APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
i	131
ii	131
iii	131
iv	131
v	90, 130, 131
vi	90, 130, 131
vii	90, 102, 130, 131
viii	90, 130,
ix	90, 91, 102, 130, 131
x	90, 91, 130, 131
xi	55, 92, 102, 110, 111, 130, 131
xii	56, 58, 71, 79, 83, 104, 117, 130, 131
xiii	31, 56, 58, 79, 83, 84, 91, 117, 130, 131
xiv	131
1.0-1	31, 56, 59 83, 117, 130
1.0-2	31, 56, 59, 83, 104, 117, 130
1.0-3	31, 56, 59, 79, 83, 104, 117, 130
1.0-4	31, 56, 59, 79, 83, 117, 130
1.0-5	77, 79, 83, 117, 130
1.0-6	79, 83, 96 117, 130
1.0-7	58, 83, 117, 130
1.0-8	56, 83, 117, 130
2.1-1	43, 55, 97, 117, 130
2.1-2	43, 97, 117, 121, 130
2.1-3	43, 117, 121, 122, 130
2.1-4	43, 117, 121, 122, 130
2.1-5	43, 97, 117, 121 122, 130
2.1-6	55, 130
3.0-1	43, 56, 64, 83, 130
3.0-2	56, 64, 83, 130
3.1-1	29, 38, 70, 83, 91, 96, 130
3.1-2	29, 83, 96, 130
3.1-3	43, 77, 103, 130
3.1-4	77, 130
3.1-5	77, 125, 130
3.1-6	77, 102, 130
3.1-7	77, 102, 103, 130
3.1-8	43, 102, 103, 130
3.1-9	77, 102, 130
3.1-10	Change No. 14 38, 102, 130
3.1-11	Change No. 14 38, 102, 130
3.1-12	Change No. 14 92, 130
3.1-13	Change No. 14 92, 130
3.1-14	Change No. 14 102, 130
3.1-15	Change No. 14 102, 130
3.1-16	Change No. 14 102, 130
3.1-17	Change No. 14 102, 130
3.1-18	Change No. 7 37, 55, 91, 119, 130
3.1-19	Change No. 7 55, 119, 130
3.1-20	37, 55, 119, 130
3.1-21	58, 59, 83, 130
3.1-22	58, 130
3.1-23	83, 130
3.2-1	102, 130
3.2-2	25, 102, 130
3.3-1	25, 37, 86, 124, 130
3.3-2	25, 130
3.3-3	25, 38, 86, 124, 130
3.3-4	25, 37, 124, 130
3.3-5	25, 130
3.3-6	25, 102, 120, 130
3.3-7	25, 102, 130
3.3-8	25, 38, 122, 130
3.3-9	25, 38, 122, 130
3.3-10	25, 130
3.3-11	NRC order 4/20/81, 130
3.3.12	NRC order 4/20/81, 130
3.4-1	29, 82, 125, 130
3.4-2	29, 130
3.4-3	82, 125, 130
3.4-4	82, 125, 130
3.5-1	83, 117, 130
3.5-2	43, 56, 58, 83, 117, 128, 130
3.5-3	43, 56, 58, 83, 117, 121, 122, 130
3.5-4	55, 58, 83, 117, 118, 121, 128, 130
3.5-5	83, 117, 130
3.5-6	7, 11, 25, 35, 55, 56, 111, 130
3.5-7	7, 11, 25, 35, 49, 55, 56, 111, 122, 130
3.5-8	11, 49, 111, 122, 130
3.5-9	11, 25, 56, 111, 130
3.5-10	56, 130
3.5-11	56, 130
3.5-12	56, 130
3.5-13	56, 130

