APR 25 1983

Docket No.: 50-361

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Mr. Gary D. Cotton
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Gentlemen:

Subject: Issuance of Amendment No. 16 to Facility Operating License No. NPF-10, San Onofre Nuclear Generating Station, Unit 2

The Nuclear Regulatory Commission has issued Amendment No. 16 to Facility Operating License No. NPF-10 for the San Onofre Nuclear Station, Unit 2, located in San Diego County, California.

This amendment is in response to your letters dated September 3, October 21, and December 1, 1982 and February 4, 1983. The amendment updates the Unit 2 Technical Specifications to make them consistent with the San Onofre Unit 3 Technical Specifications.

A copy of the related safety evaluation supporting Amendment No. 16 to Facility Operating License No. NPF-10 is enclosed. 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:
George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosures:
1. Amendment No. 16 to NPF-10
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

No legal objection to Notice of Amendment form 4/22

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SAN ONOFRE NUCLEAR GENERATING STATION , UNIT 2 (AMENDMENT NO. 16)

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

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. 16
License No. NPF-10

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

   A. The applications for amendment for the San Onofre Nuclear Generating Station, Unit 3 (the facility) 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 September 3, October 21 and December 1, 1982, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations 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. 16, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment becomes effective on May 16, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Date of Issuance: APR 25 1983

4/22/83
DWO from Amend & Date

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ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-362

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amendment pages.

Amendment Page	Overleaf Page
I	II
VIII	VII
IX	-
X	-
XI	-
XII	-
XIII	-
XIV	-
XVIII	XVII
XIX	-
XX	-
XXI	-
XXII	-
1-3	1-4
3/4 1-5	3/4 1-6
3/4 2-2	3/4 2-1
3/4 3-3	-
3/4 3-4	-
3/4 3-8	3/4 3-7
3/4 3-9	-
3/4 3-10	-
3/4 3-11	-
3/4 3-12	-
3/4 3-15	3/4 3-16
3/4 3-17	-
3/4 3-18	-
3/4 3-24	3/4 3-23
3/4 3-25	-
3/4 3-26	-
3/4 3-28	3/4 3-27
3/4 3-29	3/4 3-30
3/4 3-31	-
3/4 3-32	-
3/4 3-33	3/4 3-34

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Amendment Page	Overleaf Page
3/4 3-53	-
3/4 3-53a	-
3/4 3-56	3/4 3-55
3/4 3-57	-
3/4 3-58	-
3/4 3-67	3/4 3-68
3/4 3-70	3/4 3-69
3/4 3-74	3/4 3-73
3/4 3-75	3/4 3-76
3/4 4-2	3/4 4-1
3/4 4-3	3/4 4-4
3/4 4-5	-
3/4 4-6	-
3/4 4-18	3/4 4-17
3/4 4-27	3/4 4-28
3/4 4-29	-
3/4 4-30	-
3/4 5-3	-
3/4 5-4	-
3/4 6-6	3/4 6-5
3/4 6-13	-
3/4 6-14	-
3/4 6-15	3/4 6-16
3/4 6-18	3/4 6-17
3/4 6-21	-
3/4 6-22	-
3/4 7-13	3/4 7-14
3/4 7-26	3/4 7-25
3/4 7-27	3/4 7-28
3/4 7-29	3/4 7-30
3/4 7-34	3/4 7-33
3/4 7-35	-
3/4 8-4	3/4 8-3
3/4 8-18	3/4 8-17
3/4 8-19	-
3/4 8-20	-
3/4 8-21	-
3/4 8-22	-
3/4 8-23	-
3/4 8-24	-
3/4 8-25	3/4 8-26
3/4 10-1	-
3/4 10-2	-

OFFICE ▶							
SURNAME ▶							
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Amendment Page	Overleaf Page
3/4 11-7	3/4 11-8
3/4 11-9	3/4 11-10
3/4 11-15	3/4 11-16
3/4 12-11	-
3/4 12-12	-
B 3/4 1-3	B 3/4 1-4
B 3/4 4-1	B 3/4 4-2
B 3/4 4-7	B 3/4 4-8
B 3/4 4-9	-
B 3/4 5-3	-
B 3/4 6-2	B 3/4 6-1
B 3/4 7-2	B 3/4 7-1
B 3/4 9-2	B 3/4 9-1
B 3/4 9-3	-
B 3/4 10-1	-
B 3/4 10-2	-
6-2	6-1
6-4	6-3
6-5	6-6
6-11	-
6-12	-
6-14	6-13
6-19	-
6-20	-
6-25	6-26

OFFICE ▶							
SURNAME ▶							
DATE ▶							

INDEX

DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

INDEX

BASES

INDEX

BASES

INDEX

ADMINISTRATIVE CONTROLS

INDEX

LIST OF TABLES

LIST OF TABLES

INDEX

LIST OF TABLES

LIST OF FIGURES

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.20 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

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

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 520°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 520°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 520°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 535°F.

#With K_{eff} greater than or equal to 1.0.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

b. At least once per 31 days by 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed 13.9 kw/ft.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kw/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is within the limit of 13.9 kw/ft.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on kw/ft.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC and COLSS addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTORS to a factor greater than or equal to (F^m_{xy}/F^c_{xy}); or

b. Adjust only the CPC addressable constants as in (a). Restrict subsequent operation so that a margin to the COLSS operating limits of at least [(F^m_{xy}/F^c_{xy}) - 1.0] x 100% is maintained; or

c. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

d. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 EFPD.

*See Special Test Exception 3.10.2.

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	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	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	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 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

*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 NOTATION

maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 1. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

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

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.45 seconds*
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 0.90 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	≤ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. Cold Leg Temperature	≤ 0.68 seconds##
d. Hot Leg Temperature	≤ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	≤ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.68 seconds
g. CEA positions: CEAC Penalty Factor	≤ 0.53 seconds

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4), R(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

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),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

* - 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) - Deleted.

(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.

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, 4	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)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. CONTROL ROOM ISOLATION (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*#
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*#
ii. Gaseous	2	1	1	All	13*#
d. Automatic Actuation Logic	1/train	1	1	All	13*#
10. TOXIC GAS ISOLATION (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*#, 15*#
b. Chlorine - High	2	1	1	All	14*#, 15*#
c. Ammonia - High	2	1	1	All	14*#, 15*#
d. Butane/Propane - High	2	1	1	All	14*#, 15*#
e. Carbon Dioxide - High	2	1	1	All	14*#, 15*#
f. Automatic Actuation Logic	1/train	1	1	All	14*#, 15*#

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
11. FUEL HANDLING ISOLATION (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. CONTAINMENT PURGE ISOLATION (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17*#
b. Airborne Radiation					
i. Gaseous	2	1	1	All	17, 17a, 17b
ii. Particulate	2	1	1	All	17, 17a, 17b
iii. Iodine	2	1	1	All	17, 17b
c. Containment Area Radiation (Gamma)	2	1	1	6	17*#
d. Automatic Actuation Logic	1/train	1	1	All	17, 17a, 17b*#

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. CONTAINMENT COOLING (CCAS)		
a. Manual CCAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Automatic Actuation Logic	Not Applicable	Not Applicable
7. LOSS OF POWER (LOV)		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	See Fig. 3.3-1 (4)	See Fig. 3.3-1 (4)
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level-Low	$\geq$ 25% (3)	$\geq$ 24.23% (3)
c. Steam Generator ΔP-High (SG-A > SG-B)	$\leq$ 50 psi	$\leq$ 66.25 psi
d. Steam Generator ΔP-High (SG-B > SG-A)	$\leq$ 50 psi	$\leq$ 66.25 psi
e. Steam Generator (A&B) Pressure - Low	$\geq$ 729 psia (2)	$\geq$ 711 psia (2)
f. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. CONTROL ROOM ISOLATION (CRIS)		
a. Manual CRIS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Airborne Radiation		
i. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
ii. Gaseous	$\leq 3.8 \times 10^2$ cpm**	$\leq 4.0 \times 10^2$ cpm**
d. Automatic Actuation Logic	Not Applicable	Not Applicable
10. TOXIC GAS ISOLATION (TGIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Chlorine - High	$\leq$ 6.0 ppm	$\leq$ 6.2 ppm
c. Ammonia - High	$\leq$ 42.4 ppm	$\leq$ 44.7 ppm
d. Butane/Propane - High	$\leq$ 84.8 ppm	$\leq$ 89.3 ppm
e. Carbon Dioxide - High	$\leq$ 4061.3 ppm	$\leq$ 4275.0 ppm
f. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^2$ cpm**	$\leq 1.4 \times 10^2$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq$ per ODCM	$\leq$ per ODCM
ii. Particulate	$\leq$ per ODCM	$\leq$ per ODCM
iii. Iodine	$\leq$ per ODCM	$\leq$ per ODCM
c. Containment Area Radiation (Gamma)	$\leq$ 2.4 mR/hr	$\leq$ 2.5 mR/hr
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

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

*Variable setpoints are for use only during normal, controlled plant heatups and cooldowns.

