JUL 1 6 1982

Docket No.: 50-361

Mr. Robert Dietch
Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
Post Office Box 800
Rosemead, California 91770

Mr. Gary D. Cotton
Mr. Louis Bernath
San Diego Gas & Electric Company
101 Ash Street
Post Office Box 1831
San Diego, California 92112

Gentlemen:

Subject: Issuance of Amendment No. 4 to Facility Operating License NPF-10
San Onofre Nuclear Generating Station, Unit 2

The Nuclear Regulatory Commission has issued Amendment No. 4 to Facility Operating License NPF-10 for the San Onofre Nuclear Station, Unit 2, located in San Diego County, California.

This amendment is in response to your letter, dated May 14, 1982. The amendment makes a number of modifications to the Technical Specifications, Appendix A to the operating license.

The modifications:

(1) Add three valves that were inadvertently omitted from a table of motor operated valves requiring surveillance to verify bypassing of thermal overload protection.

(2) Add special test exceptions to allow performance of natual circulation tests.

(3) Make various editorial and typographical corrections.

(4) Clarify certain administrative controls.

(5) Require sampling of the milk injestion pathway when it is available.

(6) Clarify the containment air lock door seal pressure requirement.

The other changes requested in the May 14, 1982 letter are currently under review by the staff and may be addressed in future amendments to the operating license.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

A copy of the related safety evaluation supporting Amendment No. 4 to Facility Operating License MPF-10 is enclosed. The staff safety evaluation of the special test exceptions to allow performance of natural circulation tests is given in Supplement No. 6 to the Safety Evaluation Report (NUREG-0712). Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

Enclosures:
1. Amendment No. 4
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
As stated

Distribution:
See attached sheet

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DATE	6/30/82	7/6/82	7/8/82				

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMAPNY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the San Onofre Nuclear Generating Station, Unit 2 (the facility) Facility Operating License No. NPF-10 filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated May 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulation as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

OFFICE							
SURNAME							
DATE							

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 4, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUL 1 6 1982

no legal objection as to form subject to exemption [illegible]

OFFICE	DL:LB#3	DL:LB#3	OELD				
SURNAME	HRood:ph	FMiraglia	LChandler				
DATE	6/30/82	7/15/82	7/8/82				

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMAPNY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. NPF-10

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the San Onofre Nuclear Generating Station, Unit 2 (the facility) Facility Operating License No. NPF-10 filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated May 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulation as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 4, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUL 1 6 1982

AMENDMENT TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following pages of the Appendix A Technical Specification with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Overleaf Page	Amended Page
VI	V
XVII	XVIII
3/4 1-11	3/4 1-12
3/4 1-13	3/4 1-14
3/4 1-19	3/4 1-20
	3/4 1-25
3/4 3-3	3/4 3-4
3/4 3-12	3/4 3-11
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3/4 3-16	3/4 3-15
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3/4 3-21	3/4 3-22
	3/4 3-51
	3/4 3-52
	3/4 3-53
	3/4 3-53a
3/4 4-2	3/4 4-1
3/4 4-4	3/4 4-3
3/4 4-6	3/4 4-5
3/4 4-7	3/4 4-8
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REPLACEMENT PAGES (CONTINUED)

AMENDMENT NO. 4

Overleaf Page	Amended Page
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	6-4
	6-4a
	6-5
	6-6
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	7-2
B3/4 10-2	B3/4 10-1

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

ADMINISTRATIVE CONTROLS

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

1. A minimum borated water volume of 5465 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F or greater than 100°F.

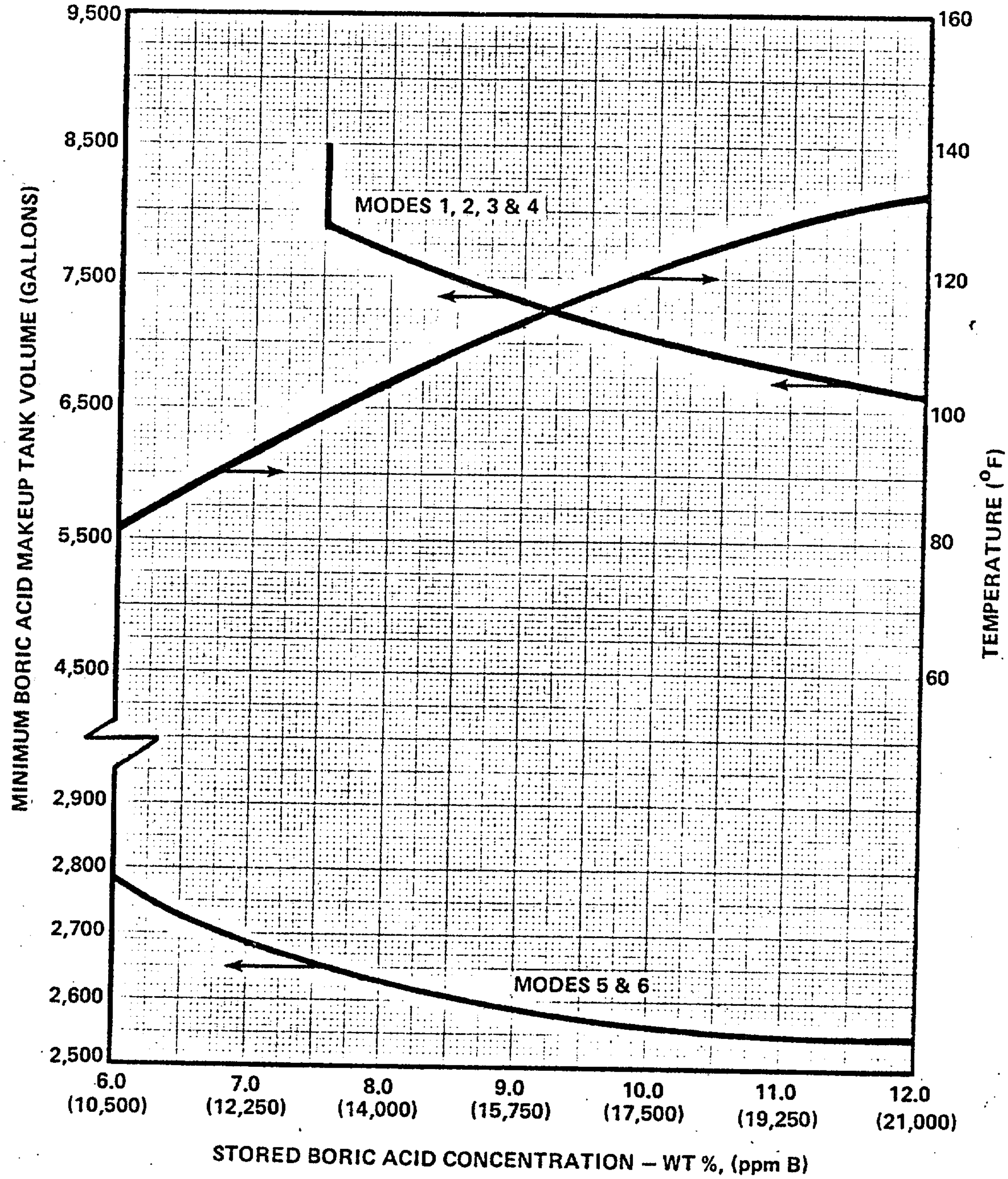

Figure 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND TEMPERATURE AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

2. Between 1720 and 2300 ppm of boron, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

POSITION INDICATOR CHANNEL - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4* and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and regulating) CEA drop time, from a withdrawn position greater than or equal to 145 inches, shall be less than or equal to 3.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. T_{avg} greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length CEA determined to exceed the above limit, be in at least HOT STANDBY within six hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be:

a. withdrawn to $\geq$ 145" or;

b. restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment ($\leq$ 145") of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1, 2 3*, 4*, 5*	1 7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4 4	2(a)(d) 2	3 3	1, 2 3*, 4*, 5*	2#, 3# 7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	2#, 3# 7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	5 7A
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
@17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

@To be OPERABLE prior to first exceeding 5% RATED THERMAL POWER.

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),S(7) R(4,5,10),M(8)	M(11),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow-Low	S	R	M	1, 2
@17. Seismic-High	S	R	M	1, 2
18. Loss of Load	S	N.A.	M	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

@ - To be OPERABLE prior to first exceeding 5% RATED THERMAL POWER.

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data.