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PDR ADOCK 05000206
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APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
3.5-14	56, 130
3.5-15	58, 83, 130
3.5-16	58, 83, 130
3.5-17	58, 83, 130
3.5-18	58, 72, 130
3.5-19	58, 83, 125, 130
3.5-20	125, 130
3.5-21	58, 83, 117, 124 125, 130
3.5-22	64, 82, 125, 130
3.5-23	58, 82, 125, 130
3.5-24	58, 82, 125, 130
3.5-25	58, 82, 125, 129, 130
3.5-26	79, 90, 105, 130
3.5-27	79, 90, 130
3.5-28	79, 130
3.5-29	79, 90, 105, 130
3.5-30	79, 90, 130
3.5-31	79, 90, 130
3.5-32	83, 130
3.5-33	83, 130
3.5-34	83, 91, 130
3.6-1	25, 56, 58, 73, 130
3.6-2	Change No. 7 56, 130
3.6-3	58, 99, 130
3.6-4	58, 71, 99, 130
3.6-5	58, 59, 71, 99, 130
3.6-6	58, 59, 83, 130
3.7-1	25, 52, 68, 84, 130
3.7-2	25, 84, 106, 130
3.7-3	25, 84, 130
3.7-4	25, 52, 68, 84, 106, 130
3.7-5	25, 52, 68, 84, 106, 120, 130
3.8-1	25, 36, 37, 73, 77, 130
3.8-2	36, 77, 130
3.8-3	25, 36, 73, 77, 130
3.8-4	36, 73, 77, 130
3.8-5	36, 73, 77, 130
3.9-1	3, 10, 112, 130
3.9-2	3, 10, 112, 130
3.10-1	7, 8, 112, 122, 130
3.10-2	7, 112, 122, 130
3.11-1	7, 8, 11, 35, 55, 112, 117, 122, 130
3.11-2	7, 11, 112, 117, 122, 130
3.12-1	14, 130
3.13-1	21, 63, 81, 130
3.13-2	21, 63, 81, 130
3.14-1	31, 44, 93, 130, 131
3.14-2	31, 130, 131
3.14-3	31, 130, 131
3.14-4	31, 93, 130, 131
3.14-5	31, 130, 131
3.14-6	130, 131
3.14-7	130, 131
3.14-8	130, 131
3.14-9	130, 131
3.14-10	130, 131
3.14-11	31, 93, 130, 131
3.14-12	31, 130, 131
3.14-13	31, 130, 131
3.14-14	31, 130, 131
3.14-15	31, 93, 130, 131
3.14-16	31, 130, 131
3.14-17	31, 130, 131
3.14-18	31, 130, 131
3.14-19	130, 131
3.14-20	130, 131
3.14-21	130, 131
3.14-22	130, 131
3.14-23	130, 131
3.14-24	130, 131
3.14-25	130, 131
3.14-26	130, 131
3.15-1	79, 130
3.15-2	79, 90, 91, 105, 130
3.15-3	79, 90, 91, 105, 130
3.16-1	79, 130
3.16-2	79, 90, 91, 105, 130
3.16-3	79, 90, 91, 105, 130
3.16-4	79, 90, 91, 105, 130
3.16-5	79, 130
3.16-6	79, 90, 130
3.16-7	79, 90, 130
3.17-1	79, 90, 91, 105, 130
3.17-2	79, 90, 130
3.18-1	79, 91, 105, 130
3.18-2	79, 90, 130
3.18-3	79, 130
3.18-4	79, 130
3.18-5	79, 130
3.18-6	79, 130
3.18-7	79, 91, 105, 130
3.18-8	79, 90, 130
3.18-9	79, 90, 105, 130
3.19-1	79, 90, 105, 130
3.20-1	102, 130
3.20-2	102, 130
4.0-1	83, 130
4.0-2	83, 130
4.1-1	29, 56, 83, 117, 130
4.1-2	7, 22, 83, 117, 122, 130
4.1-3	117, 130
4.1-4	25, 29, 70, 96, 117, 130
4.1-5	25, 29, 70, 96, 130
4.1-6	25, 29, 56, 70, 100, 122, 130
4.1-7	25, 29, 36, 77, 103, 130
4.1-8	77, 125, 130

