8	77
4.1-9	77
4.1-10	117
4.1-11	25
4.1-12	Change No. 5 25
4.1-13	79
4.1-14	79, 90
4.1-15	79
4.1-16	79
4.1-17	79, 90
4.1-18	79
4.1-19	58
4.1-20	58
4.1-21	58
4.1-22	58, 83
4.1-23	58, 83, 117
4.1-24	58, 109
4.1-25	58
4.1-26	58, 82
4.1-27	58, 82
4.1-28	65, 82
4.1-29	65
4.1-30	82
4.1-31	83
4.1-32	83
4.1-33	83
4.1-34	119
4.1-35	119

APPENDIX A TO LICENSE DPR-13

SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page No.	Amendment No.
4.2-1	25, 37, 54, 114
4.2-2	25, 37, 114
4.2-3	25, 37, 114
4.2-4	25, 37, 114
4.2-5	25, 37, 114
4.2-6	NRC order 4/20/81 54
4.3-1	24, 58, 75, 118
4.3-2	5, 24, 58, 75, 87, 118
4.3-3	24, 58, 87, 118
4.3-4	24, 87
4.3-5	24, 75, 87
4.3-6	75, 87
4.3-7	58
4.3-8	58, 59, 83
4.3-9	58, 83
4.4-1	Change No. 12 25, 56, 82, 84, 95, 104
4.4-2	25, 84
4.4-3	25, 56, 82, 84, 117
4.4-4	25, 82, 84, 95, 104
4.4-5	84
4.4-6	25, 56, 84, 104
4.5-7	25, 56, 84
4.5-1	Change No. 10 38, 79
4.5-2	79, 90
4.5-3	79
4.5-4	79
4.5-5	79
4.5-6	79
4.6-1	38, 79
4.6-2	79, 90
4.6-3	79
4.6-4	79
4.6-5	79

Page No.	Amendment No.
4.6-6	79
4.6-7	79
4.6-8	79
4.6-9	79
4.7-1	Change No. 14 46
4.8-1	91
4.9-1	Change No. 14 NONE
4.9-2	NONE
4.10-1	13, 94
4.10-2	13, 94
4.11-1	14, 109
4.11-2	14, 109
4.12-1	91
4.13-1	18, 113
4.14-1	21, 33, 63, 81
4.14-2	21, 63, 81
4.14-3	21, 63, 81
4.14-4	21, 63, 81, 105
4.15-1	31, 37, 38, 44
4.15-2	31, 44
4.15-3	31
4.16-1	31, 101, 115
4.16-2	37, 101, 115
4.16-3	37, 101, 115
4.16-4	37, 55, 101, 115
4.16-5	37, 91, 101, 105, 115
4.16-6	37, 55, 101, 115
4.16-7	37, 55, 101
4.17-1	79
4.18-1	79
4.18-2	79
4.18-3	79
4.18-4	79
4.18-5	79
4.18-6	79

Page No.	Amendment No.
4.19-1	79, 105
4.20-1	102
5.1-1	25, 72
5.1-2	72, 79
5.2-1	72
5.2-2	72
5.2-3	72
5.3-1	3, 72
5.3-2	3, 37, 72
5.3-3	72
5.4-1	NONE
6.1-1	12, 39, 66, 91
6.2-1	44, 66, 91, 110
6.2-2	88, 91
6.2-3	66, 88, 91, 105, 110
6.2-4	12, 58, 91
6.3-1	12, 27, 39, 54, 58, 66, 91
6.4-1	31, 39, 54, 91
6.5-1	Change No. 69 12, 39, 42, 54, 66, 91
6.5-2	39, 54, 66, 91
6.5-3	12, 16, 39, 50, 54, 66, 91
6.5-4	39, 66, 69, 79, 91
6.5-5	12, 50, 90, 91
6.5-6	12, 16, 69, 91, 105
6.5-7	12, 31, 91, 110
6.6-1	12, 91

APPENDIX A TO LICENSE DPR-13

SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page No.	Amendment No.
6.7-1	39, 54, 66, 91
6.8-1	39, 54, 66, 79, 91, 105
6.8-2	39, 91
6.8-3	91
6.9-1	12, 16, 30, 91
6.9-2	12, 16, 30, 79, 91, 96
6.9-3	12, 15, 16, 30, 79, 91, 96
6.9-4	12, 30, 79, 91, 96
6.9-5	12, 16, 30, 38, 79, 90, 91, 96, 105
6.10-1	12, 15, 16, 91
6.10-2	12, 63, 81, 91
6.11-1	58, 61, 91
6.12-1	23, 38, 58, 91
6.12-2	23, 38, 91
6.13-1	58, 79, 91
6.14-1	58, 79, 91
6.15-1	58, 79, 91
6.16-1	90, 91
6.16-2	90, 91
6.16-3	90, 91, 105
6.16-4	90, 91

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds with the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

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 instrument channels measuring the same parameter.

CHANNEL TEST

A CHANNEL TEST shall be the injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action. The CHANNEL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints, such that the setpoints are within the required range and accuracy.

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

(1) All non-automatic containment isolation valves (or blind flanges) are closed.

(2) The equipment door is properly closed.

(3) At least one door in each personnel air lock is properly closed.

(4) All automatic containment isolation valves are operable.

CORE ALTERATION

CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORRELATION CHECK

A CORRELATION CHECK shall be an engineering analysis of an incore flux map wherein at least one point along the incore versus excore correlation data plot is obtained.

CORRELATION VERIFICATION

A CORRELATION VERIFICATION shall be the engineering analysis of incore flux maps wherein multiple points along the incore versus excore correlation data plot are obtained.

DG FAST START

A DG FAST START shall be an automatic or manual start of an emergency diesel generator in which the steady state voltage and frequency is achieved within 10 seconds.

DG SLOW START

A DG SLOW START shall be an automatic or manual start of an emergency diesel generator in which steady state voltage and frequency is achieved in not less than 24 seconds.

DOSE EQUIVALENT I-131

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."

E - AVERAGE DISINTEGRATION ENERGY

E is 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 and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of a water source(s), pump(s), and distribution piping with associated isolation valves (i.e., system header, hose standpipe and spray header isolation valves).

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

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

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include nonemployees of the licensee who are permitted to use portions of the site for recreational, occupational, or purposes not associated with plant functions. This category shall not, include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

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).

OPERATIONAL MODE - MODE

An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PROCESS CONTROL PROGRAM

The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

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 condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 1347 Mwt.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESIDUAL HEAT REMOVAL (RHR) TRAIN

An RHR TRAIN shall be a train of components that includes: one RHR pump aligned with one RHR heat exchanger; one component cooling water pump aligned with the same RHR heat exchanger and with one component cooling water heat exchanger; and one salt water pump aligned with the same component cooling water heat exchanger.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

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

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into an equal subintervals,

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

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 effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

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

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
BW	At least once per 14 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release
N/A	Not applicable.

TABLE 1.2

OPERATIONAL MODE

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	>5%	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	<0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	≤ 0.95	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	≤ 0.95	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat
**Reactor vessel head unbolted or removed and fuel in the vessel

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

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

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded.

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1.

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3)

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

   In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

   In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less.

2. Variable Low Pressure Loss of Flow and Nuclear Overpower Trips

   These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, control rod ejection, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)(7)

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 5%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above.

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

<u>References</u>:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

(7) NIS Safety Review Report, April 1988

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

Function	Setting
*1. Pressurizer High Level	$\leq$ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip <u>is not</u> credited, or $\leq$ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip <u>is</u> credited
2. Pressurizer Pressure: High	$\leq$ 2220 psig
**3. Nuclear Overpower	
a. High Setting**	$\leq$ 109% of indicated full power
b. Low Setting	$\leq$ 25% of indicated full power
***4. Variable Low Pressure	$\geq$ 26.15 (0.894 ΔT+T avg.) - 14341
***5. Coolant Flow	$\geq$ 85% of indicated full loop flow

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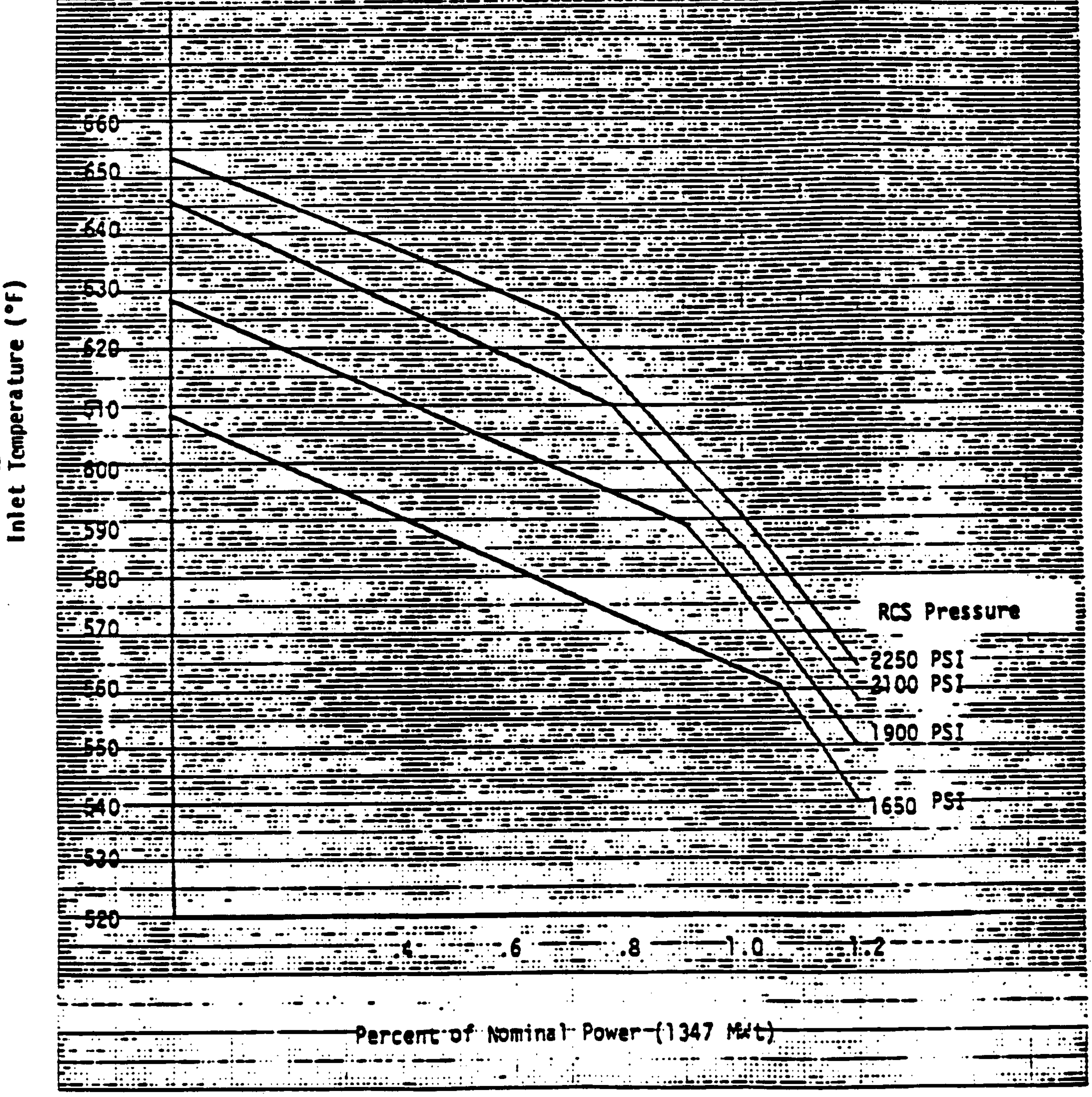
* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function.

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 5%.

***May be bypassed at power levels below 10% of full power.

Figure 2.1.1
Safety Limits
Temperature, Power, Pressure
RCS Flow ≥ 195,000 GPM



3.0 LIMITING CONDITIONS FOR OPERATION (GENERAL)

APPLICABILITY: Applies to the Operational requirements to be implemented when specific actions are not identified within individual Limiting Conditions for Operation.

OBJECTIVE: To ensure that the station is placed in a safe condition when circumstances arise which are not identified within individual Limiting Conditions for Operation.

SPECIFICATION: 3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE 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 specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

BASIS:

Specification 3.0.1 defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

Specification 3.0.2 defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

Specification 3.0.3 delineates the action to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Technical Specification 3.3 requires in part that two recirculating pumps be OPERABLE in order for the reactor to be made or maintained critical and provides explicit ACTION requirements if one recirculation pump is inoperable. Under the terms of Specification 3.0.3, if more than one recirculation pump is inoperable, action shall be initiated within one hour to place the unit in at least HOT STANDBY within the next 6 hours and at least HOT SHUTDOWN within the following 24 hours and at least COLD SHUTDOWN in the following 24 hours unless corrective measures are completed. It is assured that the unit is brought to the required MODE of operation within the required times by promptly initiating and carrying out the appropriate action statement.

Specification 3.0.4 provides that entry into the OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.1 REACTOR COOLANT SYSTEM

3.1.1 MAXIMUM REACTOR COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum activity in the reactor coolant system at any time.

OBJECTIVE: To limit the consequences of an accidental release of reactor coolant to the environment.

SPECIFICATION: The specific activity of the reactor coolant shall be limited to:

1. ≤ 1.0 μCi/gm DOSE EQUIVALENT I-131.

2. ≤ 100/$\bar{E}$ μCi/gm.

ACTION:

A. With the specific activity of the reactor coolant determined to be >1 μCi/gm DOSE EQUIVALENT 1-131 for more than 48 hours during one continuous time interval or >60 μCi/gm DOSE EQUIVALENT I-131 or >100/$\bar{E}$ μCi/gm, be in at least HOT STANDBY with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.

B. With the specific activity of the reactor coolant > 1.0 μCi/gm DOSE EQUIVALENT 1-131 or >100/$\bar{E}$ μCi/gm, perform the sampling and analysis requirements of item 1a.4.a of Table 4.1.2 until the specific activity of the reactor coolant is restored to within its limits.

C. The provisions of Specification 3.0.4 are not applicable.

Specific Activity

BASIS:

The limitations on the specific activity of the reactor Coolant ensure that the resulting 2 hour doses at the site boundary will not exceed the guidelines of 10 CFR Part 100 following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity > 1.0 μCi/gm DOSE EQUIVALENT 1-131, accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER.

Reducing T_{avg} to < 535°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor 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 reactor coolant will be detected in sufficient time to take corrective action. Increased surveillance for performing isotopic analyses for iodine is required whenever the DOSE EQUIVALENT I-131 exceeds 1.0 μCi/gram and following a significant change in power level to monitor possible iodine spiking phenomena to assure the activity remains < 60 μCi/gm DOSE EQUIVALENT I-131.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.1.2 OPERATIONAL COMPONENTS

APPLICABILITY: Applies to the operating status of the reactor coolant system equipment and related equipment. For the applicable surveillance requirements, see Table 4.1.2.

OBJECTIVE: To identify those conditions of the reactor coolant system necessary to ensure safe reactor operation.

SPECIFICATIONS:

A. At least one pressurizer safety valve shall be OPERABLE or open when the reactor head is on the vessel, except for hydrostatic tests.

B. The reactor shall not be made critical or maintained critical unless both pressurizer safety valves are OPERABLE.

C. During MODES 1 and 2 and in MODE 3 with reactor trip breakers closed, all three reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation. With less than the above required coolant loops in operation, be in at least HOT STANDBY with reactor trip breakers open within 1 hour, except as modified by Specification D below.

D. The limitations of Specification C may be suspended as follows:

   1. During MODES 1 and 2, operation may be conducted with 0, 1, 2 or 3 reactor coolant pumps operating at less than 5% of RATED THERMAL POWER for purposes of conducting low power physics testing.

   2. During MODES 1 and 2 and in Mode 3 with reactor trip breakers closed, operation may be conducted for less than 24 consecutive hours with one or two reactor coolant pumps operating if THERMAL POWER is less than 10% of RATED THERMAL POWER.

E. During MODE 3 with the reactor trip breakers open, the following specifications shall apply:

   1. At least two of the reactor coolant loops listed below shall be OPERABLE:

      a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

      b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

      c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

2. At least one of the above reactor coolant loops shall be in operation.*

3. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

4. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required reactor coolant loop to operation.

F. During MODE 4, the following specifications shall apply:

1. At least two of the reactor coolant loops/RESIDUAL HEAT REMOVAL (RHR) TRAINS listed below shall be OPERABLE:

   a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

   b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

   c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

   d. Residual heat removal (RHR) pump G-14A and one associated RHR TRAIN.

   e. Residual heat removal (RHR) pump G-14B and one associated RHR TRAIN.

2. At least one of the above loops/trains shall be in operation.**

* All reactor coolant pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

** All reactor coolant pumps and residual heat removal pumps may be deenergized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

3. With less than the above required loops/trains OPERABLE immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop/train is an RHR TRAIN, be in COLD SHUTDOWN within 24 hours.

4. With no loop or train in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return one required loop or train to operation.

G. During MODE 5 with reactor coolant loops filled, the following specifications shall apply:

1. At least one RESIDUAL HEAT REMOVAL (RHR) TRAIN shall be OPERABLE and in operation*, and either

   a. One additional RHR TRAIN shall be OPERABLE,** or

   b. The secondary side water level of at least two steam generators shall be greater than or equal to 256 inches of narrow range on cold calibrated scale.

2. With less than the above required loops/trains OPERABLE, or with less than the required steam generator level, immediately initiate corrective action to return the required loops/trains to OPERABLE status or to restore the required level as soon as possible.

3. With no RHR TRAIN in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required RHR TRAIN to operation.

* The RHR pump may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

** One RHR TRAIN may be inoperable for up to 2 hours for surveillance testing, provided the other RHR TRAIN is OPERABLE and in operation.

H. During MODE 5 with reactor coolant loops not filled, the following specifications shall apply:

1. Two RESIDUAL HEAT REMOVAL (RHR) TRAINS shall be OPERABLE* and at least one RHR TRAIN shall be in operation**.

2. With less than the above required RHR TRAINS OPERABLE, immediately initiate corrective action to return the required RHR TRAINS to operable status as soon as possible.

3. With no RHR TRAIN in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required RHR TRAIN to operation.

I. A reactor coolant pump shall not be started with the RCS pressure $\leq$ 400 psig unless:

1. the pressurizer water level is less than 80%, or

2. the potential for having developed reactor coolant system temperature gradients has been evaluated.

BASIS:

One pressurizer safety valve is sufficient to prevent over-pressurizing when the reactor is subcritical, since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., residual heat, pump energy and pressurizer heaters.

Prior to reducing boron concentration by dilution with make up water either a reactor coolant pump or a residual heat removal pump is specified to be in operation in order to provide effective mixing. During boron injection, the operation of a pump, although desirable, is not essential. The boron is injected into an inlet leg of the reactor coolant loop. Thermal circulation which exists whenever there is residual heat in the core and the reactor coolant system is filled and vented, will cause the boron to flow to the core.

* One RHR TRAIN may be inoperable for up to 2 hours for surveillance testing provided the other RHR TRAIN is operable and in operation.

** The RHR pump may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

Lack of further mixing cannot result in areas of reduced boron concentration within the core. Prior to criticality the two pressurizer safety relief valves are specified in service in order to conform to the system relief capabilities.(1)

The plant is designed to have all three reactor coolant loops operational during normal power operation (MODES 1 and 2). Under these conditions, the DNB ratio will not drop below 1.30 after a loss of flow with a reactor trip.(2)(3) With one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY with reactor trip breakers open within one hour (for the significance of the trip breaker position, see below). However, exception is taken whenever reactor power is less than 10% of RATED THERMAL POWER. Heat transfer analyses show that reactor heat equivalent to 8% of RATED THERMAL POWER can be removed with natural circulation only; hence, for up to 24 hours the specified upper limit of 10% of RATED THERMAL POWER with 1 or 2 reactor coolant pumps operating provides a substantial safety factor.

In MODES other than MODES 1 and 2, functional redundancy in the core heat removal methods (not necessarily system redundancy) is specified to satisfy single failure considerations. Functional redundancy, as applied to the San Onofre Unit 1 power plant, includes use of diverse heat removal methods. Furthermore, single failure considerations apply only to active components.

For operation in MODE 3 under all design basis conditions, it has been determined that one reactor coolant (RC) loop generally provides the required decay heat removal capability, the only exception to this being the control rod bank withdrawal from subcritical accident, when the DNB design basis may not be met. Since power to the gripper and lift coils of the control rod drive mechanism is carried through two reactor trip circuit breakers connected in series with the coils, both breakers must be manually closed before any control rod motion out of the core can take place. In light of this design feature, these Technical Specifications require that all three RC loops be in operation in MODE 3 if the reactor trip breakers are closed. Whenever the reactor trip breakers are open, the design feature would prevent any control rod motion, even though single failure considerations* require that at least two loops be operable. For the same reasons and subject to the same limitations that are stated in the preceding paragraph, exception is taken whenever reactor power is less than 10% of RATED THERMAL POWER.

*Single failure considerations apply to active components.

In MODES 4 and 5, the Technical Specifications permit functional redundancy in the core heat removal methods (not necessarily system redundancy) to satisfy single failure considerations. Functional redundancy, as applied to the San Onofre Unit 1 power plant, includes use of diverse heat removal methods.

In MODE 4 and MODE 5 (reactor coolant loops filled), a single reactor coolant loop or RHR TRAIN provides sufficient capability for removing decay heat; but single failure considerations* require that at least two methods (either RCS loop or RHR TRAIN) be OPERABLE.

In MODE 5 (reactor coolant loops not filled), a single RHR TRAIN provides sufficient heat removal capability for removing decay heat; but single failure considerations,* and the unavailability of any of the steam generators as a heat removing component, require that at least two RHR TRAINS be OPERABLE.

The operation of one reactor coolant pump or one residual heat removal pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control(4).

The limitation on reactor coolant pump operation with the RCS pressure $\leq$ 400 psig ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50(5). A pressurizer water level of less than 80% ensures that the start of a reactor coolant pump, with a temperature differential of 100°F will not result in 10 CFR Part 50 Appendix G limits being exceeded.

There are several means available for determining that there is not a temperature differential of > 50°F between the secondary and primary systems with $\leq$ 400 psig primary system pressure. These methods may include but are not necessarily limited to the following:

1) Converting steam line pressure indication into maximum temperature of steam generator fluid.

2) Tagging RCP switches with shutoff temperatures.

*Single failure considerations apply to active components.

3) Assuring adequate time for temperature gradients to dissipate.

4) Filling steam generators with water of known temperature.

REFERENCES:

(1) Final Engineering Report and Safety Analysis, Sections 9 and 10.

(2) Final Engineering Report and Safety Analysis, Paragraph 10.2.

(3) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 3, Question 9.

(4) NRC letter dated June 11, 1980, from D. G. Eisenhut to all operating pressurized water reactors.

(5) Letter to A. Schwencer from K. Baskin dated October 12, 1977.

3.1.3 COMBINED HEATUP, COOLDOWN AND PRESSURE LIMITATIONS

APPLICABILITY: Applies to heatup and cooldown of the reactor coolant system.

OBJECTIVE: To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.

SPECIFICATION:

A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 16 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 16 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly (by formal license amendment application).*

B. Figures 3.1.3a and 3.1.3b shall be updated in accordance with the following criteria and procedures:

(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature relationships for the reactor coolant system.

(2) The curves in Figure 3.1.3c shall be used in predicting the reference nil-ductility temperature increase, RT_{NDT} unless measurements on the irradiation specimens show RT_{NDT}s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.

C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 200°F/hour shall not be exceeded.

D. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line as shown in Figures 3.1.3a.

BASIS: The initial Reference Nil Ductility Temperature (RT_{NDT}) for all reactor vessel material based on Charpy V-notch data, drop weight tests, and conservative estimates** is 82°F or less. The RT_{NDT} at the 1/4 thickness location (location of Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 240°F for the core region of the reactor vessel after 30 years of operation.

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* Technical Specification 3.20.A(1) should be reevaluated for continued applicability of the low pressure PORV overpressure setpoint at any time the heatup and cooldown curves are changed.

** NRC Standard Review Plan Branch Technical Position MTEB 5-2.

A sixteen (16) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting RT_{NDT} of the reactor vessel at the 1/4 thickness location is approximately 217°F in the core region. This RT_{NDT} is at least 50°F above the RT_{NDT} of all other regions in the primary reactor coolant system.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in the Table 3.1.3.1. The fast neutron ($E > 1$Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the RTNDT is obtained from Figure 3.1.3c.

Values of ΔRT_{NDT} may continue to be determined in this manner unless measurements on the irradiation specimens show ΔRT_{NDT}s greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from non-mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.

The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The vertical line portion of the criticality limit given in Figures 3.1.3a is at the minimum permissible temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10CFR Part 50. The non-vertical portion of the criticality limit is shifted 40°F to the right of the heatup curve as required by Appendix G to 10CFR Part 50.

REFERENCES:

(1) "Pressure Temperature Limits" Section 5.3.2 of Standard Review Plan, NUREG-751087, 1975.

(2) S. E. Yanichko, et al, "Analysis of Capsule F from the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program", WCAP 9520, May 1979.

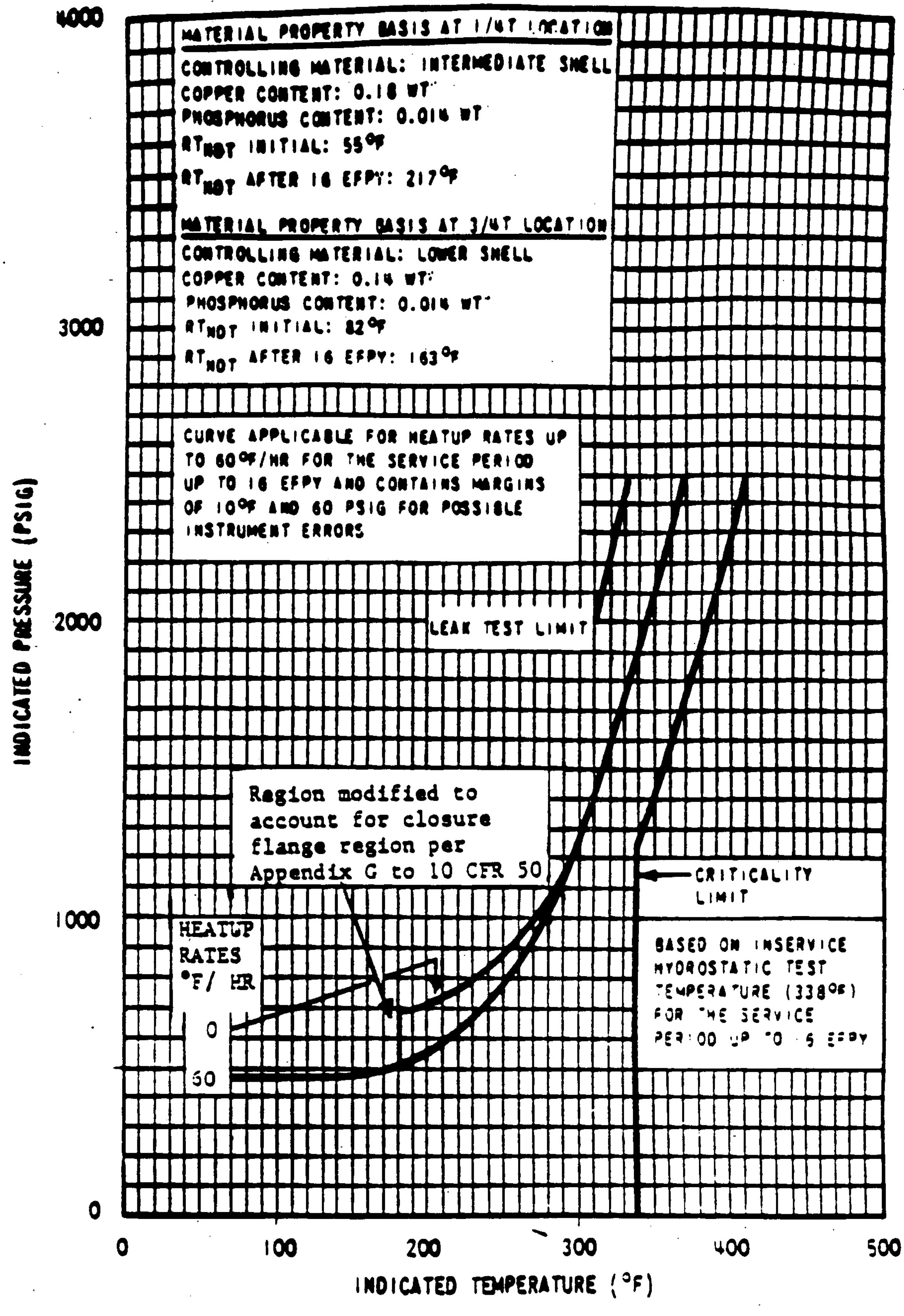

FIGURE 3.1.3a San Onofre Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY.

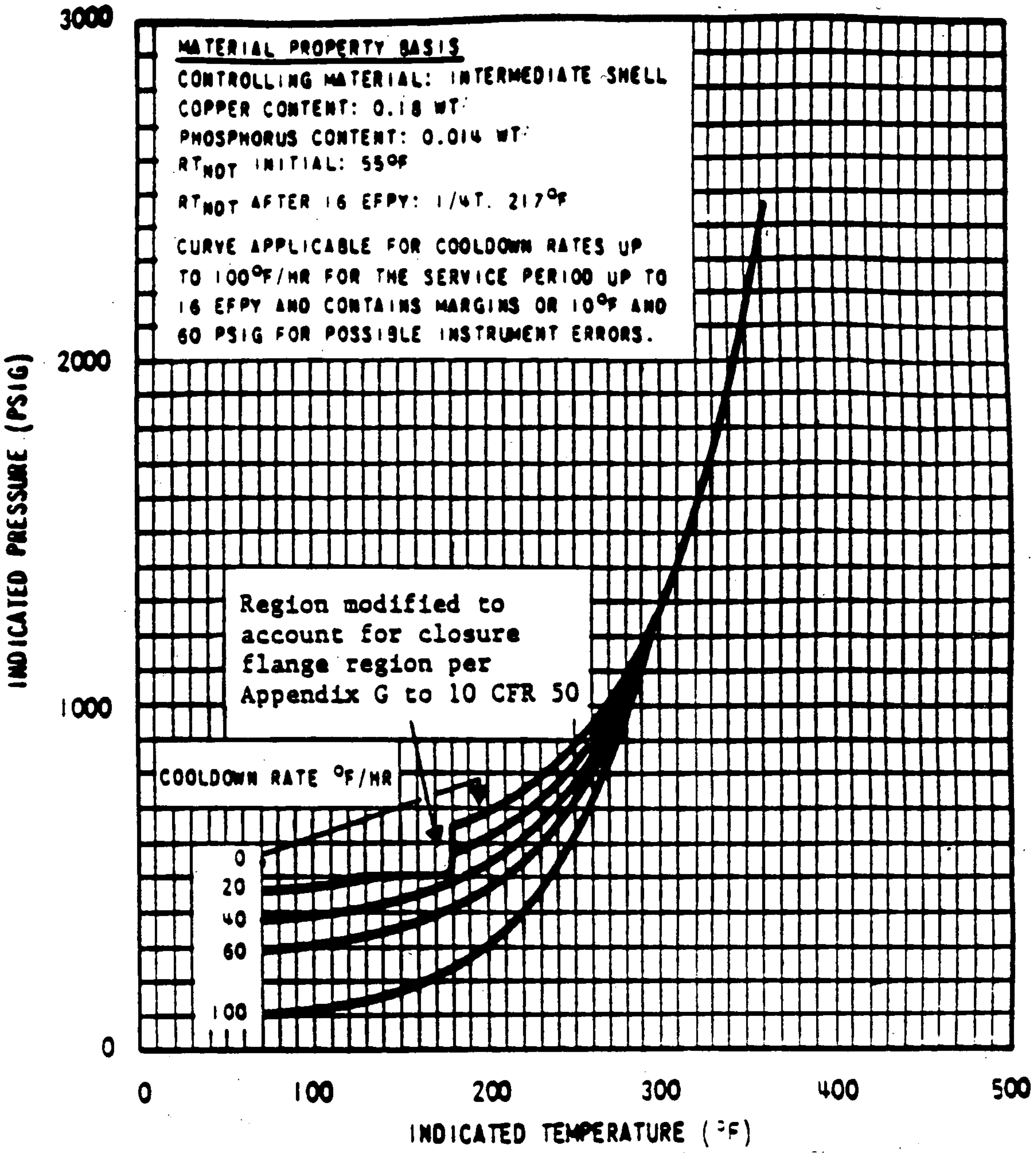

FIGURE 3.1.3b San Onofre Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY.