(11) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.*

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. 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.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

*See Special Test Exception 3.10.5

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sest of 2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	9*, 10*
d. Automatic Actuation - Logic	4	2	3	1, 2, 3, 4	9*, 10*
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3	8
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	9*, 10*
c. Automatic Actuation Logic	4	2	3	1, 2, 3	9*, 10*
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
b. Manual SIAS (Trip Buttons) (c)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
c. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
d. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3	11
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	9*, 10*
c. Automatic Actuation Logic	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	9*, 10*
5. RECIRCULATION (RAS)					
a. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3, 4	9*, 10*
b. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*
6. CONTAINMENT COOLING (CCAS)					
a. Manual CCAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
b. Manual SIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3	8
c. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER (LOV)					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3, 4	9*, 10*
8. EMERGENCY FEEDWATER (EFAS)					
a. Manual (Trip Buttons)	2 sets of 2 per S/G	1 set of 2 per S/G	2 sets of 2 per S/G	1, 2, 3	11
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	9*, 10*
c. SG Level (A/B) - Low and ΔP (A/B) - High	4/SG	2/SG	3/SG	1, 2, 3	9*, 10*
d. SG Level (A/B) - Low and No S/G Pressure - Low Trip (A/B)	4/SG	2/SG	3/SG	1, 2, 3	9*, 10*

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 400 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Actuated equipment only; does not result in CIAS.

The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - 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.

ACTION 9 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

	Process Measurement Circuit	Functional Unit Bypassed
1.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Containment Pressure Circuit	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level - Low	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 9 are satisfied.

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

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, 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 1 hour for surveillance testing provided the other channel is OPERABLE.

Table 3.3-3 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 14 - With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9. (Mode 6 only)

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.9. (At all times)

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
c. Pressurizer Pressure - Low	$\geq$ 1806 psia (1)	$\geq$ 1763 psia (1)
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 16.14 psig	$\leq$ 16.83 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)(5)	Not Applicable	Not Applicable
c. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 729 psia (2)	$\geq$ 711 psia (2)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
5. RECIRCULATION (RAS)		
a. Refueling Water Storage Tank	18.5% of tap span	19.27% $\geq$ tap span $\geq$ 17.73%
b. Automatic Actuation Logic	Not Applicable	Not Applicable

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one or more radiation monitoring alarm channels inoperable, take the ACTION shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 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 4.3-7.

*See Special Test Exception 3.10.5.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure - Narrow Range	2	1	20, 21
2. Containment Pressure - Wide Range	2	1	20, 21
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	20, 21
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	20, 21
5. Pressurizer Pressure - Wide Range	2	1	20, 21
6. Pressurizer Water Level	2	1	20, 21
7. Steam Line Pressure	2/steam generator	1/steam generator	20, 21
8. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	20, 21
9. Refueling Water Storage Tank Water Level	2	1	20, 21
10. Auxiliary Feedwater Flow Rate	1/steam generator	N.A.	20
11. Reactor Coolant System Subcooling Margin Monitor	2	1	20, 21
12. Safety Valve Position Indicator	1/valve	N.A.	20
13. Spray System Pressure	2	1	20, 21
14. LPSI Header Temperature	2	1	20, 21
15. Containment Temperature	2	1	20, 21
16. Containment Water Level - Narrow Range	2	1	20, 21

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19. Containment Area Radiation - High Range	2	1	22, 23
20. Main Steam Line Area Radiation	1/steam line	N.A.	22
21. Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	22
22. Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	22, 23
23. Cold Leg HPSI Flow	1/cold leg	N.A.	20
24. Hot Leg HPSI Flow	1/hot leg	N.A.	20

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of channels OPERABLE less than the Required Number of Channels, comply with the ACTION requirements of Specification 3.3.3.6.

ACTION 23 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both Reactor Coolant loops and both Reactor Coolant pumps in each loop shall be in operation.

APPLICABILITY: 1 and 2.*

ACTION:

With less than the above required Reactor Coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required Reactor Coolant loops shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The Reactor Coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump.

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump.

b. At least one of the above Reactor Coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

a. 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.

b. 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.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one Reactor Coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be $\geq$ 10% (wide range) at least once per 12 hours.

*All Reactor Coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the loop(s)/train(s) listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,**

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,**

3. Shutdown Cooling Train A,

4. Shutdown Cooling Train B.

b. At least one of the above Reactor Coolant loops and/or shutdown cooling trains shall be in operation.*

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required Reactor Coolant loops and/or shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling train, be in COLD SHUTDOWN within 24 hours.

b. With no Reactor Coolant loop or shutdown cooling 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 coolant loop/ train to operation.

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*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 235°F unless 1) the pressurizer water volume is less than 900 cubic feet or 2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% (wide range) at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling train shall be OPERABLE and in operation,* and either:

a. One additional shutdown cooling train shall be OPERABLE,# or

b. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5#, with Reactor Coolant loops filled.

ACTION:

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

b. With no shutdown cooling 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 shutdown cooling train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators, when required, shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 The shutdown cooling train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling train is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling trains shall be OPERABLE# and at least one shutdown cooling train shall be in operation.*

APPLICABILITY: MODES 5 with Reactor Coolant loops not filled.

ACTION:

a. With less than the above required trains OPERABLE, immediately initiate corrective action to return the required trains to OPERABLE status as soon as possible.

b. With no shutdown cooling trains 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 shutdown cooling train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling train is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 900 cubic feet and at least two groups of pressurizer heaters powered from the 1E busses, each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups 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 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.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually energizing the heaters.

4.4.3.3 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

REACTOR COOLANT SYSTEM

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE $\leq$ 235°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

   1) A lift setting of 406 ± 10 psig*, and

   2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to 235°F; MODE 5; MODE 6 with the reactor vessel head on.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) within 7 days or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. In the event either the SDCS Relief Valve or an RCS vent is used to mitigate an 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 initiating the transient, the effect of the SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open.

*For valve temperatures less than or equal to 130°F.

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 235°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

   1) A lift setting of 406 ± 10 psig*, and

   2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 1%**.

APPLICABILITY: MODE 4 with RCS temperature above 235°F.

ACTION:

a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. In the event the SDCS Relief Valve or an RCS vent is used to mitigate an 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 initiating the transient, the effect of the SDCS Relief Valve code safety valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.

b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

4.4.8.3.2.3 The RCS vent shall be verified to be open at least once per 12 hours when the vent is being used for overpressure protection, except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

*For valve temperatures less than or equal to 130°F.

**The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, (55.7 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that seal leakage is less than or equal to 0.01 La when determined by flow measurement, with the volume between the door seals pressurized to greater than or equal to 9.5 ± 0.5 psig for at least 15 minutes,

b. By conducting overall air lock leakage tests at not less than P_a (55.7 psig), and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months,# and

2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, perform an engineering evaluation of the containment to demonstrate its structural integrity within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 Containment Tendons The containment's structural integrity shall be demonstrated at the end of one, three and five years after the initial structural integrity test (ISIT) and every five years thereafter with the exception of tendon lift off force and tendon detensioning and material tests and inspections which shall be determined at the end of one, five and ten years following the ISIT and every ten years thereafter in accordance with Table 4.6-1. The structural integrity shall be demonstrated by:

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the maximum first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the minimum lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to obtain a lift off force equal to +0, -5% of the prescribed upper limit. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a specific load differs by more than 5% from that recorded during installation of tendons, an investigation should be made to ensure that such difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be restored to the required level of integrity. More than one tendon below the predicted bounds out of the original sample population or the lift off force of a selected tendon lying below 90% of the prescribed lower limit is evidence of abnormal degradation of the containment structure.

b. Performing tendon detensioning and material tests and inspections of a previously stressed tendon wire or strand from one tendon of each group (hoop and U), and determining over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. A minimum tensile strength value of 270 ksi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing a visual inspection of the following:

1. Containment Surfaces - The structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined during the shutdown for, and prior to, each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation (e.g., widespread cracking, spalling and/or grease leakage).

2. End Anchorages - The structural integrity of the end anchorages (e.g., bearing plates, stressing washers, shims, wedges and anchorheads) of all tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by inspection that no apparent changes have occurred in the visual appearance of the end anchorage.

3. Concrete Surfaces - The structural integrity of the concrete surfaces adjacent to the end anchorages of tendons inspected

pursuant to Specification 4.6.1.6a shall be demonstrated by visual examination of the crack patterns to verify no abnormal material behavior.

d. Verifying the OPERABILITY of the sheathing filler grease by the following:

   1. No significant voids (in excess at 5% of the net duct volume), or the presence of free water, within the grease filler material, taking into account temperature variations.