**Above normal background.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
1. Manual	
a. SIAS	
Safety Injection	Not Applicable
Control Room Isolation	Not Applicable
Containment Isolation (3)	Not Applicable
Containment Emergency Cooling	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. CCAS	
Containment Emergency Cooling	Not Applicable
g. EFAS	
Auxiliary Feedwater	Not Applicable
h. CRIS	
Control Room Isolation	Not Applicable
i. TGIS	
Toxic Gas Isolation	Not Applicable
j. FHIS	
Fuel Handling Building Isolation	Not Applicable
k. CPIS	
Containment Purge Isolation	Not Applicable

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) CCW Valves (Note 4a)	20.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation	20.9
(2) Main Feedwater Isolation	10.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (steam/DC train)	30.9 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (Steam/DC train)	30.9 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs.

4a. CCW non-critical loop isolation valves 2HV-6212, 2HV-6213, 2HV-6218 and 2HV-6219.

4b. Containment emergency cooler CCW isolation valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3, 4
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Containment Pressure -- High - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)(5)	N.A.	N.A.	R	1, 2, 3, 4
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3, 4
4. MAIN STEAM ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
5. RECIRCULATION (RAS)				
a. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3, 4
6. CONTAINMENT COOLING (CCAS)				
a. Manual CCAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
c. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. CONTROL ROOM ISOLATION (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. TOXIC GAS ISOLATION (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Carbon Dioxide - High	S	R	M	All
f. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	(2)	(2)	(2)	All
ii. Particulate	(2)	(2)	(2)	All
iii. Iodine	(2)	(2)	(2)	All
c. Containment Area Radiation (Gamma)	S	R	M	6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

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

* With irradiated fuel in the storage pool.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING ALARM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring alarm instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.*

ACTION:

a. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring alarm channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring alarm 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 4.3-3.

*See Special Test Exception 3.10.5.

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	20, 21
20. Main Steam Line Area Radiation	1/steam line	N.A.	20
21. Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	20
22. Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	22
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 OPERABLE accident monitoring channels less than the Required Number of Channels, 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.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Containment Area Radiation - High Range	(a)	(a)
20. Main Steam Line Area Radiation	(a)	(a)
21. Condenser Evacuation System Radiation Monitor - Wide Range	M	R
22. Purge/Vent Stack Radiation Monitor - Wide Range	M	R
23. Cold Leg HPSI Flow	M	R
24. Hot Leg HPSI Flow	M	R

NOTES:

(a) In accordance with Table 4.3-3.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.7.4 Following a seismic event (basemat acceleration greater than or equal to 0.05 g):

a. Within 2 hours each zone shown in Table 3.3-11 shall be inspected for fires, and

b. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"			10			
	Cable Tray Areas Elev 45'			9			
	Cable Tray Areas Elev 30'			4			
	Elevator Machinery Room			1			
	Combustible Oil Area Two steam generator rooms					32	
	Charcoal Filter Area Elev 45'	2					
2	Penetration Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas						
	Spent Fuel Pool		4				
	New Fuel Pool		3				
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2		24	
	Cable Riser Gallery Rm 449			3		24	
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A		None				
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A	1		1			
	Emgy. A.C. Unit Rm 302-Train B	1		1			
10	Penetration Elev 45'			6			

*The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None					
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3		42	
	Cable Riser Gallery Rm 315			3		40	
13A	Control Building Elev 50' Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50' Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50' Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3			4
	Train B			3			4
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2			6
23	Fuel Handling Bldg Elev 30'						
	Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'	2					

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13*

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

*See Special Test Exception 3.10.5

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT - 7814 or 2/3 RT - 7808	1	*	35
b. Effluent System Flow Rate Measuring Device	1	*	36
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor - 2RT - 7818 or 2RT - 7870-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
4. PLANT VENT STACK			
a. Noble Gas Activity Monitor - 2/3 RT - 7808, or 2RT-7865-1 and 3RT-7865-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2RT - 7804-1	1	*	38, (b),(c)
b. Iodine Sampler	1	*	40, (c)
c. Particulate Sampler	1	*	40, (b), (c)
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

a) In accordance with Table 3.3-6 ACTION 19

b) In accordance with the ACTION Requirements of Specification 3.4.5.1 (Modes 1, 2, 3 and 4)

c) In accordance with the ACTION Requirement of Specification 3.9.9 (Mode 6)

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:#

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists#:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) Prior to each release and at least once per month.

(7) Surveillance of containment airborne monitor 2RT-7807-2 and its associated sampling media, when required OPERABLE by other Specifications, shall be in accordance with the Surveillance Requirement for Containment Purge Effluent monitoring.

#If the instrument controls are not set in the operate mode, procedures shall call for declaring the channel inoperable.

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2* and 3.*

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

   1. Four high pressure turbine stop valves.
   2. Four high pressure turbine control valves.
   3. Six low pressure turbine reheat stop valves.
   4. Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

SURVEILLANCE REQUIREMENTS (Continued)

**For performance of surveillance testing of the valves in the unaffected low pressure turbine steam lead, the valve or valves closed in an affected low pressure turbine steam lead may be re-opened following successful completion of surveillance testing of that valve in accordance with 4.3.4.2a. for a period not to exceed one hour. The provisions of Specification 3.0.4 are not applicable to the low pressure turbine valves. This license amendment is in effect during the time interval of December 6, 1982 until the completion of 50% power startup testing or until January 31, 1983, whichever occurs first.

3/4.4 REACTOR COOLANT SYSTEM

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.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

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

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

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 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

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

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

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

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.

HOT SHUTDOWN

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.

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: MODE 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.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator.

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inlet flow at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying valve leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to declaring the valve operable following maintenance, repair or replacement work on the valve.

d. Within 48 hours following valve actuation due to automatic or manual action or flow through the valve (for valves in Section B of Table 3.4-1).

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 and Figure 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 280°F. A maximum heatup of 60°F in any one hour period with RC cold leg temperature greater than 280°F.

b. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 280°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 280°F.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

a. The actual shift in reference temperature for plates C-6404-2 as determined by impact testing, or