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
4.1-9	77, 122, 125, 130
4.1-10	117, 130
4.1-11	25, 130
4.1-12	Change No. 5 25, 130
4.1-13	79, 130
4.1-14	79, 90, 130
4.1-15	79, 130
4.1-16	79, 130
4.1-17	79, 90, 130
4.1-18	79, 130
4.1-19	58, 130
4.1-20	58, 130
4.1-21	58, 130
4.1-22	58, 83, 130
4.1-23	58, 83, 117, 125, 130
4.1-24	58, 109, 130
4.1-25	58, 130
4.1-26	58, 82, 125, 130
4.1-27	58, 82, 125, 130
4.1-28	65, 82, 125, 130
4.1-29	65, 125, 130
4.1-30	82, 130
4.1-31	83, 130
4.1-32	83, 130
4.1-33	83, 130
4.1-34	119, 130
4.1-35	119, 130
4.2-1	25, 37, 54, 114, 130
4.2-2	25, 37, 114, 130
4.2-3	25, 37, 114, 130
4.2-4	25, 37, 114, 124, 130
4.2-5	25, 37, 114, 124, 130
4.2-6	37, 114, 130
4.2-7	NRC order 4/20/81 54, 130
4.3-1	24, 58, 75, 130
4.3-2	5, 24, 58, 75, 87, 118, 130
4.3-3	24, 58, 87, 118, 130
4.3-4	24, 87, 130
4.3-5	24, 75, 87, 130
4.3-6	75, 87, 130
4.3-7	58, 130
4.3-8	58, 59, 83, 130
4.3-9	58, 83, 130
4.4-1	Change No. 12 25, 56, 82, 84, 95, 104, 123, 130
4.4-2	25, 84, 130
4.4-3	25, 56, 82, 84, 117, 130
4.4-4	25, 82, 84, 95, 104, 123, 124, 130
4.4-5	84, 130
4.4-6	25, 56, 84, 104, 123, 130
4.5-7	25, 56, 84, 130
4.5-1	Change No. 10 38, 79, 130
4.5-2	79, 90, 130
4.5-3	79, 130
4.5-4	79, 130
4.5-5	79, 130
4.5-6	79, 130
4.6-1	38, 79, 130
4.6-2	79, 90, 130
4.6-3	79, 130
4.6-4	79, 130
4.6-5	79, 130
4.6-6	79, 130
4.6-7	79, 130
4.6-8	79, 130
4.6-9	79, 130
4.7-1	Change No. 14 46, 130
4.8-1	91, 130
4.9-1	Change No. 14 130
4.9-2	130
4.10-1	13, 94, 130
4.10-2	13, 94, 130
4.11-1	14, 109, 130
4.11-2	14, 109, 130
4.12-1	91, 130
4.13-1	18, 113, 130
4.14-1	21, 33, 63, 81, 130
4.14-2	21, 63, 81, 130
4.14-3	21, 63, 81, 130
4.14-4	21, 63, 81, 105, 130
4.15-1	31, 37, 38, 44, 130, 131
4.15-2	31, 44, 130, 131
4.15-3	31, 130, 131
4.15-4	130, 131
4.15-5	130, 131
4.15-6	31, 130, 131
4.15-7	31, 130, 131
4.15-8	130, 131
4.15-9	130, 131
4.15-10	130, 131
4.15-11	130, 131
4.15-12	130, 131
4.16-1	31, 101, 115, 130
4.16-2	37, 101, 115, 130
4.16-3	37, 101, 115, 130
4.16-4	37, 55, 101, 115, 130
4.16-5	37, 91, 101, 105, 115, 130
4.16-6	37, 55, 101, 115, 130
4.16-7	37, 55, 101, 130
4.17-1	79, 130
4.18-1	79, 130
4.18-2	79, 130
4.18-3	79, 130
4.18-4	79, 130
4.18-5	79, 130
4.18-6	79, 130
4.19-1	79, 105, 130

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	3.5-22
	3.5.8 Radioactive Liquid Effluent Instrumentation	3.5-26
	3.5.9 Radioactive Gaseous Process and Effluent Monitoring Instrumentation	3.5-29
	3.5.10 Radiation Monitoring Instrumentation	3.5-32
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 (MTC)	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
	3.14.1 Fire Suppression Water System	3.14-1
	3.14.2 Spray and/or Sprinkler Systems	3.14-3
	3.14.3 Foam Suppression System	3.14-7
	3.14.4 Halon Systems	3.14-9
	3.14.5 Fire Hose Stations	3.14-11

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

	Three Reactor Coolant Pumps Operating
1. Pressurizer High Level	≤ 50% Pressurizer Narrow Range Level
2. Pressurizer Pressure: High	≤ 2220 psig
3. Nuclear Overpower	
a. High Setting*	≤ 109% of indicated full power
b. Low Setting	≤ 25% of indicated full power
**4. Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341
**5. Coolant Flow	≥ 85% of indicated full loop flow
***6. Steam/Feedwater Flow Mismatch	
a. Low+ Setting:	$\frac{\text{Steam Flow - Feedwater Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$
b. High+ Setting:	$\frac{\text{Feedwater Flow - Steam Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$
**7. Reactor Coolant Pump Breaker Open	
a. Overcurrent	≤ 2400 amps
b. Undercurrent	≥ 110 amps
c. Undervoltage	≥ 60% of rated bus voltage

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* 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.

***May be bypassed at power levels below 50% of full power.

\+ High and Low feedwater flow relative to steam flow.

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 pumps and valves for MODES 1, 2 and 3.

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

SPECIFICATION: Two trains of auxiliary feedwater including associated pumps and valves, shall be OPERABLE.

ACTION:

A. With one Train of auxiliary feedwater inoperable, restore the inoperable train 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.

B. With both Trains of auxiliary feedwater inoperable, be in at least HOT STANDBY within 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 from normal operating conditions in the event of a total loss of offsite power.

Two auxiliary feedwater trains and the steam system relief valves provide core decay heat removal capability in the event of a sustained loss of off-site power. Either auxiliary feedwater train has the capability to satisfy decay heat removal requirements from the core, with a delivered flow of at least 185 gpm per train with three intact main feedwater lines and pressurized steam generators, 125 gpm per train with two intact main feedwater lines and pressurized steam generators, and 250 gpm per train with two intact main feedwater lines and depressurized steam generators.(1)

AFW System Train A pumps and valves consist of AFW pumps G-10S and G-10 and associate valves, including flow control valves FCV-2300A, FCV-2300B, and FCV-2300C.