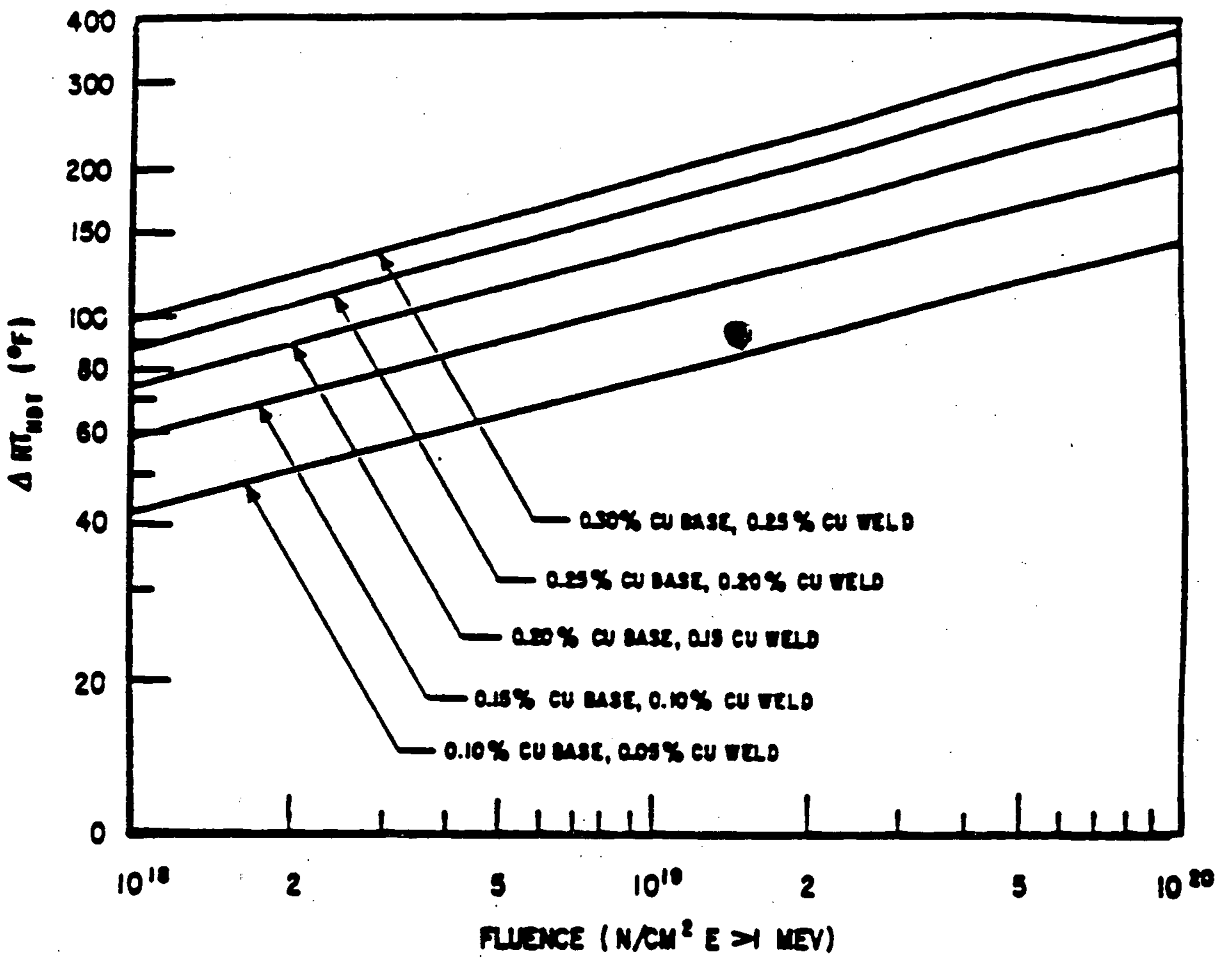

FIGURE 3.1.3c Effect of Fluence and Copper Content on ΔRT_{NDT} For Reactor Vessel Steels Exposed to Irradiation at 550°F

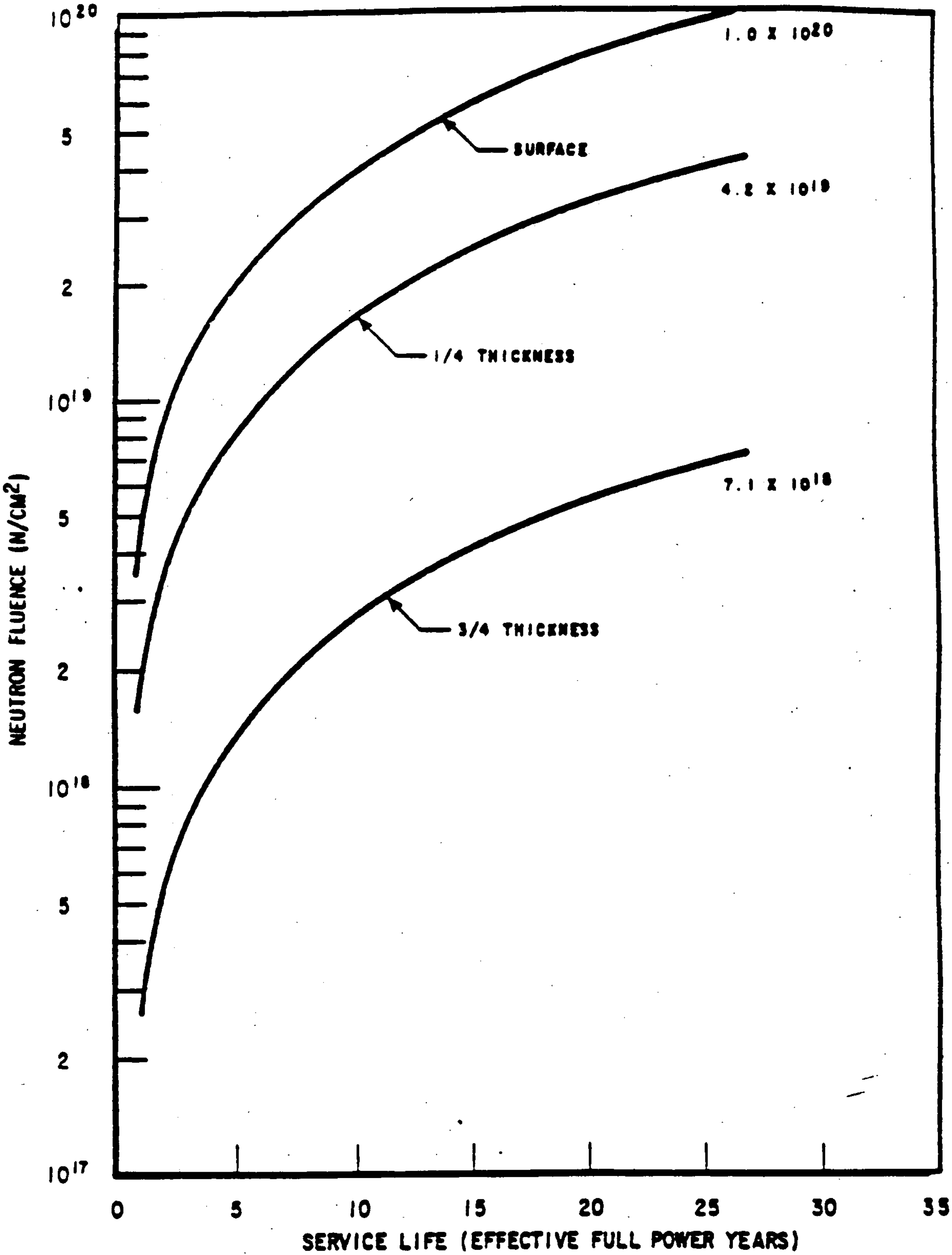

Figure 3.1.3d Fast Neutron Fluence (E 1 Mev) as a Function of Full-Power Service Life.

TABLE 3.1.3.1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp (°F) Long.	Minimum 50 ft-lb/35 mil Temp (°F) Trans.	RT_{NDT} (°F)	Average Upper Shelf Energy (ft-lb) Long.	Average Upper Shelf Energy (ft-lb) Trans.
Cl. Hd. Dome	W7604	A302B	--	--	60(a)	112	132	72	72.5	--
Peel Segment	W7605-1	A302B	--	--	-10	114	134	74	70.5	--
Peel Segment	W7605-2	A302B	--	--	-10	90	110	50	122	--
Peel Segment	W7605-3	A302B	--	--	-10	108	128	68	85	--
Peel Segment	W7605-4	A302B	--	--	-10	120	140	80	74	--
Peel Segment	W7605-5	A302B	--	--	-10	26	46	10	109	--
Peel Segment	W7605-6	A302B	--	--	-10	102	122	62	88	--
Hd. Flange	W7602	A336 mod	--	--	60(a)	(b)	--	60	--	--
Ves. Flange	W7603	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-1	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-2	A336 mod	--	--	60(a)	(b)	--	60	--	--
Inlet Nozzle	W7611-3	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-1	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-2	A336 mod	--	--	60(a)	(b)	--	60	--	--
Outlet Nozzle	W7610-3	A336 mod	--	--	60(a)	(b)	--	60	--	--
Upper Shell	W7601-3	A302B	0.15	0.014	-10	48	68	8	98.5	--
Upper Shell	W7601-6	A302B	0.16	0.012	-30	64	84	24	104	--
Upper Shell	W7601-7	A302B	0.15	0.014	-20	52	72	12	95.5	--

a. Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2.
b. Only 10°F Charpy V-notch data available. Conservative estimates for NDTT and RT_{NDT} were used.

TABLE 3.1.3.1(cont'd)

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Code No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp (°F) Long.	Minimum 50 ft-lb/35 mil Temp (°F) Trans.	RT_{NDT} (°F)	Average Upper Shelf Energy (ft-lb) Long.	Average Upper Shelf Energy (ft-lb) Trans.
Inter. Shell	W7601-1	A302B	0.17	0.013	0	57	120(a)	60	94	75
Inter. Shell	W7601-8	A302B	0.18	0.012	10	93	100(a)	40	97	79
Inter. Shell	W7601-9	A302B	0.18	0.014	0	64	115(a)	55	102	72
Lower Shell	W7601-2	A302B	0.17	0.013	-20	74	94	34	97	--
Lower Shell	W7601-4	A302B	0.14	0.014	-10	91	111	51	94	--
Lower Shell	W7601-5	A302B	0.14	0.014	10	122	142	82	87.5	--
Bot. Hd. Peel	W7607	A302B	--	--	-20	62	82	22	91	--
Bot. Hd. Dome	W7606	A302B	--	--	60(b)	99	119	60	86	--
Weld	--	--	0.19	0.017	0(b)	--	29(a)	0	--	90
HAZ	--	--	--	--	0(b)	--	-14(a)	0	--	101

a. Actual not estimated.
b. Estimated per NRC Standard Review Plan Branch Technical Position MTEB 5-2.

3.1.4 LEAKAGE AND LEAKAGE DETECTION SYSTEMS

APPLICABILITY: Applies to reactor coolant system leakage and leakage detection systems during MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure that leakage from the reactor coolant system is detected and does not exceed acceptable limits.

SPECIFICATION:

a. The reactor coolant system shall be monitored for evidence of leakage. Abnormal or significant leakage from the reactor coolant system shall be investigated and evaluated. The following reactor coolant system leakage limits shall apply:

(i) The total unidentified leakage shall not exceed 1 gpm.

(ii) The total leakage shall not exceed 6 gpm.

b. The following detection systems shall be OPERABLE:

(i) The containment atmosphere monitor R1211 or R1212, or containment atmosphere grab samples shall be taken every 12 hours and analyzed within the following 6 hours.

(ii) The sphere sump level instrumentation LIS 2001, LIS 3001 or both LS 80 and LS 82.

(iii) The steam generator blowdown effluent line monitor R1216 or steam generator blowdown effluent grab samples shall be taken every 12 hours and analyzed within the following 6 hours.

ACTION:

A. With any reactor coolant system leakage greater than the above defined limits, reduce the leakage rate to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. Upon detection and confirmation of any of the following conditions:

1. An increase in primary to secondary leakage of 140 gpd (0.1 gpm) over a period of 24 hours in any steam generator; or

2. Any primary to secondary leakage in excess of 215 gpd (0.15 gpm) in any steam generator; or

3. Measured increase in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day in any steam generator, when measured primary to secondary leakage is above 140 gpd;

the reactor will be placed in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. Following reactor shutdown, leaking tubes shall be repaired or plugged.

C. Upon detection and confirmation of primary to secondary leaks in excess of 0.3 gpm in any steam generator, the reactor will be placed in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. Following reactor shutdown, an eddy current inspection will be performed as required by Technical Specification 4.16, any leaking steam generator tubes shall be repaired or plugged and the NRC be notified pursuant to Specification 6.9.2 prior to resumption of plant operation.

D. With only two of the above required leakage detection systems/methods OPERABLE, operation may continue for up to 30 days provided a Reactor Coolant System water inventory balance is performed every 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

<u>BASIS</u>:

Two basic kinds of leakage from the reactor coolant system are possible, namely:

1. To other closed systems.

2. Directly to the containment.

Systems into which leakage from the reactor coolant system could occur are designed to accept such leakage. However, leakage directly into the containment indicates the possibility of a breach in the coolant envelope. For this reason, the acceptable value for a source of leakage not identified was set at 1 gpm.

Once the source of leakage has been identified, it can be determined if operation can safely continue. Under these conditions, an allowable leakage rate of 6 gpm has been established. This is based upon the contingency of sustained loss of all off-site power and failure of the on-site generation. With 6 gpm leakage, decay heat removal can safely be accomplished for a period in excess of 12 hours. Within the 12 hour period, the reactor coolant system can be depressurized.

To comply with Paragraph IV.C.1(b)(4) of the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors" adopted by the AEC on June 19, 1971, the maximum allowable identified leakage rate from the primary coolant system has been established as not exceeding 6 gpm. This value is based on operating experience regarding non-safety related equipment limitations which has shown that, under certain circumstances where primary system leakage is directed to the gas handling portion of the radwaste system, the capacity of this system would be exceeded during extended operation with a leakage greater than 6 gpm. The justification for the 0.3 gpm primary to secondary leakage limit is as described in the Basis for Technical Specification 4.16.

Detection of leaks from the reactor coolant system to the containment and/or secondary system is accomplished primarily through use of the following methods:

1. Sump level

2. Radiation monitoring

3. Blowdown effluent monitoring

With these methods, a leak of 1 gpm can be detected in a matter of hours. The radiation monitors can measure the presence of a leak into the containment by monitoring the change in background radiation levels. As an alternate to direct measurement, the use of grab samples at an appropriate frequency is also acceptable. The sump level control system consists of two instrumentation inputs (LS-80 and 82) which alert the operators of changing conditions at different sump levels. The sump level monitoring system (LIS 2001 and LIS 3001) is an alternate to the sump level control system, but since it is not alarmed, it is required by surveillance to be monitored every 12 hours. Additional indicators of potential RCS leakage include containment temperature, humidity and pressure. Leakage through the steam generators is detected primarily through use of the blowdown effluent monitor and alternately by grab samples. In the event of unavailability of one of the three methods of reactor coolant system leakage detection, the performance of a reactor coolant system water inventory balance at an increased frequency assures safety.

3.1.5 PRESSURIZER RELIEF VALVES

APPLICABILITY: MODES 1, 2 and 3.

OBJECTIVE: To ensure reliability of the power operated relief valves (PORVs) and their associated block valves.

SPECIFICATION: Two PORVs and their associated block valves shall be OPERABLE.

ACTION:

A. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain the block valve(s) in the closed position; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. With one or more block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. The provisions of Specification 3.0.4 are not applicable.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

3.1.6 PRESSURIZER

APPLICABILITY: Applies to the pressurizer heaters and pressurizer water level for MODES 1, 2 and 3.

OBJECTIVE: To ensure that pressurizer heaters are available during a loss of offsite power condition.

SPECIFICATION: A. The pressurizer shall be OPERABLE with at least 125 kilowatts of pressurizer heaters and a water level between 5 percent and 70 percent.

ACTION: B. With the pressurizer inoperable due to the loss of capability to energize the pressurizer heaters from an emergency diesel generator, either restore the capability to energize the pressurizer heaters from an emergency diesel generator within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The requirement that 125 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency diesel generator provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

(2) SCE letter dated October 17, 1979, from J. H. Drake to D. G. Eisenhut, "Responses to NRC Requirements Related to the Three Mile Island Accident," Item 2.1.1 of the enclosure.

3.1.7 REACTOR COOLANT SYSTEM VENTS

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To provide the capability of VENTING non-condensible gas accumulations in the primary coolant system.

SPECIFICATION: As a minimum at least one reactor coolant vent path consisting of at least two valves in series powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Reactor Vessel head
b. Pressurizer steam space

ACTION:

A. With one of the above reactor coolant system vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

BASIS: Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head, and the pressurizer steam space, ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737,[1] "Clarification of TMI Action Plan Requirements", November 1980.

REFERENCES:

(1) NRC letter dated November 1, 1983 from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specifications (Generic Letter No. 83-37).

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

APPLICABILITY: Applies to the operational status of the chemical and volume control system.

OBJECTIVE: To identify those conditions of the chemical and volume control system necessary to ensure safe reactor operation.

SPECIFICATION: A. When fuel is in the reactor, the following chemical and volume control system conditions shall be met:

(1) One charging pump or the test pump shall be OPERABLE. However, when the RCS pressure is < 400 psig and pressurizer water level is greater than 50%, a maximum of one of the two centrifugal charging pumps shall be OPERABLE. The inoperable centrifugal charging pump shall have the motor circuit breaker removed from the electrical power supply circuit and shall be condition tagged.

(2) One boric acid transfer pump or the boric acid injection pump shall be OPERABLE.

(3) A solution of at least 3450 pounds of boric acid in not less than 3500 gallons of water at a temperature of 140°F or higher, with at least one heater OPERABLE, shall be in the boric acid tank.

(4) System piping and valves shall be OPERABLE to the extent of establishing two flow paths for boric acid tanks.

(5) During periods when borated water is in the refueling cavity, the requirements in A.(1) through A.(4) may be waived provided that an alternate source of borated water is available to establish at least one flow path to the core for boric acid injection which can be initiated from the control room. The minimum capability for boric acid addition shall be equivalent to that supplied by a charging pump from the refueling water storage tank.

B. The reactor shall not be made critical unless the following additional conditions are met:

(1) One additional charging pump or test pump OPERABLE.

(2) One additional boric acid transfer pump or boric acid injection pump OPERABLE.

(3) Electrical heat tracing for boric acid piping OPERABLE.

C. After criticality is achieved, maintenance on item B.(3) will be allowed providing the boric acid temperature does not fall below 140°F.

BASIS: The Chemical and Volume Control System[1] provides control of the reactor system boron concentration. This is accomplished by using either one of the two charging pumps or the test pump (Chemical and Volume Control System test pump installed in parallel with the charging pump) to inject concentrated boric acid solution into the reactor coolant system. There are two sources of borated water available for injection through three different paths as follows:

1. The boric acid injection pump can deliver the boric acid tank contents to the charging pump and/or test pump.

2. Boric acid transfer pumps can deliver the boric acid tank contents to the charging pumps and/or test pump.

3. The charging pumps and the test pump can take suction directly from the refueling water storage tank (3750 ppm boron).

The quantity of boric acid in storage from the above two sources is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during the core life. Furthermore, if the letdown capability from the primary coolant system to the chemical and volume control system should be impaired, the pressurizer void space volume is sufficient to accommodate the required injection. This free volume will accommodate sufficient concentrated boric acid solution such that the reactor coolant water can reach a concentration of about 400 ppm above the required level to shut the plant down.

The above system assured that for Specification A, continuous borated water supply is provided to maintain the core subcritical. In Specification B, redundancy is provided for borated water injection during reactor operations.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE with an RCS pressure $\leq$ 400 psig with pressurizer water level greater than 50%, provides assurance that a mass addition pressure transient can be relieved by operation of the overpressurization mitigating system assuming a single failure of one PORV and no operator action for 10 minutes.

Tagged, as it applies to the inoperable charging pump, means tagged in accordance with current Southern California Edison procedures for tagging of equipment which must not be operated.

REFERENCE: (1) Final Engineering Report and Safety Analysis, Paragraph 3.6.

3.3 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEMS

3.3.1 OPERATING STATUS

APPLICABILITY: Applies to the operating status of the Safety Injection and Containment Spray Systems.

OBJECTIVE: To define those conditions necessary to ensure availability of the Safety Injection and Containment Spray Systems.

SPECIFICATION: A. The reactor shall not be made or maintained critical unless the following conditions are met. In addition, the reactor coolant system temperature shall not be increased above 200°F unless the containment spray system, the refueling water storage tank and the associated valves and interlocks are operable.

(1) Safety Injection Systems

a. Refueling tank water storage and boron concentration comply with Specification 3.3.3.

b. Two safety injection pumps are OPERABLE.

c. Two feed water pumps are OPERABLE.

d. Two recirculation pumps are OPERABLE, except as indicated in item D below.

e. The recirculation heat exchanger is OPERABLE.

f. Two charging pumps are OPERABLE.

g. Two component cooling water pumps are OPERABLE.

h. Two saltwater cooling pumps are OPERABLE. The reactor may be maintained critical with one saltwater cooling pump provided the auxiliary saltwater cooling pump or two screen wash pumps are available as backup. Return the inoperable pump to operable status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 30 hours. The backup pump(s) shall be demonstrated operable by test within 1 hour of declaring the saltwater cooling pump inoperable.

i. A minimum of 5400 pounds of anhydrous trisodium phosphate is stored in the containment sump in racks provided.

(2) Containment Spray System

a. Two refueling water pumps are OPERABLE.

b. Two hydrazine additive pumps are OPERABLE.

c. Hydrazine tank level and hydrazine concentration comply with Specification 3.3.4.

(3) Valves and interlocks associated with each of the above systems are OPERABLE.

(4) Effective leakage from the recirculation loop outside the containment shall be less than 625 cc/hr as calculated from the following formula.

Effective Leakage = $a_1 \times L_1 + a_2 \times L_2 + a_3 \times L_3$

where,

L_1 = pump and valve leakage which drains to auxiliary building sump

L_2 = valve leakage in auxiliary building or doghouse

L_3 = valve leakage outside

a_1 = iodine release factor for leakage in auxiliary building sump

a_2 = iodine release factor for leakage in auxiliary building or doghouse

a_3 = iodine release factor for leakage outside the auxiliary building or doghouse

If effective leakage from the recirculating loop outside the containment exceeds 625 cc/hr, make necessary repairs to limit leakage to 625 cc/hr. within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

B. During critical operation or when the reactor coolant system temperature is above 200°F, as appropriate per Item A above, maintenance shall be allowed on any one of the following items at any one time:

(1) One motor-operated valve at a time (MOV 1100B or 1100D) in the recirculation loop upstream of the charging pump suction header for a period of time not longer than 72 consecutive hours.

(2) One refueling water pump and/or its associated discharge valve at a time, for a period not longer than 72 consecutive hours.

(3) One hydrazine pump and/or its associated discharge valve (SV600 or 601) at a time, for a period of time not longer than 72 consecutive hours.

(4) One charging pump for a period of time not longer than 72 consecutive hours.

(5) One of the two required component cooling water pumps for a period of time not longer than 72 consecutive hours.

(6) One of the two saltwater cooling pumps with the auxiliary saltwater cooling pump or screen wash pumps available as backup for a period of time not longer than 72 consecutive hours. The backup pump(s) shall be demonstrated operable by test within 1 hour of declaring the saltwater pump inoperable.

C. Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to demonstrate availability.

D. In the event of a failure of a recirculating pump, plant operation may continue provided operability of the remaining pump and its associated motive and control power are satisfactorily demonstrated on a daily basis, including verification that the containment spray bypass valves (CV517 and 518) are closed.

BASIS: The requirements of Specification A assure that before the reactor can be made critical, or before the reactor coolant system heatup is initiated, adequate engineered safeguards are OPERABLE. The limit of 625 cc/hr for the recirculation loop leakage ensures that the combined 0-2 hr EAB thyroid dose due to recirculating loop leakage and containment leakage will not exceed the limits of 10 CFR 100. The formula for determining the leakage incorporates consideration of the significance of leakage in different plant areas. The iodine release factor adjusts actual pump or valve leakage to account for the fraction of the iodine in the leakage which would actually be released to the atmosphere. The iodine release factors in the auxiliary building sump, the auxiliary building or doghouse, and outside are 0.05, 0.5, and 1.0, respectively.

When the reactor is critical or the reactor coolant system temperature is above 200°F, maintenance is allowed per Specifications B and C providing requirements in

Specification C are met which assure OPERABILITY of the redundant component. The specified maintenance times are a maximum, and maintenance work will proceed with diligence to return the equipment to an operable condition as promptly as possible. OPERABILITY of the specified components shall be based on the results of Specification No. 4.2.

The allowable maintenance periods are based upon the repair of certain specific items. Based on the demonstration that equipment redundant to that removed from service is OPERABLE, it is reasonable to maintain the reactor at power over this short period of time.

In the unlikely event that the need for safety injection should occur:

-- functioning of one train will protect the core.(1)(2)(3)(4) Containment sprays alone, however, will maintain containment pressure under design pressure.(5)

-- functioning of one of the two hydrazine additive pumps and associated discharge valve will effect introduction of hydrazine into containment spray water. This provides for absorption of airborne fission products and reduction of the thyroid doses associated with the maximum hypothetical accident to within 10 CFR 100 limits.

-- dissolution of 5400 pounds of anhydrous trisodium phosphate stored in the sump will ensure that the pH of the water in the sump will be greater than 7 within four (4) hours, so as to prevent chloride stress corrosion cracking of systems and components exposed to the circulating sump water.

In the event of inoperability of a recirculation pump, plant operation may continue since either pump is sufficient and a daily OPERABILITY demonstration of the remaining pump and its associated motive and control power provides assurance that it will be OPERABLE if required.

<u>REFERENCES</u>:

(1) Final Engineering Report and Safety Analysis, Paragraph 10.1.

(2) Final Engineering Report and Safety Analysis, Paragraph 5.1.

(3) "San Onofre Nuclear Generating Station," report forwarded by letter dated December 29, 1971, from Jack B. Moore to Director, Division of Reactor Licensing, USAEC, subject: Emergency Core Cooling System Performance, San Onofre Nuclear Generating Station, Unit 1.

(4) USAEC Safety Evaluation of ECCS Performance Analysis for San Onofre Unit 1, forwarded by letter dated March 6, 1974, from Mr. Donald J. Skovholt to Mr. Jack B. Moore.

(5) Supplement No. 1 to the Final Engineering Report and Safety Analysis, Section 5, Question 3c.

3.3.2 SHUTDOWN STATUS

APPLICABILITY: Applies to piping connections between the feedwater condensate system and the reactor coolant system.

OBJECTIVE: To preclude injection of feedwater condensate into the reactor coolant system when the reactor is shut down and to preclude the potential for overpressurization when water solid.

SPECIFICATION: A. When reactor fuel assemblies are in the vessel and the reactor coolant pressure is less than 500 psig, two "positive barriers" shall be provided between the feedwater condensate system and the piping connections to the reactor coolant system. Additionally, when the reactor coolant system is water solid at less than 500 psig, two positive barriers shall be provided between the safety injection system and piping connections to the reactor coolant system. A "positive barrier" is defined as follows:

(1) Motor Operated Valves

When closed and tagged with supply breakers open, except that power may be restored during no-flow tests of the safety injection system (Specification No. 4.2).

(2) Pneumatic/Hydraulic Operated Valves

When closed and the condition tagged with the respective hydraulic block valve closed except that they may be opened during no-flow tests of the safety injection system (Specification No. 4.2).

(3) Manually Operated Valves

When closed and condition tagged.

(4) Feedwater Pump (Overpressurization Protection Only)

When shutdown with the breaker in the racked out condition.

BASIS: Under normal conditions, system operational interlocks assure that injection of feedwater condensation to the reactor by the safety injection system cannot occur.[1] These interlocks include:

1. Actuation of the safety injection relay which de-energizes the condensate and heater drain pumps and closes the flow path for condensate, thereby preventing injection of feedwater into the coolant system.

2. Interlocks between the condensate isolation valves at the feedwater pump suction and the safety injection header isolation valves at the pump discharge which prevent the opening of the one valve unless the other is closed.

Below 500 psig the Safety Injection System may be removed from service. Below 400 psig the feedwater system may be removed from service. During these low pressure shutdown reactor coolant system conditions, the interlocks may be overridden for maintenance and/or tests of components of these systems. However, it is still necessary to prevent intrusion of feedwater condensate or safety injection water into the reactor coolant system. Injection of feedwater has the potential to dilute the system and create a potential for a reactivity excursion. Injection of either safety injection water or feedwater, especially during water solid operations, creates the potential for pressurizing above limits established by 10 CFR 50 Appendix G and as reflected in Technical Specification 3.1.3.

The "two positive barriers" required by this specification provide protection of the Reactor Coolant System against boron dilution and overpressurization when in the low pressure and low temperature conditions. Two positive barriers are provided in each potential path between the Feedwater Condensate System, Safety Injection System and the RCS. During period of no-flow testing, an exception is provided on two of the positive barriers to allow the components involved in the test to perform their test functions while the remaining positive barriers (nos. 3 and 4) remain in effect.

Tagged, as used above, means tagged in accordance with current Southern California Edison Company procedures for tagging of equipment which must not be operated.

<u>REFERENCE</u>: (1) Final Engineering Report and Safety Analysis, Paragraph 5.1.

3.3.3 MINIMUM WATER VOLUME AND BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANK

APPLICABILITY: Applies to the inventory of borated refueling water.

OBJECTIVE: To ensure immediate availability of safety injection and containment spray water of required quality.

SPECIFICATION: When the Safety Injection System or the Containment Spray System is required to be operable, the refueling water tank shall be filled to at least elevation 50 feet with water having a boron concentration of not less than 3750 ppm and not greater than 4300 ppm.

BASIS: The refueling water storage tank serves two purposes; namely:

(1) As a reservoir of borated water for accident mitigation purposes,

(2) As a reservoir of borated water for flooding the refueling cavity during refueling.