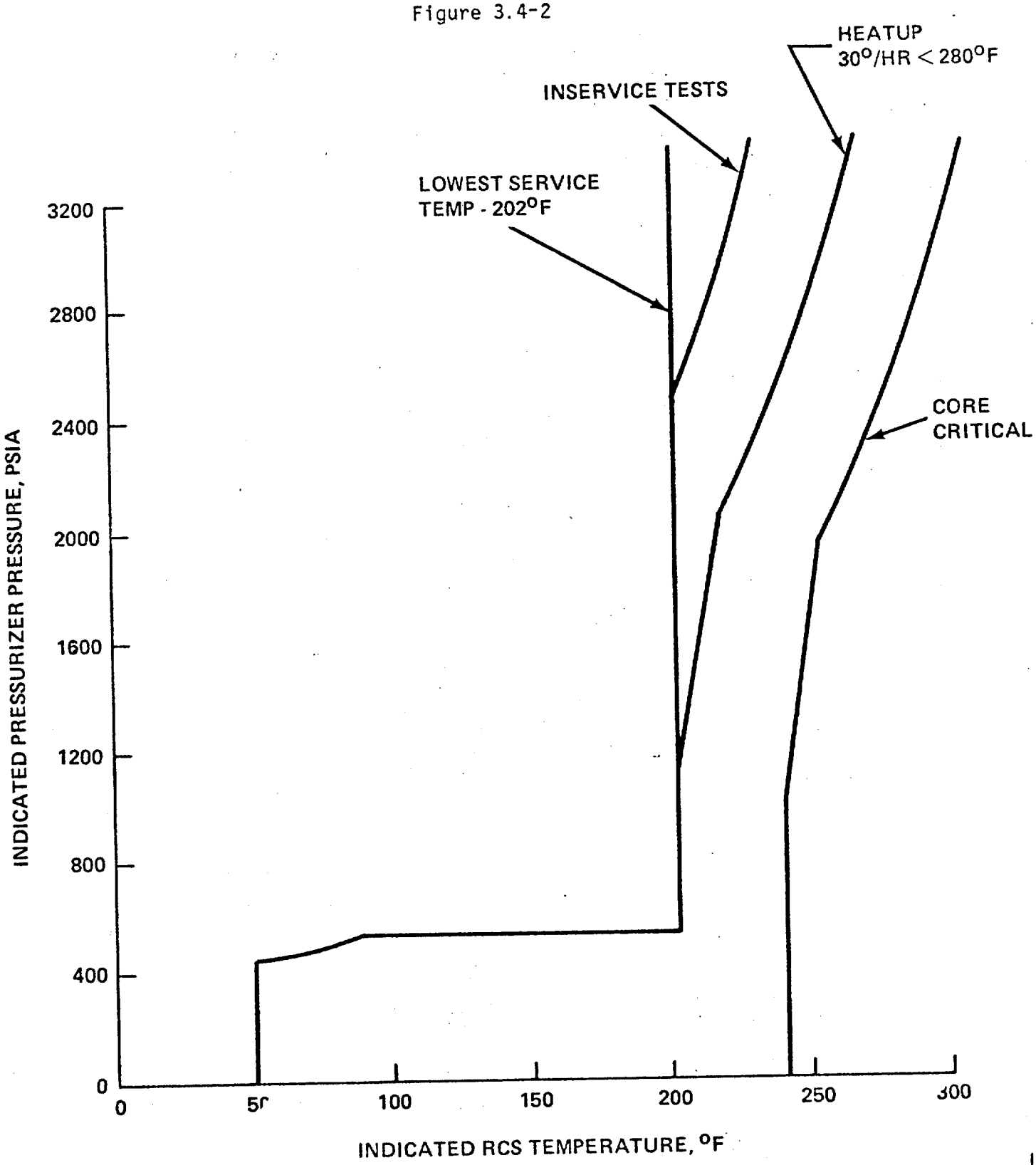
b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, Revision 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

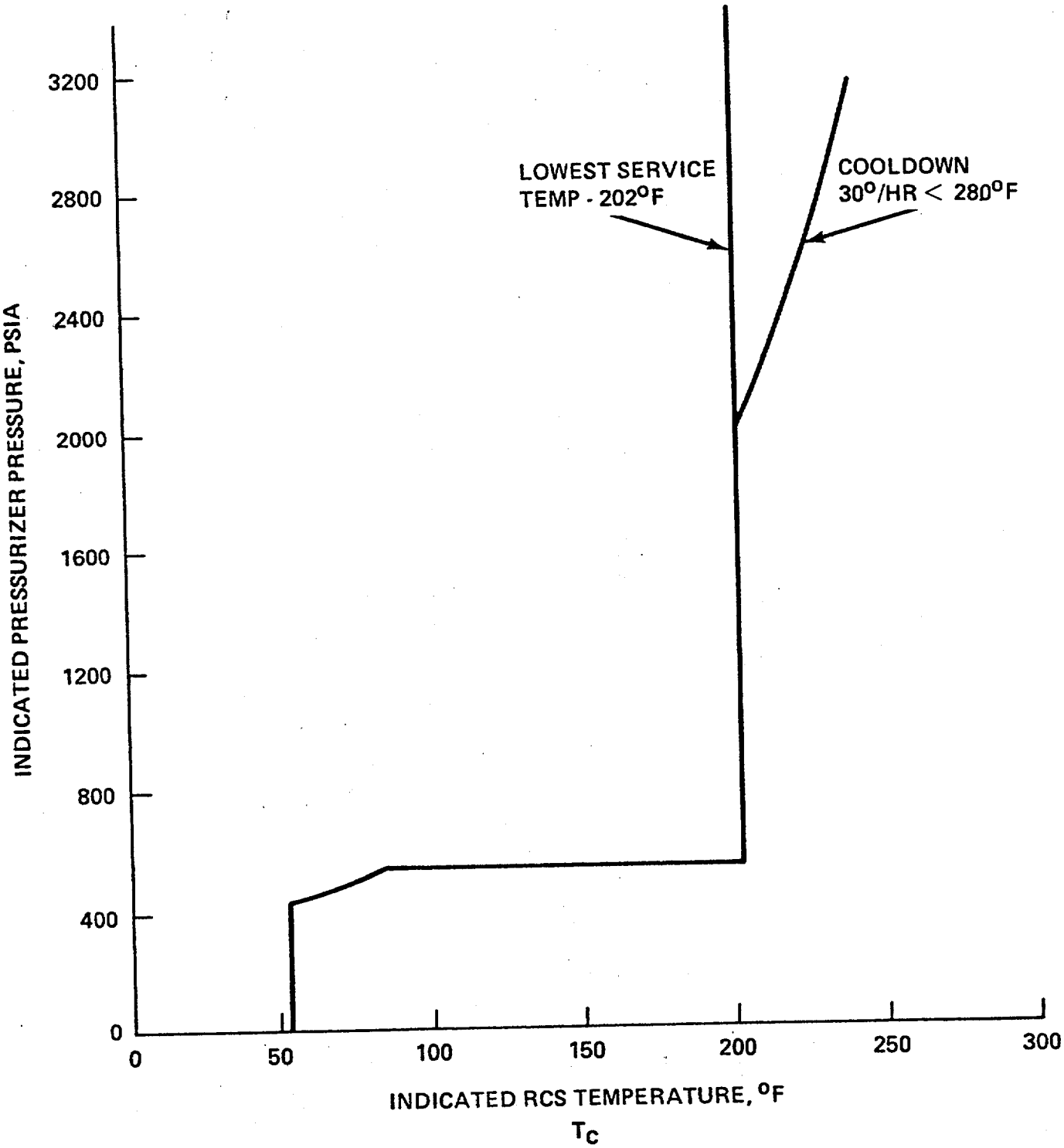
CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
1	83°	1.15	Standby
2	97°	1.15	3.2 EFPY
3	104°	1.15	13.6 EFPY
4	284°	1.15	24 EFPY
5	263°	1.15	Standby
6	277°	1.15	Standby

Figure 3.4-2



HEATUP RCS PRESSURE/TEMPERATURE LIMITATIONS FOR 0-5 YEARS

Figure 3.4-3



COOLDOWN RCS PRESSURE/TEMPERATURE LIMITATIONS FOR 0-5 YEARS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. One OPERABLE charging pump capable of taking suction from either the boric acid makeup tank or the refueling water storage tank.

d. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 400 psia.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	FV0306	SDC Bypass Flow Control	LOCKED OPEN (THROTTLED)(MANUAL)
h.	14-153		LOCKED CLOSED (MANUAL)
i.	14-081		LOCKED OPEN (MANUAL)
j.	14-082		LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV9316	SDC(HX) Flow Control	OPEN (THROTTLED)(AIR REMOVED)
n.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
o.	14-78	HV9316 Isolation	LOCKED OPEN (MANUAL)
p.	14-80	HV9316 Isolation	LOCKED OPEN (MANUAL)

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

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 the correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

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 seal leakage is less than or equal to .01 L_a when determined by flow measurement, with the volume between the door seals pressurized 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.