AFW System Train B pump and valves consist of AFW pump G-10W and associated valves, including flow control valves FCV-3300A, FCF-3300B, and FCV-3300C.

REFERENCES:

(1) SCE letter dated November 20, 1987, from M. O. Medford to NRC Document Control Desk.

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 usable water volume of at least 190,000 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. In addition, the water volume will provide sufficient margin to account for spillage that occurs during a main feedwater line break with loss of AFW flow indication prior to isolation of the broken line. Spillage is assumed to last no longer than one hour until the broken loop is identified via RCS Loop Delta-T positive indication that will be evident for the two intact steam generators. The usable water volume limit is specified relative to the bottom of the tank indicated level range (i.e., level tap). The contained water volume below this datum provides a significant margin to the NPSH and vortexing limits above the highest AFW pump suction inlet in the tank, but is not considered available for purposes of this specification.

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*		
o Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
o Steam Generator Water Level (Wide Range)	1/steam generator	1/steam generator
o Reactor Coolant System Loop Delta-T Indication	1/loop	1/loop
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
Refueling Water Storage Tank Level	2	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 is provided by one channel of auxiliary feedwater flow rate (Train B), one channel of environmentally qualified steam generator wide range level (Train A), and one channel of RCS Loop Delta-T indication. These comprise the three types of indication of auxiliary feedwater flow for each steam generator.

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

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} mCi/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

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.

3.14.5 FIRE HOSE STATIONS

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

SPECIFICATION: The following fire hose stations shall be OPERABLE:

a. See Table 3.14.5.1

ACTION:

A. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route a fire hose* to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to plant workers, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose shall be mounted at the inoperable hose station.

B. The provision of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: In the event that a fire hose station is inoperable, the establishment of backup suppression in the affected areas is required to provide fire suppression capability until the inoperable system is restored to operability.

REFERENCES:

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.

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* Fire hose will be run within 1 hour of entering the ACTION statement if an operable water supply is not available within 250 feet of the area protected by the inoperable hose station, or two 150 feet hose packs (1-3/4") on the fire truck are not operable. Fire hose will be supplied by the fire department responding to a fire if an operable water supply is available within 250 feet of the area protected by the inoperable hose station. With the required hose station inside containment inoperable and containment integrity established, fire hose will be supplied only to the nearest access point.

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.

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.
13. Reactor Coolant Pump Breaker Position*	S	R	R	N.A.	N.A.

*Applicable to Item 6 in Table 2.1

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.

4.2 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 8 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.

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.

C. RWST Low Level Trips

Monthly, perform a CHANNEL TEST and every refueling interval, perform a CHANNEL CALIBRATION, of the SI/Feedwater Pump trip and the MOV 850A, 850B and 850C automatic closure on low-low Refueling Water Storage Tank level.

BASIS: 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 8 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

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.