Approximately 220,000 gallons of borated water is required to provide adequate post-accident core cooling and containment spray to maintain calculated post-accident doses below the limits of 10 CFR 100(1). The refueling water storage tank filled to elevation 50 feet represents in excess of 240,000 gallons.

A boron concentration of 3750 ppm is required to meet the requirements of postulated steam line break.(2) A maximum boron concentration of 4300 ppm ensures that the post-accident containment sump water is maintained at a pH between 7.0 and 7.5(3).

The refueling tank capacity of 240,000 gallons is based on refueling volume requirements.

Sustained temperatures below 32°F do not occur at San Onofre. At 32°F, boric acid is soluble up to approximately 4650 ppm boron. Therefore, no special provisions for temperature control to avoid either freezing or boron precipitation are necessary.

REFERENCES:

(1) Enclosure 1 "Post-Accident Pressure Reanalysis, San Onofre Unit 1" to letter dated January 19, 1977 in Docket No. 50-206.

(2) "Steam Line Break Accident Reanalysis, San Onofre Nuclear Generating Station, Unit 1, October 1976" submitted by letter dated December 30, 1976 in Docket No. 50-206.

(3) Additional information, San Onofre, Unit 1 submitted by letter dated March 24, 1977 in Docket No. 50-206.

3.3.4 MINIMUM SOLUTION VOLUME HYDRAZINE CONCENTRATION IN THE HYDRAZINE TANK

APPLICABILITY: Applies to the inventory of spray additive solution.

OBJECTIVE: To insure availability of containment spray additive solution of required quality.

SPECIFICATION: When the reactor coolant system temperature is above 200°F, the hydrazine tank shall contain not less than 150 gallons of aqueous solution having a concentration of not less than 21 wt% N_2H_4.

BASIS: The hydrazine tank serves the purpose of acting as a reservoir of aqueous hydrazine solution for post-accident iodine removal.

100 gallons of N_2H_4 solution are required to reduce airborne iodine concentration in the event of a loss of coolant accident. By adding a 50% margin to this figure to enshire that NPSH to the spray addition pumps is maintained at all times, a total of 150 gallons is required. This amount fulfills requirements for safety injection operations.

3.3.5 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

APPLICABILITY: Applies to the operational status of the primary coolant system pressure isolation valves during MODES 1, 2, 3.

OBJECTIVE: To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

SPECIFICATION: 1. The integrity of all pressure isolation valves listed in Table 3.3.5-1 shall be demonstrated by Specification 4.2.2. Valve leakage shall not exceed the amounts indicated in Table 3.3.5-1.

ACTION: 2. If Specification 1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN condition within 24 hours.

TABLE 3.3.5-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[a] Allowable Leakage
Safety Injection		
Loop A, cold leg	867a	≤5.0 GPM
Loop B, cold leg	867b	≤5.0 GPM
Loop C, cold leg	867c	≤5.0 GPM

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.

3.4 TURBINE CYCLE

3.4.1 OPERATING STATUS

APPLICABILITY: Applies to the operating status of turbine cycle in MODES 1, 2, and 3.

OBJECTIVE: To define conditions of the turbine cycle necessary to ensure the capability to remove decay heat from the core.

SPECIFICATION:

(A) A minimum turbine cycle steam-relieving capability of 5,706,000 lb/hr (except for testing of the main steam safety valves).

(B) The auxiliary feedwater pumps OPERABLE as specified in 3.4.3.

(C) The auxiliary feedwater storage tank OPERABLE as specified in 3.4.4.

(D) System piping and valves directly associated with the above components OPERABLE.

BASIS: A reactor shutdown from power requires subsequent removal of core decay heat. In the event of a reactor trip from high power levels, immediate decay heat removal requirements are satisfied by the steam bypass to the condensers, supplemented by release to the atmosphere. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser or steam dump to atmosphere as feed water in the steam generator is converted to steam by heat absorption. In the event of a planned shutdown, steam release to atmosphere is not required. In either case, feedwater to the steam generators is normally supplied by operation of the turbine cycle feedwater pumps.

The power operated relief valves and the main steam safety valves have a total combined relief capability of 7,629,432 lb/hr. A capability of 5,706,000 lb/hr is required to maintain the pressure in turbine cycle components within ASME Code allowable values in the event of full load rejection. Therefore the limiting conditions for operation can be met with less than the full number of valves in service.

Two auxiliary feedwater pumps, one steam driven and one electric driven, together with the steam system relief valves, provide core decay heat removal capability in the event of a sustained loss of off-site power. The electric driven pump is capable of being powered from the diesel. Either auxiliary feedwater pump has the capability to satisfy decay heat removal requirements from the core.(1)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions (including cooldown) for 32 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

REFERENCES: (1) Supplement No. 1 to the Final Engineering Report and Safety Analysis, Section 3, Question 6.

3.4.2 MAXIMUM SECONDARY COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum radioiodine activity in the secondary coolant of the steam generators any time the primary coolant system temperature exceeds 200°F.

OBJECTIVE: To limit the consequences of an accidental release of secondary coolant to the environment.

SPECIFICATION: A. The specific activity of radioiodine in the secondary coolant shall be limited to 0.1 μCi/gm DOSE EQUIVALENT I-131.

ACTION: B. With the specific activity of the secondary coolant in excess of 0.1 μCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in cold shutdown within 36 hours.

BASIS: 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. The restriction of 0.1 μCi/gram DOSE EQUIVALENT I-131 in the secondary system limits the 2 hour thyroid exposure dose to well within the guidelines of 10 CFR Part 100 at the site boundary under these accident conditions. This thyroid dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analysis.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.4.3 AUXILIARY FEEDWATER SYSTEM

APPLICABILITY: Applies to the motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat.

SPECIFICATION: A. Both steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE as follows:

1. One auxiliary feedwater pump capable of being powered from an emergency electrical power source, and

2. One auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

ACTION: B. With one auxiliary feedwater pump inoperable, restore both auxiliary feedwater pumps (one capable of being powered from an emergency electrical power source and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours'.

BASIS: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F for normal operating conditions in the event of a total loss of offsite power.

REFERENCE: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

3.4.4 AUXILIARY FEEDWATER STORAGE TANK

APPLICABILITY: Applies to the auxiliary feedwater storage tank for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat.

SPECIFICATION: A. The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a contained water volume of at least 150,000 gallons of water.

ACTION: B. With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions (including cooldown) for 32 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3.5 INSTRUMENTATION AND CONTROL

3.5.1 REACTOR TRIP SYSTEM INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.1-1.

OBJECTIVE: To delineate the conditions of the Plant instrumentation and safety circuits necessary to ensure reactor safety.

SPECIFICATION: As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.5.1-1 shall be OPERABLE.

ACTION: As shown in Table 3.5.1-1.

BASIS: During plant operations, the complete instrumentation systems will normally be in service.(1) Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits.(2) Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design.(1)(3) This Standard outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

REFERENCES:

(1) Final Engineering Report and Safety Analysis, Section 6.

(2) Final Engineering Report and Safety Analysis, Section 6.2.

(3) NIS Safety Review Report, April 1988

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	7
2. Power Range, Neutron Flux, Overpower Trip	4	2	3	1, 2	2#
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	4	1**	4	1, 2	28#
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
6. NIS Coincidentor Logic	2	1	2	1, 2	29
				3*, 4*, 5*	7
7. Pressurizer Variable Low Pressure	3	2	2	1####	6#
8. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
9. Pressurizer High Level	3	2	2	1	6#

TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
11. Steam/Feedwater Flow Mismatch	3	2	2	1,2	6#
12. Turbine Trip-Low Fluid Oil Pressure	3	2	2	1####	6#

TABLE 3.5.1-1 (Continued)

TABLE NOTATION

*	With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.
**	A "TRIP" will stop all rod withdrawal.
#	The provisions of Specification 3.0.4 are not applicable.
##	Below the Source Range High Voltage Cutoff Setpoint.
###	Below the P-7 (At Power Reactor Trip Defeat) Setpoint.
####	Above the P-7 (At Power Reactor Trip Defeat) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are met:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be returned to the untripped condition for up to 2 hours for surveillance testing of other channels per Specification 4.1.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint.

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, within one hour reduce THERMAL POWER such that T_{ave} is less than or equal to 551.5°F, and place the rod control system in manual mode.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, be in at least HOT STANDBY within 6 hours; however, one channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

3.5.2 CONTROL ROD INSERTION LIMITS

APPLICABILITY: MODES 1 and 2

OBJECTIVE: This specification defines the insertion limits for the control rods in order to ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

SPECIFICATION:

A. Except during low power physics tests or surveillance testing pursuant to Specification 4.1.1.G, the Shutdown Groups and Control Group 1 shall be fully withdrawn, and the position of Control Group 2 shall be at or above the 21-step uncertainty limit shown in Figure 3.5.2.1.

B. The energy weighted average of the positions of Control Group 2 shall be at least 90% (i.e. > Step 288) withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all Control Group 2 positions during the core cycle.

ACTION:

A. With the control groups inserted beyond the above insertion limits either:

   1. Restore the control groups to within the limits within 2 hours, or

   2. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

   3. Be in at least HOT STANDBY within 6 hours.

B. With a single dropped rod from a shutdown group or control group, the provisions of Action A are not applicable, and retrieval shall be performed without increasing THERMAL POWER beyond the THERMAL POWER level prior to dropping the rod. An evaluation of the effect of the dropped rod shall be made to establish permissible THERMAL POWER levels for continued operation. If retrieval is not successful within 3 hours from the time the rod was dropped, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% Δ k/k.

BASIS: During STARTUP and POWER OPERATION, the shutdown groups and control group 1 are fully withdrawn and control of the reactor is maintained by control group 2. The control group insertion limits are set in consideration of maximum specific power, shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.7 kW/ft, 1.57 and 2.89, respectively. At partial power, the $F^N_{\Delta H}$ maximum values (limits) increase according to the following equation,
   $F^N_{\Delta H}(P) = 1.57\,[1 + 0.2\,(1-P)]$, where P is the fraction of RATED THERMAL POWER. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% $\Delta\rho$ at BOL, 1.9% $\Delta\rho$ at EOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available SHUTDOWN MARGIN is greater than the above values.

3. The worst case ejected rod accident (8) covering HFP-BOL, HZP-BOL, HFP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 220 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

   Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% RATED THERMAL POWER. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

   It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available SHUTDOWN MARGIN.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

The presence of a dropped rod leads to abnormal power distribution in the core. The location of the rod and its worth in reactivity determines its effect on the temperatures of nearby fuel. Under certain conditions, continued operation could result in fuel damage, and it is the intent of ACTION B to avoid such damage.

<u>References</u>:

(1) Final Engineering Report and Safety Analysis, revised July 28, 1970.

(2) Amendment No. 18 to Docket No. 50-206.

(3) Amendment No. 22 to Docket No. 50-206.

(4) Amendment No. 23 to Docket No. 90-206.

(5) Description and Safety Analysis, Proposed Change No. 7, dated October 22, 1971.

(6) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 4, WCAP 8131, May, 1973.

(7) Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

(8) An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January, 1975.

CONTROL GROUP INSERTION LIMITS

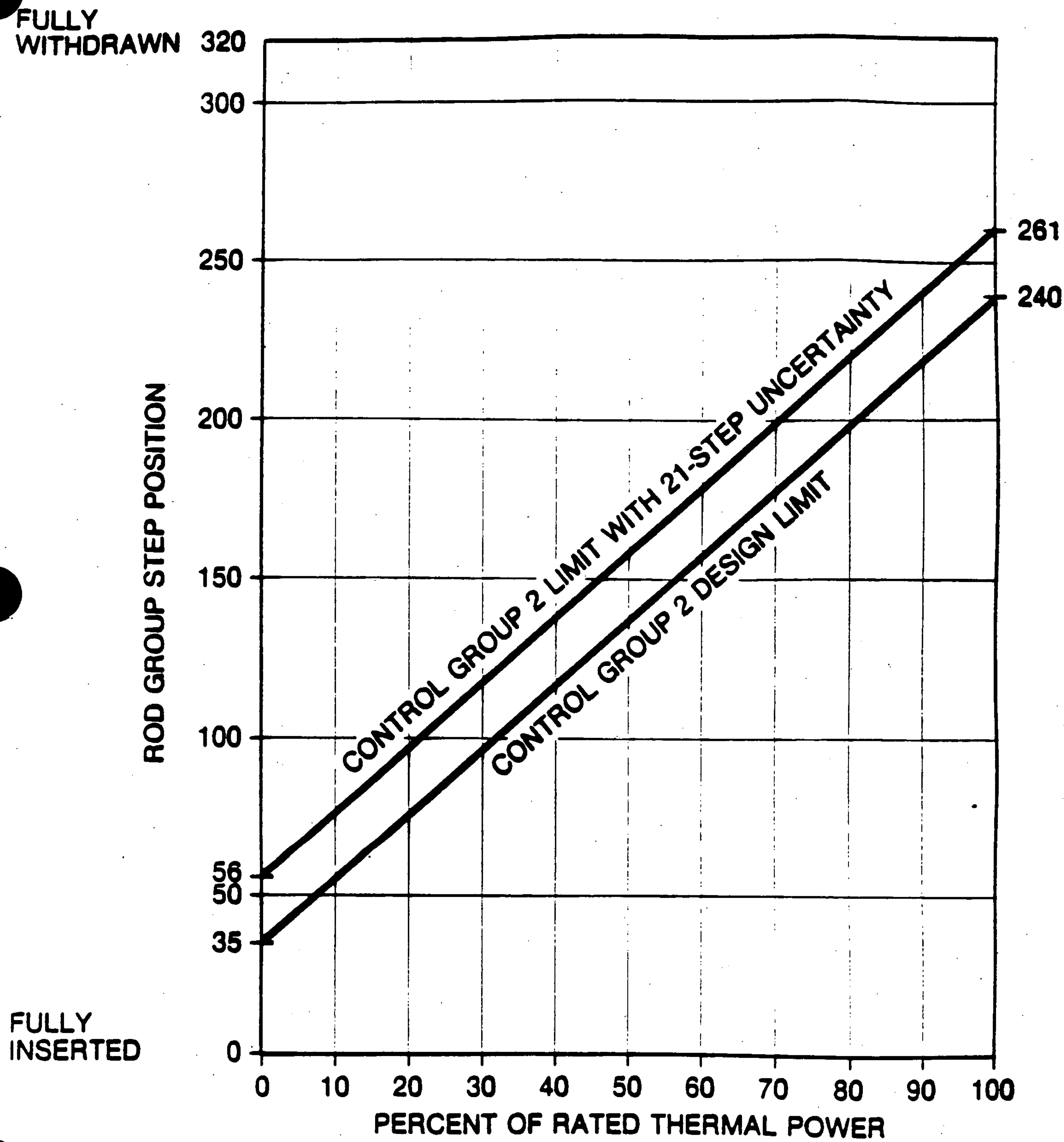

FIGURE 3.5.2.1

3.5.3 CONTROL AND SHUTDOWN ROD MISALIGNMENT

APPLICABILITY: Applies to the number of steps an individual control or shutdown rod may be misaligned from its group position during STARTUP and POWER OPERATION.

OBJECTIVE: To ensure that the effects of rod misalignment from the group position do not exceed the core design margins.

SPECIFICATION: A. During STARTUP and POWER OPERATION, all rods shall be OPERABLE and maintained within ±35 steps (indicated by the Analog Detection System) of their step counter indicated bank position (indicated by the Digital Detection System), except during low power physics tests.

ACTION: B. With Specification A, above, not met, the following specifications are applicable.

1. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN BASIS of Specification 3.5.2 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

2. With more than one rod inoperable or misaligned from the step counter indicated position by more than ±35 steps (indicated by the Analog Detection System), be in HOT STANDBY within 6 hours.

3. With one rod inoperable due to causes other than addressed by Specification B.1, above, or misaligned from its step counter indicated height by more than ±35 steps (indicated by the Analog Detection System), POWER OPERATION may continue provided that within one hour either:

   a. The rod is restored to OPERABLE status within the above alignment requirements, or

   b. The rod is declared inoperable and the SHUTDOWN MARGIN BASIS of Specification 3.5.2 is satisfied. POWER OPERATION may then continue provided that:

      1) A reevaluation of each accident analysis of Table 3.5.3-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

2) The SHUTDOWN MARGIN BASIS of Specification 3.5.2 is determined at least once per 12 hours.

3) A power distribution map is obtained from the movable incore detectors and F_Q (Z) and F^N_H are verified to be within their limits within 72 hours.

4) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

5) The remainder of the rods in the group with the inoperable rod are aligned to within ±35 steps of the inoperable rod within one hour while maintaining the rod insertion limits of Figure 3.5.2.1.

BASIS: The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential affects of rod misalignment on associated accident analyses.

The misalignment allowance of Specification B, assures core performance within allowed design margins including allowance for the inaccuracy of the position signals.

TABLE 3.5.3-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks In Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

3.5.4 ROD POSITION INDICATING SYSTEM

APPLICABILITY: Applies to the operating status of the Rod Position Indicating System.

OBJECTIVE: To ensure the ability to accurately detect the position of control and shutdown rods.

SPECIFICATION:

A. During STARTUP and POWER OPERATION the Analog Detection System and the Digital Detection System shall be OPERABLE and capable of determining the control rod positions within ±21 steps.

B. The Analog Detection System remains OPERABLE if the specified rod position indications can be obtained from direct LVDT voltage measurements.

ACTION:

C. With specifications A and B, above, not met, the following specifications are applicable.

1. With a maximum of one rod position indicator (Analog Detection System) per bank inoperable either:

   a. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors within 8 hours, and at least once per 8 hours thereafter and immediately after any motion of the non-indicating rod which exceeds 56 steps in one direction since the last determination of the rod's position, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

2. With a maximum of one step counter indicator (Digital Detection System) per bank inoperable either:

   a. Verify that all rod position indicators (Analog Detection System) for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 35 steps of each other at least once per 8 hours, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. With more than one rod position indicator (Analog Detection System) per bank inoperable or more than one step counter indicator (Digital Detection System) per bank inoperable be in HOT STANDBY within 6 hours.

<u>BASIS</u>:

Control rod position and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per shift with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The indicator inoperability allowance of Specification C requires indirect measurement of rod position or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation.

3.5.5 CONTAINMENT ISOLATION INSTRUMENTATION

APPLICABILITY: Applies to instrumentation which actuates the containment sphere isolation valves, containment sphere purge and exhaust valves, and containment sphere instrumentation vent header valves.

OBJECTIVE: To ensure reliability of the containment sphere isolation provisions.

SPECIFICATION: The instrumentation channels shown in Table 3.5.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5.5-2.

ACTION:

A. With an instrumentation channel trip setpoint less conservative than the Allowable Values column of Table 3.5.5-2, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.5.5-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

B. With an instrumentation channel inoperable, take the ACTION shown in Table 3.5.5-1.

BASIS: The OPERABILITY of these instrumentation systems ensure that 1) the associated action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 3.5.5-1

CONTAINMENT ISOLATION INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Containment Isolation (Valves listed In Table 3.6.2-1)					
a) Manual	2	1	2	1, 2, 3, 4	11
b) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	9
c) Sequencer Subchannels	2/sequencer	2/sequencer	2/sequencer	1, 2, 3, 4	8
d) Safety Injection					
1) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	9*
2) Pressurizer Pressure-Low	3/train	2/train	2/train	1, 2, 3	9*
Purge and Exhaust Isolation (POV-9, POV-10, CV-10, CV-40, CV-116)					
a) Manual	1	1	1	1, 2, 3, 4	10
b) Containment Radioactivity-High	1	1	1	1, 2, 3, 4	10

TABLE 3.5.5-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

* The provisions of Specification 3.0.4 are not applicable.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1.4.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

ACTION 10 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves (POV-9 & POV-10) are maintained closed.

ACTION 11 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.5.5-2

CONTAINMENT ISOLATION INSTRUMENTATION TRIP SET POINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
Containment Isolation		
a) Manual	Not Applicable	Not Applicable
b) Containment Pressure-High	$\leq$ 1.4 psig	$\leq$ 2.0 psig
c) Sequencer Subchannels	Not Applicable	Not Applicable
d) Safety Injection		
1) Containment Pressure-High	$\leq$ 1.4 psig	$\leq$ 2.0 psig
2) Pressurizer Pressure-Low	$\geq$ 1735 psig	$\geq$ 1675 psig
Purge and Exhaust Isolation		
a) Manual	Not Applicable	Not Applicable
b) Containment Radioactivity-High	$\leq$ 2 x Background	$\leq$ 2.5 x Background

3.5.6 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3.

OBJECTIVE: To ensure reliability of the accident monitoring instrumentation.

SPECIFICATION: The accident monitoring instrumentation channels shown in Table 3.5.6-1 shall be OPERABLE.

ACTION:

A. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

B. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

C. The provisions of Specification 3.0.4 are not applicable.

BASIS: The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and 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," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

(2) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 3.5.6-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
Pressurizer Water Level	3	2
Auxiliary Feedwater Flow Indication*	2/steam generator	1/steam generator
Reactor Coolant System Subcooling Margin Monitor	2	1
PORV Position Indicator (Limit Switch)	1/valve	1/valve
PORV Block Valve Position Indicator (Limit Switch)	1/valve	1/valve
Safety Valve Position Indicator (Limit Switch)	1/valve	1/valve
Containment Pressure (Wide Range)	2	1
Steam Generator Water Level (Narrow Range)	1/steam generator	1/steam generator
Refueling Water Storage Tank Level	1	1
Containment Sump Water Level (Narrow Range)**	2	1
Containment Water Level (Wide Range)	2	1
Neutron Flux (Wide Range)	2	1

* Auxiliary feedwater flow indication for each steam generator to provided by one channel of steam generator level (Wide Range) and one channel of auxiliary feedwater flow rate. These comprise the two channels of auxiliary feedwater flow indication for each steam generator.

** Operation may continue up to 30 days with one less than the total number of channels OPERABLE.

3.5.7 AUXILIARY FEEDWATER INSTRUMENTATION

APPLICABILITY: Applies to automatic initiation of the auxiliary feedwater pumps.

OBJECTIVE: To ensure reliability of automatic initiation of the auxiliary feedwater pumps.

SPECIFICATIONS: A. The instrumentation channels shown in Table 3.5.7-1 shall be OPERABLE with their trip setpoints set consistent with the Trip Setpoint column of Table 3.5.7-2.

ACTION: B. With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5.7-2, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.5.7-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

C. With one instrumentation channel inoperable, take the action shown in Table 3.5.7-1.

BASIS: The OPERABILITY of the auxiliary feedwater instrumentation ensures that 1) the associated action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of this instrumentation is required to provide the overall reliability, redundancy, and diversity assumed available for the protection and mitigation of accident and transient conditions. The operation of this instrumentation is consistent with the assumptions used in the accident analyses.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 3.5.7-1

AUXILIARY FEEDWATER INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
a. Manual Actuation	2	1	2	1, 2, 3	12
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Steam Generator Water Level-Low					
i. Start Motor Driven Pump	3	2	2	1, 2, 3	14, 15
ii. Start Turbine-Driven Pump	3	2	2	1, 2, 3	14, 15

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 13 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hour for surveillance testing per Specification 4.1.8 provided the other channel to OPERABLE.

ACTION 14 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL TEST provided the inoperable channel to placed in the tripped condition within 1 hour, or an operator shall assume continuous surveillance and actuate manual initiation of auxiliary feedwater, if necessary.

ACTION 15 - With more than one channel inoperable, an operator shall assume continuous surveillance and actuate manual initiation of auxiliary feedwater, if necessary. Restore the system to no more than one channel inoperable within 7 days, or be in HOT STANDBY within the following 6 hours and in HOT SHUTDOWN within the following 6 hours.

TABLE 3.5.7-2

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
a. Manual Actuation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Steam Generator Water Level-Low	$\geq$ 5% of narrow range instrument span each steam generator	$\geq$ 0% of narrow range instrument span each steam generator

3.5.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

APPLICABILITY: During releases via this pathway.

OBJECTIVE: Monitor and control radioactive liquid effluent releases.

SPECIFICATION:

A. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5.8.1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.15.1 are not exceeded.

B. ACTION:

1. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.15.1 are met, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

2. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.5.8.1. If the inoperable instruments remain inoperable for greater than 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments are calculated in accordance with methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE 3.5.8.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactive Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (R-1218)	(1)	16
b. Steam Generator Blowdown [(a)] Effluent Line (R-1216)	(1)	17
c. Turbine Building Sumps Effluent Line (Reheater Pit Sump) (R-2100)	(1)	18
d. Yard Sump (R-2101)	(1)	18
e. Component Cooling Water System [(b)] (R-1217)	(1)	19
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (FE-16, FE-18)	(1)	20
b. Circulating Water Outfall*		
c. Steam Generator Blowdown Effluent* Line		

* Pump status, valve turns or calculations are utilized to estimate flow.

(a) Secondary coolant samples and activity analysis performed in accordance with T.S. 4.1, Table 4.1.2.

(b) Closed loop system. Monitor closes vent valve to isolate surge tank.

TABLE 3.5.8.1
(Continued)

TABLE NOTATION

ACTION 16 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two separate samples which can be taken by a single person are analyzed in accordance with Specification 4.5.1.A., and;

2. At least two technically qualified persons verify the release rate calculations and discharge valving.

ACTION 17 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml;

1. At least once per 12 hours when the specific activity of the secondary coolant is > 0.01 μCi/gram DOSE EQUIVALENT I-131.

2. At least once per 24 hours when the specific activity of the secondary coolant is ≤ 0.01 μCi/gram DOSE EQUIVALENT I-131.

ACTION 18 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

ACTION 19 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, determine if there is leakage from the Component Cooling Water System to the Salt Water Cooling System. If leakage exists sample the Component Cooling Water System to estimate the activity being released via the Salt Water Cooling System at least once per 24 hours for gross activity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

ACTION 20 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in-situ may be used to estimate flow.

3.5.9 RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

APPLICABILITY: During releases via this pathway.

OBJECTIVE: Monitor and control radioactive gaseous releases.

SPECIFICATION:

A. The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.5.9.1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.16.1 are not exceeded.

B. ACTION:

1. With a radioactive gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.16.1 are met, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

2. With less than the minimum number of radioactive gaseous process or effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.5.9.1. If the inoperable instruments remain inoperable for greater than 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments are calculated in accordance with methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE 3.5.9.1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Stack Monitoring System[1]		
a. Gross Activity Monitor - Providing Alarm (R-1214 or (R-1219[2], 1220 and 1221))	(1)	21
b. Noble Gas Activity Monitor (R1219[2] or 1212[3] or 1254*)	(1)	22
c. Iodine Sampler Cartridge (R1221 or 1254*)	(1)	23
d. Particulate Sampler Filter (R-1211 or 1220 or 1254*)	(1)	23
e. Stack Fan Flow Indication (R-1254*)	(1)	24
f. Sampler Flow Rate Measuring Device	(1)	24

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1. Includes the following subsystems:
   a. Spent Fuel Building Ventilation, Auxiliary Building Ventilation, and Waste Gas Treatment (CVI) Building Ventilation system.
   b. Containment Monitoring System.
   c. Air Ejector System.

2. Provides for auto-termination of release from the Waste Gas Holdup System.

3. Provides for auto-termination of containment purge.

* Does not perform any isolation function. Does not provide control room alarm annunciation when the instrument controls are set in the "not operate" mode.

TABLE 3.5.9.1
(Continued)

TABLE NOTATION

ACTION 21 With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement and instrument 1b inoperable the contents of a waste gas decay tank may be released to the environment provided that prior to initiating the release:

1. At least two separate samples which can be taken by a single person of the tank's contents are analyzed; and

2. At least two technically qualified persons verify the release rate calculations and discharge valve lineup.

All other effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 22 With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement and instrument 1a inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 23 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.6.1.1.

ACTION 24 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 8 hours.

3.5.10 RADIATION MONITORING INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.10-1.

OBJECTIVE: To ensure reliability of the radiation monitoring instrumentation.

SPECIFICATION: The radiation monitoring instrumentation shown in Table 3.5.10-1 shall be OPERABLE with their alarm setpoints within the specified limits.

ACTION:

A. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.5.10-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

B. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.5.10-1.

C. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: The OPERABILITY of the radiation monitoring channels ensures that (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm is initiated when the radiation level trip setpoint is exceeded.

REFERENCES:

(1) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 3.5.10-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Control Room Area (R-1231)	1	All	1 mR/hr	10^{-2} - 10^{2} mR/hr	25
b. Spent Fuel Pool Area (R-1236)	1	*	25 mR/hr	10^{-2} - 10^{2} mR/hr	26
c. Containment Radiation Monitor-High Range (R-1255, R-1257)	2	1, 2, 3 & 4	10 R/hr	1-10^{8} R/hr	27
2. PROCESS MONITORS					
a. Wide Range Gas Monitor (R-1254)	1	1, 2, 3 & 4	per ODCM	10^{-7}-10^{5} μCi/cc	27
b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B)	1/steamline	1, 2, 3 & 4	1mR/hr (low) 1 R/hr (high)	10^{-1}-10^{4} mR/hr 10^{-1}-10^{4} R hr	27

* With fuel in the spent fuel pool or building

TABLE 3.5.10-1 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement; within 1 hour: (1) either initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation, or (2) initiate the preplanned alternate method of monitoring and alarming the area radiation.

ACTION 26 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 27 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement of Table 3.5.10-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

(1) either restore the inoperable channel(s) to OPERABLE status within 7 days of initiating the preplanned alternate method, or

(2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, within 14 days following initiating the preplanned alternate method, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3.6 CONTAINMENT SYSTEMS

3.6.1 CONTAINMENT SPHERE

APPLICABILITY: Applies to the operating status of the containment sphere.

OBJECTIVE: To ensure containment integrity.

SPECIFICATION: A. Leakage

The reactor coolant system temperature shall not be increased above 200°F if the containment leakage exceeds the maximum acceptable values specified in Specification 4.3.

B. Access to Containment

(1) Containment integrity shall not be violated unless the reactor coolant system is below 500 psig and a SHUTDOWN MARGIN greater than 1% $\Delta k/k$ with all rods inserted is maintained for the most reactive temperature.

(2) Containment integrity shall not be violated when the reactor coolant system is open to the containment atmosphere unless a SHUTDOWN MARGIN greater than 5% $\Delta k/k$ is maintained with all control rods inserted.

(3) Positive reactivity changes shall not be made by rod drive motion whenever the containment integrity is not intact. Boron dilution (resulting in positive reactivity) may be made when the containment integrity is not intact if a SHUTDOWN MARGIN greater than 5% $\Delta k/k$ is maintained.

C. Internal Pressure

The reactor shall not be made critical, nor be allowed to remain critical, if the containment sphere internal pressure exceeds 0.4 psig, or the internal vacuum 2.0 psig.

BASIS: The basis for the SHUTDOWN MARGINS and 500 psig pressure are as follows:

$\Delta k/k$	Event	Basis for Adequacy
1% (below 500 psig)	Violation of Containment	Safety injection system disarmed; no credible automatic or operator action could cause return to criticality.

Δk/k	Event	Basis for Adequacy
5%	Open reactor coolant	Provides adequate margin so that maintenance activities can be carried out with the reactor head removed.[1]

Regarding internal pressure limitations, the containment design pressure of 46.4 psig would not be exceeded if the sphere internal pressure before a major loss of coolant accident was no greater than 0.4 psig. The design criteria also allows an internal vacuum not in excess of 2.0 psig. Thus, the specified limiting conditions for internal pressure are consistent with the design basis.[2] Although such design values could be exceeded without damage to the structure, it is considered that the importance of the containment function warrants the specified values.