   2. No significant changes have occurred in the physical appearance of the sheathing filler grease.

   3. Complete grease coverage exists for the anchorage system.

   4. Chemical properties are within the tolerance limits specified by the sheathing filler grease manufacturer.

TABLE 4.6-1

TENDON SURVEILLANCE

Years After Initial Structural Integrity Test	1		3		5		10		15	
	TENDON NUMBERS									
Type of Inspection	H	U	H	U	H	U	H	U	H	U
Visual Inspection of End Anchorages and Adjacent Concrete Surface	20 86 97 53 64	31-121 9-143 66-176 88-154	5 36 79 113 87	13-139 35-117 4-58 78-164	42 86 75 9 108	64-178 9-143 94-148 19-133	20 86 53	66-176 9-143 39-113	50 114 13	12-140 5-57 96-146
Prestress Monitoring Tests	20 86 97 53 64	31-121 9-143 66-176 88-154			42 86 75 9 108	64-178 9-143 94-148 19-133	20 86 53	66-176 9-143 39-113		
Dentensioning and Material Tests	97	88-154			42	19-133	20	66-176		

Years After Initial Structural Integrity Test	20		25		30		35		40	
	TENDON NUMBERS									
Type of Inspection	H	U	H	U	H	U	H	U	H	U
Visual Inspection of End Anchorages and Adjacent Concrete Surface	75 86 9	86-156 9-143 43-109	12 90 25	24-128 70-172 76-166	86 31 64	9-143 69-178 94-148	81 109 31	41-111 90-152 50-102	97 86 108	9-143 31-121 86-156
Prestress Monitoring Tests	75 86 9	86-156 9-143 43-109			86 31 64	9-143 64-178 94-148			97 86 108	9-143 31-121 86-156
Detensioning and Material Tests	75	43-109			31	64-178			86	9-143

TABLE 4.6-2

TENDON LIFT-OFF FORCE

U TENDONS

Tendon Number	Ends	First Year Maximum (kips)	First Year Minimum (kips)	X1	X2
43-109	43	1634	1457	21.2	31.2
	109	1604	1431	20.6	30.0
39-113	39	1625	1449	21.8	31.8
	113	1601	1428	20.0	30.0
31-121	31	1574	1406	21.2	29.3
	121	1586	1415	21.2	30.0
19-133	19	1644	1465	22.5	31.3
	133	1593	1423	20.6	30.0
9-143	9	1618	1444	21.8	31.2
	143	1598	1428	20.6	30.0
94-148	94	1560	1394	19.4	29.3
	148	1570	1403	20.6	28.7
88-154	88	1588	1415	21.2	30.0
	154	1568	1399	19.4	28.7
86-156	86	1567	1400	20.6	30.0
	156	1568	1399	19.4	28.7
66-176	66	1577	1407	20.6	30.0
	176	1579	1409	20.0	30.0
64-178	64	1560	1393	20.0	28.1
	178	1582	1412	20.6	28.7

HOOP TENDONS

Tendon Number	Buttress	First Year Maximum (kips)	First Year Minimum (kips)	Y1	Y2
9	2	1528	1348	26.8	36.8
	3	1502	1328	25.6	31.8
20	1	1569	1383	28.1	39.3
	3	1527	1348	25.6	36.2
31	1	1443	1281	23.1	31.8
	2	1502	1349	24.3	46.2
42	2	1577	1398	26.2	36.2
	3	1549	1395	24.3	46.2
53	1	1597	1416	26.2	36.2
	3	1564	1390	25.6	35.0
64	1	1607	1426	26.2	37.5
	2	1570	1396	25.6	35.6
75	2	1553	1374	26.2	36.2
	3	1525	1371	24.3	35.6
86	1	1600	1423	21.2	31.2
	3	1527	1362	20.6	29.3
97	1	1563	1393	20.6	29.3
	2	1546	1380	19.4	29.3
108	2	1626	1450	21.8	30.6
	3	1587	1418	20.6	28.7

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER			
3	3"-018-A-551#	High pressure safety injection	NA
3	HV-9323#	High pressure safety injection	NA
3	HV-9324#	High pressure safety injection	NA
5	3"-019-A-551#	High pressure safety injection	NA
5	HV-9326#	High pressure safety injection	NA
5	HV-9327#	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
9	PSV-9349#	Shutdown cooling relief	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
17	HV-4058#*	Steam generator secondary coolant sample	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
32	HV-8421#	Mainsteam atmospheric dump	NA
32	PSV-8410#	Mainsteam relief	NA
32	PSV-8411#	Mainsteam relief	NA
32	PSV-8412#	Mainsteam relief	NA
32	PSV-8413#	Mainsteam relief	NA
32	PSV-8414#	Mainsteam relief	NA
32	PSV-8415#	Mainsteam relief	NA
32	PSV-8416#	Mainsteam relief	NA
32	PSV-8417#	Mainsteam relief	NA
32	PSV-8418#	Mainsteam relief	NA
32	HV-8249B#	Mainsteam trap isolation	NA
32	HV-8202#	Mainsteam isolation bypass	NA
32	HV-8200#	Mainsteam to auxiliary feedwater turbine	NA
33	HV-8419#	Mainsteam atmospheric dump	NA
33	PSV-8401#	Mainsteam relief	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
33	PSV-8402#	Mainsteam relief	NA
33	PSV-8403#	Mainsteam relief	NA
33	PSV-8404#	Mainsteam relief	NA
33	PSV-8405#	Mainsteam relief	NA
33	PSV-8406#	Mainsteam relief	NA
33	PSV-8407#	Mainsteam relief	NA
33	PSV-8408#	Mainsteam relief	NA
33	PSV-8409#	Mainsteam relief	NA
33	HV-8248B#	Mainsteam trap isolation	NA
33	HV-8203#	Mainsteam isolation bypass	NA
33	HV-8201#	Mainsteam to auxiliary feedwater turbine	NA
36	HV-4054#*	Steam generator blowdown	NA
37	HV-4053#*	Steam generator blowdown	NA
39	3"-020-A-551#	High pressure safety injection	NA
39	HV-9329#	High pressure safety injection	NA
39	HV-9330#	High pressure safety injection	NA
41	3"-021-A-551#	High pressure safety injection	NA
41	HV-9332#	High pressure safety injection	NA
41	HV-9333#	High pressure safety injection	NA
42	HV-6223	Component cooling water inlet	NA
43	HV-6236	Component cooling water inlet	NA
44	HV-4057#*	Steam generator secondary coolant sample	NA
48	8"-072-A-552#@	Low pressure safety injection	NA
48	HV-9322#@	Low pressure safety injection	NA
49	8"-073-A-552#@	Low pressure safety injection	NA
49	HV-9325#@	Low pressure safety injection	NA
50	8"-074-A-552#@	Low pressure safety injection	NA
50	HV-9328#@	Low pressure safety injection	NA
51	8"-075-A-552#@	Low pressure safety injection	NA
51	HV-9331#@	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA

TABLE 3.7-2

MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Linear Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	98.9
2	86.6
3	74.2
4	61.8

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps 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 two auxiliary pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T-121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generators shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F.

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection shall be bypassed by a bypass device integral with the motor starter of each valve listed in Table 3.8.2 .

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection not bypassed by the integral bypass device, bypass the thermal overload protection within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection shall be verified to be bypassed by integral bypass devices:

a. At least once per 18 months,

b. Following maintenance on the valve motor starter, and

c. Following any periodic testing during which the thermal overload device was temporarily placed in force.

TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION BYPASS DEVICES

Permanently Bypassed

VALVE NUMBER	FUNCTION
HV-9339	Shutdown cooling flow from reactor coolant loop 2
HV-9340	SI tank T008 to reactor coolant loop 1A
HV-9370	SI tank T010 to reactor coolant loop 2B
HV-9347	SI pump minimum recirculation
HV-9322	LPSI Header to reactor coolant loop 1A
HV-9331	LPSI Header to reactor coolant loop 2B
HV-9348	SI pump minimum recirculation
HV-9323	HPSI Header #2 to reactor coolant loop 1A
HV-9332	HPSI Header #2 to reactor coolant loop 2B
HV-9217	RCP bleed off to volume control tank - cont. isol.
HV-9326	HPSI Header #2 to reactor coolant loop 1B
HV-9329	HPSI Header #2 to reactor coolant loop 2A
HV-7258	Waste gas surge tank header containment isolation
HV-0508	Reactor coolant hot leg #1 sample containment isolation
HV-0517	Reactor coolant hot leg #2 sample containment isolation
HV-9368	Shutdown HX E003 to containment spray Header #2
HV-0510	Pressurizer vapor sample containment isolation
HV-0512	Pressurizer surge line liquid sample containment isolation
HV-9950	Containment purge outlet to exhaust unit A060 - cont. isol.
HV-9917	Hydrogen purge exhaust unit A082 inlet - containment isol.
HV-9946	Hydrogen purge supply unit A080 discharge - containment isol.