ATTACHMENT 2

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.	Page	Amendment No.	Page	Amendment No.
i	131, ___	2.1-6	55, 130	3.3-2	25, 130
ii	131, ___			3.3-3	25, 38, 86, 124, 130
iii	131, ___	3.0-1	43, 56, 64, 83, 130	3.3-4	25, 37, 124, 130
iv	131, ___	3.0-2	56, 64, 83, 130	3.3-5	25, 130
v	90, 130, 131	3.1-1	29, 38, 70, 83, 91, 96, 130	3.3-6	25, 102, 120, 130
vi	90, 130, 131, ___	3.1-2	29, 83, 96, 130	3.3-7	25, 102, 130
vii	90, 102, 130, 131	3.1-3	43, 77, 103, 130	3.3-8	25, 38, 122, 130
viii	90, 130, 131	3.1-4	77, 130	3.3-9	25, 38, 122, 130
ix	90, 91, 102, 130, 131	3.1-5	77, 125, 130	3.3-10	25, 130
x	90, 91, 130, 131	3.1-6	77, 102, 130	3.3-11	NRC order 4/20/81, 130
xi	55, 92, 102, 110, 111, 130, 131	3.1-7	77, 102, 103, 130	3.3.12	NRC order 4/20/81, 130
xii	56, 58, 71, 79, 83, 104, 117, 130, 131	3.1-8	43, 102, 103, 130	3.4-1	29, 82, 125, 130
xiii	31, 56, 58, 79, 83, 84, 91, 117, 130, 131	3.1-9	77, 102, 130	3.4-2	29, 130, ___
xiv	131	3.1-10	Change No. 14 38, 102, 130	3.4-3	82, 125, 130, ___
1.0-1	31, 56, 59 83, 117, 130	3.1-11	Change No. 14 38, 102, 130	3.4-4	82, 125, 130, ___
1.0-2	31, 56, 59, 83, 104, 117, 130	3.1-12	Change No. 14 92, 130	3.5-1	83, 117, 130
1.0-3	31, 56, 59, 79, 83, 104, 117, 130	3.1-13	Change No. 14 92, 130	3.5-2	43, 56, 58, 83, 117, 128, 130
1.0-4	31, 56, 59, 79, 83, 117, 130	3.1-14	Change No. 14 102, 130	3.5-3	43, 56, 58, 83, 117, 121, 122, 130
1.0-5	77, 79, 83, 117, 130	3.1-15	Change No. 14 130	3.5-4	55, 58, 83, 117, 118, 121, 128, 130
1.0-6	79, 83, 96 117, 130	3.1-16	Change No. 14 102, 130	3.5-5	83, 117, 130
1.0-7	58, 83, 117, 130	3.1-17	Change No. 14 102, 130	3.5-6	7, 11, 25, 35, 55, 56, 111, 130
1.0-8	56, 83, 117, 130	3.1-18	Change No. 7 37, 55, 91, 119, 130	3.5-7	7, 11, 25, 35, 49, 55, 56, 111, 122, 130
2.1-1	43, 55, 97, 117, 130	3.1-19	Change No. 7 55, 119, 130	3.5-8	11, 49, 111, 122, 130
2.1-2	43, 97, 117, 121, 130	3.1-20	37, 55, 119, 130	3.5-9	11, 25, 56, 111, 130
2.1-3	43, 117, 121, 122, 130	3.1-21	58, 59, 83, 130	3.5-10	56, 130
2.1-4	43, 117, 121, 122, 130	3.1-22	58, 130	3.5-11	56, 130
2.1-5	43, 97, 117, 121 122, 130, ___	3.1-23	83, 130	3.5-12	56, 130
		3.2-1	102, 130		
		3.2-2	25, 102, 130		
		3.3-1	25, 37, 86, 124, 130		

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
3.5-13	56, 130
3.5-14	56, 130
3.5-15	58, 83, 130
3.5-16	58, 83, 130
3.5-17	58, 83, 130
3.5-18	58, 72, 130
3.5-19	58, 83, 125, 130
3.5-20	125, 130
3.5-21	58, 83, 117, 124 125, 130, ___
3.5-22	64, 82, 125, 130
3.5-23	58, 82, 125, 130
3.5-24	58, 82, 125, 130
3.5-25	58, 82, 125, 129, 130
3.5-26	79, 90, 105, 130
3.5-27	79, 90, 130
3.5-28	79, 130
3.5-29	79, 90, 105, 130
3.5-30	79, 90, 130
3.5-31	79, 90, 130
3.5-32	83, 130
3.5-33	83, 130, ___
3.5-34	83, 91, 130
3.6-1	25, 56, 58, 73, 130
3.6-2	Change No. 7 56, 130, ___
3.6-3	58, 99, 130
3.6-4	58, 71, 99, 130
3.6-5	58, 59, 71, 99, 130
3.6-6	58, 59, 83, 130
3.7-1	25, 52, 68, 84, 130
3.7-2	25, 84, 106, 130
3.7-3	25, 84, 130
3.7-4	25, 52, 68, 84, 106, 130
3.7-5	25, 52, 68, 84, 106, 120, 130
3.8-1	25, 36, 37, 73, 77, 130
3.8-2	36, 77, 130
3.8-3	25, 36, 73, 77, 130
3.8-4	36, 73, 77, 130
3.8-5	36, 73, 77, 130
3.9-1	3, 10, 112, 130
3.9-2	3, 10, 112, 130
3.10-1	7, 8, 112, 122, 130
3.10-2	7, 112, 122, 130
3.11-1	7, 8, 11, 35, 55, 112, 117, 122, 130
3.11-2	7, 11, 112, 117, 122, 130
3.12-1	14, 130
3.13-1	21, 63, 81, 130
3.13-2	21, 63, 81, 130
3.14-1	31, 44, 93, 130, 131
3.14-2	31, 130, 131
3.14-3	31, 130, 131
3.14-4	31, 93, 130, 131
3.14-5	31, 130, 131
3.14-6	130, 131
3.14-7	130, 131
3.14-8	130, 131
3.14-9	130, 131
3.14-10	130, 131
3.14-11	31, 93, 130, 131, ___
3.14-12	31, 130, 131
3.14-13	31, 130, 131
3.14-14	31, 130, 131
3.14-15	31, 93, 130, 131
3.14-16	31, 130, 131
3.14-17	31, 130, 131
3.14-18	31, 130, 131
3.14-19	130, 131
3.14-20	130, 131
3.14-21	130, 131
3.14-22	130, 131
3.14-23	130, 131
3.14-24	130, 131
3.14-25	130, 131
3.14-26	130, 131
3.15-1	79, 130
3.15-2	79, 90, 91, 105, 130
3.15-3	79, 90, 91, 105, 130
3.16-1	79, 130
3.16-2	79, 90, 91, 105, 130
3.16-3	79, 90, 91, 105, 130
3.16-4	79, 90, 91, 105, 130
3.16-5	79, 130
3.16-6	79, 90, 130
3.16-7	79, 90, 130
3.17-1	79, 90, 91, 105, 130
3.17-2	79, 90, 130
3.18-1	79, 91, 105, 130
3.18-2	79, 90, 130
3.18-3	79, 130
3.18-4	79, 130, ___
3.18-5	79, 130
3.18-6	79, 130
3.18-7	79, 91, 105, 130
3.18-8	79, 90, 130
3.18-9	79, 90, 105, 130
3.19-1	79, 90, 105, 130
3.20-1	102, 130
3.20-2	102, 130
4.0-1	83, 130
4.0-2	83, 130
4.1-1	29, 56, 83, 117, 130
4.1-2	7, 22, 83, 117, 122, 130, ___
4.1-3	117, 130
4.1-4	25, 29, 70, 96, 117, 130
4.1-5	25, 29, 70, 96, 130
4.1-6	25, 29, 56, 70, 100, 122, 130
4.1-7	25, 29, 36, 77, 103, 130
4.1-8	77, 125, 130