Opening of the ventilation system backup valves, POV 9A and POV 10A, is not considered a violation of containment integrity during startup conditions provided that their corresponding in-line valves POV 9 and POV 10 are closed.

REFERENCES:

(1) Supplement No. 3 to Final Engineering Report and Safety Analysis, Question No. 2.

(2) Final Engineering Report and Safety Analysis, Paragraph 5.3.

3.6.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To provide assurance that the containment isolation valves listed in Table 3.6.2-1 will function when initiated by appropriate sensors.

SPECIFICAIION: The containment isolation valves specified in Table 3.6.2-1 shall be OPERABLE.

ACTION:

A. With one or more of the isolation valve(s) specified in Table 3.6.2-1 inoperable, for each affected penetration that is provided with two isolation valves and is open maintain at least one valve OPERABLE, and for all affected penetrations with either one or two isolation valves, one of the following Actions shall be taken:

1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

2. Isolate each affected penetration within 4 hours by use of at least one deactivated* power operated valve secured in the isolation position, or

3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

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

B. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with Action A.2 or A.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are taken.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

* Valve may be temporarily activated for valve position verification and testing. While the valve is activated by this note, Action A.1 shall be applied and any system(s) declared inoperable pursuant to Action B shall not be declared OPERABLE.

The isolation valves of the Sphere Purge Air Supply (POV-9) and Air Outlet (POV-10) lines have not been demonstrated capable of closure under the differential pressures generated by a design basis accident. For this reason, containment isolation in these lines shall be maintained. This configuration shall be accomplished by locking closed manual isolation valves CVS-301 and CVS-313 of these lines. These valves shall remain locked closed during MODES 1, 2, 3 and 4 until POV-9 and POV-10 can be demonstrated capable of performing their containment isolation function under post accident conditions.

Temporary activation of a secured closed containment isolation valve permits position indication of certain types of valves and allows maintenance testing of isolation valves.

<u>REFERENCE</u>: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 3.6.2-1

REMOTE MANUAL (RM) OR AUTOMATIC CONTAINMENT ISOLATION VALVE SUMMARY

	DESCRIPTION	INSIDE SPHERE	OUTSIDE SPHERE
1.	Sphere Sump Discharge	CV-102	CV-103
2.	RCS Or Tk Discharge	CV-104	CV-105
3.	RCS Or Tk Vent	CV-106	CV-107
4.	N_2 to RCS Drain Tank and PRT	CV-536	CV-535
5.	ORMS 1211/1212 Sphere Sample Supply	CV-147	SV-1212-9
6.	ORMS 1211/1212 Sphere Sample Return	CV-146	SV-1212-8
7.	A Stm. Gen. Stm. Sample	-	SV-119
8.	B Stm. Gen. Stm. Sample	-	SV-120
9.	C Stm. Gen. Stm. Sample	-	SV-121
10.	A Stm. Gen. Blowdown Sample	-	SV-123
11.	B Stm. Gen. Blowdown Sample	-	SV-122
12.	C Stm. Gen. Blowdown Sample	-	SV-124
13.	Service Water to Sphere	CV-537	CV-115
14.	Service Air to Sphere	Check Valve	SV-125
15.	SI Loop C Vent	SV-702B	SV-702A
16.	SI Loop B Vent	SV-702D	SV-702C
17.	H2 Calibration Gas	SV-3004	SV-2004
18.	RC Loop Sample	(CV-955, CV-956, CV-962) RM	CV-957 SV-3302
19.	Pressurizer Sample	(CV-951, CV-953) RM	CV-992
20.	Sphere Purge Air Supply*	-	POV-9
21.	Sphere Purge Air Outlet*	-	POV-10
22.	Sphere Equalizing/Vent Inst. Air Vent	CV-116 CV-40	CV-10
23.	Primary Makeup to PRT	CV-533	CV-534
24.	Cont. Cooling Out	-	CV-515 RM
25.	Cont. Cooling In	-	CV-516 RM
26.	N2 Supply to PORV	Check Valve	CV-532 RM
27.	Letdown	CV-525 RM	CV-526 RM
28.	Seal Water Return	CV-527 RM	CV-528 RM
29.	RC Loop Sample Return	Check Valve	SV-3303
30.	PRT Gas Sample	CV-948 RM	CV-949

* Manual valves CVS-301 and CVS-313 of the Sphere Purge Air Supply and Air Outlet lines, respectively, shall be locked closed during MODES 1, 2, 3 and 4.

3.6.3 HYDROGEN MONITORS AND HYDROGEN RECOMBINERS

APPLICABILITY: MODES 1 and 2.

OBJECTIVE: To ensure the capability of the hydrogen monitors and hydrogen recombiners to maintain the hydrogen concentration within the containment sphere below its flammable limit during post-LOCA conditions.

SPECIFICATION:

a. Two independent containment hydrogen monitors shall be OPERABLE.

b. Two independent containment hydrogen recombiner systems shall be OPERABLE.

ACTION:

A. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

B. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

C. With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

BASIS: The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that the equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with radiolytic decomposition of water and corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

REFERENCES:

(1) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March, 1971.

(2) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees NUREG-0737 Technical Specifications (Generic Letter No. 83-37)

3.7 AUXILIARY ELECTRICAL SUPPLY

APPLICABILITY: Applies to the availability of electrical power for the operation of the plant auxiliaries.

OBJECTIVE: To define those conditions of electrical power availability necessary (1) to provide for safe reactor operation, (2) to provide for the continuing availability of engineered safeguards, and (3) to ensure that the station can be maintained in the shutdown or refueling condition for extended time periods.

SPECIFICATION: I. In MODES 1, 2, 3 and 4 the following specifications shall apply:

A. As a minimum the following shall be OPERABLE:

1. One Southern California Edison Company and one San Diego Gas & Electric Company high voltage transmission line to the switchyard and two transmission circuits from the switchyard, one immediate and one delayed access, to the onsite safety-related distribution system. This configuration constitutes the two required offsite circuits.

2. Two separate and independent diesel generators each with:

   a. A separate day tank containing a minimum of 290 gallons of fuel,

   b. A separate fuel storage system containing a minimum of 37,500 gallons of fuel, and

   c. A separate fuel transfer pump.

3. AC Distribution

   a. 4160 Volt Bus 1C and 2C,

   b. 480 Volt Bus No. 1, Bus No. 2 and Bus No. 3, and

   c. Vital Bus 1, 2, 3, 3A, 4, 5 and 6.

4. DC Bus No. 1 and DC Bus No. 2 (including at least one full capacity charger and battery supply per bus).

5. The two Safety Injection System Load Sequencers.*

* The automatic load function may be blocked in Mode 3 at a pressure $\leq$ 1900 psig.

B. ACTION:

1. With one of the required offsite circuits inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter and SURVEILLANCE REQUIREMENT B.1.a within 24 hours; restore an additional offsite circuit to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

2. If one diesel generator in declared inoperable, demonstrate the OPERABILITY of the two offsite transmission circuits and the remaining diesel generator by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter and SURVEILLANCE REQUIREMENT B.1.a within 24 hours; restore the inoperable diesel generator to service within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

3. With one offsite circuit and one diesel generator of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per eight (8) hours thereafter and SURVEILLANCE REQUIREMENT B.1.a within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in COLD SHUTDOWN within the next 36 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from, the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

4. With two required offsite circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement B.1.a of Technical Specification 4.4 within 8 hours, unless the diesel generators are already operating; restore at least one of the inoperable sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 4 hours. With only one of the required offsite circuits restored, restore the remaining offsite circuit to OPERABLE status within 72 hours from the time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

5. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite circuits by performing Periodic Testing Requirement A of Technical Specification 4.4 within one hour and at least once per two (2) hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore both diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

6. With less than the above complement of AC buses OPERABLE, restore the inoperable bus within 8 hours or be in COLD SHUTDOWN within the next 36 hours.

7. With one required DC bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours.

8. With a required DC bus battery and both of its chargers inoperable, restore the inoperable battery and one of its chargers to operable status within 2 hours or be in cold shutdown within the next 36 hours.

9. With one Safety Injection Load Sequencer inoperable, restore the inoperable sequencer to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

II. Additionally, in MODES 1, 2 and 3 the following specifications shall apply:

A. As a minimum, the following shall be OPERABLE:

1. The MOV850C Uninterruptable Power Supply (UPS).

B. ACTION:

1. With the MOV850C UPS inoperable, restore the UPS to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

III. In MODES 5 and 6 the following specifications shall apply:

A. As a minimum, the following shall be OPERABLE:

1. One Southern California Edison Company or San Diego Gas and Electric Company high voltage transmission line to the switchyard and one transmission circuit from the switchyard, immediate or delayed access, to the onsite safety-related distribution system.

2. One diesel generator (capable of automatic start) with:

a. A day tank containing a minimum 290 gallons of fuel,

b. A fuel storage system containing a minimum of 37,500 gallons of fuel, and

c. A fuel transfer pump.

3. The electrical Buses associated with the operable power sources as follows:

a. One 4,160 Volt AC Bus

b. One 480 Volt AC Bus

c. AC Vital Buses 1, 2 and 4, and

d. One DC Bus (including at least one full capacity charger and battery supply per Bus).

B. ACTION:

1. With less than the minimum required AC and DC electrical sources specified in III.A above, suspend all operations involving core alterations or positive reactivity changes.

<u>BASIS</u>: The station is connected electrically to the Southern California Edison Company and San Diego Gas & Electric Company system via either of two physically independent high voltage transmission routes composed of four Southern California Edison Company high voltage lines and four San Diego Gas & Electric Company high voltage lines.

Of the four Southern California Edison Company lines, any one can serve as a source of power to the station auxiliaries at any time. Similarly, any of the four San Diego Gas & Electric Company lines can serve as a source of power to the station auxiliaries at any time. By specifying one transmission line from each of the two physically independent high voltage transmission routes, redundancy of sources of auxiliary power for an orderly shutdown is provided.

Similarly, either transformer A or B, along with transformer C, provide redundancy of 4160 volt power to the auxiliary equipment, and in particular to the safety injection trains. Correct operation of the safety injection system is assured by the operability of the load sequencers and the UPS for MOV 850C. Correct operation of the recirculation system is assured by the operability of the UPS for MOV 850C which also supplies MOV 358. In addition, each 4160 volt bus has an onsite diesel generator as backup.

In MODES 1, 2, 3 and 4, two diesel generators provide the necessary redundancy to protect against a failure of one of the diesel generator systems or in case one diesel generator system is taken out for maintenance, without requiring a reactor shutdown. This also eliminates the necessity for depending on one diesel generator to operate for extended periods without shutdown if it were required for post-accident conditions.

In MODES 5 and 6, the requirement for one source of offsite power and one diesel generator to be OPERABLE will provide diverse and redundant electrical power sources in order that the station can be maintained in the COLD SHUTDOWN or REFUELING condition for extended time periods. Additionally, this requirement will assure that operations involving core alterations or positive reactivity changes can be conducted safely.

3.8 FUEL LOADING AND REFUELING

APPLICABILITY: Applies to fuel handling and refueling operations. For the applicable surveillance requirements, see Table 4.1.2.

OBJECTIVE: To prevent incidents during fuel handling operations that could affect public health and safety.

SPECIFICATIONS: A. During refueling operations (MODE 6):

1. Radiation levels in the containment and spent fuel building shall be monitored.

2. Core subcritical neutron flux shall be continuously monitored during the entire refueling period by not less than two neutron monitors, each with continuous visual indication and one with continuous audible indication.

3. For water levels in the refueling pool, greater than elevation 40 feet, 3 inches (See 7. below for reference evaluation), the following specifications shall apply:

   a. At least one of the following methods of decay heat removal shall be in operation and circulating reactor coolant at a flow rate of $\geq$ 400 gpm:

      (1) One RHR TRAIN.

      (2) One refueling water pump taking suction from the refueling pool through the recirculation heat exchanger (with supporting heat removal systems operating), and discharging via the safety injection system piping to one reactor coolant loop cold leg.

   b. With less than one method of decay heat removal in operation, except as provided in c. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system. Immediately initiate corrective action to return the required decay heat removal method to operating status as soon as possible. In addition, within four hours, close all containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

   c. The decay heat removal capability may be removed from operation for up to one hour per eight hour period.

4. Whenever the water level in the refueling pool is less than elevation 40 feet 3 inches (see (7) below for reference elevation), the following specifications shall apply:

   a. Two of the following methods of decay heat removal shall be OPERABLE, and at least one shall be in operation circulating reactor coolant:

      (1) Residual heat removal (RHR) pump G-14A with one RHR TRAIN.

      (2) Residual heat removal (RHR) pump G-14B with a second RHR TRAIN.

      (3) One refueling water pump taking suction from the refueling pool through the recirculation heat exchanger (with supporting heat removal systems operating), and discharging via the safety injection system piping to one reactor coolant loop cold leg.

   b. With less than the required methods of decay heat removal OPERABLE, immediately initiate corrective action to return the required methods of decay heat removal to OPERABLE status as soon as possible or to establish water level in the refueling pool of at least 40 feet 3 inches (see (7) below for reference elevation).

   c. With none of the required methods of decay heat removal in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system. Immediately initiate corrective action to return the required decay heat removal method to operation. In addition, within four hours, close all containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

5. With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the more restrictive of the following reactivity conditions shall be met:

   a. A SHUTDOWN MARGIN greater than 5% $\Delta k/k$.

   b. A boron concentration greater than or equal to 2,000 ppm.

6. The reactor shall be subcritical for at least 148 hours prior to movement of irradiated fuel in the reactor pressure vessel.

7. Borated water to insure the SHUTDOWN MARGIN as specified in Item A.(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the refueling pool during movement of fuel assemblies and RCC's. Reference elevation is sea level, mean lower low water.

8. If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be carried out.

B. With fuel assemblies in the spent fuel storage pool:

1. Loads in excess of 1,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

2. Borated water to insure the SHUTDOWN MARGIN as specified in Item A(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the spent fuel storage pool. Reference elevation is sea level, mean lower low water.

3. With the requirement of B(2) above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limits within four hours.

<u>BASIS</u>: During refueling the reactor cavity is filled with approximately 240,000 gallons of borated water whose concentration is sufficient to maintain the reactor subcritical by greater than 5% $\Delta k/k$ or to a boron concentration greater than or equal to 2,000 ppm, whichever is more restrictive. Operation of one method of decay heat removal is provided to assure continuous mixing flow of refueling water through the reactor vessel during the refueling period.(1) Borated water injection capability is provided as per Specification 3.2 Part A in the unlikely event there is any need during the refueling period.

The requirement that at least one method of decay heat removal be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during refueling, and (2) sufficient cooling circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two methods of decay heat removal OPERABLE when the refueling pool water level is less than elevation 40 feet 3 inches ensures that a single failure of an operating component will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling; thus, in the event of a failure of an operating component, adequate time is available to initiate alternate means to cool the core.[2]

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling.[3] These include:

(1) An interlock on the lifting hoist to prevent lifting of more than one fuel assembly at any one time.

(2) The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The restrictions on movement of loads in excess of 1,500 pounds (i.e., the nominal weight of a fuel assembly, RCC, and associated handling tool) over fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analysis.

Requiring a minimum water elevation of 40 feet 3 inches in the refueling pool, and similarly in the spent fuel storage pool, ensures that (1) at least 23 feet of water would be available to remove 99% of the iodine gap activity assumed to be released in the event of a dropped and damaged fuel assembly, and (2) there will be at least twelve feet of water above the top of the fuel rods of a withdrawn fuel assembly so as to limit dose rates at the top of the water in accordance with Section 4.2.6 of the facility FSA. Reference elevation is sea level, mean lower low water.

Finally, detailed written procedures are provided, and are carried out under close supervision by licensed personnel.

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel assures that sufficient time has elapsed to allow the radioactive decay of short-lived fission products.

REFERENCES:

(1) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 5, Questions 8 and 9.

(2) NRC letter dated June 11, 1980, from D. G. Eisenhut to all operating pressurized water reactors.

(3) Final Safety Analysis, Paragraph 2.9.

3.9 MODERATOR TEMPERATURE COEFFICIENT (MTC)

APPLICABILITY: Applies to negative moderator temperature coefficient (MTC) during core operations whenever the nominal reactor coolant inlet temperature is greater than or equal to 528°F.

OBJECTIVE: To establish negative MTC limits for the core.

SPECIFICATION:

a. The MTC shall be less negative than -3.8×10^{-4} $\Delta k/k/°F$ for all rods withdrawn, end of cycle life (EOL) and the RATED THERMAL POWER condition.

b. In order to assure that the above negative MTC limit is not exceeded, the MTC shall be measured at any THERMAL POWER and compared to the predetermined, calculated negative MTC within 7 effective full power days (EFPD) of reaching an equilibrium boron concentration of 300 ppm. The predetermined calculated MTC value (at RATED THERMAL POWER conditions with all rods withdrawn) is determined as follows:

   -3.1×10^{-4} $\Delta k/k/°F$ MTC at a nominal core inlet coolant temperature of 551.5°F, and MTC increasing linearly with decreasing inlet coolant temperature to -2.5×10^{-4} $\Delta k/k/°F$ at a nominal core inlet coolant temperature of 528.0°F.

ACTION: In the event this comparison indicates the MTC is more negative than the applicable value given above, the MTC shall be remeasured, and compared to the EOL MTC limit of -3.8×10^{-4} $\Delta k/k/°F$ at least once per 14 EFPD during the remainder of the cycle. If the measured MTC is more negative than the -3.8×10^{-4} $\Delta k/k/°F$ limit any time during the remainder of the cycle, the reactor shall be in HOT SHUTDOWN within 12 hours after exceeding the negative MTC limit.

BASIS: The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the San Onofre Unit 1 accident and transient analyses.

The limiting MTC used in the steam line break accident analysis is given as a function of k_{eff} and average moderator temperature in Figure 14 of Amendment 18 to the FSAR. In order to ensure that the safety analysis remains valid. The reactor should not be operated with an MTC more negative than the limit implied by Figure 14 of Amendment 18.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the San Onofre Unit 1 analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition, and a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.0×10^{-4} $\Delta k/k/°F$. The MTC predetermined calculated value for surveillance purposes represent conservative values (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and are obtained by making these corrections to the limiting MTC value of -4.0×10^{-4} $\Delta k/k/°F$. In order to provide a margin of safety, the reactor should not be operated with an MTC more negative than -3.8×10^{-4} $\Delta k/k/°F$.

The measurement of the MTC at the end of the fuel cycle is adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in boron concentration associated with fuel burnup.

References:

(1) Final Engineering Report and Safety Analysis, Paragraph 3.4

(2) Supplement No. 1 to Final Engineering Report and Safety Analysis Section 5, Question 8 and 9

(3) Final Engineering Report and Safety Analysis, Paragraph 3.9

3.10 INCORE INSTRUMENTATION

APPLICABILITY: MODE 1 above 90% RATED THERMAL POWER

OBJECTIVE: To specify the type and frequency of incore measurements used to verify linear power density values.

SPECIFICATION:

a. A power distribution measurement shall be performed every 30 effective full power days (EFPD) and after attainment of equilibrium xenon upon return to power following a refueling shutdown.

b. The incore instrumentation system shall be used to accomplish the Correlation Verification of incore versus excore data for the axial offset monitoring system prior to exceeding 90% of RATED THERMAL POWER following each refueling and at least once per 180 effective full power days (EFPD) thereafter. Subsequent to the Correlation Verification and for the duration of each cycle, incore instrumentation shall be used to perform a Correlation Check of the axial offset monitoring system every 30 EFPD.

c. A core thermocouple map shall be taken every 30 EFPD and after attainment of equilibrium xenon upon return to power following a refueling shutdown.

ACTION:

A. If the correlation check, power distribution measurement or core thermocouple map described above cannot be made within the prescribed time, a maximum of 15 EFPD will be allowed for equipment correction.

B. In the event that Specification a, b and c cannot be met during the 15 EFPD allowed for corrective action, within one hour action shall be taken such that THERMAL POWER is restricted to less than or equal to 90% of RATED THERMAL POWER until these specifications can be met.

BASIS: The flux mapping system is used to measure the core power distribution and to correlate incore versus excore data for the axial offset system. Measurements made with the flux mapping system every 30 effective full power days and upon return to power following a refueling shutdown will monitor the core power distribution to confirm that the maximum linear power density remains below allowable values. The axial offset system will monitor the axial core power distribution in a continuous manner. If the Correlation Verification or Correlation Check is not performed, the 90% of full thermal power restriction assures safe operation of the reactor. In addition, core thermocouples provide an independent means of measuring the balance of power among the core quadrants.

The flux mapping system and the thermocouple system are not integral parts of the Reactor Protection System. These systems are, rather, surveillance systems which may be required in the event of an abnormal condition such as a power tilt or a control rod misalignment. Since such a condition cannot be predicted, it is prudent to have the surveillance systems in an operable state. The 90% of full power restriction, used when these measurements cannot be taken as scheduled, is applied to minimize the probability of exceeding allowed peaking factors.

Operation for a 180 effective full power day period prior to reperforming the correlation verification is acceptable on the basis that the allowed incore axial offset limits are reduced by the amount in percent of incore axial offset that the monthly correlation check differs from the correlation.

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

APPLICABILITY: MODE 1 above 90% RATED THERMAL POWER

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $$IAO = \frac{2.89/P - 2.1225}{0.03021} - FCC$$

For negative offsets: $$IAO = \frac{2.89/P - 2.1181}{-.03068} + FCC$$

where

IAO = incore axial offset

P = fraction of RATED THERMAL POWER

FCC = The larger of 3.0 or the value in percent of incore axial offset by which the current correlation check differs from the incore-excore correlation.

ACTION:

A. With IAO exceeding the limit defined by the specification, within 1 hour action shall be taken to reduce THERMAL POWER until IAO is within specified limits or such that THERMAL POWER is restricted to less than 90% of RATED THERMAL POWER.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

C. With no method for determining IAO available, within 1 hour action shall be taken such that THERMAL POWER is reduced to less than 90% of RATED THERMAL POWER until a method of determining axial offset is restored.

BASIS: The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating incore axial offset to the axial offset monitoring systems. The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed incore axial offset limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and incore axial offset has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power in cases when limits are approached or exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to reduce THERMAL POWER to 90% of RATED THERMAL POWER. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored.

References:

(1) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(2) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974

(3) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1 Cycle 5, January, 1975, Westinghouse Non-Proprietary Class 3.

3.12 CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM

APPLICABILITY: Applies to the operational status of the control room emergency air treatment system.

OBJECTIVE: To identify those conditions of the control room emergency air treatment system which will ensure reliable and efficient operation, should the system be needed.

SPECIFICATION: Effective upon completion of field testing to the modified filter system.

A. Except as specified in Specification 3.12.B below, the control room emergency air treatment system shall be OPERABLE whenever the reactor is to be made or maintained critical. The system will be considered OPERABLE as long as the tests and analyses specified in Specification 4.11 are satisfactorily completed at the required intervals and the system is not removed from service.

B. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days.

C. If the conditions in 3.12.B cannot be met, reactor shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN condition with 24 hours.

BASIS: The control room emergency air treatment system is designed to filter the control room intake air during control room isolation. The system is placed in operation under administrative control when conditions warrant its use.

The system utilizes a fan, a high efficiency particulate absolute (HEPA) filter, pre-filters and a charcoal absorber bed. The pre-filters are installed before the charcoal bed to prevent clogging of the iodine adsorbers. The charcoal adsorbers reduce the potential intake of radioiodine to the control room.

The OPERABILITY requirements of this Specification in conjunction with the surveillance requirement of Specification 4.11 provide reasonable assurance that the system will operate, if needed, at a degree of efficiency equal to or better than that assumed in the Final Safety Analysis.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited time while repairs are made. If the system cannot be repaired within seven days, the reactor is shut down and brought to COLD SHUTDOWN within 24 hours.

3.13 SHOCK SUPPRESSORS (SNUBBERS) OPERABILITY

APPLICABILITY: Applies to safety related shock suppessors (snubbers).

OBJECTIVE: To define operability requirements of snubbers required to protect safety related piping from unrestricted motion when subjected to dynamic loading as might occur during a seismic event or severe transient.

SPECIFICATION:

A. During MODES 1, 2, 3, and 4 (MODES 5 and 6 for snubbers located in the systems required OPERABLE in those MODES) all snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

B. 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 4.14.C on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

BASIS: Snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an 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.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by in-service functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly 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, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling outage intervals. Observed failures of these sample snubbers will require functional testing of additional units. Snubbers of rated capacity greater than 120,000 pounds are exempt from functional testing requirements because of the impracticability of testing such large units.

Hydraulic snubbers and mechanical snubber may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.14 FIRE PROTECTION SYSTEMS OPERABILITY

APPLICABILITY: Applies to the operating status of the fire detection and extinguishing systems and equipment at all times.

OBJECTIVE: To ensure availability of fire protection systems.

SPECIFICATION: A. As a minimum, the following fire detection and extinguishing systems and equipment shall be OPERABLE.

1. The Fire Suppression Water Systems[1] with:

   a. Any two of the following four pumps OPERABLE each with a capacity of 1000 gallons per minute with their discharge aligned to the fire main:

      (1) San Onofre Unit 1 fire water pumps (2)

      (2) San Onofre Units 2&3 motor driven fire water pumps (2)

   b. With San Onofre Unit 1 fire water pumps satisfying the pump requirement, the San Onofre Unit 1 service water reservoir supply available containing a minimum of 300,000 gallons reserved for fire fighting.

   c. With San Onofre Units 2&3 fire pumps satisfying the pump requirement, the San Onofre Units 2&3 service and fire water storage tanks available with 300,000 gallons reserved for fire fighting.

   d. With a combination of the four pumps satisfying the pump requirement, the separate water supplies for each pump(s) available as indicated in A(1)b and A(1)c above.

   e. An OPERABLE flow path capable of taking suction from the separate supplies per A(1)b or A(1)c above and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve upstream of each sprinkler, hose standpipe or spray system riser required to be OPERABLE per Specifications 3.14.A.(2) and 3.14.A(3).

2. The Spray and/or Sprinkler Systems located in the following areas:

   a. Containment sphere. This includes a refueling water pump, 240,000 gallons of water in the Refueling Water Storage Tank and associated system valves. During refueling operations, when the Refueling Water Storage Tank water has been transferred to the refueling cavity, backup fire suppression equipment shall be provided.

   b. Lube oil reservoir and conditioner.

   c. Hydrogen seal oil.

   d. Diesel generator building.

3. The Fire Hose Stations indicated in Table 3.14.1.

4. The Fire Detection Instrumentation for each fire detection area or zone indicated in Table 3.14.2.

B. In the event a limiting condition for operation for the fire detection and extinguishing systems and equipment indicated in A above is not met, the following corrective measure shall be taken:

1. The Fire Suppression Water System

   a. With less than the required equipment indicated in A(1) above, restore the inoperable equipment to OPERABLE status within seven days or in lieu of any other report required by Specification 6.9 prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.3.c within the next thirty days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

   b. With no Fire Suppression Water System operable;

      (1) Establish a backup Fire Suppression Water System within 24 hours, and

      (2) In lieu of any other reports required by Specification 6.9, submit a special report in accordance with Specification 6.9.3.c;

         (a) By telephone within 24 hours,

(b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

(c) In writing within 14 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.

(3) If B.(1)b.1 and 2.(a) above cannot be fulfilled, place the reactor in Hot Standby within six (6) hours and in Cold Shutdown within the following thirty (30) hours.

2. The Spray and/or Sprinkler System

a. With a spray and/or sprinkler system inoperable establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s), within one hour.

b. Restore the system to OPERABLE status within fourteen days or in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.3.c within the next thirty days outlining the action taken, the cause of inoperability and the plans and schedule for restoring the system to operable status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3. The Fire Hose Stations

With one or more of the fire hose stations indicated in Table 3.14.1 inoperable, route an additional equivalent capacity fire hose to the unprotected area from an OPERABLE hose station within one hour. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

4. The Fire Detection Instrumentation

With one or more of the fire detection instruments shown in Table 3.14.2 inoperable.

a. Within one hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour with the exception of the zones inside containment where the following alternative instrumentation shall be utilized:

(1) Inside the secondary Shield: temperature indication of air after primary coolant Motor cooling fan unit, primary coolant motor space, and reactor coolant pump lower bearing coolant temperature, motor winding temperature and oil lubricated bearing temperature.

(2) Outside the secondary shield: temperature of control rod cooler discharge, control rod shroud air inlet, sphere space, and control rod cooler inlet; closed circuit television camera.

b. Restore the inoperable instrument(s) to OPERABLE status within fourteen days, or in lieu of any other report required by Specification 6.9, prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.3.c within the next thirty days outlining the course of action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Basis: The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The Fire Suppression Systems consist of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression system is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment is restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measure to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

REFERENCE: (1) Fire Protection Program Review, BTP APCSB 9.5-1, San Onofre Nuclear Generating Station, Unit 1, March 1977; submitted to the NRC by letter dated March 16, 1977 in Docket No. 50-206.

Table 3.14.1

FIRE HOSE STATIONS

Fire Area or Zone	
Inside Sphere	One
Reactor Auxiliary Building, Lower Level	One
Boric Acid Injection Pump Room	One
Turbine Plant Cooling Water Area	One
Chemical Feed and Lubricating-Oil Reservoir Area	One
East Feedwater Pump/Condenser Area	Three
West Feedwater Pump/Condenser Area	Two
Turbine and Heater Decks	Six
Administration/Control Building, First Floor Single-Story Office Area	One
Administration/Control Building, First Floor Health Physics and Locker Area	One
Control Room Area	One
Administration/Control Building, Third Floor East Office Space and Storage	One
Diesel-Generator Room No. 1	One
Diesel-Enclosure Room No. 2	One
Sphere Enclosure Cable Penetration Area	Four
Administration/Control Building, Second Floor North Stairwell	One

Table 3.14.2
FIRE DETECTION INSTRUMENTS

Zone	Location	Minimum Instruments Operable Infrared Scanners	Minimum Instruments Operable Smoke
1	DC switchgear and battery room		3
2	480-V Switchgear room		7
3	4160-V SWitchgear room		15
4	Exciter and MCC3 area		15
7	Control room and third floor administration building		12
8	Turbine lube oil reservoir		28
9	Containment sphere inside secondary shield		5
10	Containment sphere outside secondary shield		5
11	Reactor auxiliary building and storage rooms		7
16	Sphere enclosure building		11
DG 1	Diesel Generator Room No. 1	2	2
DG 2	Diesel Generator Room No. 2	2	2

NOTE: Fire Detection Zones 5, 6, 12, 13, 14 and 15 do not contain safety related equipment, nor do they contain potential fire hazards to safety related equipment.

3.15 RADIOACTIVE LIQUID EFFLUENTS

3.15.1 LIQUID EFFLUENTS CONCENTRATION

APPLICABILITY: At all times.

OBJECTIVE: Maintain the concentration of radioactive liquid material released from the site below 10 CFR 20 limits.

SPECIFICATION:

A. The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix 8, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} μCi/ml.

B. ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

BASIS: This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3.15.2 LIQUID EFFLUENT DOSE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the release of radioactive liquid effluents from the site as low as is reasonably achievable.

SPECIFICATION: A. The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

1. During any calendar quarter to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and

2. During any calendar year to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.

B. ACTION:

1. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to implement the requirements of Section II.A and IV.A of Appendix I, 10 CFR Part 50. Specification A implements the guides set forth in Section II.A of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable."

3.15.3 LIQUID WASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: Maintain radioactive releases from the site as low as is reasonable achievable by use of the liquid radwaste treatment system.