LIMITING CONDITION FOR OPERATION

3.6.1.7 Containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 42-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. Each 8-inch containment purge supply and exhaust isolation valve may be open for less than or equal to 1000 hours per 365 days.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the 42-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, or with the 8-inch purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close and/or seal closed the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 42-inch or 8-inch containment purge supply and/or exhaust isolation valve having a measured leakage rate exceeding the limits of Surveillance Requirement 4.6.1.7.3, restore the inoperable valve(s) 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

4.6.1.7.1 The 42-inch containment purge supply and exhaust isolation valves shall be verified sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 8-inch purge supply and exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 3 months each 42 inch and each 8 inch purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by 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 the correct position with suction aligned to the RWST.

b. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Spray Actuation test signal.

   2. Verifying that upon a Recirculation Actuation Test Signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a Safety Injection Actuation test signal.

4. Verifying that each containment spray header riser is filled with water to within 10 feet of the lowest spray ring.

c. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray additive tank containing a minimum solution volume of 1456 gallons of between 40 and 44% by weight NaOH solution with a minimum solution temperature between 82°F and 88°F and

b. Two spray chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b. At least once per 31 days by 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.

c. At least once per 6 months by:

   1. Verifying the contained solution volume in the tank, and

   2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by verifying that (1) each automatic valve in the flow path actuates to its correct position and (2) that each spray chemical addition pump starts automatically on a Containment Spray Actuation test signal.

e. At least once per 5 years by verifying a minimum solution flow rate of 20 gpm through all piping sections from the spray additive tank to the suction at the containment spray pumps.

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable, and both containment spray systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each fan group from the control room and verifying that each fan group operates for at least 15 minutes.

   2. Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a Containment Cooling Actuation test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a CIAS or SIAS test signal, each isolation valve actuates to its isolation position.

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
28	HV-4052#	Steam generator feedwater	10
29	HV-4048#	Steam generator feedwater	10
30A	HV-7802	Containment air radioactivity monitor inlet	1
30A	HV-7803	Containment air radioactivity monitor inlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30B	HV-7816	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
32	HV-8204#	Mainsteam isolation	5
33	HV-8205#	Mainsteam isolation	5
42	HV-6211	Component cooling water inlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water inlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40
B. CONTAINMENT PURGE (CPIS)			
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-334*	Safety injection drain to RWST	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
9	HV-9337#@	Shutdown cooling to LPSI pumps	NA
9	HV-9377#@	Shutdown cooling to LPSI pumps	NA
9	HV-9336#@	Shutdown cooling to LPSI pumps	NA
9	HV-9379#@	Shutdown cooling to LPSI pumps	NA
10A	HV-0352A#	Containment pressure detectors	NA
10C	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10C	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
16A	HV-0500*	Post LOCA hydrogen monitor	NA
16A	HV-0501*	Post LOCA hydrogen monitor	NA
16B	HV-0502*	Post LOCA hydrogen monitor	NA
16B	HV-0503*	Post LOCA hydrogen monitor	NA
20	2"-321-C-376*	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
27A	HV-0352D#	Containment pressure detectors	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA
40A	HV-0352B#	Containment pressure detectors	NA
67	HV-9434	Hot leg injection	NA
68	2"-130-C-334	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C#	Containment pressure detectors	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA

3/4.7.5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM*

LIMITING CONDITION FOR OPERATION

3.7.5 Two independent control room emergency air cleanup systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

Unit 2 or 3 in MODE 1, 2, 3 or 4:

With one control room emergency air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Units 2 and 3 in MODE 5 or 6:

a. With one control room emergency air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air cleanup system in the recirculation mode.

b. With both control room emergency air cleanup systems inoperable, or with the OPERABLE control room emergency air cleanup system required to be in the recirculation mode by ACTION (a), not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.3 are not applicable in MODE 6.

SURVEILLANCE REQUIREMENTS

4.7.5 Each control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 110°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that with the system operating at a flow rate of 35485 cfm ± 10% for the air conditioning unit, and 2050 ± 150 cfm for the ventilation unit and recirculating through the respective HEPA filters and charcoal adsorbers, leakage through the system diverting valves is less than or equal to 1% air conditioning unit and 1% ventilation unit when the system is tested by admitting cold DOP at the respective intake.

*Shared system with San Onofre - Unit 3.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2050 ± 150 cfm for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 2050 ± 150 cfm for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.0 inches Water Gauge ventilation unit and less than 7.3 inches Water Gauge air conditioning unit while operating the system at a flow rate of 2050 ± 150 cfm for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

2. Verifying that on a control room isolation test signal, the system automatically switches into the emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that on a toxic gas isolation test signal, the system automatically switches into the isolation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

4. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch W.G. relative to the outside atmosphere during system operation in the emergency mode.

5. Verifying that the heaters dissipate 4.8 kw ± 5% when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.7.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 The fire suppression water system shall be OPERABLE with:

a. Two electric motor-driven fire pumps, each with a capacity of 1500 gpm and one diesel-driven fire pump with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Two separate water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path capable of taking suction from each water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first valve upstream of the water flow alarm device on each spray and/or sprinkler or fire hose station required to be OPERABLE per Specifications 3.7.8.2 and 3.7.8.3.

APPLICABILITY: At all times.

ACTION:

a. With one required electric motor-driven/diesel-driven pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying performance of the fire pumps as follows:

a. Diesel engine drive pump develops at least 2500 gpm at a system head of 283 feet.

b. Electric motor driven pumps each develop at least 1500 gpm at a system head of 289 ft.

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 95 psig.

f. At least once per 3 years by performance of a system flush.

SURVEILLANCE REQUIREMENTS (Continued)

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.8.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The diesel fuel oil day storage tank contains at least 225 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.8.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas outside containment in which redundant systems or components could be damaged; for other areas outside containment, establish an hourly fire watch patrol.

b. With one or more of the above required spray and/or sprinkler systems inside containment inoperable, restore the system to OPERABLE status within 24 hours or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN or REFUELING by verifying that each valve (manual, power operated or automatic) inside containment in the flow path is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and wet pipe spray sprinkler headers to verify their integrity, and

3. By a visual inspection of each spray/sprinkler head to verify the spray pattern is not obstructed.

e. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	STATION NUMBER
Containment Bldg. - Unit 2	63'-6"	130
Containment Bldg. - Unit 2	63'-6"	1
Containment Bldg. - Unit 2	63'-6"	8
Containment Bldg. - Unit 2	45'-0"	2
Containment Bldg. - Unit 2	45'-0"	5
Containment Bldg. - Unit 2	45'-0"	9
Containment Bldg. - Unit 2	30'-0"	3
Containment Bldg. - Unit 2	30'-0"	6
Containment Bldg. - Unit 2	30'-0"	10
Containment Bldg. - Unit 2	17'-6"	4
Containment Bldg. - Unit 2	17'-6"	7
Containment Bldg. - Unit 2	17'-6"	11
Electrical Penetration Area - Unit 2	45'-0"	120
Electrical Penetration Area - Unit 2	45'-0"	121
Electrical Penetration Area - Unit 2	63'-6"	122
Electrical Penetration Area - Unit 2	63'-6"	123
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	9'-0"	109
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	9'-0"	114
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	108
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	113
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	48
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	60
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	30'-0"	110
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	30'-0"	115
Corridor (North)-Auxiliary Bldg. Control Area	30'-0"	49
Corridor (South)-Auxiliary Bldg. Control Area	30'-0"	61
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	50'-0"	111
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	50'-0"	116
Corridor (North)-Auxiliary Bldg. Control Area	50'-0"	50
Corridor (South)-Auxiliary Bldg. Control Area	50'-0"	62
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	56
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	57
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	70'-0"	112
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	70'-0"	117
Fuel Handling Bldg.-Unit 2	63'-6"	118
Fuel Handling Bldg.-Unit 2	63'-6"	119

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and that doors with automatic hold-open and release mechanisms are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Performing a CHANNEL FUNCTIONAL TEST at least once per 31 days of the fire door supervision system.

d. Inspecting at least once per 6 months the automatic hold-open, release and closing mechanism and latches.

e. Performing a functional test at least once per 18 months of automatic hold open, release, closing mechanisms and latches.