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.
4.1-9	77, 122, 125, 130
4.1-10	117, 130
4.1-11	25, 130
4.1-12	Change No. 5 25, 130
4.1-13	79, 130
4.1-14	79, 90, 130
4.1-15	79, 130
4.1-16	79, 130
4.1-17	79, 90, 130
4.1-18	79, 130
4.1-19	58, 130
4.1-20	58, 130
4.1-21	58, 130
4.1-22	58, 83, 130
4.1-23	58, 83, 117, 125, 130
4.1-24	58, 109, 130, ___
4.1-25	58, 130
4.1-26	58, 82, 125, 130
4.1-27	58, 82, 125, 130
4.1-28	65, 82, 125, 130
4.1-29	65, 125, 130
4.1-30	82, 130
4.1-31	83, 130
4.1-32	83, 130
4.1-33	83, 130
4.1-34	119, 130
4.1-35	119, 130
4.2-1	25, 37, 54, 114, 130, ___
4.2-2	25, 37, 114, 130, ___
4.2-3	25, 37, 114, 130
4.2-4	25, 37, 114, 124, 130, ___
4.2-5	25, 37, 114, 124, 130
4.2-6	37, 114, 130
4.2-7	NRC order 4/20/81 54, 130
4.3-1	24, 58, 75, 130
4.3-2	5, 24, 58, 75, 87, 118, 130
4.3-3	24, 58, 87, 118, 130
4.3-4	24, 87, 130
4.3-5	24, 75, 87, 130
4.3-6	75, 87, 130
4.3-7	58, 130
4.3-8	58, 59, 83, 130
4.3-9	58, 83, 130
4.4-1	Change No. 12 25, 56, 82, 84, 95, 104, 123, 130
4.4-2	25, 84, 130
4.4-3	25, 56, 82, 84, 117, 130
4.4-4	25, 82, 84, 95, 104, 123, 124, 130
4.4-5	84, 130
4.4-6	25, 56, 84, 104, 123, 130
4.5-7	25, 56, 84, 130
4.5-1	Change No. 10 38, 79, 130
4.5-2	79, 90, 130
4.5-3	79, 130
4.5-4	79, 130
4.5-5	79, 130
4.5-6	79, 130
4.6-1	38, 79, 130
4.6-2	79, 90, 130
4.6-3	79, 130
4.6-4	79, 130
4.6-5	79, 130
4.6-6	79, 130
4.6-7	79, 130
4.6-8	79, 130
4.6-9	79, 130
4.7-1	Change No. 14 46, 130
4.8-1	91, 130
4.9-1	Change No. 14 130
4.9-2	130
4.10-1	13, 94, 130
4.10-2	13, 94, 130
4.11-1	14, 109, 130
4.11-2	14, 109, 130
4.12-1	91, 130
4.13-1	18, 113, 130
4.14-1	21, 33, 63, 81, 130
4.14-2	21, 63, 81, 130
4.14-3	21, 63, 81, 130
4.14-4	21, 63, 81, 105, 130
4.15-1	31, 37, 38, 44, 130, 131
4.15-2	31, 44, 130, 131
4.15-3	31, 130, 131
4.15-4	130, 131
4.15-5	130, 131
4.15-6	31, 130, 131
4.15-7	31, 130, 131
4.15-8	130, 131
4.15-9	130, 131
4.15-10	130, 131
4.15-11	130, 131
4.15-12	130, 131
4.16-1	31, 101, 115, 130
4.16-2	37, 101, 115, 130
4.16-3	37, 101, 115, 130
4.16-4	37, 55, 101, 115, 130
4.16-5	37, 91, 101, 105, 115, 130
4.16-6	37, 55, 101, 115, 130
4.16-7	37, 55, 101, 130
4.17-1	79, 130
4.18-1	79, 130
4.18-2	79, 130
4.18-3	79, 130
4.18-4	79, 130
4.18-5	79, 130
4.18-6	79, 130
4.19-1	79, 105, 130
4.20-1	102, 130