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION
HV-9302	Containment emergency sump outlet
HV-9304	Containment emergency sump outlet
HV-6211	CCW Non-Critical Loop to containment - isolation valve
HV-6368	CCW Loop B to emergency cooling unit E400
HV-6369	CCW from emergency cooling unit E400 to loop B
HV-6216	CCW Non-Critical Loop from containment - isolation valve
HV-6372	CCW Loop B to emergency cooling unit E402
HV-6373	CCW Loop B from emergency cooling unit E402
HV-9900	Containment normal cooling supply isolation
HV-9971	Containment normal cooling return isolation
LV-0227C	Boric Acid makeup control
HV-4713	Aux. F.W. P141 discharge to steam generator E089 control valve
HV-9334	SI tank drain Header to refueling water tank - cont. isol.
HV-9350	SI tank T007 to reactor coolant loop 1B
HV-9360	SI tank T009 to reactor coolant loop 2A
HV-9325	LPSI Header to reactor coolant loop 1B
HV-9328	LPSI Header to reactor coolant loop 2A
HV-9201	Aux. spray to pressurizer
HV-9327	HPSI Header #1 to reactor coolant loop 1B
HV-9330	HPSI Header #1 to reactor coolant loop 2A
HV-6223	CCW Non-Crit Loop Containment inlet isolation
HV-9324	HPSI Header #1 to reactor coolant loop 1A
HV-9333	HPSI Header #1 to reactor coolant loop 2B
HV-9337	Shutdown coolant flow from reactor coolant loop 2
HV-9377	Shutdown coolant flow from reactor coolant loop 2

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION
HV-9378	Shutdown coolant flow from reactor coolant loop 2
HV-0516	Reactor coolant drain tank sample containment isolation
HV-7512	Containment isolation reactor coolant drain tank to R.W. system
HV-9367	Shutdown HX E004 to containment spray header #1
HV-0514	Quench tank vapor sample containment isolation
HV-5803	Containment sump to R.W. sump
HV-9949	Containment purge inlet from supply unit A374 isol.
HV-9303	Containment emergency sump outlet
HV-9305	Containment emergency sump outlet
HV-6366	CCW Loop A to emergency cooling unit E401
HV-6367	CCW Loop A from emergency cooling unit E401
HV-6236	CCW Non-crit. containment outlet isolation valve
HV-6370	CCW Loop A to emergency cooling unit E399
HV-6371	CCW Loop A from emergency cooling unit E399
HV-8150	Shutdown cooling HX E004 outlet isolation valve
HV-8151	Shutdown cooling HX E003 outlet isolation valve
HV-9306	SI pump minimum recirculation
HV-9307	SI pump minimum recirculation
HV-9247	Boric acid pumps to CVC charging pump suction
HV-9379	Shutdown cooling flow to LPSI
HV-9353	Shutdown cooling warm-up valve
HV-9420	HPSI Header #1 to reactor coolant loop 2 hot leg
HV-6497	Saltwater from CCW HX-E001
HV-9300	Refueling water tank east (T-005) outlet

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION
HV-5686	Firewater to containment isolation
HV-0227B	Volume control tank (T077) drain return
HV-9240	Boric acid makeup tank (T071) to charging pump suction
HV-9235	Boric acid makeup tank (T072) to charging pump suction
HV-9336	Shutdown cooling flow to LPSI pump suction
HV-9359	Shutdown cooling warm up valve
HV-9301	Refueling water tank west (T-006) outlet
HV-6495	Saltwater from CCW HX-E002
TV-9267	Letdown line containment isolation valve
HV-9434	HPSI Header #2 to reactor coolant loop 1 hot leg
HV-8152	Shutdown cooling HX inlet isolation valve
HV-8153	Shutdown cooling HX inlet isolation valve
HV-4712	Aux F.W. pump P504 discharge to Steam gen. control

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1.1 and the noted requirements of Table 2.2-1 and Table 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

SPECIAL TEST EXCEPTIONS

3/4.10.5 RADIATION MONITORING/SAMPLING

LIMITING CONDITION FOR OPERATION

3.10.5 The OPERABILITY requirements of Specifications 3/4.3.2, 3/4.3.3.1, 3/4.3.3.6, 3/4.3.3.8, and 3/4.3.3.9 for the radiation monitoring and sampling instrumentation listed in Table 3.10-1 may be modified per Table 3.10-1 provided the requirements listed in Table 3.10-1 are met.

APPLICABILITY: As shown in Table 3.10-1.

ACTION:

With the THERMAL POWER or criticality condition exceeding the limit for monitoring/sampling instrumentation as shown in Table 3.10-1, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.5 The monitoring/sampling instrumentation listed in Table 3.10-1 shall be demonstrated OPERABLE accordance with Specification 4.3.2, 4.3.3.1, 4.3.3.6, 4.3.3.8 or 4.3.3.9, as applicable, except as modified by Table 3.10-1.

TABLE 3.10-1

RADIATION MONITORING/SAMPLING EXCEPTIONS

1. Testing performed pursuant to FSAR Section 11.5.2.1.5.2 in startup program shall satisfy the initial CHANNEL CALIBRATION for the following monitors prior to first exceeding 5% RATED THERMAL POWER:

a.	Control Room Airborne Monitors	2RT-7824 2RT-7825
b.	Containment Airborne Monitors	2RT-7804-1 2RT-7807-2
c.	Containment Purge Area Monitors	2RT-7856-1 2RT-7857-2
d.	Containment Area Radiation - High Range Monitors	2RT-7820-1 2RT-7820-2
e.	Plant Vent Stack Airborne Monitor	2/3RT-7808
f.	Radwaste Discharge Line Monitor	2/3RT-7813
g.	Blowdown Neutralization Sump Monitor	2RT-7817
h.	Turbine Building Sump Monitor	2RT-7821

2. The following monitors and samplers shall be OPERABLE prior to first exceeding 5% RATED THERMAL POWER:

a.	Main Steam Line Area Monitors	2RT-7874A1 2RT-7874B1 2RT-7875A1 2RT-7875B1
b.	Condenser Evacuation System - Wide Range Monitor	2RT-7870-1
c.	Purge/Vent Stack Monitors - Wide Range	2RT-7865-1 3RT-7865-1
d.	Plant Vent Stack	Flow Rate Monitor
e.	Containment Purge	Flow Rate Monitor
f.	Condenser Evacuation System	Iodine Sampler Particulate Sampler Flow Rate Monitor

3. The Steam Jet Air Ejector Monitor (2RT-7818) shall be OPERABLE prior to initial criticality.

TABLE 3.10-1 (Continued)

4. Testing performed pursuant to FSAR Sectino 14.2.12 in startup program is acceptable for the initial CHANNEL FUNCTIONAL TEST for a period up to 30 days following initial criticality for the following liquid effluent monitors:

a.	Radwaste Discharge Line Monitor	2/3 RT-7813
b.	Blowdown Neutralization Sump Monitor	2RT-7817
c.	Turbine Building Sump Monitor	2RT-7821

5. Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 2RT-7804-1 or 2RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 2RT-7804-1 and associated sampling media shall perform the above required functions.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.6 The minimum temperature for criticality limits of Specification 3.1.1.4 and the MODE 2 definition of Table 1.1 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 320°F provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Linear Power Level - High neutron flux monitoring channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship and the minimum temperature for criticality are maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: MODE 2.*

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately trip the reactor.

b. With the Reactor Coolant System temperature and pressure relationship and/or the minimum temperature for criticality within the region of unacceptable operation on Figure 3.4-2, immediately trip the reactor and, if necessary, restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.6.1 At least once per hour:

a. The Reactor Coolant System temperature and pressure relationship and the minimum temperature for criticality shall be verified to be within the acceptable region for operation of Figure 3.4-2.

b. The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER.

c. The Reactor Coolant System temperature shall be verified to be greater than or equal to 320°F.

4.10.6.2 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

TABLE 3.12-1 (Continued)

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
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
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 semiannually and I-131 analysis for leafy crops.

TABLE 3.12-1 (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

b. Gamma isotopic analysis means the identification and quantification of gamma-emitting 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 ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposed 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.

f. Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the 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 measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The San Onofre Unit 2&3 fire pumps and water supplies, supply water to the San Onofre Unit 1 fire system.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements to verify OPERABILITY of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system allows for one of two alternatives. The connection can be made by back-feeding from Unit 3. Alternatively, the Unit 2 auxiliary transformer also may provide an alternate means of operation during low power PHYSICS TESTS. With the Unit 2 isolated-phase bus links removed, if preferred power from the Unit 2 reserve auxiliary transformer is lost, the 4.16 kV feeder circuit breaker can be inserted into the auxiliary transformer position to reestablish power to the Class 1E bus. Breaker controls for this connection,

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975 rather than re-issued. The reverified 1965 standard is therefore the appropriate standard to be used.

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kw, while the calculated accident loading is 4000 kw. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kw is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

The Surveillance Requirement 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 onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures 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 4.8-2 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 .015 below the manufacturer's full charge specific gravity or a battery charger current that had 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 the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 RADIATION MONITORING/SAMPLING

This special test exception permits fuel loading and reactor operation with radiation monitoring/sampling instrumentation calibration and quality assurance conforming to either FSAR procedures or Regulatory Guide 4.15 Rev. 1, February 1979. This test exception is required to allow for a phased implementation of Regulatory Guide 4.15 Rev. 1, February 1979. Equivalent instrumentation, quality assurance and/or calibration is provided until full implementation of Regulatory Guide 4.15 Rev. 1, February 1979.