SURVEILLANCE REQUIREMENTS

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank,
2. Verifying the fuel level in the fuel storage tank,
3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,
4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4360 ± 436 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using the manual start signal.
5. Verifying the generator is synchronized, loaded to greater than or equal to 4700 kw in less than or equal to 77 seconds, and operates with a load greater than or equal to 4700 kw for at least an additional 60 minutes, and
6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. At least once per 92 days and from new fuel oil prior to addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @40°C of greater than or equal to 1.9 but less than or equal to 4.1 when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.
2. Verifying the generator capability to reject a load of greater than or equal to 655.7 kw while maintaining voltage at 4360 ± 436 volts and frequency at 60 ± 6.0 Hz.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the generator capability to reject a load of 4700 kW without tripping. The generator voltage shall not exceed 5450 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the permanently connected loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ± 436 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady state generator voltage and frequency shall be 4360 ± 436 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and freqency shall be maintained within these limits during this test.

6. Verifying that on a simulated loss of the diesel generator (with offsite power not available), the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an ESF test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto connected emergency (accident) loads through the load sequence and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After loading, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ± 436 volts and 60 + 1.2/-0.3 Hz during this test.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers' nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2B0106	2BLP0101	Containment Normal Cooling Fan E-397
2B0107	2BLP0102	CEDM Cooling Supply Fan E-403B
2B0109	2BLP0103	CEDM Cooling Supply Fan E-403A
2B0111	2BLP0104	Standby Containment Normal Cooling Fan E-393
2B0209	2BLP0201	Containment Normal Cooling Fan E-394
2B0406	2BLP0301	Hydrogen Recombiner E-145 Power Panel L-180
2B0409	2BLP0302	Upper Dome Air Circulator A-071
2B0410	2BLP0303	Containment Emergency Fan E-399
2B0411	2BLP0304	Containment Emergency Fan E-401
2B0419	2BLP0305	Standby Upper Dome Air Circulator A-074
2B0606	2BLP0401	Hydrogen Recombiner E-146 Power Panel L-181
2B0609	2BLP0402	Upper Dome Air Circulator A-072
2B0610	2BLP0403	Containment Emergency Fan E-400
2B0611	2BLP0404	Containment Emergency Fan E-402
2B0619	2BLP0405	Standby Upper Dome Air Circulator A-073
2B0809	2BLP0501	Containment Normal Cooling Fan E-396
2B0811	2BLP0601	Containment Normal Cooling Fan E-398
2B0903	2BLP0701	Containment Recirculation Unit E-333
2B0906	2BLP0702	Polar Crane (Containment) R001 (C)
2B0907	2BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
2B0909	2BLP0704	Standby CEDM Cooling Supply Fan E-404B
2B0911	2BLP0705	Containment Recirculating Unit Heater E-568
2BA02	2BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
2BA03	2BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
2BA04 (2BA04-A)	2BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BA04 (2BA04-B)	2BLP0802	CEDM Cooling Supply Fan E-403B (Enclosure Heater)
2BA04 (2BA04-C)	2BLP0814	Standby Containment Normal Cooling Fan E-393 (Enclosure Heater)
2BA04 (2BA04-D)	2BLP0826	Containmnt Normal Cooling Fan E-394 (Enclosure Heater)
2BA04 (2BA04-E)	2BLP0828	Containment Normal Cooling Fan E-397
2BA08	2BLP0803	Movable Incore Detector Drive Package W338A
2BA11	2BLP0905	Cont. Structure Electric Heater E-467
2BA25	2BLP0910	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FB
2BA26	2BLP0911	Cont. Cooling Unit E-394 Circ. Water Outlet HV-9940EB
2BA27	2BLP0912	Cont. Cooling Unit E-397 Circ. Water Outlet HV-9940D8
2BA31	2BLP0913	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FC
2BA32	2BLP0914	Cont. Cooling Unit E-394 Circ. Water Inlet HV-9940EC
2BA33	2BLP0915	Cont. Cooling Unit E-397 Circ. Water Inlet HV-9940DC
2BA36	2BLP0808	RCP 1A Oil Lift Pump 1A1 P-260
2BA37	2BLP0809	RCP 1B Oil Lift Pump 1B1 P-264
2BA38	2BLP0810	RCP 2B Oil Lift Pump 2B1 P-262
2BA39	2BLP0901	Reactor Coolant Drain Pump (W) P-023
2BA40	2BLP0811	RCP 2A Oil Lift Pump 2A1 P-266
2BA41	2BLP0817	RCP 1A Anti Rev. Rotation Device Lube Pump 1 P-399
2BA42	2BLP0818	RCP 2B Anti Rev. Rotation Device Lube Pump 1 P-401
2BA43	2BLP0819	RCP 1B Anti Rev. Rotation Device Lube Pump 1 P-403
2BA44	2BLP0820	RCP 2A Anti Rev. Rotation Device Lube Pump 1 P-405
2BA45	2BLP0902	Reactor Cavity Cooling Fan A-319
2BA46	2BLP0903	Standby Reactor Cavity Cooling Fan A-321

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BA47	2BLP0807	Charging Line to Reactor Cooling Loop 1A HV-9203
2BA49	2BLP0821	Reactor Cavity Cooling Unit C HV-9905C
2BA50	2BLP0822	Reactor Cavity Cooling Unit A HV-9905A
2BA51	2BLP0804	Quench Tank to Reactor Drain Tank HV-9101
2BA55	2BLP0805	RCP Bleed Off to Quench Tank HV-9216
2BA57	2BLP0916	CEDM Cooling Unit E-403 CCW Outlet HV-9907AA
2BA58	2BLP0917	CEDM Cooling Unit E-403 CCW Inlet HV-9907AC
2BA59	2BLP0806	Safety Injection Tank to Reactor Drain Tank HV-9335
2BA60	2BLP0904	Welding Receptacles Containment (50 KVA)
2BA62	2BLP0824	Recept. for Portable Cont. Sump Pump (H.P.) P-005
2BA63	2BLP0906	Containment Elevator P-003
2BA65	2BLP0815	Lower Level Air Circulator A-031
2BA66	2BLP0816	Lower Level Air Circulator A-033
2BE09	2BLP1001	Saf. Inj. Tank Drain to Refueling Wtr Tank HV-9334
2BE10	2BLP1002	Saf. Inj. Tk T-007 to Reactor Coolant Loop 1B HV-9350
2BE11	2BLP1003	Saf. Inj. Tk T-009 to Reactor Coolant Loop 2A HV-9360
2BE17	2BLP1010	Auxiliary Spray to Pressurizer HV-9201
2BE21	2BLP1012	CCW Noncritical Cont. Inlet Isolation Valve HV-6223
2BE25	2BLP1005	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9337
2BE26	2BLP1015	Reac. Coolant Drain Tk Sample Cont. Isolation HV-0516
2BE27	2BLP1016	Containment Isolation Reactor Coolant Drain to Radwaste System HV-7512
2BE30	2BLP1017	Quench Tank Vapor Sample Cont. Isol. HV-0514
2BE31	2BLP1004	Containment Sump to Radwaste Sump HV-5803
2BE33	2BLP1021	Containment Purge Inlet HV-9949
2BE35	2BLP1018	Containment Emergency Sump Oulet HV-9305

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BE46	2BLP1011	CCW Noncritical Containment Isolation Valve HV-6336
2BF08	2BLP0823	Containment Sump Pump P-008
2BF09	2BLP1220	Containment Sump Pump P-007
2BJ05	2BLP1101	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9339
2BJ06	2BLP1104	Saf. Inj. Tk T-008 to Reactor Coolant Loop 1A HV-9340
2BJ07	2BLP1105	Saf. Inj. Tk T-010 to Reactor Coolant Loop 2B HV-9370
2BJ17	2BLP1123	RCP Bleed off to Volume Control Tank HV-9217
2BJ21	2BLP1106	Cont. Isol. Safety Injection Tank Vent Header HV-7258
2BJ22	2BLP1115	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0508
2BJ23	2BLP1116	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0517
2BJ26	2BLP1117	Pressurizer Vapor Sample Containment Isol. HV-0510
2BJ27	2BLP1121	Pressur. Surge Line Liquid Smpl. Cont. Isol. HV-0512
2BJ29	2BLP1110	Containment Purge Outlet HV-9950
2BJ30	2BLP1102	Hydrogen Purge Exhaust Unit Inlet HV-9917
2BJ31	2BLP1103	Hydrogen Purge Supply Unit Discharge HV-9946
2BJ34	2BLP1118	Containment Emergency Sump Outlet HV-9304
2BJ47	2BLP1124	Containment Normal Cooling Supply Isol. Valve HV-9900
2BJ48	2BLP1125	Containment Normal Cooling Return Isol. Valve HV-9971
2BN04	2BLP1201	Movable Incore Detector Drive Pack W-338B
2BN07	2BLP1304	Containment Structure Electric Heater E-466
2BN21	2BLP1206	Charging Line to Reactor Coolant Loop 2A HV-9202
2BN24	2BLP1301	Reactor Cavity Cooling Fan A-320
2BN25	2BLP1302	Standby Reactor Cavity Cooling Fan A-322
2BN26	2BLP1226	CCW from RCP P-004 Seal Heat Exchanger TV-9164
2BN27	2BLP1227	CCW from RCP P-002 Seal Heat Exchanger TV-9174