APPENDIX A

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

Page	Amendment No.	Page	Amendment No.	Page	Amendment No.
5.1-1	25, 72, 130	6.8-3	91, 130		
5.1-2	72, 79, 130	6.9-1	12, 16, 30, 91, 130		
5.2-1	72, 130	6.9-2	12, 16, 30, 79, 91, 96, 130		
5.2-2	72, 130	6.9-3	12, 15, 16, 30, 79, 91, 96, 130		
5.2-3	72, 130	6.9-4	12, 30, 79, 91, 96, 130		
5.3-1	3, 72, 130	6.9-5	12, 16, 30, 38, 79, 90, 91, 96, 105, 130		
5.3-2	3, 37, 72, 130	6.10-1	12, 15, 16, 91, 130		
5.3-3	72, 130	6.10-2	12, 63, 81, 91, 130		
5.4-1	130	6.11-1	58, 61, 91, 130		
6.1-1	12, 39, 66, 91, 130	6.12-1	23, 38, 58, 91, 130		
6.2-1	44, 66, 91, 110, 130	6.12-2	23, 38, 91, 130		
6.2-2	88, 91, 130	6.13-1	58, 79, 91, 130		
6.2-3	66, 88, 91, 105, 110, 126, 130	6.14-1	58, 79, 91, 130		
6.2-4	12, 58, 91, 130	6.15-1	58, 79, 91, 130		
6.3-1	12, 27, 39, 54, 58, 66, 91, 130	6.16-1	90, 91, 130		
6.4-1	31, 39, 54, 91, 130, ___	6.16-2	90, 91, 130		
6.5-1	Change No. 69 12, 39, 42, 54, 66, 91, 130	6.16-3	90, 91, 105, 130		
6.5-2	39, 54, 66, 91, 130	6.16-4	90, 91, 130		
6.5-3	12, 16, 39, 50, 54, 66, 91, 130				
6.5-4	39, 66, 69, 79, 91, 130				
6.5-5	12, 50, 90, 91, 130				
6.5-6	12, 16, 69, 91, 105, 130				
6.5-7	12, 31, 91, 110, 130				
6.6-1	12, 91, 130				
6.7-1	39, 54, 66, 91, 130				
6.8-1	39, 54, 66, 79, 91, 105, 130				
6.8-2	39, 91, 130				

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	3.4-2
	3.4.3 Auxiliary Feedwater System	3.4-3
	3.4.4 Auxiliary Feedwater Storage Tank	3.4-4
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	3.5-22
	3.5.8 Radioactive Liquid Effluent Instrumentation	3.5-26
	3.5.9 Radioactive Gaseous Process and Effluent Monitoring Instrumentation	3.5-29
	3.5.10 Radiation Monitoring Instrumentation	3.5-32
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 (MTC)	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
	3.14.1 Fire Suppression Water System	3.14-1
	3.14.2 Spray and/or Sprinkler Systems	3.14-3
	3.14.3 Foam Suppression System	3.14-7
	3.14.4 Halon Systems	3.14-9
	3.14.5 Fire Hose Stations	3.14-11

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

	Three Reactor Coolant Pumps Operating
1. Pressurizer High Level	≤ 50% Level
2. Pressurizer Pressure: High	≤ 2220 psig
3. Nuclear Overpower	
a. High Setting*	≤ 109% of indicated full power
b. Low Setting	≤ 25% of indicated full power
**4. Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341
**5. Coolant Flow	≥ 85% of indicated full loop flow
***6. Steam/Feedwater Flow Mismatch	
a. Low+ Setting:	$\frac{\text{Steam Flow - Feedwater Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$
b. High+ Setting:	$\frac{\text{Feedwater Flow - Steam Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$
**7. Reactor Coolant Pump Breaker Open	
a. Overcurrent	≤ 2400 amps
b. Undercurrent	≥ 110 amps
c. Undervoltage	≥ 60% of rated bus voltage

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* 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.

***May be bypassed at power levels below 50% of full power.

\+ High and Low feedwater flow relative to steam flow.

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 mCi/gm DOSE EQUIVALENT I-131.