The containment airborne monitors and associated sampling media test exception is required to allow for operation prior to and during installation of upgraded monitors/media. Adequate monitoring is provided until and subsequent to the completion of the upgraded installation. Extensive containment air mixing during high volume purge (MODES 5 and 6) occurs as a result of containment HVAC and fans resulting in representative air monitoring via either 2RT-7804-1 or 2RT-7807-2. During low volume purge operations (MODES 1, 2, 3 and 4) 2RT-7804-1 provides representative indication of purged air due to its location in the immediate vicinity of the low volume purge exhaust.

3/4.10.6 MINIMUM TEMPERATURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{avg} below 520°F during PHYSICS TESTS which provide data that can be used to verify the adequacy of design codes for new fuel designs for reduced temperature conditions. The Low Power Physics Testing program at low temperature (320°F) is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. Regulatory CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (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 of Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. 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 identified as such on Table 6.2-1.

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 either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include the Shift Supervisor and the 2 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.

#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.

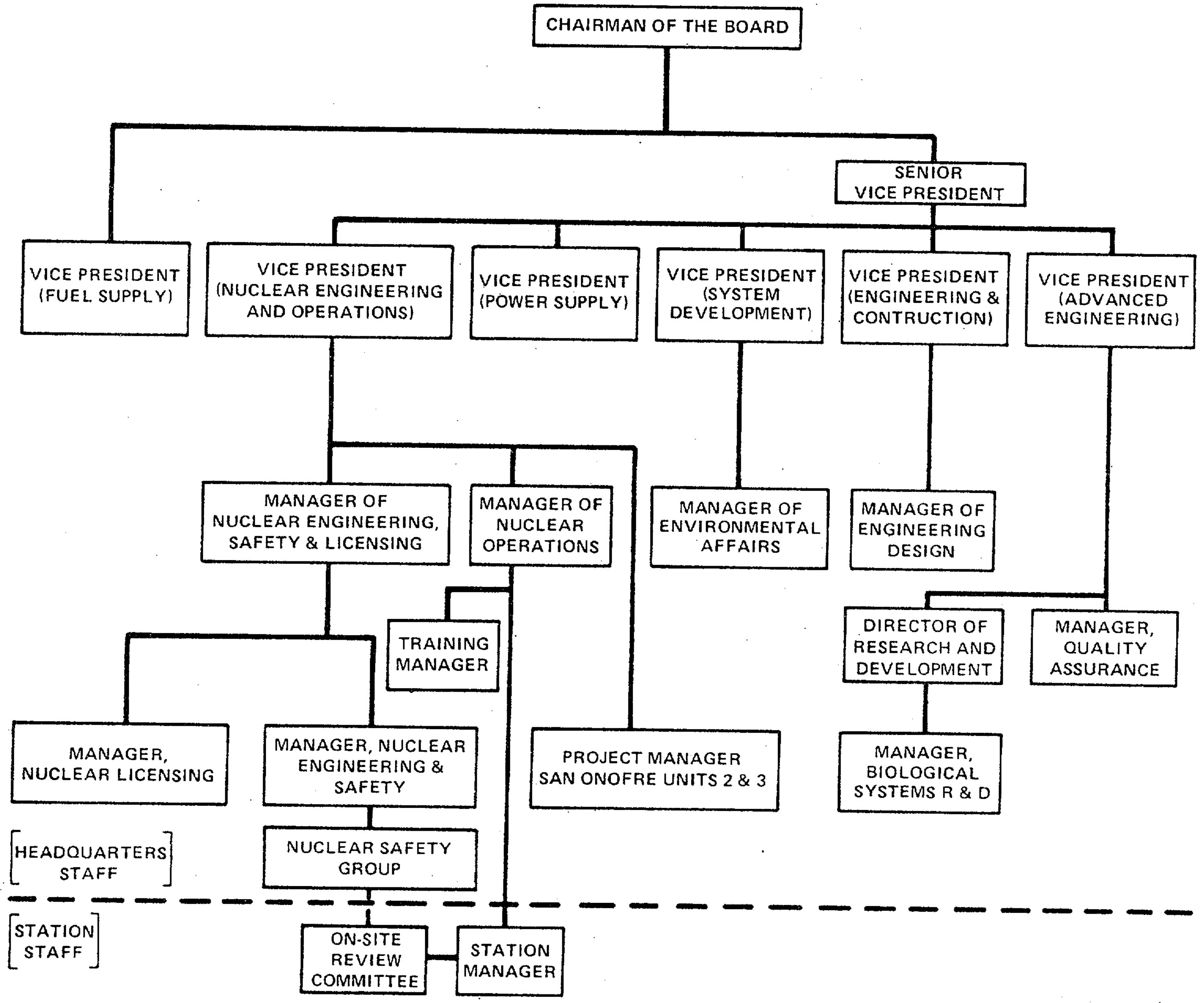
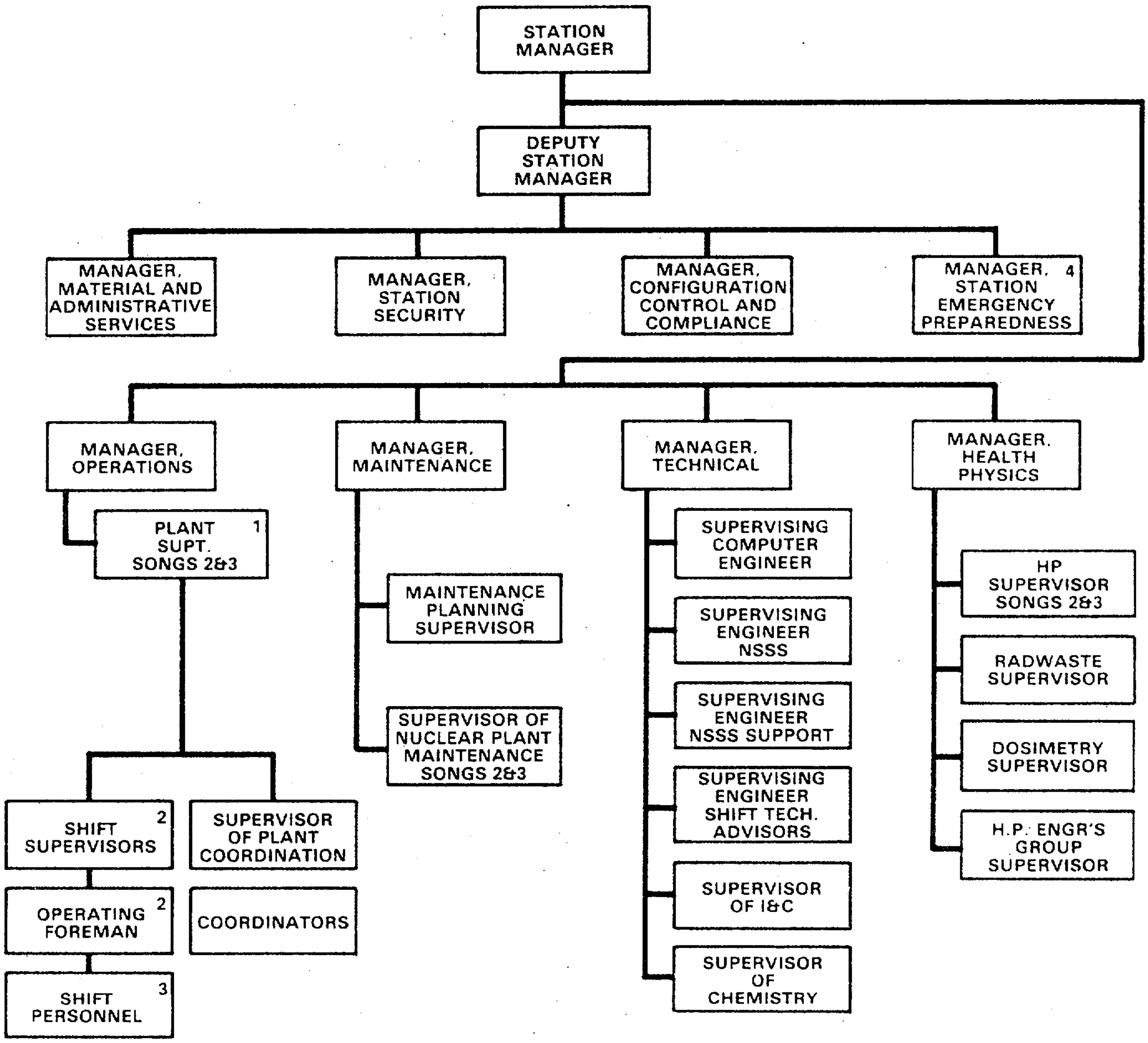


Figure 6.2-1
OFFSITE ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION – UNIT 2



1. AT TIME OF APPOINTMENT TO THE POSITION, SENIOR REACTOR OPERATOR LICENSE REQUIRED.
2. SENIOR REACTOR LICENSE REQUIRED.
3. CONTROL AND ASSISTANT CONTROL OPERATORS ARE HOLDERS OF REACTOR OPERATOR LICENSES.
4. INCLUDES FIRE PROTECTION.