SPECIFICATION:

A. The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected dose due to the liquid effluent from San Onofre Unit 1, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

B. ACTION:

1. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   a. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reason for inoperability.

   b. Action(s) taken to restore the inoperable equipment to OPERABLE status.

   c. Summary description of action(s) taken to prevent a recurrence.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonable achievable." This specification implements the requirements of 10 CFR Part 50.36a and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3.16 RADIOACTIVE GASEOUS EFFLUENTS

3.16.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose rate at the exclusion area boundary from radioactive gaseous effluents within 10 CFR 20 limits.

SPECIFICATION:

A. The dose rate to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following values:

1. The dose rate limit for noble gases shall be ≤ 500 mrem/year to the total body and ≤ 3000 mrem/year to the skin, and

2. The dose rate limit for I-131, I-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days shall be ≤ 1500 mrem/year to any organ.

B. ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrating of 10 CFR Part 20, Appendix B, Table 11, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limit also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to ≤ 1500 mrem/year.

3.16.2 DOSE, NOBLE GASES

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to noble gases in gaseous effluents as low as is reasonable achievable.

SPECIFICATION: A. The air dose due to noble gases released in gaseous effluents, from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

1. During any calendar quarter: $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation.

2. During any calendar year: $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation.

B. ACTION:

1. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to implement the requirements of Section II.B and IV.A of Appendix I, 10 CFR Part 50. Specification A implements the guides set forth in Section II.B of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonable achievable."

3.16.3 DOSE, IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to radioiodine, radioactive materials in particulate form and radionuclides other than noble gases in gaseous effluents as low as is reasonable achievable.

SPECIFICATION:

A. The dose to a MEMBER OF THE PUBLIC from I-131, I-133, from tritium and from all radionuclides in particulate form with half-lives greater than 8 days-in gaseous effluents released from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

   1. During any calendar quarter: $\leq$ 7.5 mrem to any organ; and

   2. During any calendar year: $\leq$ 15 mrem to any organ.

B. ACTION:

   1. With the calculated dose from the release of I-131, I-133, tritium and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

   2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to implement the requirements of Sections II.C and IV.A of Appendix I, 10 CFR Part 50. Specification A is the guide set forth in Section II.C of Appendix I. Specification B provides the required operating flexibility and at the same time implements the guides set forth in Section IV.a of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonable achievable."

3.16.4 GASEOUS RADWASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: Maintain radioactive gaseous releases from the site as low as is reasonable achievable by use of the GASEOUS RADWASTE and VENTILATION EXHAUST TREATMENT SYSTEMS.

SPECIFICATION:

A. The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from San Onofre Unit 1 to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.3 mrem to any organ over 31 days.

B. ACTION:

1. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   a. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems and the reasons for the inoperability.

   b. Action(s) taken to restore the inoperable equipment to OPERABLE status.

   c. Summary description of action(s) taken to prevent a recurrence.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.,

BASIS: The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This

Specification implements the requirements of 10 CFR Part 50.36a, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3.16.5 GAS STORAGE TANK

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of radioactivity contained in gas storage tanks.

SPECIFICATION:

A. The quantity of radioactivity contained in each gas storage tank shall be limited to ≤ 56,000 curies noble gases (considered as Xe-133).

B. ACTION:

1. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS:

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 mrem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG 0800, July 1982.

3.16.6 EXPLOSIVE GAS MIXTURE

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of explosive gases contained in the gas storage tanks.

SPECIFICATION:

A. The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

B. ACTION:

1. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

2. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

3. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3.17 DOSE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose due to the release of radioactive materials within specified limits.

SPECIFICATION: A. The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and to radiation, from uranium fuel cycle sources shall be limited to ≤ 25 mrem to the total body or any organ (except the thyroid which shall be limited to ≤ 75 mrem).

B. ACTION:

1. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.15.2.A, 3.16.2.A or 3.16.3.A, calculations should be made to determine whether the above limits of Specification 3.17 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases, to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. The Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification is provided to meet the reporting requirements of 40 CFR 190. In complying with 40 CFR 190, nuclear fuel cycle facilities over five miles away are not considered to contribute to the dose assessment.

3.18 RADIOLOGICAL ENVIRONMENTAL MONITORING

3.18.1 MONITORING PROGRAM

APPLICABILITY: At all times.

OBJECTIVE: Monitor exposure pathways for radiation and radioactive material.

SPECIFICATION:

A. The radiological environmental monitoring program shall be conducted as specified in Table 3.18.1.

B. ACTION:

1. With the radiological environmental monitoring program not being conducted as specified in Table 3.18.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

2. With the level of radioactivity as the result of plant effluents in an environmental sampling medium exceeding the reporting levels of Table 3.18.2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification 6.9.2. When more than one of the radionuclides in Table 3.18.2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.18.2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.15.2, 3.16.2 and 3.16.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3. With fresh leafy vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 3.18.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.18.1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

4. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of a MEMBER OF THE PUBLIC resulting from the station operation. This monitoring program thereby supplements the radiological effluents monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

TABLE 3.18.1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
1. AIRBORNE Radioiodine and Particulates	Samples from at least 5 locations 3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q. 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. 1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction.[c]	Continuous operation of sampler with sample collection an required by dust loading but at least once per 7 days.[d]	Radioiodine cartridge. Analysis at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample When gross beta activity is $\geq$ 10 time the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION[e]	At least 30 locations including an inner ring of stations in the general area of the SITE BOUNDARY and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	At least once per 92 days.	Gamma dose. At least once per 92 days.

	Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
3.	WATERBORNE			
	a. Ocean	4 Locations	At least once per month and composited quarterly	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days.
	b. Drinking	2 Locations	Monthly at each location.	Gamma isotopic and tritium analyses of each sample.
	c. Sediment	4 Locations from Shoreline	At least once per 184 days.	Gamma isotopic analysis of each sample.
	d. Ocean	5 Locations Bottom Sediments	At least once per 184 data.	Gamma isotopic analysis of each sample.
4.	INGESTION			
	a. Nonmigratory Marine Animals	3 Locations	One sample from each group (listed below) will be collected in season, or &t least once per 184 days it not seasonal. Groups to be sampled: 1. Fish-2 adult species such as flatfish, bass, parch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Hollunks-such as limpets, clams or seahares.	Gamma isotopic analysis an edible portions.

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
b. Local Crops	2 Locations	Representative vegetables, normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semi-annually an I-131 analysis for leafy crops.

TABLE NOTATION

(a) Sample locations are indicated in the ODCM.

(b) Gamma isotopic analysis means the identification and quantification of gamma-omitting radionuclides that may be attributable to the effluents from the facility.

(c) The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

(d) Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

(e) Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

TABLE 3.18.2

REPORTING LEVELS FOR RADIOACTIVITY
CONCENTRATION IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals (pCi/Kg, wet)	Local Corps (pCi/Kg, wet)
H-3	2×10^4 (a)			
Mn-54	1×10^3		3×10^4	
Fe-59	4×10^2		1×10^4	
Co-58	1×10^2		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^3	
Zr-Nb-95	4×10^2			
I-131	2	0.9		1×10^2
Cs-134	30	10	1×10^3	1×10^3
Cs-137	50	20	2×10^3	2×10^3
Ba-La-140	2×10^2			

(a) For drinking water samples. This is 40 CFR Part 141 value.

3.18.2 LAND USE CENSUS

APPLICABILITY: At all times.

OBJECTIVE: Monitor the UNRESTRICTED AREAS surrounding the site for potential changes to the radiological monitoring program as necessary.

SPECIFICATION: A. A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

B. ACTION:

1. With the land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.6.3, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new locations. Identify the new locations in the next Semiannual Radioactive Effluent Release Report.

2. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.18.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new locations. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31, of the year in which this land use census was conducted.

*Broad leaf vegetation sampling may be performed at the SITE BOUNDARY in the direction section with the highest D/Q in lieu of the garden census.

3. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This Specification is provided to ensure that changes in the use of UNRESTRICTED AREAS are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (25 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, (1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/square meter.

3.18.3 INTERLABORATORY COMPARISON PROGRAM

APPLICABILITY: At all times.

OBJECTIVE: To ensure laboratory analysis of radiological environmental monitoring samples is correct and accurate.

SPECIFICATION:

A. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

B. ACTION:

1. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

3.19 SOLID RADIOACTIVE WASTE

APPLICABILITY: At all times.

OBJECTIVE: Ensure meeting the requirements for the SOLIDIFICATION and shipment of solid radwaste.

SPECIFICATION:

A. The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

B. ACTION:

1. With the provisions of the PROCESS CONTROL PROGRAM satisfied suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification/agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to operability of the overpressurization protection systems.

OBJECTIVE: To preclude the potential for exceeding 10 CFR 50, Appendix G, in the event of a pressure transient while water-solid.

SPECIFICATION:

A. When the RCS pressure is ≤ 400 psig* and pressurizer water level is greater than 50%, at least one of the following overpressure protection systems shall be OPERABLE:

(1) Two power operated relief valves (PORVs) with a lift setting of ≤ 500 psig,** or

(2) A reactor coolant system vent(s) of ≥ 1.75 square inches.

ACTION:

B. With one PORV inoperable when required in accordance with Specification A above, either restore the inoperable PORV to OPERABLE status within seven days or depressurize and vent the RCS through a 1.75 square inch vent(s) within the next eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

C. With both PORVs inoperable when required in accordance with Specification A above, depressurize and vent the RCS through at least a 1.75 square inch vent(s) within eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

D. In the event either the PORVs or the RCS vent(s) are used to mitigate a potential RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances indicating transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

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* The placing in service of the OMS at ≤ 400 psig is intended to assure that protection is provided whenever temperature is below 360°F. The alarm to arm the OMS being keyed to pressure assures that inadvertent opening of the PORVs does not occur due to placing the OMS into service with RCS pressure above the 500 psig initiation setpoint.

** The 500 psig setpoint is based on the current heatup and cooldown curves for 16 EFPY. The setpoint requires reevaluation for acceptability any time the curves are changed.

BASIS:

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.75 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the initial RCS pressure is ≤ 400 psig and the pressurizer water level is greater than 50%, assuming a single failure of one PORV and no operator action for 10 minutes. Either PORV has adequate relieving capability to protect the RCS from overpressurization due to a design basis transient as described in submittal to the NRC dated October 12, 1977.

Tagged as it refers to the RCS vent, means tagged in accordance with current Southern California Edison procedures for tagging of equipment which must not be operated.

4.0 SURVEILLANCE REQUIREMENTS (GENERAL)

APPLICABILITY: Applies to the surveillance requirements to be implemented in these specifications.

OBJECTIVE: To define the conditions under which the surveillance requirements of Section 4 Specifications are applicable.

SPECIFICATION: 4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance-intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

BASIS: Specification 4.0.1 provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for, Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

The provisions of Specification 4.0.2 provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

The provisions of specification 4.0.3 set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components, OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

Specification 4.0.4 ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an, OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.1.1 OPERATIONAL SAFETY ITEMS

APPLICABILITY: Applies to surveillance requirements for items directly related to Safety Standards and Limiting Conditions for Operation.

OBJECTIVE: To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

SPECIFICATION:

A. Reactor Trip System instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.1.

B. Equipment and sampling tests shall be as specified in Table 4.1.2.

C. The specific activity and boron concentration of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1.2., Item 1a.

D. The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1.2., Item 1b.

E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for the instrument inaccuracies, at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours.

G. During MODE 1 or 2 operation each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

H. Instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.3.

TABLE 4.1.1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.
2. Power Range, Neutron Flux	S	D (2,3) R (3,4)	M	N.A.	N.A.
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	N.A.	N.A.	M	N.A.	N.A.
4. Intermediate Range, Neutron Flux	S	R (3,4)	S/U (1), M	N.A.	N.A.
5. Source Range, Neutron Flux	S	R (3)	S/U (1), M	N.A.	N.A.
6. NIS Coincidentor Logic	N.A.	N.A.	N.A.	N.A.	M (5)
7. Pressurizer Variable Low Pressure	S	R	M	N.A.	N.A.
8. Pressurizer Pressure	S	R	M	N.A.	N.A.
9. Pressurizer Level	S	R	M	N.A.	N.A.
10. Reactor Coolant Flow	S	R	Q	N.A.	N.A.
11. Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.
12. Turbine Trip-Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U (1,6)	N.A.

TABLE 4.1.1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) - The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Setpoint verification is not applicable.

TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during MODES 1, 2, 3 and 4.
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during MODE 1.
	3.	Spectroscopic for E(1) Determination	1 per 6 months(2) Required only during MODE 1.
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours,(3) whenever the specific activity exceeds 1.0 µCi/gram DOSE EQUIVALENT I-131 or 100/ E (1) µCi/gram. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.
	5.	Boron concentration	Twice/Week

(1) E is defined in Section 1.0.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.2 (continued)

		Check	Frequency
1.b	Secondary Coolant Samples	1. Gross Activity Determination	At least once per 72 hours. Required only during MODES 1, 2, 3 and 4.
		2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. Required only during MODES 1, 2, 3 and 4.
			b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Required only during MODES 1, 2, 3, and 4.

TABLE 4.1.2 (continued)

			Check	Frequency
2.	Safety Injection Water Samples	a.	Boron Concentration	Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test beyond 1 month
3.	Control Rod Drop	a.	Verify that all rods move from full out to full in, in less than 2.44 seconds	At each refueling shutdown
4.	(Deleted)			
5.	Pressurizer Safety Valves	a.	Pressure Setpoint	At each refueling shutdown
6.	Main Steam Safety Valves	a.	Pressure Setpoint	At each refueling shutdown
7.	Main Steam Power Operated Relief Valves	a.	Test for OPERABILITY	At each refueling shutdown
8.	Trisodium Phosphate Additive	a.	Check for system availability as delineated in Technical Specification 4.2	At each refueling shutdown
9.	Hydrazine Tank Water Samples	a.	Hydrazine concentration	Once every six months when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test interval beyond six months
10.	Transfer Switch No. 7	a.	Verify that the fuse block for breaker 8-1181 to MCC 1 is removed	Monthly, when the reactor is critical and prior to returning reactor to critical when period of subcriticality extended the test interval beyond one month

TABLE 4.1.2 (continued)

		Check	Frequency
11.	MOV-LCV-1100 C Transfer Switch	a. Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above
12.	Emergency Siren Transfer Switch	a. Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed	Same as Item 10 above
13.	Communication Power Panel Transfer Switch	a. Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed	Same as Item 10 above
14a.	Spent Fuel Pool Water Level	Verify water level per Technical Specification 3.8	a. Once every seven days when spent fuel is being stored in the pool.
b.	Refueling Pool Water Level		b. Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.
15.	Reactor Coolant Loops/ Residual Heat Removal Loops	a. Per Technical Specifications 3.1.2.C and 3.1.2.D, in MODE 1 and MODE 2 and in MODE 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant.	a. Once per 12 hours
		b. Per Technical Specification 3.1.2.E, in MODE 3 with the reactor trip breakers open, verify	

TABLE 4.1.2 (continued)

Check	Frequency
1. At least two required reactor coolant pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The steam generators associated with the two required reactor coolant pumps are operable with secondary side water level ≥ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop is in operation and circulating reactor coolant.	3. Once per 12 hours
c. Per Technical Specification 3.1.2.F, in MODE 4 verify	
1. At least two required (RC or RHR) pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days
2. The required steam generators are operable with secondary side water level ≥ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours
3. At least one reactor coolant loop/RHR TRAIN is in operation and circulating reactor coolant.	3. Once per 12 hours
d. Per Technical Specifications 3.1.2.G and 3.1.2.H, in MODE 5 verify, as applicable:	

TABLE 4.1.2 (continued)

Check	Frequency
1. At least one RHR TRAIN is in operation and circulating reactor coolant.	1. Once per 12 hours
2. When required, one additional RHR TRAIN is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days
3. When required, the secondary side water level of at least two steam generators is $\geq$ 256 inches of narrow range on cold calibrated scale.	3. Once per 12 hours
e. Per Technical Specification 3.8.A.3, in MODE 6, with water level in refueling pool greater than elevation 40 feet 3 inches, verify that at least one method of decay heat removal is in operation and circulating reactor coolant at a flow rate of at least 400 gpm.	e. Once per 12 hours
f. Per Technical Specification 3.8.A.4, in MODE 6, with water level in refueling pool less than elevation 40 feet 3 inches, verify	
1. At least one decay heat removal method is in operation and circulating reactor coolant.	1. Once per 12 hours
2. One additional decay heat removal method is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days

TABLE 4.1.3

MINIMUM FREQUENCIES FOR TESTING, CALIBRATING, AND/OR CHECKING OF INSTRUMENT CHANNELS

	Channels	Surveillance	Minimum Frequency
1.	Axial Offset	Calibration	At each refueling shutdown
		Check	Once per shift
2.	Reactor Coolant Temperature	Calibration	At each refueling shutdown
		Test	Once per month
		Check	Once per shift
3.	Pressurizer Pressure Input to Safety Injection Actuation	Calibration	At each refueling shutdown
		Test	Once per month
4.	Rod Position Recorder	Calibration	At each refueling shutdown
		Check, comparison with digital readouts	Once per shift during operation
5.	Charging Flow	Calibration	At each refueling shutdown
6.	Boric Acid Tank Level	Calibration	At each refueling shutdown
		Test	Once per month
7.	Residual Heat Pump Flow	Calibration	At each refueling shutdown
8.	Volume Control Tank Level	Calibration	At each refueling shutdown.
		Test	Once per month during MODES 1 and 2
9.	Hydrazine Tank Level	Calibration	At each refueling shutdown
		Test	One per month during operation

BASIS: **CALIBRATION**

CALIBRATION should be performed at every reasonable opportunity in order to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels should be calibrated daily against a heat balance standard to account for errors induced by, changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between CALIBRATION. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of approximately one year.

Substantial CALIBRATION shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum CALIBRATION frequencies of once-per-day for the nuclear flux (linear level) channels, and once-per-year (approximately) for the process system channels is considered acceptable.

TESTING

The minimum testing frequency for those instrument channels connected to the safety system is based on an assumed "unsafe failure" rate of one per channel every four years. This assumption is, in turn, based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

The failure rate of one per channel every four years and the testing interval of two weeks imply that, on the average, each channel will be inoperable for 1.75 days per year, or 1.75/365 year. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two channels (assuming only two to be in service) is 1.75/365 squared, or 2.3×10^{-5}. From this it can be inferred that in a three channel system the probability of simultaneous failure of two channels is approximately 6.9 x 10^{-5}. This represents the fraction of time in which each three channel system would have one operable and two inoperable channels, and equals 6.9 x 10^{-5} x 8760 hours per year, or (approximately) 36 minutes/year.

It must also be noted that to thoroughly and correctly test a channel, the channel components must be made to respond in the same manner and to the same type of input as they would be expected to respond to during their normal operation. This, of necessity, requires that during the test the channel be made inoperable for a short period of time. This factor must be, and has been, taken into consideration in determining testing frequencies.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

During a 2-year testing period, the Reactor Coolant Flow Trips for each loop were tested 40 times. In all the tests the trips operated precisely on set point. Also, during this period, there were no 'unsafe failures' as defined above in the Reactor Coolant Flow Trips or any similar trip circuitry. All of these channels represent more than 30 years of service without a single 'unsafe failure'. Because of the demonstrated reliability of these instrument channels and particularly the Reactor Coolant Flow Trip, the testing interval of the Reactor Coolant Flow Trip has been extended to 3 months.

CHECK

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication, etc. 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, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, the minimum checking frequency of once per shift is deemed adequate.

4.1.2 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

APPLICABILITY: During releases via this pathway.

OBJECTIVE: To specify the minimum frequency and type of surveillance to be applied to the radioactive liquid instrumentation.

SPECIFICATION:

A. The setpoints shall be determined in accordance with procedures as described in the ODCM.

B. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations at the frequencies shown in Table 4.1.2.1.

BASIS: The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments are calculated in accordance with methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE 4.1.2.1

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL TEST
1. Gross Beta or Gamma Radioactivity Monitoring Providing Alarm and Automatic Isolation				
a. Liquid Radwaste Effluents Line (R-1218)	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line (R-1216)	D	M	R(3)	Q(1)
c. Turbine Building Sumps Effluent Line (Reheater Pit Sump R-2100*)	D	M	R(3)	Q(1)
d. Yard Sump (R-2101*)	D	M	R(3)	Q(1)
e. Component Cooling Water System (R-1217)	D	M	R(3)	Q(1)
2. Flow Rate Monitors				
Liquid Radwaste Effluent Line (FE 16 and FE 18)	D(4)	N/A	R	N/A

*Does not provide control room alarm annunciation when the instrument controls are set in the "not operate" mode.

TABLE 4.1.2.1
(Continued)

TABLE NOTATION

(1) The CHANNEL TEST also demonstrates the following:

1. Automatic isolation of this pathway and control room alarm annunciation occurs when the instrument indicates measured levels above the alarm/trip setpoint.

2. Control Room alarm annunciation when the instrument controls are not set in the operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from the suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

4.1.3 RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

APPLICABILITY: During releases via this pathway.

OBJECTIVE: To specify the minimum frequency and type of surveillance to be applied to the radioactive gaseous monitoring instrumentation.

SPECIFICATION:

A. The setpoints shall be determined in accordance with procedures as described in the ODCM.

B. Each radioactive gaseous process or effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations at the frequencies shown in Table 4.1.3.1.

BASIS: The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments are calculated in accordance with methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE 4.1.3.1

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL TEST
1. Stack Monitoring System				
a. Gross Activity Monitor (R-1214**)	D	M	R(2)	Q
b. Noble Gas Activity Monitor (R-1219, 1212, 1254*)	D	M	R(2)	Q(1)
c. Iodine Sampler Cartridge (R-1221, 1254*)	W	N/A	N/A	N/A
d. Particulate Sampler Filter (R-1211, 1220, 1254*)	W	N/A	N/A	N/A
e. Stack Fan Flow Indication (R-1254*)	D	N/A	Q	Q
f. Sampler Flow Rate Measuring Device	D	N/A	R	N/A

*Does not perform any isolation function. Does not provide control room alarm annunciation when the instrument controls are set in the "not operate" mode.

**Alarm only, does not perform any isolation function.

TABLE 4.1.3.1
(Continued)

TABLE NOTATION

(1) The CHANNEL TEST also demonstrates the following:

1. Automatic isolation of this pathway and control room alarm annunciation occurs when the instrument indicates measured levels above the alarm/trip setpoint.

2. Control room alarm annunciation when the instrument controls are not set in the operate mode.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)

4.1.4 CONTAINMENT ISOLATION INSTRUMENTATION

APPLICABILITY: Applies to instrumentation which actuates the containment sphere isolation valves, containment sphere purge and exhaust valves, and containment sphere instrumentation vent header valves.

OBJECTIVE: To ensure reliability of the containment sphere isolation provisions.

SPECIFICATION:

A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.4-1.

B. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

BASIS: The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standard. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.4-1

CONTAINMENT ISOLATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
Containment Isolation (Valves listed in Table 3.6.2-1)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
c) Sequencer Subchannels	N.A.	N.A.	M	1, 2, 3, 4
d) Safety Injection				
1) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
2) Pressurizer Pressure-Low	N.A.	R	M	1, 2, 3, 4
Purge and Exhaust Isolation (POV-9, POV-10, CV-10, CV-40, CV-116)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Radioactivity-High	S	R	M	1, 2, 3, 4

TABLE 4.1.4-1 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL TEST at least once per 31 days.

(2) The CHANNEL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

4.1.5 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3.

OBJECTIVES: To ensure the reliability of the accident monitoring instrumentation shown in Table 4.1.5-1.

SPECIFICATION: Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.1.5-1.

BASIS: The surveillance requirements specified for these systems ensure that the overall functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
Pressurizer Water Level	M	R
Auxiliary Feedwater Flow Indication*	M	R
Reactor Coolant System Subcooling Margin Monitor	M	R
PORV Position Indicator	M	R
PORV Block Valve Position Indicator	M	R
Safety Valve Position Indicator	M	R
Containment Pressure (Wide Range)	M	R
Steam Generator Water Level (Narrow Range)	M	R
Refueling Water Storage Tank Water Level	M	R
Containment Sump Water Level (Narrow Range)	M	R
Containment Water Level (Wide Range)	M	R
Neutron Flux (Wide Range)	M	R**

* See footnote of Table 3.5.6-1.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) an their associated block valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the reliability of the PORVs and block valves.

SPECIFICATION:

A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and

2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is being maintained closed in order to meet the requirements of Specification 3.1.5.A.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the getting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.7 PRESSURIZER

APPLICABILITY: Applies to pressurizer heaters and pressurizer water level for MODES 1, 2 and 3.

OBJECTIVE: To ensure proper pressurizer water volume and to ensure the capability to energize the pressurizer heaters from the emergency diesel generator.

SPECIFICATION:

A. The pressurizer water level shall be determined to be between 5% and 70% at least once per 12 hours.

B. The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal supply to the emergency diesel generator and energizing the heaters.

BASIS: The requirement that the pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency diesel generator provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

REFERENCES:

(1) NRC Letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.8 AUXILIARY FEEDWATER INSTRUMENTATION

APPLICABILITY: Applies to the instruments shown in Table 4.1.8-1.

OBJECTIVE: To ensure reliability of automatic initiation of the auxiliary feedwater pumps.

SPECIFICATION: A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the surveillance requirements specified in Table 4.1.8-1.

BASIS: The surveillance requirements specified for this instrumentation ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.8-1

AUXILIARY FEEDWATER INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
a. Manual	N/A	N/A	N/A	R	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M	N/A	1, 2, 3
c. Steam Generator Water Level-Low	S	R	M	N/A	1, 2, 3

4.1.9 AUXILIARY FEEDWATER SYSTEM SURVEILLANCE

APPLICABILITY: Applies to the motor driven auxiliary feedwater pump, the turbine driven auxiliary feedwater pump, and auxiliary feedwater valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the reliability of the auxiliary feedwater system.

SPECIFICATION:

A. Each auxiliary feedwater pump shall be demonstrated OPERABLE by testing each pump in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

B. At least once per 31 days an inspection shall be made to verify that each non-automatic valve in the emergency flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

C. Each auxiliary feedwater pump shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal. Within 72 hours after entering MODE 3, the steam driven auxiliary feedwater pump shall be similarly tested.

D. When the reactor coolant system pressure remains less than 500 psig for a period longer than thirty (30) days flow test shall be performed to verify the emergency flow path from the auxiliary feedwater storage tank to each steam generator, using the motor driven auxiliary feedwater pump prior to increasing reactor coolant system pressure above 500 psig. The flow test shall be conducted with the auxiliary feedwater system valves in their emergency alignment. Within 72 hours after entering MODE 3, the steam driven auxiliary feedwater pump shall be similarly tested.

BASIS: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

The electric driven auxiliary feedwater pump and the steam driven auxiliary feedwater pump are both capable of delivering a total feedwater flow of 165 gpm at a pressure of 1015 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

REFERENCES:

(1) NRC letter dated July 2, 1980 from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.10 AUXILIARY FEEDWATER STORAGE TANK SURVEILLANCE

APPLICABILITY: Applies to the auxiliary feedwater storage tank for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of an adequate auxiliary feedwater supply.

SPECIFICATION: A. The auxiliary feedwater storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

BASIS: See Basis for 3.4.4.

4.1.11 RADIATION MONITORING INSTRUMENTATION

APPLICABILITY: As shown in Table 4.1.11-1.

OBJECTIVE: To ensure the reliability of the radiation monitoring instrumentation.

SPECIFICATION: A. Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.11-1.

BASIS: The surveillance requirements specified for these systems ensure that the overall functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

REFERENCES: (1) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 4.1.11-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	APPLICABLE MODES
1. AREA MONITORS				
a. Spent Fuel Pool Area (R-1236)	S	R	M	*
b. Control Room Area (R-1231)	S	R	M	All
c. Containment Radiation Monitor-High Range (R-1255, R-1257)	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Wide Range Gas Monitor (R-1254)	**	**	**	1, 2, 3 & 4
b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B)	S	R	M	1, 2, 3 & 4

* See footnote of Table 3.5.10-1

** In accordance with Table 4.1.3.1 surveillance requirements for this instrument channel.

4.1.12 REACTOR COOLANT SYSTEM VENTS

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure the reliability of the reactor coolant vent system.

SPECIFICATION: Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.
3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN.

BASIS: See basis for 3.1.7, Reactor Coolant System Vents.

REFERENCES: NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG 0737 Technical Specifications (Generic Letter No. 83-37).

4.1.13 LEAKAGE AND LEAKAGE DETECTION SYSTEMS

APPLICABILITY: Applies to the reactor coolant leakage and detection systems delineated in Specification 3.1.4.

OBJECTIVE: To ensure the reactor coolant system leakage limits are maintained and to ensure the OPERABILITY of those systems that are used to detect leakage from the reactor coolant system.

SPECIFICATION: A. Reactor Coolant System leakage shall be demonstrated to be within limits by:

1. Monitoring the containment atmosphere radioactivity at least once per 12 hours.

2. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

3. Monitoring the steam generator blowdown effluent radioactivity at least once per 12 hours.

4. Monitoring the containment sump level indicator (LIS 2001 or 3001)at least once per 12 hours.

B. The leakage detection systems shall be demonstrated OPERABLE by the performance of CHANNEL CHECK, SOURCE CHECK, CHANNEL TEST, and CHANNEL CALIBRATION at the frequencies specified in Table 4.1.13-1;

BASIS: The monitoring of reactor coolant system leakage and maintenance of OPERABILITY of the reactor coolant leakage detection systems will assure that the sources of leakage are monitored and/or identified. The methods described above provide an acceptable means of verifying the OPERABILITY required by Specification 3.1.4.

REFERENCES:

1. SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage, NUREG-0829, December 1986

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973

3. Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 4, NUREG-0452

TABLE 4.1.13-1

LEAKAGE DETECTION SYSTEMS

	INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL TEST	CHANNEL CALIBRATION
1.	Containment Atmosphere Particulate Monitor (R1211)	D	M	N/A	R
2.	Containment Atmosphere Gaseous Monitor (R1212)	*	*	*	*
3.	Sphere Sump Level Control System (LS80 and 82)	N/A	N/A	N/A	R
4.	Containment Sphere Sump Level Monitor (LIS 2001 and 3001)	**	N/A	N/A	**
5.	Steam Generator Blowdown Effluent Monitor (R1216)	***	***	***	***

* In accordance with Table 4.1.3.1, surveillance requirements for this instrument channel.
** In accordance with Table 4.1.5-1, surveillance requirements for these instrument channels.
*** In accordance with Table 4.1.2.1, surveillance requirements for this instrument channel.

4.2 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM

4.2.1 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Safety Injection System and the Containment Spray System.

OBJECTIVE: To verify that the Safety Injection System and the Containment Spray System will respond promptly and properly if required.

SPECIFICATION: I. System Tests

A. Hot Safety Injection System Test

(1) When the plant is planned to be shutdown from MODE 1 operation and is planned to enter MODE 5 operation, a Hot SIS Test shall be performed in MODE 3 while RCS pressure is above 1500 psi but not more often than once every 9 months. The test shall include a determination of the force required to open valves NV 851 A and 8 and the margin of available actuation force.

(2) The test will be considered satisfactory if:

(a) control board indication and visual observations indicate all components have operated and sequenced properly. That is, the appropriate pumps have started and/or stopped and started, and all valves have completed their travel.