ACTION: B. With the specific activity of the secondary coolant in excess of 0.1 mCi/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 mCi/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 pumps and valves for MODES 1, 2 and 3.

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

SPECIFICATION: Two trains of auxiliary feedwater including associated pumps and valves, shall be OPERABLE.

ACTION:

A. With one Train of auxiliary feedwater inoperable, restore the inoperable train 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.

B. With both Trains of auxiliary feedwater inoperable, be in at least HOT STANDBY within 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 from normal operating conditions in the event of a total loss of offsite power.

Two auxiliary feedwater trains and the steam system relief valves provide core decay heat removal capability in the event of a sustained loss of off-site power. Either auxiliary feedwater train has the capability to satisfy decay heat removal requirements from the core, with a delivered flow of at least 185 gpm per train with three intact main feedwater lines and pressurized steam generators, 125 gpm per train with two intact main feedwater lines and pressurized steam generators, and 250 gpm per train with two intact main feedwater lines and depressurized steam generators.(1)

AFW System Train A pumps and valves consist of AFW pumps G-10S and G-10 and associate valves, including flow control valves FCV-2300A, FCV-2300B, and FCV-2300C.

AFW System Train B pump and valves consist of AFW pump G-10W and associated valves, including flow control valves FCV-3300A, FCF-3300B, and FCV-3300C.

REFERENCES:

(1) SCE letter dated November 20, 1987, from M. O. Medford to NRC Document Control Desk.

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 usable water volume of at least 190,000 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. In addition, the water volume will provide sufficient margin to account for spillage that occurs during a main feedwater line break with loss of AFW flow indication prior to isolation of the broken line. Spillage is assumed to last no longer than one hour until the broken loop is identified via RCS Loop Delta-T positive indication that will be evident for the two intact steam generators. The usable water volume limit is specified relative to the bottom of the tank indicated level range (i.e., level tap). The contained water volume below this datum provides a significant margin to the NPSH and vortexing limits above the highest AFW pump suction inlet in the tank, but is not considered available for purposes of this specification.

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*		
o Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
o Steam Generator Water Level (Wide Range)	1/steam generator	1/steam generator
o Reactor Coolant System Loop Delta-T Indication	1/loop	1/loop
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
Refueling Water Storage Tank Level	2	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 is provided by one channel of auxiliary feedwater flow rate (Train B), one channel of environmentally qualified steam generator wide range level (Train A), and one channel of RCS Loop Delta-T indication. These comprise the three types of indication of auxiliary feedwater flow for each steam generator.

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

AMENDMENT NO: 58, 83, 117, 124, 125, 130,

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

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, CVS-301 and CVS-313, 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.

3.14.5 FIRE HOSE STATIONS

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

SPECIFICATION: The following fire hose stations shall be OPERABLE:

a. See Table 3.14.5.1

ACTION:

A. With one or more of the fire hose stations shown in Table 3.14.5.1 inoperable, route a fire hose* to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to plant workers, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose shall be mounted at the inoperable hose station.

B. The provision of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: In the event that a fire hose station is inoperable, the establishment of backup suppression in the affected areas is required to provide fire suppression capability until the inoperable system is restored to operability.

REFERENCES:

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.

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* Fire hose will be run within 1 hour of entering the ACTION statement if an operable water supply is not available within 250 feet of the area protected by the inoperable hose station, or two 150 feet hose packs (1-3/4") on the fire truck are not operable. Fire hose will be supplied by the fire department responding to a fire if an operable water supply is available within 250 feet of the area protected by the inoperable hose station. With the required hose station inside containment inoperable and containment integrity established, fire hose will be supplied only to the nearest access point.

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 at least once per 184 days it not seasonal. Groups to be sampled: 1. Fish-2 adult species such as flatfish, bass, perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets, clams or seahares.	Gamma isotopic analysis on edible portions.

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.
13. Reactor Coolant Pump Breaker Position*	S	R	R	N.A.	N.A.

*Applicable to Item 7 in Table 2.1

AMENDMENT NO: 7, 22, 83, 117, 122, 130,

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) and 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.

4.2 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 HV-851 A and B 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.

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.

C. RWST Low Level Trips

Monthly, perform a CHANNEL TEST and every refueling interval, perform a CHANNEL CALIBRATION, of the SI/Feedwater Pump trip and the MOV 850A, 850B and 850C automatic closure on low-low Refueling Water Storage Tank level.

BASIS:

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

6.4 TRAINING

6.4.1 A retraining and replacement training program for the NRC licensed unit staff shall be maintained under the direction of Manager, Nuclear Training and shall meet or exceed the requirements of 10 CFR Part 55. These programs shall include familiarization with relevant industry operational experience identified by the ISEG.

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 - 1975.