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BN28	2BLP1207	Reactor Cavity Cooling Unit D HV-9905D
2BN29	2BLP1208	Reactor Cavity Cooling Unit B HV-9905B
2BN30	2BLP1209	RCP 1A Oil Lift Pump 1A2 P-261
2BN31	2BLP1210	RCP 1B Oil Lift Pump 1B2 P-265
2BN32	2BLP1211	RCP 2B Oil Lift Pump 2B2-263
2BN33	2BLP1212	RCP 2A Oil Lift Pump 2A2-267
2BN34	2BLP1303	Reactor Coolant Drain Tank Pump (E) P-022
2BN37	2BLP1213	RCP 1A Anti Rev. Rotation Device Lube Pump 2 P-400
2BN38	2BLP1214	RCP 2B Anti Rev. Rotation Device Lube Pump 2 P-402
2BN39	2BLP1215	RCP 1B Anti Rev. Rotation Device Lube Pump 2 P-404
2BN40	2BLP1216	RCP 2A Anti Rev. Rotation Device Lube Pump 2 P-406
2BN42	2BLP1305	Welding Recpt. Cont. (50KVA) 2R005A, 2R005b, 2R005C
2BN43	2BLP1217	CEA Change Mechanism Transfer Machine Control Console (8 KVA) L-023
2BN44	2BLP1306	Welding Recpt. Cont. (50 KVA) 2R007A, 2R007B, 2R007C
2BN45	2BLP1218	Refueling Pool End Junction Box (8KVA) L-371
2BN46	2BLP1308	Welding Recpt. Cont. (50KVA) 2R013A, 2R013B, 2R013C
2BN47	2BLP1219	Receptable for Portable Cont. Sump Pump (1hp) P-005
2BN49	2BLP1319	Equipment Hatch 200R, Electrical Hoist Z-028, Z-029
2BN52	2BLP1221	Lower Level Air Circulator A-032
2BN53	2BLP1222	Lower Level Air Circulator A-034
2BN56	2BLP1310	Cont. Cooling Unit E-396 Circ. Water Outlet HV-9940BB
2BN57	2BLP1311	Cont. Cooling Unit E-396 Circ. Water Inlet HV-9940BC
2BN58	2BLP1312	Cont. Cooling Unit E-398 Circ. Water Outlet HV-9940CB
2BN59	2BLP1313	Cont. Cooling Unit E-398 Circ. Water Inlet HV-9940CC
2BN60	2BLP1314	CEDM Cooling Unit E-404 CCW Outlet HV-9907BA

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BN61	2BLP1315	CEDM Cooling Unit E-404 CCW Inlet HV-9907BC
2BN62 (2BN62-A)	2BLP1223	Containment Recirculation Unit A-353 (Motor Enclosure Heater)
2BN62 (2BN62-B)	2BLP1224	CEDM Cooling Supply Fan E-404A (Motor Enclosure Heater)
2BN62 (2BN62-C)	2BLP1225	CEDM Cooling Supply Fan E-404B (Motor Enclosure Heater)
2BN62 (2BN62-H)	2BLP1202	Containment Normal Cooling Fan A-398 (Motor Enclosure Heater)
2BN62 (2BN62-G)	2BLP1228	Containment Normal Cooling Fan E-396 (Motor Enclosure Heater)
L0108	L0101	Panel 2LP4 Emergency Lighting
L0118	L0101	Panel 2LP11 Emergency Lighting
L0120	L0101	Panel 2LP16 Emergency Lighting
2BHP0201	2B0205	Backup Pressurizer Heater E-607
2BHP0202	2B0205	Backup Pressurizer Heater E-608
2BHP0203	2B0205	Backup Pressurizer Heater E-609
2BHP0204	2B0205	Backup Pressurizer Heater E-610
2BHP0301	2B0206	Backup Pressurizer Heater E-611
2BHP0302	2B0206	Backup Pressurizer Heater E-612
2BHP0303	2B0206	Backup Pressurizer Heater E-613
2BHP0304	2B0206	Backup Pressurizer Heater E-614
2BHP0101	2B0210	Proportional Pressurizer Heater E-601
2BHP0102	2B0210	Proportional Pressurizer Heater E-602
2BHP0103	2B0210	Proportional Pressurizer Heater E-603
2BHP0401	2B0402	Backup Pressurizer Heater E-615

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BHP0402	2B0402	Backup Pressurizer Heater E-616
2BHP0403	2B0402	Backup Pressurizer Heater E-617
2BHP0404	2B0402	Backup Pressurizer Heater E-618
2BHP0601	2B0805	Backup Pressurizer Heater E-619
2BHP0602	2B0805	Backup Pressurizer Heater E-620
2BHP0603	2B0805	Backup Pressurizer Heater E-621
2BHP0604	2B0805	Backup Pressurizer Heater E-622
2BHP0701	2B0806	Backup Pressurizer Heater E-623
2BHP0702	2B0806	Backup Pressurizer Heater E-624
2BHP0703	2B0806	Backup Pressurizer Heater E-625
2BHP0704	2B0806	Backup Pressurizer Heater E-626
2BHP0501	2B0810	Proportional Pressurizer Heater E-604
2BHP0502	2B0810	Proportional Pressurizer Heater E-605
2BHP0503	2B0810	Proportional Pressurizer Heater E-606
2BHP0801	2B0602	Backup Pressurizer Heater E-627
2BHP0802	2B0602	Backup Pressurizer Heater E-628
2BHP0803	2B0602	Backup Pressurizer Heater E-639
2BHP0804	2B0602	Backup Pressurizer Heater E-630
2BY40	2BLP1013	Cont. Bldg. Emer. A/C Unit E-399 (Motor Enclos. Htr.)
2BY40	2BLP1014	Cont. Bldg. Emer. A/C Unit E-401 (Motor Enclos. Htr.)
2BZ32	2BLP1111	Reactor Coolant Regen. Heat Exch. Isol. Valve TV-9267
2BZ38	2BLP1112	Containment Bldg. Emergency A/C Unit E-400
2BZ38	2BLP1126	Containment Bldg. Emergency A/C Unit E-403
2Q01704	2Q017 (Main Breaker)	Containment Reactor Cavity Cooling Fan A-319 (Motor Enclosure Heater)
2Q01706	2Q017 (Main Breaker)	Containment Reactor Cavity Cooling Fan A-321 (Motor Enclosure Heater)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2Q01724	2Q017 (Main Breaker)	Containment Sump Inlet Flow 2FT5799A/B, 2FT5802A/B
2Q02801	2Q028 (Main Breaker)	RCP P-001 (Motor Enclosure Heater)
2Q02802	2Q028 (Main Breaker)	RCP P-004 (Motor Enclosure Heater)
2Q02803	2Q028 (Main Breaker)	RCP P-002 (Motor Enclosure Heater)
2Q02804	2Q028 (Main Breaker)	Containment Reactor Cavity Cooling Fan A-320 (Motor Enclosure Heater)
2Q02805	2Q028 (Main Breaker)	RCP P-003 (Motor Enclosure Heater)
2Q02808	2Q028 (Main Breaker)	Containment Reactor Cavity Cooling Fan (Motor Enclosure Heater)
2Q03904	2Q039 (Main Breaker)	Dome Circulating Fan A-071 (Motor Enclosure Heater)
2Q03906	2Q039 (Main Breaker)	Dome Circulating Fan A-074 (Motor Enclosure Heater)
2Q04104	2Q041 (Main Breaker)	Standby Dome Circulating Fan A-072 (Motor Enclosure Heater)
2Q04106	2Q041 (Main Breaker)	Standby Dome Circulating Fan A-073
2D5P108	2D503	Panel 2LP4 Emergency Lighting
2D5P109	2D503	Panel 2LP11 Emergency Lighting
2D5P118	2D503	Panel 2LP16 Emergency Lighting
2A0101	2A0102	Reactor Coolant Pump P-001
	2A0104	Reactor Coolant Pump P-001
	2A0105	Reactor Coolant Pump P-001