(b) the measured actuator force for both the HV-851 A and B valves is equal to or less than 10,000 lb_f.*

(3) If the measured actuator force of either HV-851 A or B is between 10,000 and 22,000 lb_f, the HV-851 A and B valves shall be considered OPERABLE but the future testing interval shall be accelerated as determined by the following equation:

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*Upon receipt of satisfactory data from continuing testing and analysis, the NRC staff will consider a request from Southern California Edison Company to change this number to more accurately reflect existing conditions.

$$T = T_L \frac{(22,000 - F)}{12,000}$$

where: T = maximum time in days of operation allowed before next surveillance test is required

T_L = time in days of operation since the last surveillance test

F = measured actuator force

(4) If the measured actuator force of either HV-851 A or B is greater than 22,000 lb_f, test results shall be reported to the NRC pursuant to Specification 6.9.2 along with proposed corrective actions. NRC approval shall be obtained prior to returning the unit to service.

B. Trisodium Phosphate Test

(1) A test of the trisodium phosphate additive shall be conducted once every refueling to demonstrate the availability of the system. The test shall be performed in accordance with the following procedure:

(a) The three (3) storage racks are visually observed to have maintained their integrity.

(b) The three (3) racks, each with a storage capacity of 1800 pounds of anhydrous trisodium phosphate additive, are visually observed to be full.

(c) Trisodium phosphate from one of the sample storage racks inside containment shall be submerged without agitation, in 25±0.5 gallons of 150°F to 175°F distilled water borated to 3900±100 ppm boron.

(2) The test shall be considered satisfactory if the racks have maintained their integrity, the racks are visually observed to be full, and the trisodium phosphate dissolves to the extent that a minimum pH of 7.0 is reached within 4 hours of the start of the test.

C. Containment Spray System Test

(1) During reactor shutdown at intervals not longer than the normal plant refueling intervals, a "no-flow" system test shall be conducted to demonstrate proper availability of the system. The test shall be performed either by closing a manual valve in the system or electrically disabling the refueling water pumps and initiating the system by tripping the normal actuation instrumentation.

(2) The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

(3) At least once every second refueling outage an air flow test shall be performed to demonstrate the absence of blockage at each containment spray nozzle.

II. <u>Component Tests</u>

A. Pump Tests

(1) In addition to the above test, the safety injection, recirculation, spray additive and refueling water pumps shall be started at intervals not to exceed one month to verify that they are in satisfactory running order.

(2) Acceptable levels of performance shall be as follows:

(1) The safety injection pumps shall reach and be capable of maintaining 95% of their rated shutoff head within 10 seconds after starting.

(2) The refueling water pumps shall be capable of maintaining 90% of their rated shutoff head.

(3) The recirculation pumps shall be run dry. Proper starting of the pump is confirmed by observation of the running current on the ammeter.

(4) The spray additive pumps shall be capable of maintaining their rated flow at a discharge pressure not less than 90% of their rated discharge pressure.

B. Leakage Testing

(1) The recirculation loop outside containment (including the Containment Spray System) shall be pressurized at a pressure equal to or greater than the operating pressure under accident conditions at intervals not to exceed the normal plant refueling interval. Visual inspections for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months. If leakage can be detected, measurements of such leakage shall be made.

(2) The non-redundant Containment Spray System piping shall be visually inspected at intervals not to exceed the normal plant refueling interval. Observations made as part of compliance with Paragraph C, above, or Paragraph I.C(2) of Technical Specification 4.2 will be acceptable as visual inspection of portions of non-redundant Containment Spray System piping.

<u>BASIS</u>: The Safety Injection System is a principal plant safeguard. It provides means to insert negative reactivity and limits core damage in the event of a loss of coolant or steam break accident.[(1)(2)(3)]

Preoperational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper dynamic functioning of the system. Thereafter, periodic tests demonstrate that all components are functioning properly. For these tests, flow through the system is generally not required. However, in the case of the "Hot SIS Test," actual conditions of an SI event are simulated. This test is performed to assure that long-term set of the valve seat faces on HV-851 A and B has not caused the valves to become inoperable. The test is required to be performed as the plant is shutting down from MODE 1 in order to assure that the valves have not been disturbed (i.e., the long-term set is still in effect) and that full dynamic conditions that would occur during an actual SI event are simulated. When possible the test should be performed prior to stopping the feedwater pumps (this is not a requirement). This will further assure that the valves will be in the same condition as when required for an actual Safety Injection event since the discharge pressure of the feedwater pumps acting on the valves will keep them seated even considering any backpressure built up in the downstream SI header. The equation used to determine future intervals if actuator force is between 10,000 lb_f and 22,000 lb_f is developed by shortening the interval in direct proportion to the degree to which the force exceeds 10,000 lb_f. During the test, all components are verified to have operated and sequenced properly.

The tests required in this specification will demonstrate that all components which do not normally and routinely operate will operate properly and in sequence if required. The portion of the Recirculation system outside the containment sphere is effectively an extension of the boundary of the containment. The measurement of the recirculation loop leakage ensures that the calculated EAB 0-2 hr. thyroid dose does not exceed 10 CFR 100 limits.

The trisodium phosphate stored in storage racks located in the containment is provided to minimize the possibility of stress corrosion cracking of metal components during operation of the ECCS following a LOCA. The trisodium phosphate provides this protection by dissolving in the sump water and causing its final pH to be raised to 7.0 - 7.5. The requirement to dissolve trisodium phosphate from one of the sample storage racks in distilled water heated and borated, to the extent recirculating post LOCA sump water is projected to be heated and borated, provides assurance that the stored trisodium phosphate will dissolve as required following a LOCA. The sample storage racks are sized to contain 0.5 pounds of trisodium phosphate. Trisodium phosphate stored in the sample storage racks has a surface area to volume ratio of 1.33 whereas the trisodium phosphate stored in the main racks has a surface area to volume ratio of 1.15.

Visual inspection of the non-redundant piping in the Containment Spray System provides additional assurance of the integrity of that system.

REFERENCES:

(1) Final Engineering Report and Safety Analysis, Paragraph 5.1.

(2) "San Onofre Nuclear Generating Station", report forwarded by letter dated December 29, 1971 from Jack B. Moore to Director, Division of Reactor Licensing, USAEC, subject: Emergency Core Cooling System Performance, San Onofre Nuclear Generating Station, Unit 1.

(3) USAEC Safety Evaluation of ECCS Performance Analysis for San Onofre Unit 1, forwarded by letter dated March 6, 1974 from Mr. Donald J. Skovholt to Mr. Jack B. Moore.

(4) Letter, K. P. Baskin, SCE, to D. M. Crutchfield, NRC, dated October 16, 1981.

4.2.2 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

APPLICABILITY: Applies to the operational status of the primary coolant system pressure isolation valves during MODES 1, 2 and 3.

OBJECTIVE: To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

SPECIFICATION: 1. Periodic leakage testing(a) on each valve listed in Table 3.3.5-1 shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished n accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. The minimum test differential pressure shall not be less than 150 psid.

4.3 CONTAINMENT SYSTEMS

4.3.1 CONTAINMENT TESTING

APPLICABILITY: Applies to containment leakage.

OBJECTIVE: To verify that leakage from the containment sphere is maintained within specified values.

SPECIFICATION: 1. Integrated Leakage Rate Tests, Type A

A. Test Pressure

In order to verify leakage from the containment sphere, a Type A test shall be performed. Type A tests shall consist of a peak pressure test or a reduced pressure test.

Peak pressure tests are conducted at a test pressure greater than or equal to 49.4 psig, and reduced pressure tests are conducted at a test pressure greater than or equal to 24.7 psig.

B. Acceptance Criteria

For the peak pressure test program the containment sphere leakage rate measured is less than 0.090 wt%/24 hours of the initial content of the containment air at the calculated peak pressure of 49.4 psig. For the reduced pressure test program to be conducted at 24.7 psig, the measured leakage rate shall be less than 0.064 wt%/24 hours of the initial content of the containment atmosphere at the calculated peak pressure of 49.4 psig.

The accuracy of each Type A test is verified by a supplemental test which (1) confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 25% of 0.12 wt%/24 hours for the peak pressure test or 0.085 wt%/24 hours for the reduced pressure test, and (2) requires the quantity of air bled from or injected into the containment during the supplemental test to be equivalent to at least 75 percent of the total allowable leakage rate at 49.4 psig.

C. Frequency

A set of 3 periodic Type A tests are performed at 40 ± 10 month intervals during each 10-year service period. The third test of each set is performed when the plant is shut down for the 10-year plant inservice inspection. The permissible period for Type A testing shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition.

If any periodic Type A test fails to meet the acceptance criteria above, the test schedule applicable to subsequent Type A tests shall be submitted to the NRC for review and approval. If two consecutive periodic Type A tests fail to meet the above acceptance criteria, a Type A test is performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria, after which time the normal test schedule may be resumed.

II. Containment Penetration Leakage Rate Tests (Type B)

A. Test Pressure

Type B tests are conducted at a test pressure at or above 49.4 psig. Personnel airlocks are tested every six months at or above 49.4 psig. In addition, a lower pressure test at or above 10 psig is performed on the personnel airlocks as required by Section II.C.

B. Acceptance Criteria

The combined leakage rate of all penetrations subject to Type B tests and all containment isolation valves subject to Type C tests is less than 0.072 wt%/24 hours of the initial content of the containment atmosphere at the calculated peak pressure of 49.4 psig.

C. Test Schedule

Type B tests, except for airlocks, are performed during every reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years.

Airlock volumes between the doors are tested:

(1) at least every six months and

(2) within 72 hours following each closing, except when the airlock is being used for multiple entries, then at least once per 72 hours, at or above 10 psig test pressure, and

(3) prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the airlock that could affect the airlock sealing capability at 49.4 psig.*

III. Containment Isolation Valve Leakage Rate Tests (Type C)

A. Test Pressure

Type C tests are conducted in accordance with the criteria specified in Appendix J of 10 CFR 50.

These Type C tests are conducted at a test pressure at or above 49.4 psig.

B. Acceptance Criteria (Maximum acceptable value)

The combined leakage rate of all penetrations subject to Type B tests and all containment isolation valves subject to Type C tests is less than .072 wt%/24 hours of the initial content of the containment atmosphere at the calculated peak pressure of 49.4 psig.

C. Test Schedule

Type C tests are performed during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years.

Seal tests conducted on active containment ventilation isolation valves shall be performed every three months.

Seal tests conducted on passive containment ventilation isolation valves shall be performed every six months.

IV. Recirculation System

A. Test Pressure

Leak tests shall be performed on portions of the Safety Injection System used for recirculation at a pressure equal to or greater than the operating pressure under accident conditions. The test fluid shall be water.

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*Exemption to Appendix J of 10 CFR 50

B. <u>Acceptance Criteria</u>

Visual inspection for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. The maximum effective leakage shall be maintained in accordance with Section 3.3.1.A(4) of Appendix A Technical Specifications.

C. <u>Test Schedule</u>

Visual inspections of the recirculation loop outside containment (including the Containment Spray System)) shall be made at intervals not to exceed the normal plant refueling interval. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months.

V. <u>Test Result Report</u>

The results of Type A, B, and C leakage rate tests are submitted to the NRC in a summary technical report approximately three months after the conduct of the Type A tests. This report contains an analysis and interpretation of the Type A test results and a summary of periodic Type B and C tests performed since the last Type A test. Leakage rate test results from Type A tests that fail to meet the acceptance criteria specified in Section I.B above are reported in a separate attached summary report that includes an analysis of the test data, an instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to failure in meeting the acceptance criteria. Results and analysis of the supplemental verification test used to demonstrate the validity of the Type A test measurements are included.

VI. <u>Containment Modification</u>

Any major modification or replacement of a component that is part of the containment boundary is followed by Type A, B, or C tests as applicable. The results of such tests are included in the test result report described above and meet the respective acceptance criteria. Minor modifications or replacements performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

BASIS: The containment system is one of the major engineered safety features and is a consequence-limiting system, it represents the final physical barrier that, in the event of a loss-of-coolant accident (LOCA), protects against the inadvertent release of fission products.

I. Leakage Rate Testing

Periodic containment integrated leakage rate tests are performed at or above 49.4 psig or at or above 24.7 psig for the reduced pressure test program. The leak rate will be calculated using the formulas of Reference 2 (Total Time) and Reference 3 (Mass Point).

Test schedules and the acceptance criteria specified herein are established based on the requirements of 10 CFR 50, Appendix J.(1) A containment leakage rate of 0.12 wt% of the initial content of containment atmosphere at 49.4 psig/24 hours maintains public exposure well below 10 CFR 100 values in the event of a hypothetical LOCA.(4) This leakage rate also limits public exposure to 10 CFR 100 values even if a complete core meltdown is postulated.

The acceptance criteria for

(1) Type A test is 75% of the containment leakage rate specified above

(2) Type B and Type C tests combined is 60% of the containment leakage rate specified above.

to allow for possible deterioration of the containment boundary between tests.

II. Recirculation System Testing

The portion of the Recirculation system outside the containment sphere is effectively an extension of the boundary of the containment.

Leakage from this system shall be maintained at as low as practical levels. The effective leakage of this system shall be maintained in accordance with the maximum leakage limitations established in Section 3.3.1.A(4) of Appendix A Technical Specifications.

The piping configurations of the recirculation and containment spray lines assure that leakage within Technical Specification limits will not deplete the isolation valve seal water system fluid inventory for at least 30 days at a pressure of 1.10 Pa. Therefore, leakage from the isolation valves and containment penetrations for these systems is not added to the combined leakage rate for all penetrations and valves subject to Type B and C tests.

The containment penetrations encompassed by the recirculation and containment spray systems include penetrations for one containment spray line, three reactor coolant pump seal water injection lines, and the recirculation pump discharge line to the recirculation heat exchanger.

REFERENCES:

(1) 10 CFR 50, Appendix J.

(2) ANSI N45.4-1972

(3) ANSI/ANS 56.8-1981

(4) Final Engineering Report and Safety Analysis, Paragraph 5.3

4.3.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: Applies to the containment isolation valves listed in Table 3.6.2-1 for MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure reliability of containment isolation valves.

SPECIFICATION:

A. The isolation valves specified in Table 3.6.2-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test.

B. Each isolation valve specified in Table 3.6.2-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

1. Verifying that on containment isolation test signal, each automatic isolation valve actuates to its isolation position.

2. Verifying that on a containment radiation-high test signal, each purge supply and purge outlet automatic valve actuates to its isolation position.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

REFERENCES:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.3.3 HYDROGEN MONITORS AND HYDROGEN RECOMBINERS

APPLICATION: MODES 1 and 2.

OBJECTIVE: To ensure reliability of the hydrogen monitors and hydrogen recombiners required for the detection control of hydrogen gas.

SPECIFICATION:

A. Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

1. One volume percent hydrogen, balance nitrogen.
2. Four volume percent hydrogen, balance nitrogen.

B. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per year by verifying that a heater sheath temperature of at least 1225 ± 10°F can be attained.

C. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.
2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits or foreign materials, etc.), and
3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the test in Specification B above. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

BASIS: The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with radiolytic decomposition of water and corrosion of metals within containment.

These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentration in Containment Following a LOCA," March, 1971.

REFERENCES:

(1) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment following a LOCA," March, 1971."

(2) NRC letter dated November 1, 1983, from D. E. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

4.4 EMERGENCY POWER SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Emergency Power System.

OBJECTIVE: To verify that the Emergency Power System will respond promptly and properly when required.

SPECIFICATION:

A. The required offsite circuits shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and power availability.

B. The required diesel generators shall be demonstrated OPERABLE:

1. At least once per 31 days on a STAGGERED TEST BASIS by:

   a. Verifying the diesel performs a DG SLOW START from standby conditions,

   b. Verifying a fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

   c. Verifying the diesel generator is synchronized and running at 4500 kW ± 5% for $\geq$ 60 minutes, to include a brief load increase to 5250 kW ± 5%,

   d. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses,

   e. Verifying the day tank contains a minimum of 290 gallons of fuel, and

   f. Verifying the fuel storage tank contains a minimum of 37,500 gallons of fuel.

2. At least once per 3 months by verifying that a sample of diesel fuel from the required fuel storage tanks is within the acceptable limits as specified by the supplier when checked for viscosity, water and sediment.

C. AC Distribution

1. The required buses specified in Technical Specification 3.7, Auxiliary Electrical Supply, shall be determined OPERABLE and energized from AC sources other than the diesel generators with tie breakers without automatic SIS/SISLOP tripping circuitry open between redundant buses at least once per 7 days by verifying correct breaker alignment and power availability.

D. The required DC power sources specified in Technical Specification 3.7 shall meet the following:

1. Each DC Bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability.

2. Each 125 volt battery bank and charger shall be demonstrated OPERABLE:

   a. At least once per 7 days by verifying that:

      (1) The parameters in Table 4.4-1 meet the Category A limits, and

      (2) The total battery terminal voltage is greater than or equal to 129 volts on float charge.

   b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

      (1) The parameters in Table 4.4-1 meet the Category B limits,

      (2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

      (3) The average electrolyte temperature of ten connected cells is above 61°F for battery banks associated with DC Bus No. 1 and DC Bus No. 2 and above 48°F for the UPS battery bank.

   c. At least once per 18 months by verifying that:

      (1) The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

      (2) The cell-to-cell and terminal connections are clean, tight and coated with anticorrosion material,

      (3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms,

(4) The battery charger for 125 volt DC Bus No. 1 will supply at least 800 amps DC at 130 volts DC for at least 8 hours,

(5) The battery charger for 125 volt DC Bus No. 2 will supply at least 45 amps DC at 130 volts DC for at least 8 hours, and

(6) The battery charger for the UPS will supply at least 10 amps AC at 480 volts AC for at least 8 hours as measured at the output of the UPS invertor.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80%, 85% for Battery Bank No. 1, of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.4.D.2.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

E. The required Safety Injection System Load Sequencer shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS, by simulating SISLOP* conditions and verifying that the resulting interval between each load group is within ± 10% of its design interval.

F. The required diesel generators and the Safety Injection System Load Sequencer shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Simulating SISLOP*, and:

   a. Verifying operation of circuitry which locks out non-critical equipment,

   b. Verifying the diesel performs a DG FAST START from standby condition on the auto-start signal, energizes the emergency buses with permanently connected loads and the auto connected emergency loads** through the load sequencer (with the exception of the feedwater, safety injection, charging and refueling water pumps whose respective breakers may be racked-out to the test position) and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads,

   c. Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed.

3. Verifying the generator capability to reject a load of 3220 kW without tripping.

<u>BASIS</u>:

The normal plant Emergency Power System is normally in continuous operation, and periodically tested.(1)

The tests specified above will be completed without any preliminary preparation or repairs which might influence the results of the test except as required to perform the DG SLOW START test set forth in T.S. 4.4.B.1.a. The tests will demonstrate that components which are not normally required will respond properly when required. Test loading of the generator to 4500 kW and 5250 kW corresponds to approximate engine brake mean effective pressures of 116 psi and 135 psi, respectively.

DG SLOW STARTS are specified for the monthly surveillances in order to reduce the cumulative fatigue damage to the engine crankshafts to levels below the threshold of detection under a program of augmented inservice inspection. In the event that the DG SLOW START inadvertently achieves steady state voltage and frequency in less than 24 seconds, the surveillance will not be considered a failure and require restart of the diesel generator.

* SISLOP is the signal generated by coincident loss of offsite power (loss of voltage on Buses 1C and 2C) and demand for safety injection.

** The sum of all loads on the engine shall not exceed 5250 kW + 5%.

TABLE 4.4-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable [3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$1/4" above maximum level indication mark	>Minimum level indication mark, and $\leq$1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$2.13 volts	$\geq$2.13 volts [c]	>2.07 volts
Specific[a] Gravity	$\geq$1.200[b]	$\geq$1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells >1.205	Average of all connected cells $\geq$1.195[b]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature in accordance with IEEE STD 450-1980.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameter(s) are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

The monthly surveillance specified by T.S. 4.4.B.1.c includes a "brief" load increase to 5250 kW ± 5%. This requirement verifies the ability to function under the maximum possible loading conditions. A "brief" test is required to demonstrate operability while minimizing the increased stresses from the higher load. The duration of this test is expected to be approximately three to five minutes.

The surveillance requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensure the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.4-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .020 below normal full charge specific gravity or a battery charger current that has stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below normal full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below normal full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operating with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.4-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below normal full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below normal full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

REFERENCE:

(1) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 3, Questions 6 and 8.

4.5 RADIOACTIVE LIQUID EFFLUENTS

4.5.1 LIQUID EFFLUENTS CONCENTRATION

APPLICABILITY: At all times.

OBJECTIVE: To verify that discharge of radioactive liquid material to UNRESTRICTED AREAS is maintained below 10 CFR 20 limits.

SPECIFICATION:

A. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.5.1.1.

B. The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.15.1.

BASIS: This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR Part 20.106(e) to the population.

TABLE 4.5.1.1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)
A. Batch Waste Release Tanks (1) Holdup Tanks(b) (2) Monitor Tanks(b) (3) Sewage Sludge (Offsite Shipment)	P Each Batch	P Each Batch	Principal Gamma Emitters(c)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite(d)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite(d)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases(e) (1) Steam Generator Blowdown (2) Reheater Pit sump (3) Yard Drain Sump	3 x W Grab Sample	W Composite(f)	Principal Gamma Emmiters(c)	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	3 x W Grab Sample	M Composite(f)	H-3 Gross Alpha	1×10^{-5} 1×10^{-7}
	3 x W Grab Sample	Q Composite(f)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.5.1.1
(Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\ \Delta t)}$$

where,

LLD is "a priori" lower limit of detection as defined above (as microcuries per unit mass or volume).

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per disintegration),

V is the sample size (in units of mass or volume),

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide,

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting,

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a</u> <u>priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a</u> <u>posteriori</u> (after the fact) limit for a particular measurement.

b. A batch release is the dicharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

d. A composite sample is one which results in a specimen that is representative of the liquids released.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

f. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

4.5.2 LIQUID EFFLUENT DOSE

APPLICABILITY: At all times.

OBJECTIVE: To verify that doses due to the release of radioactive liquid effluents are as low as is reasonably achievable.

SPECIFICATION: Cumulative does contributions from liquid effluents shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM) at least once per 31 days.

BASIS: This specification is provided to implement the requirements of Section III.A of Appendix I, 10 CFR Part 50. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated.

4.5.3 LIQUID WASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: To verify the operability and potential use of the liquid radwaste treatment system.

SPECIFICATION: Doses due to liquid releases shall be projected at least once per 31 days in accordance with the ODCM.

BASIS: The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in the liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

4.6 RADIOACTIVE GASEOUS WASTE RELEASE

4.6.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose rate due to the discharge of radioactive gaseous effluents is maintained within 10 CFR 20 limits.

SPECIFICATION:

A. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM.

B. The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.6.1.1.

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive materials discharged in gaseous effluents will not result in the exposure of an individual in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 [10 CFR 20.106(b)]. For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above "background to an individual at or beyond the exclusion area boundary to $\leq$ 500 mrem/year to the total body or to $\leq$ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to $\leq$ 1500 mrem/year for the nearest cow to the plant.

TABLE 4.6.1.1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(b)	1×10^{-4}
B. Containment Purge	P Each Purge(c) Grab Sample	P Each Purge(c)	Principal Gamma Emitters(b)	1×10^{-4}
			H-3	1×10^{-6}
C. Plant Stack	M(c) Grab Sample	M(c)	Principal Gamma Emitters(b)	1×10^{-4}
			H-3(d,e)	1×10^{-6}
	Continuous(f)	W(g) Charcoal Sample	I-131	1×10^{-12}
	Continuous(f)	W(g) Particulate Sample	Principal Gamma Emitters (b) (I-131, Others)	1×10^{-11}
	Continuous(f)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(f)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous(f)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.6.1.1
(Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\ \Delta t)}$$

Where,

LLD is the "a priori" lower limit of detection as defined above (as microcuries per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per disintegration),

V is the sample size (in units of mass or volume),

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide,

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

b. The principal gamma emitters for which the LLD specification applies are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

c. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within one hour unless (1) analysis shows the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

d. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the timer period covered by each dose or dose rate calculation made in accordance with Specifications 3.16.1, 3.16.2, and 3.16.3.

g. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that the effluent activity has not increased more than a factor of 3.

4.6.2 DOSE NOBLE GASES

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose due to noble gases in radioactive gaseous effluents is maintained as low as is reasonably achievable.

SPECIFICATION: Cumulative dose contributions for noble gases for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days..

BASIS: This specification implements the requirements in Section III.A of Appendix I, 10 CFR Part 50, that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through the appropriate pathways is unlikely to be substantially underestimated. The ODCM equations provided for determining the air doses at the SITE BOUNDARY will be based upon the historical average atmospheric conditions.

4.6.3 DOSE, IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose due to iodine-131, iodine-133, tritium an radionuclies in particulate form with half-lives greater than 8 days is maintained as low as is reasonably achievable.

SPECIFICATION: Cumulative does contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the ODCM at least once per 31 days.

BASIS: This specification implements the requirements in Section III.A of Appendix I, 10 CFR Part 50, that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated.

The ODCM equations provided for determining the actual doses are based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation and subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

4.6.4 GASEOUS RADWASTE TREATMENT

APPLICABILITY: At all times.

OBJECTIVE: To verify the OPERABILITY and Potential use of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM.

SPECIFICATION: Doses due to gaseous releases from San Onofre Unit 1 shall be projected at least once per 31 days in accordance with the ODCM.

BASIS: The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEMS ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50.

4.6.5 GAS STORAGE TANK

APPLICABILITY: At all times.

OBJECTIVE: To verify the quantity of radioactive material contained within the gas storage tanks.

SPECIFICATION: The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limit specified in Specification 3.16.5 at least once per 24 hours when radioactive materials are being added to the tank.

BASIS: Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

4.6.6 EXPLOSIVE GAS MIXTURE

APPLICABILITY: At all times.

OBJECTIVE: Limit the amount of explosive gases contained in the gas storage tanks.

SPECIFICATION: The concentrations of hydrogen and/or oxygen in the waste gas holdup system shall be determined to be within the limits specified in Specification 3.16.6 by analyzing grab samples of the waste gas holdup system contents at the waste gas decay tank in service daily and every 4 hours during degassing. Degassing is defined as the process to reduce reactor coolant system (RCS) dissolved H_2 gas concentration in preparation for refueling or for opening the reactor coolant system.

BASIS: This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the release of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

4.7 INSERVICE INSPECTION REQUIREMENTS

APPLICABILITY: Applies to Class 1, 2, and 3 pressure retaining components and their supports.

OBJECTIVE: To examine and/or test applicable components to ensure system integrity and/or operability.

SPECIFICATION: Commencing at the start of the second 120-month period of operation from start of commercial operation, the following inservice inspection requirements shall apply:

(1) Inservice inspection of 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 pursuant to 10 CFR 30, Section 50.55a(g)(6)(i).

BASIS: This specification ensures that inservice inspection of components 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 50.55a(g). Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specifications.

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

4.8 REACTIVITY ANOMALIES

APPLICABILITY: Applies to potential reactivity anomalies.

OBJECTIVE: To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION: A. Following a normalization of the computer 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 concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made within 30 days and reported to the NRC pursuant to Specification 6.9.2.

BASIS: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up 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 burn-up and reactivity compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. 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 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

4.9 REACTOR VESSEL SURVEILLANCE PROGRAM

APPLICABILITY: Applies to the reactor vessel irradiation surveillance capsules.

OBJECTIVE: To monitor the effect of radiation on the reactor vessel core region materials.

SPECIFICATION: Irradiation surveillance capsules shall be withdrawn in accordance with the following schedule:

Capsule Type	Capsule Identity	Capsule Withdrawal	Time to Removal
II	A	First	First refueling outage.
I	D	Second	Second refueling outage.
II	F	Third	First refueling outage following approximately 7 Effective Full Power Years (EFPY) of operation.
II	C	Fourth	First refueling outage following 14 EFPY of operation.
I	E	Fifth	First refueling outage following 23 EFPY of operation.

Results of the reactor vessel surveillance program shall be incorporated into the updating of Figures 3.1.3a and 3.1.3b of the Technical Specification 3.1.3, COMBINED HEATUP, COOLDOWN AND PRESSURE LIMITATIONS, in accordance with Specification 3.1.3.B.(2).

BASIS: Irradiation surveillance provides the capability of determining the radiation induced changes occurring in the mechanical properties in the region of the reactor pressure vessel surrounding the core.

The surveillance program is based on ASTM E 185-70, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors", and proposed Appendix H, "Reactor Vessel Material Surveillance Program Requirements", to 10 CFR Part 50. Test specimens of base metal, deposited weld metal, and the heat affected zone have been installed in the various capsule assemblies placed in sample holders alongside the reactor vessel.

Capsules A, D, F, C and E will give information equivalent to the complete life of the plant, and each will provide a check on the previous data points by the measurement of RT_{NDT} shift with neutron exposure.

REFERENCES: (1) Final Safety Analysis Report, Part II, Section 2.2.11.

4.10 AUGMENTED INSERVICE INSPECTION OF HIGH ENERGY LINES OUTSIDE CONTAINMENT

APPLICABILITY: Applies to welds in piping systems or portions of systems located outside containment where protection from the consequences of postulated pipe breaks is not provided by a system of pipe whip restraints, protective enclosures, or other measures specifically designed to cope with such breaks.

OBJECTIVE: To provide assurance of the continued integrity of the piping systems over their service lifetime.

SPECIFICATION: A. For the welds in the main steam, main feedwater, and first point extraction lines identified in Reference 1, Table 1 and Table IA, Column: "Break Point Location", for which inservice inspection is specified in Column: "Solution",

(1) At refueling outage No. 4, a baseline inspection consisting of a volumetric examination of all specified welds shall be performed in accordance with the requirements of ASME Section XI Code, "Inservice Inspection of Nuclear Reactor Coolant Systems" 1971, up to and including 1972 addenda.

(2) Subsequent to the baseline examination, the inservice inspection of each weld shall be performed in accordance with the requirements of the Edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" as required by Technical Specification 4.7, with the following schedule:

(a) During the first 3-1/3 years (or nearest refueling outage), volumetric examination of 100% all welds.

(b) Every 10 years thereafter (or nearest refueling outage), volumetric examination of 33-1/3% of the welds at the expiration of each 1/3 of the inspection interval with a cumulative 100% coverage of all welds every 10 years.

NOTE: The welds selected during each inspection Period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of all specified welds.

(3) Any evidence revealed by the examinations specified in (1) or (2) that indications have developed or grown shall be investigated, including evaluation of comparable areas of the applicable system. It may be determined that the condition can be tolerated or that repair is necessary. In the event that repair is required, restoration shall be governed by the original acceptance standards.

(4) In the event repair of any weld is required following an examination, the inspection schedule for the repaired weld will be changed to provide for its inspection at the following inspection.

B. For all welds in the main steam lines, main feedwater lines, and first point extraction lines located outside containment:

(1) A visual inspection of the surface at the insulation joints nearest to all weld locations shall be performed on a monthly basis for detection of leaks. Any detected leaks shall be investigated and evaluated. If the leakage is caused by a through-wall flaw, either the plant shall be shut down or the leaking piping isolated. Repairs shall be performed prior to return of this line to service.

<u>BASIS</u>: Under normal plant operating conditions, the piping materials operate under ductile conditions and within stress limits considerably below the ultimate strength properties of the materials. Flaws which could grow under such conditions are generally associated with cyclic loads which fatigue the metal and lead to cracks. The inservice examination and the frequency of inspection will provide a means for timely detection before the flaw penetrates the wall of the piping.