Figure 6.2-2 Unit Organization, San Onofre Nuclear Generting Station-Unit 2

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
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 2
SRO - Individual with a Senior Reactor Operators License on Unit 2
RO - Individual with a Reactor Operators License on Unit 2
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, 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 Supervisor from the Control Room Area shown in Figure 6.2-3 while the unit is in MODE 1, 2, 3 or 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 Supervisor from the Control Room Area shown in Figure 6.2-3 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.

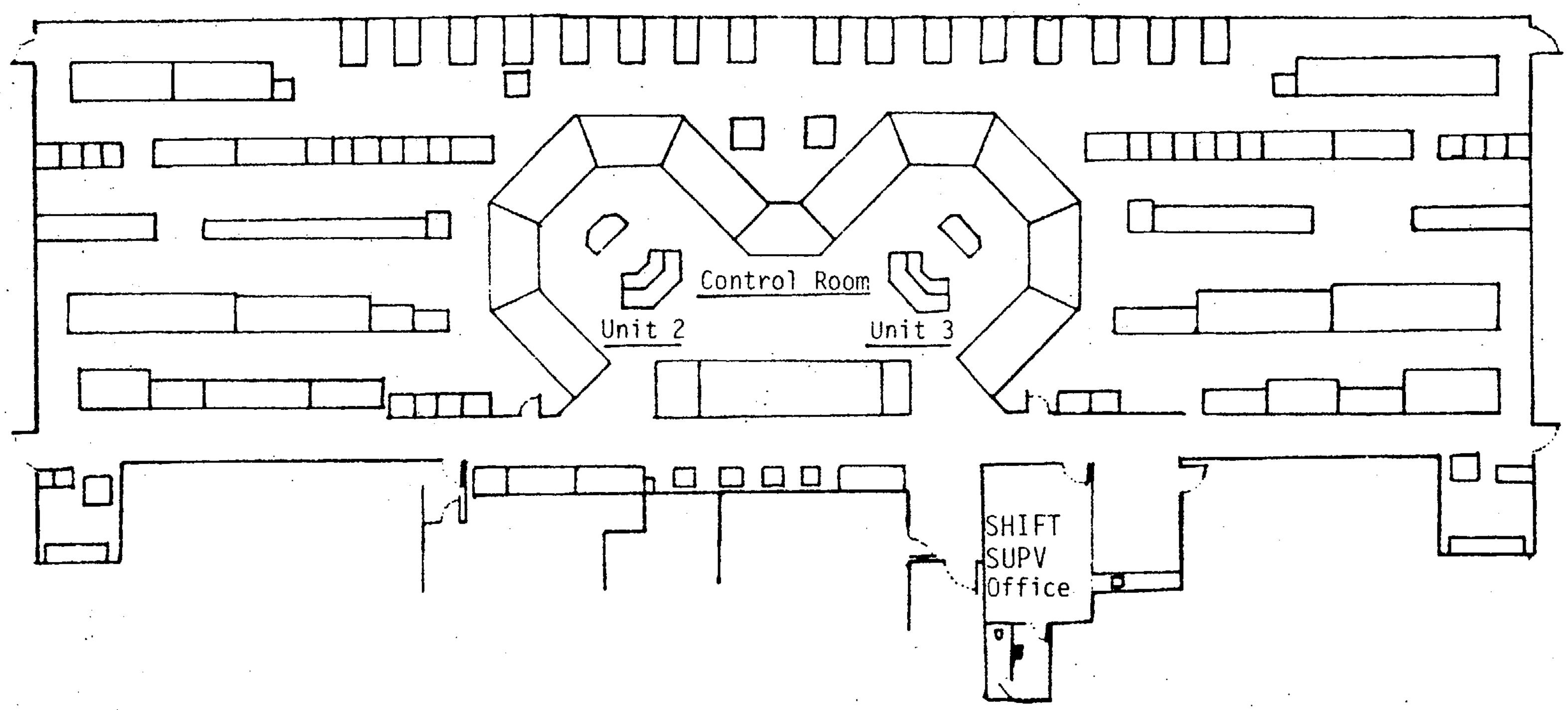


figure 6.2-3

CONTROL ROOM AREA

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 2

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five dedicated full-time engineers. Each shall have a Bachelor's Degree in Engineering or Physical Science and at least two years professional level experience in his field. Off-duty qualified Shift Technical Advisors may be used to fulfill this requirement.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Supervisor, Nuclear Safety Group.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit. 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.

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 for comparable positions, except for the Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

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 Sections 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 identified by the ISEG.

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 SONGS Unit 2 & 3
Member:	Supervisor of I&C
Member:	Manager, Health Physics
Member:	Supervisor of Chemistry
Member:	Manager, Maintenance
Member:	Supervising Engineer (NSSS, NSSS Support, Computer, 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 serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSRC activities at any one time.

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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.

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 four members including alternates.

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 events requiring 24-hour written notification to the Commission.

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.

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.

f. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

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 Manager of Nuclear Operations and NSG of disagreement between the OSRC and the Station Manager; however, the Station Manager 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 Station Manager 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.

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.

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 Station Manager.

ACTIVITIES (Continued)

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2 and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the Station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate 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, the Manager, Technical, the Manager, Operations, the Manager, Maintenance, the Deputy Station Manager or the Manager, Health Physics as previously designated by the Station Manager.

6.5.2.6 The station security program, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Manager of Nuclear Operations and to the NSG.

6.5.2.7 The station emergency plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Manager of Nuclear Operations and to the NSG.

6.5.2.8 The Station Manager shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Manager of Nuclear Operations and to the NSG.

6.5.2.9 The Station Manager shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

6.5.2.10 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Manager of Nuclear Operations and the Nuclear Safety Group.

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:

FUNCTION (Continued)

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 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 six 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 five 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 6.5.2.1. 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.

REVIEW

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.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the OSRC and submitted to the NSG and the Manager of Nuclear Operations.

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 Manager of Nuclear Operations and the NSG Chairman 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 Manager of Nuclear Operations 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 Rev. 1, February 1979.

NOTE: Quality Assurance Program for effluent and environmental monitoring and sampling shall be in accordance with Regulatory Guide 4.15, December, 1977 prior to first exceeding 5% RATED THERMAL POWER or July 1, 1982, whichever occurs first; subsequent to this time the Quality Assurance Program shall be in accordance with Regulatory Guide 4.15, Rev. 1, February, 1979.

j. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Onsite Review Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Manager, Operations (2) the Manager, Technical (3) the Manager, Maintenance, (4) the Deputy Station Manager, or (5) the Manager, Health Physics as previously designated by the Station Manager; 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 plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; 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 those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only) and the liquid radwaste system (post-accident sampling return piping only). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC and the NSG.

l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b 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 area 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.

*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.

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 Supervisor 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, 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 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 semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. 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 by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.

**Measurement made at 18" from source of radioactivity.
#The PCP shall be submitted and approved prior to shipment of "wet" solid radioactive waste.

7.0 SPECIAL TEST PROGRAM

7.1 For conducting the special low power test program as described in Section 22.2-1.G.1 of Supplement No. 1 to the Safety Evaluation Report (SER) the Technical Specifications may be exempt (E) or modified (C) as follows:

Technical Specifications Section	Description	Test A1	Test A2	Test A3
2.2.1	Reactor Trip Setpoints			
	2. Linear Power Level-High Four Reactor Coolant Pumps Operating	C(1)	C(1)	C(1)
	3. Logarithmic Power Level-High	C(2)	C(2)	C(2)
	5. Pressurizer Pressure-Low		C(3)	
	7. Steam Gen. Pressure-Low	C(4)	C(4)	C(4)
	9. Local Power Density-High	E(5)	E(5)	E(5)
	10. DNBR-Low	E(5)	E(5)	E(5)
	11. Reactor Coolant Flow-Low	E(5)	E(5)	E(5)
3.3.1	Reactor Protective Instrumentation			
	9. Local Power Density-High	E(5)	E(5)	E(5)
	10. DNBR-Low	E(5)	E(5)	E(5)
	14. Core Protection Calculators	E(5)	E(5)	E(5)
	16. Reactor Coolant Flow-Low	E(5)	E(5)	E(5)
3.3.2	Engineered Safety Feature Actuation System Instrumentation			
	1. Safety Injection (SIAS)		C(3)	
	4. Main Steam Line Isolation	C(4)	C(4)	C(4)
	6. Containment Cooling (CCAS)		C(3)	
	8. Emergency Feedwater (EFAS)	C(4)	C(4)	C(4)

Notes:

1. Trip setpoint lowered to $<$ 9.1% RATED THERMAL POWER, allowable value $\leq$ 10.4% RATED THERMAL POWER

2. Trip setpoint raised to $<$ 100% RATED THERMAL POWER, allowable value $\leq$ 100% RATED THERMAL POWER

3. Trip setpoint lowered to $\geq$ 1,550 psia

4. Trip setpoint lowered to $\geq$ 550 psia

5. Trip bypassed

SAFETY EVALUATION
AMENDMENT 4 to NPF-10
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2
DOCKET NO. 50-361

Introduction

By letter dated May 14, 1982, the Southern California Edison Company (SCE), on behalf of itself, San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim (the Licensees), requested the following changes to the San Onofre Nuclear Generating Station, Unit 2 Technical Specifications (TS).