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2A0103	2A0102	Reactor Coolant Pump P-004
	2A0104	Reactor Coolant Pump P-004
	2A0105	Reactor Coolant Pump P-004
2A0201	2A0202	Reactor Coolant Pump P-002
	2A0204	Reactor Coolant Pump P-002
	2A0205	Reactor Coolant Pump P-002
2A0203	2A0202	Reactor Coolant Pump P-003
	2A0204	Reactor Coolant Pump P-003
	2A0205	Reactor Coolant Pump P-003
CEA04	CB3001	CEA4
CEA05	CB3001	CEA5
CEA06	CB3001	CEA6
CEA07	CB3001	CEA7
CEA08	CB3002	CEA8
CEA09	CB3002	CEA9
CEA10	CB3002	CEA10
CEA11	CB3002	CEA11
CEA12	CB3003	CEA12
CEA14	CB3003	CEA14
CEA16	CB3003	CEA16
CEA18	CB3003	CEA16
CEA13	CB3004	CEA13
CEA15	CB3004	CEA15
CEA17	CB3004	CEA17
CEA19	CB3004	CEA19

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

*Operation in MODE 3 shall be limited to 6 consecutive hours.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

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

b. Be in HOT STANDBY wthin 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 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 Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in unrestricted areas due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines, tritium and for all radioactive materials in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioiodines, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	$1x10^{-4}$
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emmitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
2. Plant Vent Stack	W[b,e]	W[b]		
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	$1x10^{-11}$
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$

TABLE 4.11-2 (Continued)

TABLE NOTATION

The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a particular measurement system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance.

In calculating the LLD for a radionuclide determine by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of the measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, HASL-300 (revised annually).

(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).

(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within one hour and less than or equal to 2% by volume within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 134,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

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*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable, CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.20 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F/hr or cooldown rate of up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper and phosphorous content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
215-01	C-6403-1	A533GRBCL-1	Upper Shell Plate	40	15	35	130
215-01	C-6403-2	"	"	0	20	25	133
215-01	C-6403-3	"	"	-10	20	45	131
215-03	C-6404-1	"	Intermediate Shell Plate	-30	10	50	145
215-03	C-6404-2	"	"	-20	20	50	155
215-03	C-6404-3	"	"	-20	10	50	131
215-02	C-6404-4	"	Lower Shell Plate	-10	-5	25	124
215-02	C-6404-5	"	"	-20	10	25	134
215-02	C-6404-6	"	"	-10	-20	0	151
238-02	C-6401	A508C1-2	Vessel Flange Forging	-10	-70	-35	148
209-02	C-6402	"	Closure Head Flange Forging	-10	-90	-40	142
205-02	C-6410-1	"	Inlet Nozzle Forging	20	-40	-35	130
205-02	C-6410-2	"	"	0	-20	-5	135
205-02	C-6410-3	"	"	0	-15	-15	140
205-02	C-6410-4	"	"	0	-65	-50	140
205-06	C-6411-1	"	Outlet Nozzle Forging	-100	-30	-10	140
205-06	C6411-2	"	"	0	-35	-10	140
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122
232-02	C-6425	"	Bottom Head Dome	-50	-30	-20	136
205-03	C-6428-1	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-2	"	"	-30	-70	-50	174
205-03	C-6428-3	"	"	-30	-70	-50	174
205-03	C-6428-4	"	"	-30	-70	-50	174
205-07	C-6429-1	"	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	"	"	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	"	"	-20	10	50	100
231-02	C-6432-1	"	"	-10	-15	45	115
231-02	C-6432	"	Closure Head Dome	-10	-15	45	115

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertant safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification). The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the mimumum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 5.0 psig, 2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA or steam line break conditions, and 3) the assumptions used for the initial conditions of the LOCA safety analysis remain valid.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is 55.7 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 57.2 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 55.7 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, the chemical and visual examination of the sheathing filler grease, and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979; and Proposed Regulatory Guide 1.35.1, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1210 psig) of its design pressure of 1100 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 15,473,628 lbs/hr which is 102.3 percent of the total secondary steam flow of 15,130,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 1 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop, four pump operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 111.3$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER.

V = maximum number of inoperable safety valves per steam line.

111.3 = Power Level-High Trip Setpoint for two-loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour (15,473,628 lbs/hr at 1190 psia).

Y = Maximum relieving capacity of any one safety valve in lbs/hour (859,646 lbs/hr at 1190 psia).

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

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

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1170 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1170 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tank T-121 with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 2 hours followed by cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of offsite power and most limiting single failure. The OPERABILITY of condensate storage tank T-120 in conjunction with tank T-121 ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 24 hours including cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of offsite power and most limiting single failure. The contained water volume limits are specified relative to the highest auxiliary feedwater pump suction inlet in the tank for T-121, and to the T-121 cross connect siphon inlet for T-120. (Water volume below these datum levels is not considered recoverable for purposes of this specification.) Vortexing, internal structure, and instrument error are considered in determining the tank levels corresponding to the specified water volume limits.

Prior to achieving 100% RATED THERMAL POWER, Figure 3.7-1 is used to determine the minimum required water volume for T-121 for the maximum power level (hence maximum decay heat) achieved.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 1720 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

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

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

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

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL-REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

The limitations on the fuel handling building post-accident cleanup filter system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

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 CEA worth measurement tests are performed. 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.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

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.

BASES

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.

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.