<u>REFERENCE</u>: (1) Report on the Effects of a Piping System Break Outside the Containment, December 1973, including November 1974 Addendum 1, and May 1975 Addendum 2.

4.11 CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM

APPLICABILITY: Applies to the testing and surveillance of the control room emergency air treatment system to determine OPERABILITY.

OBJECTIVE: To ensure that the control room emergency air treatment system will operate effectively if required.

SPECIFICATION: The control room emergency air treatment system shall be demonstrated OPERABLE:

A. At least once per refueling cycle, by verifying that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than 1.8 inches of water while operating the system at a flow rate of 900 cfm $\pm$ 10%.

B. At least once per year for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system, by verifying that:

(1) In-place cold DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorbers, at a system flow rate of 900 cfm $\pm$ 10%, show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

(2) The carbon adsorber is either replaced with an adsorbent meeting the physical property requirements of Table 5-1 of ANSI N509-1976, or a laboratory carbon sample analysis shows, within 31 days after removal, $\geq$ 90% radioactive methyl iodide removal when tested in accordance with ASTM D-3803, 1979.

(3) A system flow rate of 900 cfm $\pm$ 10% is shown when tested in accordance with ANSI N510-1975.

C. At least once per 31 days by initiating, from the control room, flow through the HEPA filter and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heater(s) on.

D. After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing by performing cold DOP testing. After each complete or partial replacement of the charcoal adsorbers or after any structural maintenance on the system housing by performing halogenated hydrocarbon testing.

E. At least once per refueling cycle by demonstrating automatic closure of the fresh air intake to the control room.

BASIS: Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 1.8 inches of water at flow rates near design levels (900 cfm ± 10%) indicates that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop is determined once per refueling cycle to verify 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. The tests are performed at a charcoal residence time consistent with the design of the filter unit (i.e., 1/8 second). The removal efficiencies stipulated are consistent with criteria established in the Final Safety Analysis and subsequent analysis; specifically, laboratory carbon test results shall meet the physical property requirements of Regulatory Guide 1.52, Table 2, and when applicable, all replaced adsorbent shall meet the physical property requirements of Table 5.1 of ANSI N509-1976 in conformance with Regulatory Position C.6.a of Regulatory Guide 1.52. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52.

Operation of the system for 10 hours every month with the heaters on will demonstrate operability of the system and serve to remove excessive moisture build-up on the adsorber.

Contaminants can be generated by painting, fire or chemical release. The fumes, chemicals or foreign materials produced could contaminate the filters or adsorbent if the release occurs in an area communicating with the system. Conducting the same tests as required at refueling intervals following a significant release of contaminants in a communicating area assures that system performance is not degraded.

4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

APPLICABILITY: Applies to the leakage of radioactive source materials.

OBJECTIVE: To verify the physical integrity of portable and fixed radioactive calibration sources.

SPECIFICATION:

A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR 30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specification B, C and O below.

Exception: Not withstanding the periodic leak test required by this specification, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements and reported in the subsequent annual report filed pursuant to Specification 6.9.1.4.

BASIS: This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39 (c) limits for plutonium.

4.13 TURBINE DECK LOAD BEARING TEST AND VISUAL INSPECTION

(DELETED)

4.14 SHOCK SUPPRESSORS (SNUBBERS) SURVEILLANCE

APPLICABILITY: Applies to surveillance of safety-related snubbers.

OBJECTIVE: To ensure the operability of safety-related snubbers.

SPECIFICATION: Each snubber shall be demonstrated OPERABLE by performance of the following augmented in-service inspection program and the requirements of Specification 3.13.

A. Visual Inspections

Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable per Inspection Period	Subsequent Visual Inspection Period*
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5,6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during normal reactor operation. Each group may be inspected independently in accordance with the above schedule. Further, the subgroups - mechanical and hydraulic - may be inspected independently.

B. 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, providing 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 4.14.D or 4.14.E, as applicable.

* The inspection interval shall not be lengthened more than one step at a time.

However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

C. Functional Tests

At least once per refueling cycle shutdown, a representative sample, 10% of the total, of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.14.D or 4.14.E, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers, exempting snubbers of capacity greater than 120,000 pounds, i.e., E-1A, E-1B and E-1C. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle;

2. Snubbers within five feet of heavy equipment (valve, pump, turbine, motor, etc.);

3. Snubbers within ten feet of the discharge from a safety relief valve

Snubbers that are especially difficult to remove or are located in high radiation zones during shutdown shall also be included in the representative sample.*

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

D. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

E. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel;

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression;

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

F. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service is based shall be maintained as required by Specification 6.10.2. Concurrent with the first in-service visual inspection and at least once per refueling cycle shutdown thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

BASIS: Refer to the Basis given in 3.13.

4.15 FIRE PROTECTION SYSTEMS SURVEILLANCE

APPLICABILITY: Applies to the surveillance of fire detection and extinguishing systems and equipment.

OBJECTIVE: To ensure the OPERABILITY of fire detection and extinguishing systems and equipment.

SPECIFICATION: A. The Fire Suppression Water System shall be demonstrated to be OPERABLE.

(1) With the San Onofre Unit 1 fire water pumps satisfying the pump requirements of Technical Specification 3.14.A(1), at least once per seven days by verifying the water supply volume in the San Onofre Unit 1 Service Water Reservoir. With the San Onofre Units 2 and 3 fire water pumps satisfying the pump requirements of Technical Specification 3.14.A(1), by initially verifying the water supply volume in the San Onofre Units 2 and 3 service and firewater storage tanks and at least once per seven days thereafter.

(2) At least once per 31 days on a STAGGERED TEST BASIS by starting each pump satisfying the pump requirements of Technical Specification 3.14.A(1) and operating it for at least fifteen minutes.

(3) At least once per thirty-one days by verifying that each valve (manual, power operated or automatic) is in its correct position. For valves located inside the containment sphere, verification shall be made consistent with the 31-day requirement when possible during available plant outages or during containment entrances for other reasons.

(4) At least once per 12 months by cycling each testable valve through one complete cycle of full travel.

(5) At least once per 18 months by performing a system functional test which includes simulated actuation of the system and:

a. Verifying that each valve in the flow path is in its correct position.

b. Verifying that each pump develops at least 90% of the flow and head at some point on the manufacturer's pump performance curves.

c. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

d. Verifying that each pump starts to supply the fire suppression water system at $\geq$50 psig.

(6) At least once per 36 months by performing flow tests of the system in accordance with Chapter 5, Section 11 of Fire Protection Handbook, 14th Edition, published by National Fire Protection Association.

B. The Spray and/or Sprinkler Systems indicated in Technical Specification 3.14.A(2) shall be demonstrated to be operable:

(1) At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel. For the valves located in the containment sphere, testing shall be performed consistent with the 12-month requirement when possible during available plant outages.

(2) At least once per 18 months.

a. By performing a system functional test which includes simulated automatic actuation of the system, and:

1. Verifying that the automatic valves in the flow path actuate to their correct positions on a smoke and infrared test signal, and

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

b. By inspection of the spray headers to verify their integrity, and

(3) By inspection of each nozzle at least once every refueling outage to verify no blockage.

(4) At least once every second refueling outage by performing an air flow test through each accessible spray/sprinkler header and verifying that the spray/sprinkler nozzles are unobstructed.

C. Each Fire Hose Station indicated in Table 3.14.1 shall be verified to be operable:

(1) At least once per 31 days by visual inspection of the station to assure all equipment is at the station. For the station located in the containment sphere, inspection shall be performed consistent with the 31 days requirement when possible during available plant outages or during containment entrances for other reasons.

(2) At least once per 18 months by removing the hose for inspection and re-racking and replacing all degraded gaskets in the couplings.

(3) At least once per 36 months, partially open each hose station valve to verify valve OPERABILITY and no blockage. For the hose station located in the containment sphere, this verification shall be performed every other refueling outage.

(4) At least once per 36 months conduct a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station. For the hose station located in the containment sphere, this test shall be performed every other refueling outage.

D. Each of the Fire Detection Instruments indicated in Table 3.14.2 shall be demonstrated to be OPERABLE:

(1) At least once per six months by performance of a channel functional test. For the instrumentation located in the containment sphere, the test shall be conducted consistent with the six-month requirement when possible during available plant outages.

BASIS: Refer to the Basis for Technical Specification 3.14.

REFERENCE: 1. Refer to Reference 1 for Technical Specification 3.14.

4.16 INSERVICE INSPECTION OF STEAM GENERATOR TUBING

APPLICABILITY: Applies to the inservice inspection and sampling selection for steam generator tubing.

OBJECTIVE: To monitor the integrity of the steam generator tube primary boundary and provide guidance for corrective action when imperfections are observed.

SPECIFICATION:

A. GENERAL STEAM GENERATOR TUBE SELECTION

The steam generators shall be inspected when shutdown by selecting steam generator tubes on the following basis:

1. Tubes for the inspection shall be selected on a random basis except where experience at San Onofre Unit 1 or experience in similar plants indicates critical areas to be inspected.

2. Each inspection shall include at least 3% of the total number of tubes in each steam generator to be inspected.

3. Inservice inspections may be limited to one steam generator on a rotating schedule encompassing 3% of the total tubes of steam generators in the plant if the results of previous inspections indicate that all steam generators are performing in a like manner.

4. Every inspection shall include all non-plugged tubes, in the steam generator(s) to be inspected, that previously had been classified as degraded tubes, except as specified in Specification C.1.

B. SUPPLEMENTARY INSPECTIONS

If the inspections in Specification A indicate imperfections, additional inspections shall be required as follows:

1. If any of the tubes inspected pursuant to Specification A.3 are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration since their last inspection, inspect 3% of the tubes in one of the uninspected steam generators.

2. If more than 10% of the tubes inspected in a steam generator are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration since their last inspection, or one or more of the tubes inspected have an imperfection in excess of the plugging limit, inspect an additional 3% of the tubes in that steam generator, concentrating on tubes in those areas of the tube sheet array where tubes with imperfections were found and on that length of tube where the imperfections were found. In addition, the rest of the steam generators shall be inspected in accordance with Specification A.2.

3. If the additional inspection in Specification B.2 indicates that more than 10% of the additionally inspected tubes are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration since their last inspection, or one or more of the additionally inspected tubes have an imperfection in excess of the plugging limit, inspect an additional 6% of the tubes in that steam generator in the area of the tubesheet array where tubes with imperfections were found and through that length of tube where the imperfections were found.

C. SPECIAL STEAM GENERATOR TUBE INSPECTIONS

In addition to the general steam generator tube inspections performed in Specifications A and B, every inspection shall include the following special inspections:

1. Every inspection shall include all nonplugged tubes in one of the steam generators that previously had been noted as having discretely quantifiable imperfections greater than 30% at antivibration bar (AVB) intersections, and all non-plugged tubes in that steam generator that previously had been noted as having imperfections at AVB intersections which were not discretely quantifiable but which were identified during previous inspections as being in the 30 to 50% range.

2. At each steam generator inspection, all previously identified restricted tubes in either steam generator A or C shall be gauged by using eddy current probes to determine restriction sizes.

D. INSPECTION FREQUENCY

The inspections in Specifications A and B above shall be performed at the following frequencies:

1. Inservice inspections shall be not less than 10 nor more than 24 calendar months after the previous inspection.

2. If two consecutive inspections indicate that less than 10% of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, the inspections shall be not less than 10 nor more than 40 calendar months after the previous inspection.

3. Unscheduled inspections shall be conducted in accordance with Specification A in the event of primary-to-secondary leaks exceeding Specification 3.1.4.C, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

E. ACCEPTANCE CRITERIA

1. As used in this specification:

   a. Imperfection means an exception to the dimensions, finish, or contour required by drawing or specification. Detectable eddy current testing indications below 20% of the nominal tube wall thickness, may be considered as imperfections.

   b. Degradation means a service-induced cracking, wastage, pitting, wear or general corrosion occurring on either inside or outside of a tube.

   c. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation above the tube roll expansion. Also, a tube with an imperfection in the region one-half inch below the uppermost one inch of sound roll, is considered a degraded tube.

   d. Defect means an imperfection of such severity that it exceeds the plugging limit or an imperfection located in the uppermost one inch of the tube roll expansion. A tube or sleeve containing a defect is defective.

e. <u>Plugging Limit</u> means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40% of the nominal sleeve wall thickness.

For the tube roll expansion region, the following criteria apply:

(i) Any tube containing an imperfection within the uppermost one inch of sound roll shall be considered defective.

(ii) Any imperfection is acceptable below the uppermost one inch of sound roll.

f. <u>Tube Roll Expansion</u> means that portion of the tube which has been increased in diameter by a rolling process.

g. <u>Sound Roll</u> means a tube roll expansion which is fully expanded such that no crevice exists between the outside diameter of the tube and the tubesheet.

2. If, in the inspections performed under Specification A,

a. Less than 10% of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

b. No tube inspected exceeds the plugging limit,

plant operation may resume.

3. If, in the inspections performed under Specification B,

a. Less than 10% of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, and

b. No more than 3 of the tubes inspected exceed the plugging limit,

plant operation may resume after performance of the corrective action in Specification F.

4. If, in the inspections performed under Specification B,

   a. More than 10% of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10% wall penetration, or

   b. More than 3 of the tubes inspected exceed the plugging limit,

   prior to resumption of plant operation, the situation shall be reported in a Special Report to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

5. The results of inspections performed under specifications A or B for all tubes in service which have defects below the uppermost one inch of tube roll expansion shall be reported to the Commission in a Special Report pursuant to Technical Specification 6.9.2 at least seven days prior to the resumption of plant operation. The report shall include:

   a. Identification of applicable tubes, and

   b. Location and size of the degradation.

6. If in the inspections performed under Specification C.1, wear rates are observed at AVB intersections which are inconsistent with the 50% plugging criterion, prior to resumption of plant operation, the situation shall be reported in a Special Report to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

7. If in the inspections performed under Specification C.2 progression of the denting process is observed to be recurring, prior to resumption of plant operation, the situation shall be reported in a Special Report to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

F. CORRECTIVE ACTION

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

BASIS: The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83, Revision 1. Inservice 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 degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. 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 plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = .3 gallons per minute per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of .3 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require shutdown during which the leaking tubes will be located and plugged and additional inspections performed.

If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50% of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 40% of the nominal sleeve wall thickness. A plugging limit of 50% for tubes and 40% for sleeves ensures that defects will not occur between inspection intervals. Tubes with defects below the uppermost one inch of a sound roll may remain in service, provided there are no imperfections in this portion of the tube. The distance of one inch includes a 0.25 inch eddy current measurement uncertainty.

The results of tube ID gauging and dent detection conducted in San Onofre Unit 1 steam generators demonstrate that the denting process has been arrested. Continuing assurance of this condition can be provided by performing a program of limited tube ID gauging and dent detection in either steam generator A or C on a refueling outage frequency. Adequate surveillance of denting related tube restrictions can be maintained at refueling intervals by noting any new restrictions during the conduct of general surveillance and AVB inspections and by gauging tubes which have previously been noted as being restricted. Progression of denting can also be monitored in either steam generator A or C by evaluating third and fourth support plate denting data obtained from the general surveillance and AVB inspections as well as from the ID gauging program and comparing these results with those of previous inspections.

The results of AVB area inspections conducted in San Onofre Unit 1 steam generators demonstrate that AVB modifications installed during the Cycle VI refueling outage were successful in eliminating significant growth of tube wall penetration indications at AVB locations. Continuing assurance of this condition can be provided by performing U-bend inspections at refueling outage intervals of tubes having wall penetration indications in excess of 30% at AVB locations.

4.17 DOSE

APPLICABILITY: At all times.

OBJECTIVE: To verify the doses due to liquid and gaseous effluents are maintained as low as is reasonably achievable.

SPECIFICATION: Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.15.2.A, 3.16.2.A and 3.16.3.A and in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

BASIS: This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix 1. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of the dose to a MEMBER OF THE PUBLIC for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of five miles must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation which is part of the nuclear fuel cycle.

4.18 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.18.1 Monitoring Program

APPLICABILITY: At all times.

OBJECTIVE: Ensure required actions of the radiological monitoring program are being performed.

SPECIFICATION: The radiological environmental monitoring samples shall be collected pursuant to Table 3.18.1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.18.1 and 4.18.1.

BASIS: The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling for the environmental exposure pathways.

The detection capabilities required by Table 4.18.1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as *a priori* (before the fact) limit representing the capability of a measurement system and not as *a posteriori* (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

TABLE 4.18.1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Marine Animals (pCi/kg, wet)	Local Crops (pCi/kg, wet)	Sediment (pCi/kG, dry)
gross beta	4	1×10^{-2}			
H-3	2000				
Mn-54	15		130		
Fe-59	30		260		
Co-58, 60	15		130		
Zn-65	30		260		
Zr-95	30				
Nb-95	15				
I-131	1[b]	7×10^{-2}		60	
Cs-134	15	5×10^{-2}	130	60	150
Cs-137	18	6×10^{-2}	150	80	180
Ba-140	60				
La-140	15				

TABLE 4.18.1
(Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\ \Delta t)}$$

where,

LLD is the "a priori" lower limit of detection as defined above (as picocurie per unit mass or volume).

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation).

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide,

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium -40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing capability of a measurement system and not as _a posteriori_ (after the fact) limit for a particular measurement.*

b. LLD for drinking water

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Curries, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93
(3) Hartwell, J. K., Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

4.18.2 LAND USE CENSUS

APPLICABILITY: At all times.

OBJECTIVE: Perform the land use census to ensure the monitoring program is appropriate for the surrounding areas.

SPECIFICATION: The land use census shall be conducted at least once per 12 months between the date of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agricultural authorities.

BASIS: This specification is provided to ensure that changes in the use of UNRESTRICTED AREAS are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

4.18.3 INTERLABORATORY COMPARISON PROGRAM

APPLICABILITY: At all times.

OBJECTIVE: To ensure laboratory analysis of radiological environmental monitoring samples is correct and accurate.

SPECIFICATION: A summary of the results obtained as part of the Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

BASIS: The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

4.19 SOLID RADIOACTIVE WASTE

APPLICABILITY: At all times.

OBJECTIVE: Ensure meeting the requirements for the SOLIDIFICATION and shipment of solid radwaste.

SPECIFICATION: The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

ACTION:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

BASIS: This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

4.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to OPERABILITY of the overpressurization protection systems.

OBJECTIVE: To verify that the overpressure protection systems will respond promptly and properly if required.

SPECIFICATION: A. Each power operated relief valve (PORV) shall be demonstrated operable by:

(1) Adjusting the pressure control bistable setpoint such that the PORVs are actuated and the annunciators alarm within 31 days prior to returning to a water-solid condition following a COLD SHUTDOWN with the RCS depressurized.

(2) Performance of a CHANNEL TEST within 31 days prior to enabling the low pressure overpressure mitigation setting of the pressurizer PORVs on cooldown.

(3) Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

(4) Verifying that position indications on the PORV isolation valves indicate that the valves are open at least once per week when the PORVs are being used for overpressure protection.

BASIS: The surveillance requirement to verify OPERABILITY of the PORVs provides assurance that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when the initial RCS pressure is $\leq$ 400 psig. Either PORV has adequate relieving capability to protect the RCS from overpressurization due to a design basis transient as discussed in Reference 1.

REFERENCE: (1) Letter to A. Schwencer from K. Baskin dated October 12, 1977.

5.1 SITE DESCRIPTION

The San Onofre Nuclear Generating Station is located on the West Coast of Southern California in San Diego County, about 62 miles southeast of Los Angeles and about 51 miles northwest of San Diego. The site is located within the U.S. Marine Corps Base, Camp Pendleton, California. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 shall be 283.5 meters from the outer edge of the Unit 1 containment sphere. For the purpose of dose assessment, a slightly reduced distance of 282 meters defined by the discontinuous line in Figure 5.1.1 is assumed.

BASIS: Leasing arrangements with the U.S. Marine Corps provide that a minimum distance to the exclusion boundary will be 283.5 meters. All dose assessments are calculated assuming 282 meters.

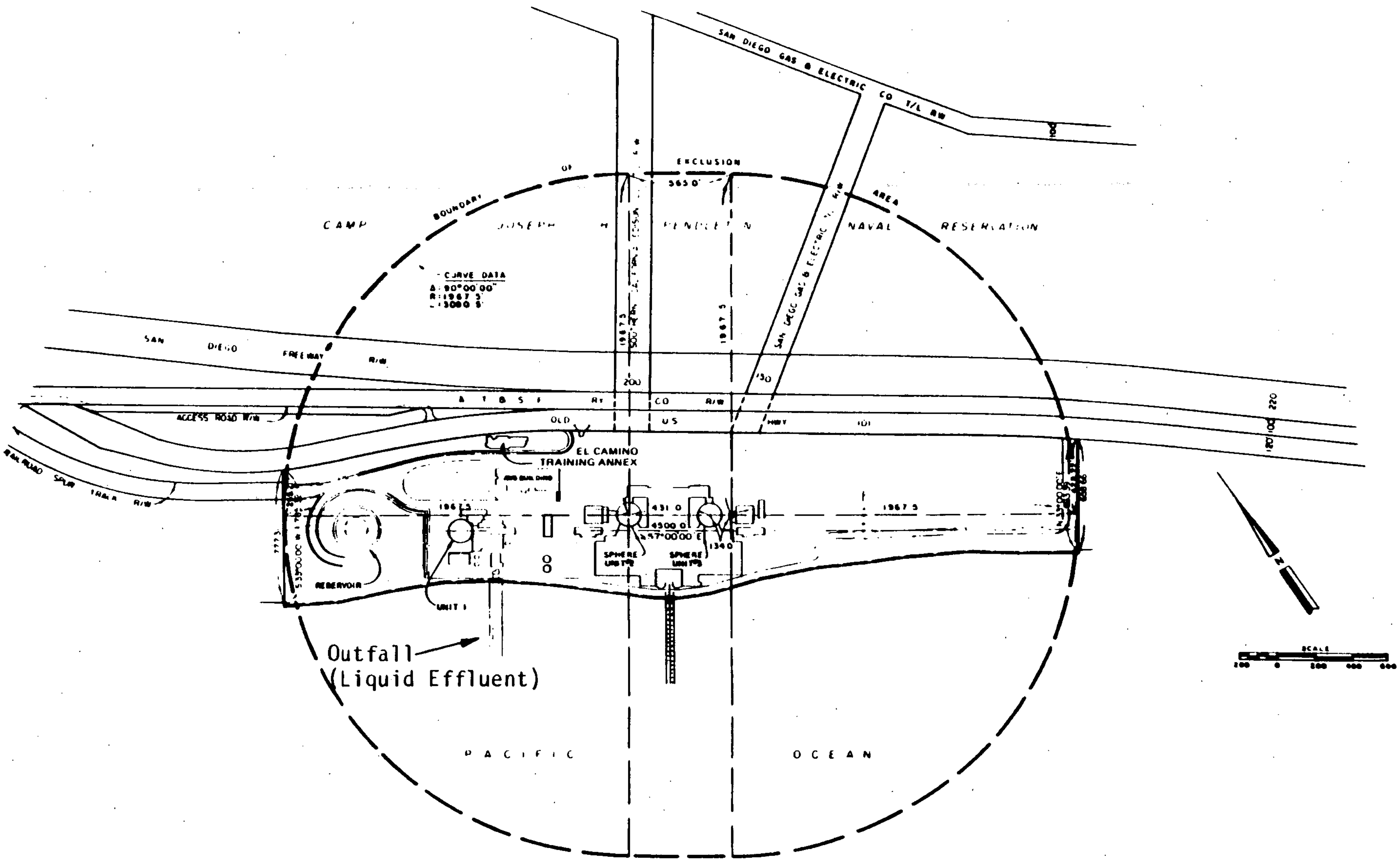

FIGURE 5.1.1
EXCLUSION AREA

5.2 CONTAINMENT

The containment vessel shall be a steel sphere having a free internal volume of approximately 1.2 x 10^6 cubic feet. Above grade, the sphere shall not be insulated and shall be protected from the environment with suitable paint. (The facility was later modified by adding a Sphere Enclosure Building around the containment. For details of this modification, see Reference (1).) The containment vessel is capable of withstanding a maximum internal pressure of 53.3 psig, a temperature of 391.5°F, and an internal vacuum of 2.0 psig. The materials used in the containment sphere and its penetrations shall have a maximum NDT of -20°F.

Penetrations added to the containment shall be designed in accordance with Section 5.3.6 of the Final Engineering Report and Safety Analysis for the appropriate class of penetration. Piping passing through such penetrations shall have isolation valves as follows:

A. Lines which penetrate the containment sphere and normally carry radioactive fluids shall have two valves in series, one of which will be located within the containment and the other outside the containment shell. These valves shall be remotely operated whenever necessary to prevent outward flow in the event of an accident. Incoming lines will be provided with a check valve inside the vapor container and will be either backed up with a closed piping system outside the vapor container or by a remotely operated valve, if necessary.

B. Lines which penetrate the sphere and open to the free volume of the sphere have two valves in series to prevent outward flow in the event of an accident. One valve closes automatically, the other can be closed from the main control room.

C. Lines which penetrate the sphere and open to the turbine cycle are equipped with one isolation valve. In the main steam lines, the turbine stop valves serve this purpose.

D. Lines which penetrate to the free volume of the sphere but which are normally closed during operation of the reactor are equipped with a single isolation valve. Depending on the service, a lock, interlock, or operating procedures ensure that these valves are closed whenever containment is required. The ventilation penetrations are included in this category.

Note 1: Lines which enter and leave the containment sphere but are not open to the containment sphere free volume or the outside atmosphere are not provided with isolation valves. These lines are either part of separate, closed systems or are not subject to damage as a result of a reactor system rupture.

Note 2: Safety injection lines must remain open in the event of an accident.

Electrical penetrations added to the containment vessel shall be designed in accordance with Section 5.3.6 of the Final Engineering Report and Safety Analysis.

The containment vessel shall have a spray system which provides a uniformly distributed borated water spray of at least 1000 gpm upon proper actuation. The system shall be automatically actuated upon actuation of the Safety Injection System and high containment pressure or it shall be capable of being manually actuated from the control room. All active components (i.e., actuating instruments, pumps, and actuated valves) shall be redundant and arranged such that a single failure of such component to respond to a demand signal will not impair the ability of the system to deliver 1000 gpm.

The automatically actuated containment isolation valves shall be designed to close upon high pressure in the containment (setpoint no higher than 1.4 psig), high radiation in the containment sphere (ventilation valves only), or safety injection actuation. The actuation system shall be designed such that no single component failure will prevent containment isolation if required.

BASIS:

The containment vessel is designed to contain the atmosphere within the vessel in the event of a rupture of the Primary Reactor Coolant System. With a free volume of 1.2×10^6 cubic feet, the containment vessel pressure resulting from a complete loss of water from the Primary Reactor System is 46.0 psig and the corresponding temperature is 271.2°F.. The containment vessel is designed to withstand these pressure-temperature conditions simultaneously with an earthquake having a maximum ground acceleration of 0.25g. The design vacuum rating of the containment vessel is 2.0 psig. No vacuum relief valves are provided for the containment vessel since no credible mechanism for creating a vacuum in excess of 2.0 psig has been identified. The materials used in construction of the containment vessel have an NDT of -20°F or less. The lowest recorded temperature at San Onofre was +25°F, hence meeting an NDT of -20°F assures that the vessel will always be operated in the ductile region and will meet NDT +30 with an adequate margin.

All plant piping penetrations have been designed in accordance with the above criteria. The basic design philosophy used to develop these criteria is recognition of the importance of isolation valves and their associated apparatus in assuring containment integrity. They must be redundant such that the failure of a single valve will not result in a release of fission products to the atmosphere. The redundant valves and their associated controls must be independent of each other.

Electrical penetrations which may have paths for leakage, such as coaxial cables, are installed in canisters which are amenable to leak rate testing during reactor operation. This design allows meaningful leak rate testing of these penetrations.

Accident evaluations indicate that a containment spray system capable of supplying 1000 gpm will provide adequate cooling to prevent post-accident pressure from exceeding the containment vessel design pressure, taking into consideration credible metal water reactions, stored energy in the Reactor Coolant System, and fission product decay heat. Calculations have shown that after spraying into the containment vessel (using the stored water in the refueling storage tank) no further credit for active heat removal systems is required since the bare steel containment vessel is capable of dissipating energy at a rate sufficient to preclude pressurization above the design limit.(2)

REFERENCES:

(1) Amendment 52 to the Final Safety Analysis, Sphere Enclosure Project, submitted December 3, 1975; Supplement to the Sphere Enclosure Project Report, submitted March 1, 1976; Second Supplement to the Sphere Enclosure Report, submitted March 25, 1978; additional information submitted by letter dated March 25, 1976 (withheld from public disclosure pursuant to 10CFR Part 2, Section 2.790(d)).

(2) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 5, Question 3.

5.3 REACTOR

The design of all components in the Reactor Coolant System shall comply with the code requirements listed in Subsection 3.5, Table 3.12 of the Final Engineering Report and Safety Analysis. Any modifications to the system shall be in accordance with these requirements and other standards imposed at the time of initial fabrication. The materials of construction shall be as indicated in this Table.

The reactor Coolant System shall be designed for a pressure of 2,485 psig and a temperature of 650°F. The maximum liquid volume of the primary system at rated conditions shall be 6800 cubic feet. Auxiliary systems which connect with the Reactor Coolant System and are exposed to the same conditions of temperature and pressure shall be designed to the same specifications as the Reactor Coolant System. Two self-actuated, spring loaded safety valves, having a combined capability of 480,000 pounds/hour, shall be provided, and shall be in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

A redundant Safety Injection System shall be designed so that each injection train can deliver at least 7000 gpm at 715 psig. The system shall be designed to be automatically actuated upon low pressurizer pressure ($\leq$1,735 psig) or high containment pressure ($\geq$1.4 psig). The system shall be capable of being manually actuated in the control room. The system shall be designed to inject into all three coolant loops and shall be provided with flow indicators to indicate the safety injection flow rate to each of the three coolant loops. The system shall be designed such that no single failure of an active component to respond to a demand signal will impair the systems capability to deliver 7000 gpm @ 715 psig.

The initial reactor core shall consist of 157 fuel assemblies containing enriched uranium dioxide pellets clad in stainless steel with the physical arrangement and dimensions of the assemblies and components as shown in Figure 3.10 of the Final Engineering Report and Safety Analysis. Fuel rods shall be held in place by spring-clip grids and 16 of the fuel rod positions shall have guide tubes which may be used to contain the absorber rods used for rod cluster control. The assemblies shall form an essentially cylindrical lattice with a height of 10 feet and an equivalent diameter of 9.2 feet. The initial core shall be divided into three concentric regions with the two outermost regions containing 52 fuel assemblies each, and the innermost region containing 53 fuel assemblies. The initial core shall contain approximately 22,000 lbs. of Type 304 stainless steel and 143,600 lbs. of UO_2.

Subsequent cores shall contain 157 fuel assemblies With each fuel assembly containing 180 fuel rods clad with type 304 stainless steel. Reload core will contain a mixture of fresh fuel assemblies and irradiated assemblies from previous cycles. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.10 weight percent U-235.*

As many as four fuel assemblies containing mixed oxide (PuO_2-UO_2) pellets clad in zircaloy may be placed in the core in lieu of an equal number of assemblies containing uranium dioxide pellets clad in stainless steel. The mixed oxide assemblies may remain in the reactor core through up to three normal reactor core cycles. The initial composition of the mixed oxide assemblies will not exceed a nominal value of 3.53 weight percent plutonium.