1. Proposed Change

The Licensee proposes to modify Table 3.8-2, MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES. This proposed change will add three valves (HV-9377, 9378, and 9235) to the list, correct two valve numbers (LV-0227C and HV-8152), and modify the listed functional description of the valves to agree with the P&ID's and the instrument index.

Staff Evaluation

This change will add three valves that were inadvertently omitted from the valve listing, correct some typographical errors, and more clearly identify the function of the valves. This change is approved because it corrects errors and improves the consistency of the Technical Specifications.

2. Proposed Changes

The Licensee proposed the following typographical changes:

a. Technical Specification 3.4.8.3.1, APPLICABILITY: change from "one any" to "any one".

b. Technical Specification 3.7.1.5, Modes 2 and 3: The word "in" was omitted. It should read "Otherwise, be in at least HOT STANDBY within the next 6 hours...."

c. Technical Specification 3.10.5, Table 3.10-1, item 2.a: Main Steam Line Area Monitor "2RT-7847B1" should be "2RT-7874B1".

d. Technical Specification 3.10.6, Line 1: the word "of" is misspelled as "fo" and should be corrected.

Staff Evaluation

These changes are typographical in nature and therefore are approved.

3. Proposed Changes

The licensee has proposed the following modifications to the BASES Sections:

a. Section 3/4.7.8 Fire Supression Systems: insert a new paragraph.

"The San Onofre Unit 2 & 3 fire pumps and water supplies supply water to the San Onofre Unit 1 fire system. Satisfactory completion of the Unit 2 & 3 fire pump and water supply surveillance requirements automatically satisfies the Unit 1 fire water supply requirements."

b. Section 3/4.8, Add the following sentence to the discussion on fuel oil sampling.

"Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975, rather than re-issued. The reverified 1965 standard is therefore the appropriate standard to be used."

Staff Evaluation

These changes are for clarification of the Bases Sections and have no safety significance and are therefore approved, except for the second sentence under a., which will not be included because it is inappropriate to discuss Unit 1 requirements in the Unit 2 Technical Specifications.

4. Proposed Changes

In Technical Specifications Sections 3.1.2.7.b.3 and 3.1.2.8.b.3, the licensee proposes to change the required refueling water storage tank temperature from "a solution temperature between 40F and 120F" to "a solution temperature between 40F and 100F, " for consistency with Technical Specification 3/4.5.4.

Staff Evaluation

This is a change in the direction of greater conservatism since it restricts the allowable temperature range. The change also results in greater consistency between sections and is therefore approved.

5. Proposed Changes

The licensee proposes to change Technical Specification 3.1.3.7 to allow the Part Length CEA group to be withdrawn to greater than or equal to 145", for consistency with Specification 3.1.3.4 and 3.1.3.5.

Staff Evaluation

This change allows the Part Length CEA's to be considered "full out" at greater than or equal to 145", the same as the Shutdown CEA's of Specification 3.1.3.5. This improves consistency between specifications and is therefore approved.

6. Proposed Change

The licensee proposes to change Technical Specification 3.3.1, Table 3.3-1, notation (c) to read "bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER", vice 1%, to be consistent with the operation of the bistable. Table 3.3-1 ACTION 2 Line 7 reads, "Specification 6.5.1.6k" and should read "6.5.1.6e" because parts of Section 6.5.1.6 were deleted and item "k" was relettered as item "e".

Staff Evaluation

The changes are typographical in nature and are approved.

7. Proposed Change

The Licensee proposes to change Technical Specification 3.3.2, Table 3.3-3, item 2, to delete the containment spray OPERABILITY requirement for Mode 4. Containment Spray (CSAS) is required to be OPERABLE in Modes 1, 2, and 3 but not Mode 4 as defined in Technical Specification 3.6.2.1.

Staff Evaluation

The containment spray system uses some of the components of the shutdown cooling system. Many of these components can be aligned only to one system at a time, either the containment spray system or the SDCS. Since the shutdown cooling system is required to be operable in Mode 4, the containment spray system can not be operable also. Therefore, this change is approved.

8. Proposed Change

The Licensee proposes to change Technical Specification 3.3.2, Table 3.3-3 Item 5 to be consistent with the implied operability requirement of Technical Specification 3.5.3 which recognizes that in addition, MODE 4 is applicable. Therefore, Mode 4 should be added to Item 5 Table 3.3-3 under Applicable Modes.

Staff Evaluation

This additional operability requirement improves consistency between the specifications and is approved.

9. Proposed Change

The licensee proposes to change Technical Specification 3.3.2, Table 3.3-3 notation "a" to state "bypass shall be automatically removed when pressurizer pressure is greater than or equal to 400 psia", vice 500 psia. This change will make Table 3.3-3 consistent with Technical Specification 3.3.1, Table 3.3-1 notation "b".

Staff Evaluation

This change is typographical in nature and is approved.

10. Proposed Change

The Licensee proposes to change Technical Specification 3.3.2, Table 3.3-4, to delete Item 5a Manual (RAS). There are no manual RAS (Trip Buttons) in the plant and therefore this should be deleted. Item 5 should be relettered as applicable.

Staff Evaluation

There are no manual RAS (Trip Buttons) nor is there a requirement to install them. Therefore, this change is approved.

11. Proposed Change

The Licensee proposes to delete the following notation from Specification 3.4.1.3 "With the Reactor Coolant System cold leg temperature less than or equal to 235F, the SDCS isolation valves HV-9337, HV-9339, HV-9377, and HV-9378 shall be open with the SDCS relief valve PSV-9349 OPERABLE."

Staff Evaluation

Specification 3.4.1.3 is the controling specification for the Reactor Coolant System in Mode 4. The Specification for the Shutdown Cooling System is 3/4.4.8.3.1 and it specifies when the SDCS isolation valves should be open. This statement in Specification 3.4.1.3 is redundant and the change is therefore approved.

12. Proposed Change

The Licensee proposes to delete the words "with all suction line valves open" from Specification 3.4.1.4.1.

Staff Evaluation

This request is similar to the above request in item 11 in that the Shutdown Cooling System suction valves are controlled by the SDCS Specification. Therefore this change is approved.

13. Proposed Change

The Licensee proposes to add the words "powered from the IE busses" to Specification 4.4.3.2.

Staff Evaluation

This change is intended to clarify the pressurizer heater power supply requirement and is in accordance with SER Supplement No. 1, Section 22.2-II.E.3.1. We concur with the proposed change but will add the required words to Specification 3.4.3.

covering the company organization or contractor organization cognizant of the work conducted under the procedure.

Staff Evaluation

The proposed change to Technical Specification 6.5.2.2 adds a new specification. It is unacceptable because it would remove the responsibility for procedure preparation and review and review of nuclear safetyrelated programs, required elsewhere in the technical specifications, from the plant management to the quality assurance program. QA programs are not set up to address technical adequacy, but may be used for audits.

Without the new proposed Technical Specification 6.5.2.2, there is no need to renumber the succeeding specifications 6.5.2.3 through 6.5.2.10.

18. Proposed Change

Technical Specification 6.5.3.4.a: The NSG shall review: 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.

Staff Evaluation

The proposed change to Technical Specification 6.5.3.4.a merely clarifies the fact that the procedures referred to are only those required by Specification 6.8. There is no safety significance to the proposed change and it is, therefore, acceptable.

19. Proposed Change

Technical Specification 6.8.2: Each procedure of 6.8.1 and changes thereto shall be approved by the Station Manager or by (1) the Manager, Operations (2) the Manager, Technical (3) the Manager, Maintenance (4) the Deputy Station Manager or (5) the Manager, Health Physics as previously designated by the Station Manager prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

Staff Evaluation

The proposed change to Technical Specification 6.8.2 merely removes the topline management from having to review each procedure or change personally. The top management still has the responsibility for the review, as required by Specification 6.5.2.1, and still must approve the procedure or change. The proposed change makes 6.8.2 consistent with 6.5.2.1, and is acceptable.

20. Proposed Change

The licensee proposes to delete the requirement of monthly sampling local vegetation and performing a gamma isotopic analysis on the sample.

14. Proposed Change

The following should be added to Specification 6.2.4: "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."