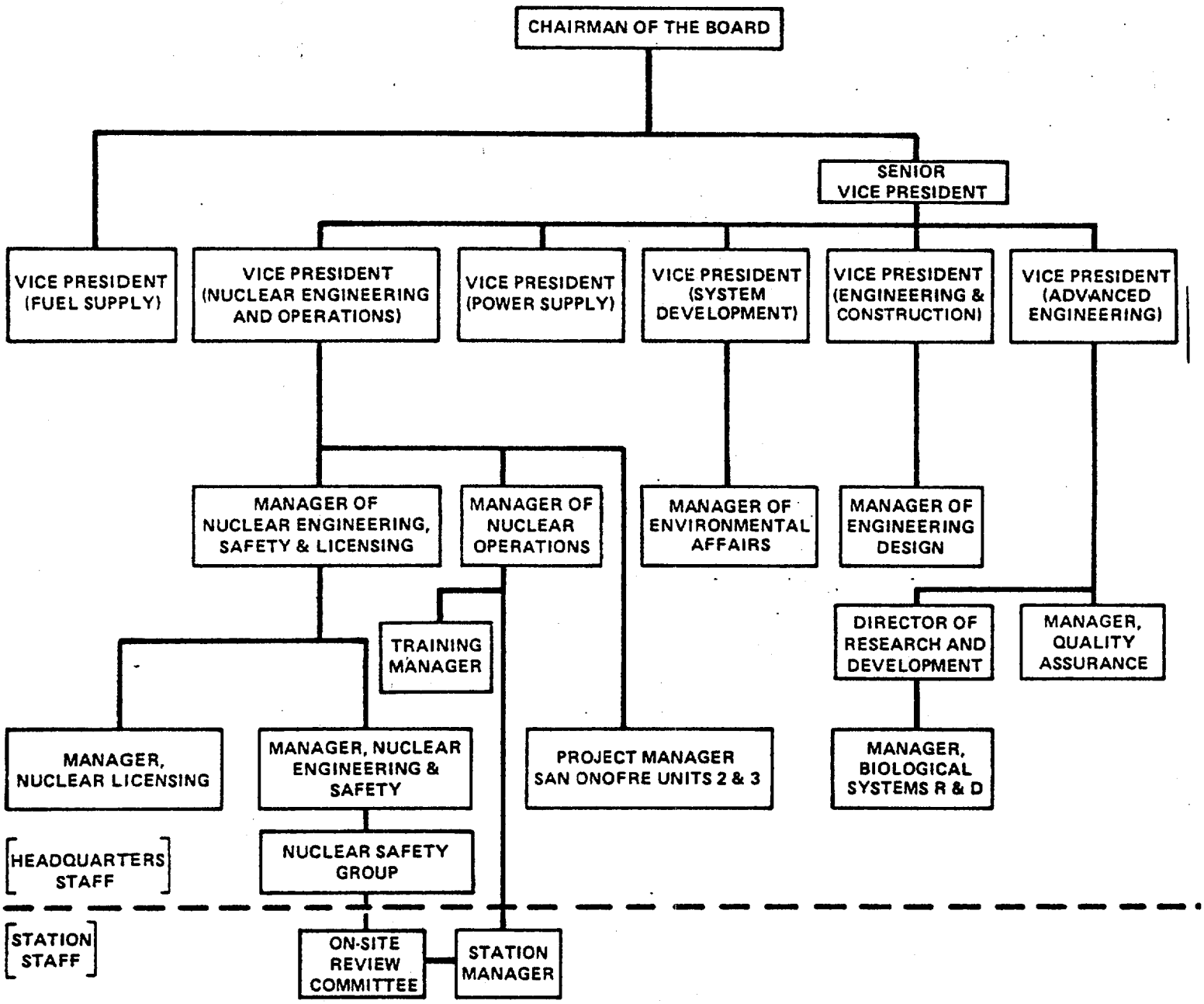
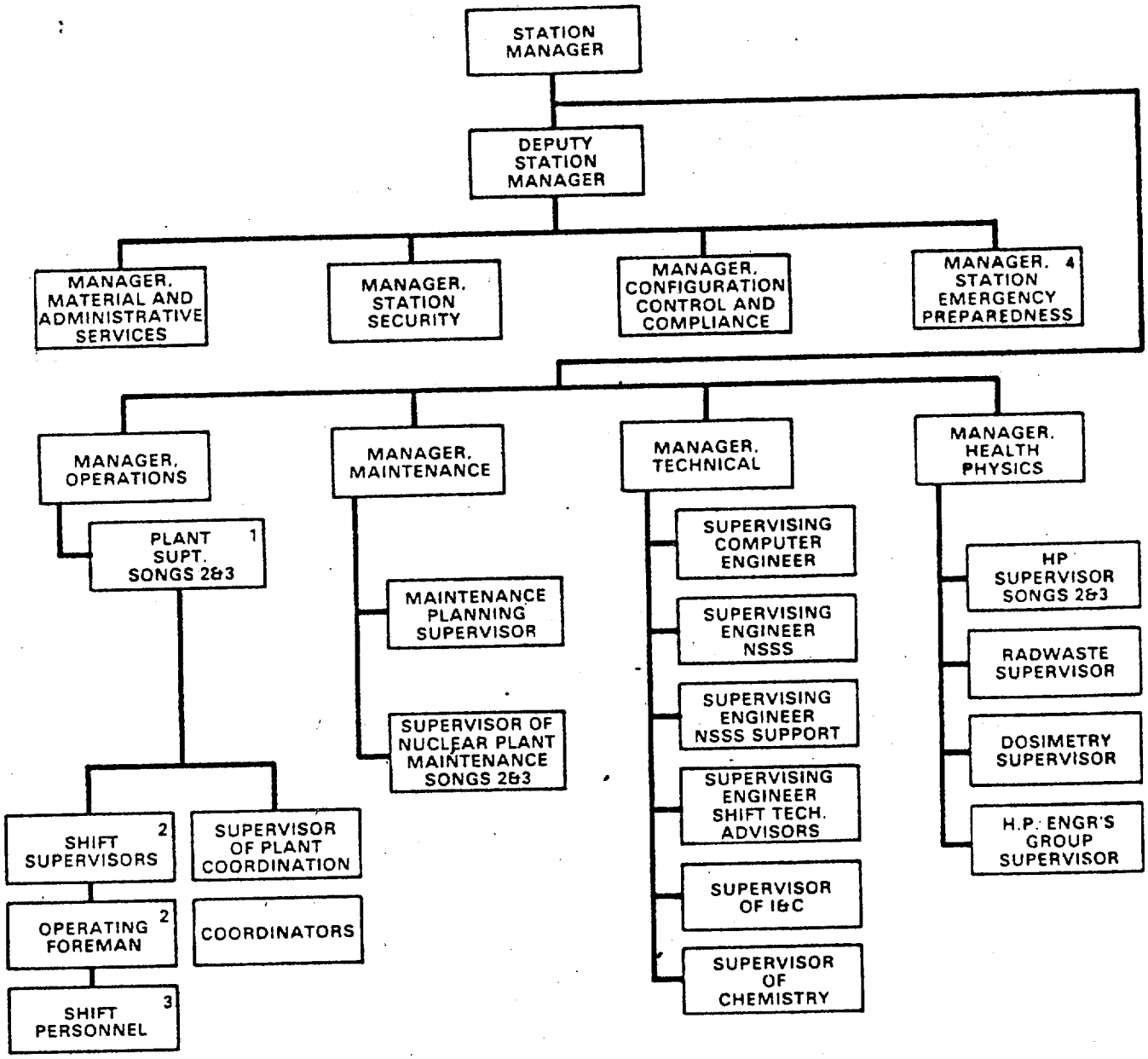


Figure 6.2-1
OFFSITE ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION – UNIT 2



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

WITH UNIT 3 IN MODE 5 OR 6 OR DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA	1	None

WITH UNIT 3 IN MODE 1, 2, 3 or 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1[a]	None
RO	2[b]	1
AO	2[b]	1
STA	1[a]	None

[a]Individual may fill the same position on Unit 3
[b]One of the two required individuals may fill the same position on Unit 3

SS - Shift Supervisor with a Senior Reactor Operators License on Units 2 and 3
SRO - Individual with a Senior Reactor Operators License on Units 2 and 3
RO - Individual with a Reactor Operators License on Units 2 and 3
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.

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 or equivalent 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.

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the NSG Supervisor.

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.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Onsite Review Committee.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the Nuclear Safety Group or Manager of Nuclear Operations.

f. The Fire Protection Program and implementing procedures at least once per 24 months.

g. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

h. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Manager, Nuclear Engineering and Safety on those areas of responsibility specified in Sections 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be distributed monthly to the Station Manager and to the management positions responsible for the areas audited.

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.

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

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the Onsite Review Committee.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the NRC Regional Administrator, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specifications 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

l. Failure of one or more pressurizer safety valves.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the OSRC.

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

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the OSRC. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the OSRC.

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

SAFETY EVALUATION

AMENDMENT NO. 16 TO NPF-10

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

DOCKET NO. 50-361

Introduction

By letters dated September 3, October 21, December 1, 1982, and February 4, 1983, the Southern California Edison Company (SCE or the licensee) requested an amendment to change the San Onofre Nuclear Generating Station, Unit 2, Technical Specifications to be consistent with those of San Onofre Unit 3.

Evaluation

The technical specification changes that were requested are identical to those that have been previously reviewed and approved by the staff with the issuance of the operating license for San Onofre Unit 3.

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 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 amendment.

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.

Dated: APR 25 1983

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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. 16 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 becomes effective on May 16, 1983.

Amendment No. 16 modifies the San Onofre Unit 2 Technical Specifications to eliminate non-plant specific differences between the Unit 2 and Unit 3 Technical Specifications.

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 pursuant 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.

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For further details with respect to this action, see (1) Southern California Edison Company's letters dated September 3, October 21, and December 1, 1982 and February 4, 1983, (2) Amendment No. 16 to Facility Operating License No. NPF-10, and (3) the Commission's related Safety Evaluation.

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 92672. 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 25th day of April, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

4/22/83
Whatever Hand Notice

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