Initial fuel enrichments of 3.15, 3.40, and 3.85 weight percent shall be used in the center, intermediate, and the outer core regions, respectively. The maximum value of temperature coefficient of reactivity shall be $+1.0 \times 10^{-4}$ delta k/k per °F and the maximum coolant void coefficient of reactivity shall be $+1.0 \times 10^{-3}$ delta k/k per % void.

Core excess reactivity shall be controlled by rod cluster control assemblies and by boron dissolved in the primary coolant. Forty-five rod cluster control assemblies shall be distributed throughout the core as shown in Figure 3.27 of the Final Engineering Report and Safety Analysis. Each assembly shall consist of sixteen silver-indium-cadmium absorber rods which shall be inserted in the guide tubes provided in the fuel assemblies. The guide tubes shall be designed such that absorber rods remain in the guide tubes when the assembly is at its upper limit of travel.

Neutron monitoring instrumentation shall be provided to continuously monitor neutron flux intensities from the fully shutdown condition to 200% of full power. Monitors in each range shall be fully redundant and shall be in continuous operation until at least one decade of reliable indication is verified on the next range of instrumentation.

The reactor protection system shall be designed and constructed such that no single failure in any of the instrument systems will prevent the desired safety action if an applicable parameter exceeds a safety setpoint.

* For Cycle 4, two Region 1 and two Region 2 assemblies have been placed in the outer region of the core (Location A-8, R-8, H-1, H-15). Four non-depleted assemblies have been placed in the inner region. (Location B-8, P-8, H-2, H- 14.)

BASIS:

Design requirements of the Reactor Coolant System and the Safety Injection System are specified to ensure that these systems adequately and reliably perform their intended functions. The maximum liquid volume of the primary system limits the containment pressure following a double-ended rupture of a main coolant line to that upon which containment design is based. Design requirements regarding the actuating mechanisms of the Safety Injection System provide assurance that action of this vital engineered safeguard will be initiated so as to protect the fuel.

5.4 AUXILIARY EQUIPMENT

The spent fuel facility shall be designed to maintain fuel element geometry such that k_∞ is < 0.9. For purposes of design, new fuel elements in unborated water shall be considered.

BASIS: The spent fuel storage facility is designed to preclude the occurrence of inadvertent criticality. Normally, this facility will contain irradiated fuel and borated water. However, since this facility is located outside the containment sphere, a substantial margin to criticality is provided.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President and Site Manager, Nuclear Generation Site shall be responsible for design, construction, operation and maintenance of Unit 1 at San Onofre Nuclear Generating Station, and all site support functions. He shall delegate in writing the succession to this responsibility in his absence.

6.1.2 The SHIFT SUPERINTENDENT (or during his absence from the Control Room Area,* a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President and Site Manager, Nuclear Generation Site shall be reissued to all site/station personnel on an annual basis.

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* "Control Room Area" is defined by the control room and the Shift Superintendent's office.

6.2 ORGANIZATION

OFFSITE AND ONSITE ORGANIZATION

6.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

b. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Engineering, Safety, and Licensing and the Vice President and Site Manager shall have corporate responsibility for overall plant nuclear safety. The Vice President and Site Manager shall take any measures needed to ensure acceptable performance of the staff in operating and maintaining the plant to ensure nuclear safety. The Vice President, Nuclear Engineering, Safety and Licensing shall take any measures needed to ensure acceptable performance of the staff in providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 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 Operator shall be at the controls when fuel is in the reactor.* During refueling operations, this operator is permitted to step outside the red line to update the refueling status board. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room Area.**

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* "At the controls" means within the area bounded by the three vertical instrumentation boards and the red line on the floor of the control room.

** "Control Room Area" is defined by the control room and the Shift Superintendent's office.

c. A health physics technician# shall be on-site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent duties during this operation.

e. A Fire Brigade of at least five members shall be maintained on site at all times.# The Fire Brigade shall not include the Shift Superintendent and the two other 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.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff in the following job classifications:

   1) Shift Superintendents, Control Room Supervisors, Control Operators, Assistant Control Operators, Nuclear Plant Equipment Operators, Plant Equipment Operators;

   2) Electricians and their first line supervisors;

   3) I&C Technicians, Computer Technicians, Test Technicians and their first line supervisors;

   4) Operational Health Physics Technicians and their first line supervisors;

   5) Boiler and Condenser Mechanics, Machinists, Welders, Crane Operators and their first line supervisors;

   6) Contractor or other Department personnel performing functions identical to those performed by personnel identified in items I through 5 above and within the organizational framework of the station.(i)

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel identified above work a normal 8-hour day, 40-hour week (excluding shift turnover and meal time) while the plant is operating (MODES 1, 2, 3 and 4). However, in the event that

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\# The health physics technician and 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.

(i) Shift Technical Advisors are exempt from the overtime guidelines specified, since sleeping accommodations are provided.

overtime which exceeds 25%(ii) of normal time is required due to unforeseen problems(iii) or during extended outages(iv), on a temporary basis, the following guidelines shall be followed:

1) An individual should not be permitted to work more than 16 hours straight (shift turnover and meal time are not included when calculating hours worked).

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period (shift turnover and meal time are not included when calculating hours worked).

3) A break (the time an individual leaves the work location to the time an individual returns to the work location) of at least 8 hours should be allowed between work periods (shift turnover time is not included when calculating the break; meal time is not included when calculating the break, unless it represents an administrative entry on the timesheet and not extra hours spent at the work location).

4) Except during extended shutdown periods, overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager, or by the Responsible Station Division Manager, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

g. The Plant Superintendent (at time of appointment), Shift Superintendents and Control Room Supervisors shall hold a senior reactor operator license. The Control Operators and Assistant Control Operators shall hold a reactor operator license.

(ii) 25% is established as a level of overtime which will not significantly reduce the effectiveness of personnel, but which requires additional management approval prior to exceeding this level.

(iii) Unforeseen problems are forced shutdowns or power reductions of any unit, equipment failure or unscheduled repair, surveillance, calibration or maintenance, entry into a Technical Specification ACTION Statement or the absence of personnel required to provide normal shift coverage.

(iv) Extended outages are periods in Modes 5 and/or 6.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
CRS	1	None Required
RO	2	1
NPEO	2	1
STA	1	None Required

SS - Shift Superintendent with a Senior Reactor Operator's License on San Onofre Unit 1
CRS- Control Room Supervisor with a Senior Reactor Operator's License on San Onofre Unit 1
RO - Individual with a Reactor Operator's License on San Onofre Unit 1
NPEO - Non-Licensed Plant Equipment Operator
STA- Shift Technical Advisor

Except for the person filling the Control Room Command function, the shift crew composition may be one less than the minimum requirements of Table 6.2-1 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 oncoming shift crewman being late or absent.

During any absence of the Shift Superintendent from the Control Room Area* while the unit is in MODE 1, 2, 3 and 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room Command function. During any absence of the Shift Superintendent from the Control Room Area* while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

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* "Control Room Area" is defined by the control room and the Shift Superintendent's office.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants," for comparable positions, except for the Manager, Health Physics, who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall provide technical support to the Shift Superintendent in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to safe operation of the unit.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager, Station Emergency Preparedness and shall meet or exceed the requirements of Section 27 of the National Fire Protection Association Code - 1976.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE REVIEW COMMITTEE (OSRC)

FUNCTION

6.5.1.1 The Onsite Review committee shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Onsite Review Committee shall be composed of the:

Chairman:	Station Manager
Member:	Deputy Station Manager
Member:	Manager, Operations
Member:	Manager, Technical
Member:	Plant Superintendent Unit 1
Member:	Supervisor of I&C
Member:	Manager, Health Physics
Member:	Supervisor of Plant Chemistry
Member:	Manager, Maintenance
Member:	Supervising Engineer (NSSS Engineering, Power Generation, Computers, or STA)
Member:	San Diego Gas & Electric Representative, Senior Engineer(1)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSRC Chairman to service on a temporary basis; however, no more than two alternates shall participate as voting members in OSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and one-half the remaining membership including alternates.

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(1) BS degree in Engineering or Physical Science plus at least four years professional level experience in his field. At least one of the four years experience shall be nuclear power plant experience.

RESPONSIBILITIES

6.5.1.6 The Onsite Review committee shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of all REPORTABLE EVENTS.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG Supervisor.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Vice President and Site Manager, Nuclear Generation Site and the NSG Supervisor of disagreement between the OSRC and the Station Manager; however, the Vice President and Site Manager, Nuclear Generation Site, shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Onsite Review Committee shall maintain written minutes of each OSRC meeting that, at a minimum, document the results of all OSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Nuclear Safety Group.

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Vice President and Site Manager, Nuclear Generation Site, shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto. Documentation of these activities shall be provided to the NSG.

6.5.2.2 Proposed changes to the Appendix "A" Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager, Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.

6.5.2.3 Proposed modifications to unit nuclear safety related structures, systems and components shall be designed by a qualified individual/ organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety related structures, systems and components shall be approved prior to implementation by the Station Manager; or by the Manager, Technical as previously designated by the Vice President and Site Manager, Nuclear Generation Site. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.

6.5.2.4 Individuals responsible for review performed in accordance with 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the site/station management staff, previously designated by the Vice President and Site Manager, Nuclear Generation Site, to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager, or members of the site/station management staff previously designated by the Vice President and Site Manager, Nuclear Generation Site. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.

6.5.2.6 Recommended changes to the station security plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.

6.5.2.7 Recommended changes to the station emergency plan shall be approved by the Station Manager and transmitted to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG; implementing procedures shall be prepared and approved in accordance with Specification 6.8.

6.5.2.8 The Station Manager shall assure the performance of a review by a qualified individual/organization of every uncontrolled or unplanned release of radioactivity to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President and Site Manager, Nuclear Generation Site and to the NSG.

6.5.2.9 The Station Manager shall assure the performance of a review by a qualified individual/organization and may designate the approval of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems. Documentation of these activities shall be provided to the Vice President and Site Manager, Nuclear Generation Site, and to the NSG.

6.5.2.10 Documentation of each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained.

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.3.1 The Nuclear Safety Group shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least three staff specialists. The Supervisor shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of 6 years of professional level managerial experience in the power field. Each staff specialist shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of 5 years of professional level experience in the field of his specialty.

The NSG shall use specialists from other technical organizations to augment its expertise in the disciplines of 6.5.3.1. Such specialists shall meet the same qualification requirements as the NSG members.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

RESPONSIBILITIES

6.5.3.4 The NSG shall review:

a. The safety evaluations for 1) changes to procedures required by Specification 6.8, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviation from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components that could affect nuclear safety.

i. Reports and meeting minutes of the Onsite Review Committee.

AUDIT

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to the provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of the actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of the activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the Nuclear Safety Group or the Vice President and Site Manager, Nuclear Generation Site.

f. The Fire Protection Program and implementing procedures at least once per 24 months.

g. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

h. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY

6.5.3.6 The NSG shall report to and advise offsite organization management on those areas of responsibility specified in Sections 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be distributed monthly to the Vice President and Site Manager, Nuclear Generation Site, and to the management responsible for the areas audited.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following action shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73, 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSRC, and the results of this review shall be submitted to the Vice President and Site Manager, Nuclear Generation Site, and the NSG.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President and Site Manager, Nuclear Generation Site, and the NSG Supervisor shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Vice President and Site Manager, Nuclear Generation Site, and the NSG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures* shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program 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, February 1979.

j. "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," June 14, 1977, as specified in Section 6 of the Fire Protection Safety Evaluation Report dated July 19, 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Vice President and Site Manager, Nuclear Generation Site; or by (1) the Station Manager; or by (2) the Responsible Station Division Manager; or by (3) Cognizant Managers reporting directly to them as previously designated by the Vice President and Site Manager, Nuclear Generation Site, 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 site/station management staff exercising responsibility in the specific area and unit or units addressed by the change, and at least one of whom holds a Senior Reactor Operator's License on the unit affected.

* Procedures and administrative policies shall meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1976, Administrative Controls for Nuclear Power Plants.

c. The change is documented, reviewed and approved by responsible management, as delineated in 6.8.2 above, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

(i) Preventative maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel
(ii) Procedures for monitoring, and
(iii) Provisions for maintenance of sampling and analysis equipment.

c. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

d. Secondary Water Chemistry Monitoring Program

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical parameters and control points for these parameters,

(ii) Identification of the procedures used to measure the values of the critical parameters,

(iii) Identification of process sampling points,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off control point chemistry conditions, and

(iv) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiated corrective action.

e. <u>Post-Accident Sampling</u>

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include:

(i) Training of personnel,

(ii) Procedures for sampling and analysis, and

(iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC Regional Administrator unless otherwise noted.

STARTUP REPORT

6.9.1.1 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.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the 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.

6.9.1.3 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.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of 10 CFR 20.407.

inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.1.1. The following information shall be included in these reports: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) 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 concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) 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 (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.18.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The annual radiological environmental operating reports shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map for all sampling locations keyed to a table giving distances and directions from the site reference point; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.18.3.

(Note: Information which may be required by Specifications 3.18.1.B.1, 2, 3.18.3.B.1 and the Basis of 4.18.1 should be included.)

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability,

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* A single submittal may be made for multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

** In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid andgaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottom),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive release reports shall include unplanned releases from the site to UNRESTRICTED AREAS of radioactive material in gaseous and liquid effluents on a quarterly basis.

The Radioactive Effluent Release Reports shall include any changes made to the, PROCESS CONTROL PROGRAM (PCP) , to the OFFSITE DOSE CALCULATION MANUAL (ODCM), or major changes to radioactive waste treatment systems during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to pressurizer safety and relief valves, shall be submitted to the Nuclear Regulatory Commission, on a monthly basis, no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principle maintenance activities, inspections, repair and replacement of principle items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and test results, in units of microcuries, for leak tests performed pursuant to Specification 4.12.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Record and drawing changes reflecting unit 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. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

f. Records of reactor tests and experiments.

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 not included in 6.10.1 that are required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews or tests and experiments pursuant to 10 CFR 50.59.

k. Records of OSRC meetings and NSG reports.

l. Records of the service lives of all safety related hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

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 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the areas and alarms when a preset integrated dose is received. 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.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Superintendent on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas,

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following approved plant radiation protection procedures for entry into high radiation areas.

** Measurement made at 18" from source of radioactivity.

such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and unless a health physics technician is in continuous attendance, a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the changes(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.9.

2. Shall become effective upon review and approval pursuant to 6.5.2.9.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s) ;

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to 6.5.2.9.

2. Shall become effective upon review and approval pursuant to 6.5.2.9.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS* (Liquid, Gaseous and Solid)

6.15.1 Licensee-initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Effluent Release Report for the period in which the change(s) was made effective pursuant to 6.5.2.9. 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 licensee application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the licensee 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 found acceptable pursuant to 6.5.2.9.

2. Shall become effective upon review and approval pursuant to 6.5.2.9.

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* Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.16 ENVIRONMENTAL PROTECTION

FACILITY DESIGN AND OPERATION

6.16.1 This section contains a description of facility design features and operating practices which, if changed, could have a significant effect on environmental impact. Any significant change in facility design features or operating practices described here must be reported to the NRC in accordance with the provisions of Section 6.16.2.a prior to the change.

a. Intake System

The circulating water system, under normal operating conditions, draws water from the ocean at a point approximately 3,200 feet offshore. The ocean bottom at this point is approximately 27 feet below mean lower low-water level. The intake structure rests on a foundation located 33 feet beneath the ocean bottom and rises vertically to a point 10 1/2 feet above the ocean floor. The inside horizontal dimensions of the intake structure are 16 to 21 feet. A velocity cap, 1-foot thick, rests on eight columns above the top of the intake structure. The top surface of the velocity cap is 15 1/2 feet above the ocean bottom and 11 1/2 feet below mean lower low-water.

A 12-foot ID reinforced concrete conduit is connected horizontally to the shoreward side of the intake structure. This conduit is buried beneath the ocean bottom, with a minimum of 4 feet of sand cover over its top and 4 feet of rock cover surrounding the intake structure. All sand cover was placed so as to approximate the local ocean bottom profile.

Water entering the top of the intake structure is accelerated to a design velocity of about 2.5 feet per second and directed into a 12- foot ID reinforced concrete conduit. As the water enters the concrete conduit from the structure, it is accelerated to a design velocity of 6.9 feet per second. The circulating water system is designed to deliver 350,000 gpm at this velocity.

The offshore system joins the onshore portion of the circulating water system at the screenwell. The screenwell is located just inside the seawall on the Station property.

Cooling water entering the onshore system passes, through a coarse bar screen, through finer traveling screens, and proceeds to two circulating water pumps designed to operate at 175,000 gpm. Water entering the screenwell structure is decelerated so that the approach velocity at the screens is approximately 2.0 feet per second.

The circulating water system uses three methods of handling the marine growth and debris associated with the flow of seawater through the plant condensers. These are heat treating, bar and traveling screens, and chlorination.

Heat treatment is used for incrustation control. This method consists of reversing the flow in the intake conduit and adjusting the temperature of the water to approximately 100°F and maintaining this temperature for approximately two hours once every five to six weeks and occasionally once every four weeks, and discharging through the intake conduit. This is accomplished by recirculating a portion of the condenser discharge back through the condenser. Cross-connections between intake and outfall conduits are provided to create the reversal of flow necessary for the treatment of the conduits. Normally only the intake conduit is treated. The water temperature in the outlet conduit can be raised for treatment when necessary. The sudden temperature increase of the cooling water causes incrustation growing in the circulating water system to expire, relax their hold, and be flushed out of the system.

Traveling and bar screens are provided to remove marine growth and debris from the seawater passing through the screenwell. The materials removed from the seawater are marine growth, shells, fish, driftwood, and other debris present in the ocean.

For chlorination, sufficient sodium hypochlorite is injected into the circulating water upstream of the circulating water pumps three times a day for each condenser half to eliminate slime-forming organisms on condenser internal surfaces.

The traveling screens and bar screens are placed in series, perpendicular to the flow. The screens are cleaned automatically, with the frequency of cleaning being dependent on the rate of material buildup on the screens. The bar screens are cleaned by a traveling mechanical rake that deposits accumulated debris, by means of a seawater jet spray washing process, into sluiceways for removal. The traveling screens are motor driven, and are capable of rotating as a unit in continuous sequence when activated by pressure differential due to trash buildup. The debris picked up by the traveling screens is also deposited in a sluiceway by means of a seawater jet spray.

b. <u>Discharge System</u>

Under normal operating conditions, the heated cooling water leaves the condenser and is discharged to the ocean through a 12-foot ID 2,600-foot-long concrete conduit. A single point discharge is effected through a discharge structure located in 24 feet of water. The dimension of the structure is the same as the intake; however, there is no velocity cap. The top of the discharge structure is about 11.5 feet below mean lower low-water.

A 12-foot ID reinforced concrete conduit is connected horizontally to the shoreward side of the discharge structure. This conduit is buried beneath the ocean bottom, with a minimum of 4 feet of sand cover over its top and 4 feet of rock surrounding the discharge structure. All sand and rock cover was placed so as to approximate the local ocean bottom profile.

The water travels through the discharge conduit with a design velocity of 6.9 feet per second and exits with a vertical velocity of about 2.5 feet per second. The vertical orientation creates a single orifice jet diffuser which entrains surrounding cooler water and assists in rapid diminution of the discharge temperature. About seven minutes is required for water to travel from the condensers to the end of the discharge.

c. Land Management

The facility occupies about 16 acres of the 84 acre site.

No use of herbicides is practiced to manage vegetation along the transmission line except in isolated cases to meet property owners' requests or permit stipulations from public agencies. Standard erosion control measures are used to minimize erosion at the facility, at tower sites, and along access roads.

REPORTS

6.16.2 The following reports shall be submitted pursuant to Specification 6.9.2.

a. A report shall be made to the NRC prior to implementation of a change in plant design, in plant operation, or in procedures described in Section 6.16.1 if the change would have a significant adverse effect on the environment or involves an environmental matter or question not previously reviewed and evaluated by the NRC. The report shall include a description and evaluation of the change and a supporting benefit-cost analysis.

b. Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to station operation shall be recorded and promptly reported to the NRC within 24 hours followed by a written report within 30 days. No routine monitoring programs are required to implement this condition.

The written report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

The following are examples of unusual or important events: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; unusual fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substances.

c. Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES Permit or State certification (pursuant to Section 401 of the Clean Water Act) shall-be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or certification. The licensee shall also provide the NRC with a copy of the results of the following studies at the same time they are submitted to the permitting agency:

Section 316(b) Demonstration Study

Changes and additions to the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY for a Class 104(b) License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 1 of the San Onofre Nuclear Generating Station	))))))	DOCKET NO. 50-206 Amendment No. 175

SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS AND ELECTRIC COMPANY, pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 175.

This amendment application consists of Proposed Technical Specification Change No. 209 to Provisional Operating License No. DPR-13. Proposed Technical Specification Change No. 209 is a request to revise San Onofre Unit 1 Radiological Effluent Technical Specifications in accordance with the guidance in NRC Generic Letter 89-01. The proposed change allows for the implementation of programmatic controls for the Radiological Effluent Technical Specifications and Radiological Environmental Monitoring Program in the administrative controls section of the Technical Specifications and the relocation of procedural details to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP).

Subscribed on this 12th day of December, 1989.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Harold B. Ray
Harold B. Ray

Subscribed and sworn to before me this 12th day of December, 1989.

Linda L. Rulon
Notary Public in and for the State of California

OFFICIAL SEAL
LINDA L. RULON
Notary Public–California
ORANGE COUNTY
My Comm. Exp. Mar. 4, 1992

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By: James A. Beoletto
James A. Beoletto

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY (San Onofre Nuclear Generating Station, Unit No. 1)	))))	Docket No. 50-206

CERTIFICATE OF SERVICE

I hereby certify that a copy of Amendment Application No. 175 was served on the following by deposit in the United States Mail, postage prepaid, on the 13th day of December, 1989.

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DESCRIPTION AND SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS
OF PROPOSED CHANGE NO. 209
TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE NO. DPR-13

This is a request to revise the following Unit 1 Technical Specification and Surveillance Standard sections:

1.0	"Definitions"
3.5	"Instrumentation and Control"
3.15	"Radioactive Liquid Effluents"
3.16	"Radioactive Gaseous Effluents"
3.17	"Dose"
3.18	"Radiological Environmental Monitoring"
3.19	"Solid Radioactive Waste"
4.0	"Surveillance Standards"
6.8	"Procedures and Programs"
6.9	"Reporting Requirements"
6.10	"Record Retention"
6.13	"Process Control Program (PCP)"
6.14	"Offsite Dose Calculation Manual (ODCM)"
6.15	"Major Changes to Radioactive Waste Treatment Systems"

DESCRIPTION:

Presently the Radiological Effluent Technical Specifications (RETS) and the Radiological Environmental Monitoring (REM) Program for San Onofre Unit 1 are located within the Appendix A Technical Specifications. Proposed Change No. 209 allows for the implementation of programmatic controls for the RETS and REM in the administrative controls section of the Technical Specifications and the relocation of the RETS and REM to the Offsite Dose Calculation Manual (ODCM) and the Process Control Program (PCP). This change is proposed in accordance with the guidelines described in the NRC's Generic Letter 89-01 dated January 31, 1989.

EXISTING TECHNICAL SPECIFICATIONS:

See Attachment 1

PROPOSED TECHNICAL SPECIFICATIONS:

See Attachment 2

DISCUSSION:

Proposed Change No. 209 incorporates programmatic controls in the Administrative Controls section of the technical specification that satisfy the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50. These changes to the Administrative Controls section replace corresponding requirements that address these items. These are proposed for incorporation as suggested in Generic Letter 89-01 without substantive change.

Procedural details of the RETS on radioactive effluent monitoring instrumentation, the control of liquid and gaseous effluents, equipment requirements for liquid and gaseous effluents, radiological environmental monitoring, and radiological reporting details have been relocated to the ODCM. Procedural details consist of the limiting conditions for operation, their applicability, remedial actions, surveillance requirements and the bases section for these requirements. The definition of solidification and existing procedural details of the RETS on solid radioactive wastes have been relocated to the PCP. The disposition of the RETS are provided in Table 1.

Specific RETS are not covered by the new programmatic controls and will be retained in the existing Technical Specifications. Retained specifications include 3.16.5, "Gas Storage Tanks" and 3.16.6, "Explosive Gas Mixture."

The proposed amendment also simplifies the associated reporting requirements, simplifies the technical specification administrative controls section describing changes to the ODCM and PCP, adds record retention requirements for changes to the ODCM and PCP, and updates the definitions of the ODCM and PCP consistent with these changes.

These actions simplify the RETS, meet the regulatory requirements for radioactive effluents and radiological environmental monitoring, and are provided as a line-item improvement of the technical specifications. Changes for relocating procedural details of current RETS to either the ODCM or PCP have been prepared in accordance with the proposed changes to the Administrative Controls section of the technical specification. Future changes will be controlled as described in the ODCM and PCP Administration Controls Section of the Technical Specifications. A copy of the revised ODCM is provided with this proposed change as Attachment 3.

These proposed changes have been developed on a generic basis by the NRC and issued for licensee use under Generic Letter 89-01. Therefore, NRC review and approval of the proposed changes has occurred on a generic basis. SCE has reviewed the changes for implementation at San Onofre Unit 1 and determined them to be applicable.

<u>SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS</u>:

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that a proposed license amendment to allow for the implementation of the Radiological Effluent Technical Specifications and Radiological Environmental Monitoring Program in the administrative controls section of the Technical Specifications and relocation of procedural details to the ODCM and the PCP in accordance with Generic Letter 89-01, does not represent a significant hazards consideration. As demonstrated below, in accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment was analyzed using the following standards and found not to: 1) involve a significant increase in the probability or consequences for an accident previously evaluated; or 2) create the possibility from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   Operation of the facility in accordance with this proposed change has been reviewed by the NRC for generic implementation. The proposed Technical Specifications amendment does not reduce the level of radiological effluent control. Rather, it will provide programmatic controls for RETS consistent with regulatory requirements and allow relocation of the procedural details of the current RETS to the ODCM and PCP. These procedural details are not required to be included in the Technical Specifications by 10 CFR 50.36a. Future changes to these procedural details will be controlled by the ODCM and PCP Administrative Controls sections of the Technical Specifications. Records of reviews performed for changes made to the ODCM an PCP will be documented and retained for the duration of the operating license. Therefore, it is concluded that operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Operation of the facility in accordance with this proposed change has been reviewed by the NRC for generic implementation. The proposed change does not modify the configuration of the facility or its mode of operation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously identified.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   Operation of the facility in accordance with this proposed change has been reviewed by the NRC for generic implementation. The proposed change does not affect the operation of the facility nor modify any method of radiological effluent monitoring or analysis. Therefore, it is concluded that operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

SAFETY AND SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Attachment 1 - Existing Specifications
Attachment 2 - Proposed Specifications
Attachment 3 - Offsite Dose Calculation Manual (ODCM)

TABLE 1.
DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
INCLUDED UNDER THE HEADING OF RETS IN THE SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
1.16	OFFSITE DOSE CALCULATION MANUAL	Definition is updated to reflect the change in scope of the ODCM.
1.19	PROCESS CONTROL PROGRAM	Definition is updated to reflect the change in scope of the PCP.
1.26	SOLIDIFICATION	Definition is relocated to the PCP.
3.5.8, 4.1.2	RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION	Programmatic controls are included in 6.8.4 f. Item 1). Existing specification procedural details are relocated to the ODCM.
3.5.9, 4.1.3	RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION	Programmatic controls are included in 6.8.4 f. Item 1). Existing specification procedural details are relocated to the ODCM. Existing requirements for explosive gas monitoring instrumentation have been retained.
3.15.1, 4.5.1	LIQUID EFFLUENTS CONCENTRATION	Programmatic controls are included in 6.8.4 f. Items 2) and 3). Existing specification procedural details are relocated to the ODCM.
3.15.2, 4.5.2	LIQUID EFFLUENT DOSE	Programmatic controls are included In 6.8.4 f. Items 4) and 5). Existing specification procedural details are relocated to the ODCM.
3.15.3, 4.5.3	LIQUID EFFLUENTS TREATMENT	Programmatic controls are included in 6.8.4 f. Item 6). Existing specification procedural details are relocated to the ODCM.

TABLE 1.
DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
INCLUDED UNDER THE HEADING OF RETS IN THE SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS (Cont.)

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
3.16.1, 4.6.1	DOSE RATE	Programmatic controls are included in 6.8.4 f. Items 3) an 7). Existing specification procedural details are relocated to the ODCM.
3.16.2, 4.6.2	DOSE, NOBLE GASES	Programmatic controls are included in 6.8.4 f. Items 5) and 8). Existing specification procedural details are relocated to the ODCM.
3.16.3, 4.6.3	DOSE, IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM	Programmatic controls are included in 6.8.4 f. Items 5) and 9). Existing specification procedural details are relocated to the ODCM.
3.16.4, 4.6.4	GASEOUS RADWASTE TREATMENT	Programmatic controls are included in 6.8.4 f. Item 6). Existing specification procedural details are relocated to the ODCM.
3.16.5, 4.6.5	GAS STORAGE TANKS	Existing specification requirements have been retained.
3.16.6, 4.6.6	EXPLOSIVE GAS MIXTURE	Existing specification requirements have been retained.
3.17, 4.17	DOSE	Programmatic controls are included in 6.8.4 f. Item 10). Existing specification procedural details are relocated to the ODCM.
3.18.1, 4.18.1	RADIOLOGICAL ENVIRONMENTAL MONITORING	Programmatic controls are included in 6.8.4 g. Item 1). Existing specification procedural details are relocated to the ODCM.
3.18.2, 4.18.2	LAND USE CENSUS	Programmatic controls are included in 6.8.4 g. Item 2). Existing specification procedural details are relocated to the ODCM.

TABLE 1.
DISPOSITION OF SPECIFICATIONS AND ADMINISTRATIVE CONTROLS
INCLUDED UNDER THE HEADING OF RETS IN THE SAN ONOFRE UNIT 1 TECHNICAL SPECIFICATIONS (Cont.)

SPECIFICATION	TITLE	DISPOSITION OF EXISTING SPECIFICATION
3.18.3, 4.18.3	INTERLABORATORY COMPARISON PROGRAM	Programmatic controls are included in 6.8.4 g. Item 3). Existing specification procedural details are relocated to the ODCM.
3.19, 4.19	SOLID RADIOACTIVE WASTE	Existing specification procedural details are relocated to the PCP.
5.1	DESIGN FEATURES: SITE DESCRIPTION	Existing specification requirements have been retained.
6.8	PROCEDURES AND PROGRAMS	Details of the Radiological Effluents Control Program and Radiological Environmental Monitoring Program are included in 6.8.4 f and 6.8.4 g.
6.9.1.6, 6.9.1.7	ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT	6.9.1.6 was simplified and existing reporting details are relocated to the ODCM. 6.9.1.7 deleted.
6.9.1.8, 6.9.1.9	SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT	6.9.1.8 was simplified and existing reporting details are relocated to the ODCM and PCP as appropriate. 6.9.1.9 deleted.
6.10	RECORD RETENTION	Record retention requirements for changes to the ODCM and PCP are included in 6.10.2.n.
6.13	PROCESS CONTROL PROGRAM	Specification requirements are simplified in 6.13.2.
6.14	OFFSITE DOSE CALCULATION MANUAL	Specification requirements are simplified in 6.14.2.
6.15	MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS AND SOLID)	Existing procedural details are relocated to the ODCM and PCP as appropriate.

ATTACHMENT 1