Staff Evaluation

This addition is needed because, while the qualifications for all other plant staff are included in the Technical Specifications, those for the Shift Technical Advisor were, inadvertently, not included.

15. Proposed Change

Change Technical Specification S 6.1-2, 6.2-2, Table 6.2-1, Figure 6.2-2 and Figure 6.2-3 to reflect current personnel titles.

Staff Evaluation

The proposed changes to Figure 6.2-2 are primarily changes in position titles, as is suggested by the Licensee's description of the change. The inclusion of note 4 merely clarifies the fact that the Manager, Station Emergency Preparedness, is responsible for plant fire protection. There is no safety significance in these changes or in the deletion of the administrative position of Supervisor, Planning and Budgeting, under the Deputy Station Manager. Therefore, these changes are acceptable.

The proposed changes to Technical Specifications 6.1.2, 6.2.2.e, Table 6.2.-1 and Figure 6.2-3 are of no safety significance and are acceptable because they only change the "Watch Engineer" position title to "Shift Supervisor."

16. Proposed Change

Technical Specification 6.5.2.1: Each procedure and program required by Specification 6.8 and changes thereto, shall be 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.

Staff Evaluation

The proposed change to Technical Specification S 6.5.2.1 would delete the Station Manager's responsibility for assuring that the processes covered by 6.5.2.1 are carried out. The Station Manager must retain this ultimate responsibility. Therefore, the proposed change is unacceptable.

17. Proposed Change

Technical Specification 6.5.2.2: Procedures and Programs which affect nuclear safety (but are not required by Specification 6.8) and changes thereto, shall be prepared and reviewed in accordance with the Quality Assurance Program

Staff Evaluation

The San Onofre area is classified as semi-arid and local vegetation does not grow most of the year. Therefore, local vegetation is not necesarily a reliable indicator of an exposure pathway. As an alternative, the licensee has added a statement to Section 5.0 of the Offsite Dose Calculation Manual which commits to sampling the milk of a milk producing animal found during land use census performed in accordance with Technical Specifications 3.12.2. Sampling this milk will adequately cover this ingestion pathway. For these reasons, this change is approved.

21. Proposed Change

Technical Specification 4.6.1.3a: The Licensee proposed to change the door seal test pressure from 10 psig or greater to 9.5+0.5 psig. This change will assure the proper seating of the door seal by verifying the seal leakage is less than or equal to 0.01La, which is consistent with industry standards, while remaining within the limits of the manufacturer's pressure recommendations to avoid door seal damage.

Staff Evaluation

Changing the door seal test pressure from 10 psig or greater to 9.5+0.5 psig is necessary to meet the door seal manufacturer's recommendation to prevent damage to the door seals. The test pressure of 9.5 psig is as reliable as 10.0 psig for testing the proper seating of the air lock door seals. Therefore, this change is acceptable and is approved.

22. Proposed Change

Add Section 7, Special Test Exceptions for Natural Circulation Tests to the Technical Specification for the duration of the special lower power test program as described in Section 22.2-I.G.1 of Supplement No. 1 to the Safety Evaluation Report, NUREG-0712.

Staff Evaluation

In Supplement No. 1 to the SER the staff requirements and the licensees' commitments regarding special natural circulation testing were discussed. The staff required that the licensee submit, four weeks prior to conducting the tests, detailed test procedures and a safety analysis. By letter dated April 15, 1982, SCE provided the required information, thereby satisfying condition 2.B.(19)g of the San Onofre Unit 2 operating license, NPF-10, issued February 16, 1982. The proposed natural circulation test was discussed with the licensees in a meeting, in Bethesda, Maryland, on May 20, 1982. The staff has concluded that the proposed natural circulation tests are acceptable. The basis for our conclusion is given in Supplement No. 6 to the San Onofre 2 and 3 Safety Evaluation Report, NUREG-0712, dated June 1982.

23. Proposed Change

Technical Specification 3.10.5, Table 3.10-1, item 5: Change the second sentence to read "Containment airborne monitor 2RT-7804-1 or 2RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality." This change corrects the Technical Specifications to agree with Supplement No. 5 to the SER, Section 11.3. The Bases section corresponding to this Specification was also corrected.

Staff Evaluation

This change corrects an error in the Technical Specifications. The corrected wording agrees with the previously issued Safety Evaluation and is acceptable.

24. Proposed Change

In several separate discussions with the staff the licensee determined that the following Technical Specifications changes were necessary for the reasons noted.

Page	Change	Reason
3/4 1-20	1. Change "Control" CEA to "Regulating CEA".	Standard Terminology
	2. Change action statement to Hot Standby within 6 hours.	Standard Terminology
	3. Change Surveillance requirement to indicate that the requirement is in force after reinstallation of the reactor vessel head.	Clarification
3/4 3-11	Add once per shift and monthly surveillance on CPC's 70% Pwr.	Clarification not a new requirement.
3/4 3-51 to 3-53A	Change method of showing action requirements for inoperable accident monitoring instruments.	Clarification.
3/4 4-1	Change 3.4.1 to 3.4.1.1 Change 4.4.1 to 4.4.1.1	Typo Typo
3/4 4-31	Change Title of Section to PRESSURIZER-HEATUP/COOLDOWN	Clarification
3/4 4-32 and 4-33	Add Relief Valve Isolation Valve requirement to LCO.	The overpressure Protection System Technical Specification (T.S. 3.4.8.3) was split into two specifications; one applicable to RCS temperature 235°F, and one applicable to RCS 235°F. This requirement was inadvertently deleted in Technical Specification 3.4.8.3.2, RCS Temperature > 235°F and is being add for consistency of the requirement of the overall specification.

Page	Change	Reason
3/4 4-33	Add 30-day report requirement to the action statement. Change surveillance method.	(Identical to 3/4 4-32 and 4-33)
3/4 6-9	Change Retensioning Specification to be in accordance with Regulatory Guide	To make this requirement in compliance with the recommenda- of Proposed Revision 3 to Regulatory Guide 1.35, "Determining Prestressing Force for Inspection of Prestressed Conrete Containments," April 1979, and Proposed Regulatory Guide 1.35.1, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979.
3/4 6-10	Change terminology to agree with Regulatory Guide	(Identical to 3/4 6-9)
3/4 6-11	Require complete grease coverage	(Identical to 3/4 6-9)
3/4 6-12a	Change method of noting Tendon Ends and Buttresses	Clarification
3/4 6-23	Add # notation to SDC Relief Valve	Typo
3/4 7-4	Change exception of Specification 4.0.4 to show that it only applies to the turbine driven Auxiliary Feedwater Pump	Clarification
3/4 10-3	Add "Table 2.2.-1" to LCO	This Special Test Exception cannot be performed without suspending the requirement of Table 2.2-1

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves action which is insignificant from the standpoint of environmental impact, and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this statement.

Conclusion

Based upon our evaluation of the proposed changes to the San Onofre, Unit 2 Technical Specifications, we have concluded that: (1) because this amendment does not involve a significant increase in the probability or consequences of accidents previously considered, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant decrease in a safety margin, this amendment does not involve a significant safety hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-361

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT

FACILITY OPERATING LICENSE NO. NPF-10

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. NPF-10, issued to Southern California Edison Company, San Diego Gas and Electric Company, The City of Riverside, California and The City of Anaheim, California (licensees) for the San Onofre Nuclear Generating Station, Unit 2 (the facility) located in San Diego County, California. This amendment is effective as of the date of issuance.

Amendment No. 4 makes the following changes to the Facility Technical Specifications:

(1) Adds three valves that were inadvertently omitted from a table of motor operated valves requiring surveillance to verify bypassing of thermal overload protection.

(2) Adds special test exceptions to allow performance of natural circulation tests.

(3) Makes various editorial and typographical corrections.

(4) Clarifies certain administrative controls.

(5) Requires sampling of the milk injestion pathway when it is available.

(6) Clarifies the containment air lock door seal pressure requirement.

Issuance of this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that prusuant to 10 CFR 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Southern California Edison Company's letter dated May 14, 1982, (2) Amendment No. 4 to Facility Operating License No. NPF-10, (3) the Commission's related Safety Evaluation, and (4) Supplement No. 6 to the Safety Evaluation Report.

These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C., and the San Clemente Library, 242 Avenida Del Mar, San Clemente, California 02672. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 16 day of July, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
Division of Licensing

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-361

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OFFICE ▶							
SURNAME ▶							
DATE ▶							

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Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that prusuant to 10 CFR 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) Southern California Edison Company's letter dated May 14, 1982, (2) Amendment No. 4 to Facility Operating License No. NPF-10, (3) the Commission's related Safety Evaluation, and (4) Supplement No. 6 to the Safety Evaluation Report.

These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C., and the San Clemente Library, 242 Avenida Del Mar, San Clemente, California 02672. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 16 day of June, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Frank J. Miraglia

Frank J. Miraglia, Chief
Licensing Branch No. 3
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