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1.0 DEFINITIONS

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

ACTION

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

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

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CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 Not Applicable.

CORE ALTERATION

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

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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 FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE 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.

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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.

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PURGE - PURGING

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

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE OCCURRENCE

1.26 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.12 and 6.9.1.13.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming (1) all full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn, and (2) no change in part length control element assembly position.

SOFTWARE

1.28 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

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

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STAGGERED TEST BASIS

1.31 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.34 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

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

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TABLE 1.1

OPERATIONAL MODES

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

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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TABLE 1.2

FREQUENCY NOTATION

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

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.20.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.20, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.20 (5)	$\geq$ 1.20 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.3%/sec (6)(8)	$\leq$ 0.315%/sec (6)(8)
b) Floor	$\geq$ 60% (6)(8)	$\geq$ 55% (6)(8)
c) Step	$\leq$ 10% (6)(8)	$\leq$ 13% (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

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TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) 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.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; 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 pressure is greater than or equal to 500 psia.

(3) 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.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. 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. The approved DNBR limit is 1.20. A DNBR trip setpoint of 1.19 is allowed provided that the difference is compensated by an increase in the addressable constant BERR1. The minimum allowable value of BERR1 is 1.15 before DNBR compensation. The BERR1 adjustment shall be

$$\mathrm{BERR1_{NEW}} = \mathrm{BERR1_{OLD}}\left[1+\Delta\mathrm{DNBR}(\%)\mathrm{x}0.01\mathrm{x}\ \frac{d(\%\mathrm{POL})}{d(\%\mathrm{DNBR})}\right]$$

where ΔDNBR(%) is the percent increase in DNBR trip setpoint requirement and d(%POL)/d(%DNBR) is the absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

(6) DN RATE, % of reference value, is the maximum decrease rate of the trip setpoint.

FLOOR, % of reference value, is the minimum value of the trip setpoint.

STEP, % of reference value, is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor. The reference value is that of the input signal at operating flow and coolant temperature.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

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TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	$\geq$1.02
64	TPC	Thermal power calibration constant	$\geq$0.90
65	KCAL	Neutron flux power calibration constant	$\geq$0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

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TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERRO	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

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TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
98	CORR1	Temperature shadowing correction factor multiplier
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient

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NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

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2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.20 for the CE-1 correlation and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kw/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

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2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.20 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

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Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA, or certain intermediate steam line breaks. This trip initiates a reactor trip at a linear power level of less than or equal to 111.3% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.96% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2389 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1763 psia. This trip's setpoint may be manually decreased, to a minimum value of 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

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Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

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Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

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DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ 495°F
b.	RCS Cold Leg Temperature-High	≤ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	≥ -0.5
e.	Pressurizer Pressure-Low	≥ 1825 psia
f.	Pressurizer Pressure-High	≤ 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ 1.28
h.	Integrated Radial Peaking Factor-High	≤ 4.28
i.	Quality Margin-Low	≥ 0

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator goes below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

Seismic - High

The Seismic - High trip is provided to trip the reactor in the event of an earthquake which exceeds 60% of the Safe Shutdown Earthquake level. This trip's setpoint does not correspond to a safety limit and no credit was taken in the accident analyses for operation of this trip.

Loss of Load

The Loss of Load trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

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2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

This specification is not applicable in MODE 5 or 6.

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

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SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

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SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, 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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

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SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 2.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% delta k/k, 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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 2.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration,and
   6. Samarium concentration.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.13×10^{-4} delta k/k/°F, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

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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.
*See Special Test Exception 3.10.6.

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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.

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Flow paths from one or both boric acid makeup tanks via

   1. The associated gravity feed connection(s) and/or

   2. The associated boric acid makeup pump(s)

   via charging pump(s) to the RCS

and/or

b. The flow path from the refueling water storage tank via charging pump(s) to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two 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 from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

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 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 40 gpm to the Reactor Coolant System.

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

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CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORIC ACID MAKEUP PUMP - SHUTDOWN

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LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1.a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1.a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

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BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

b. The refueling water storage tanks with:

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

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and ~~120°F.~~ 100°F.

X

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

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

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

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

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FIGURE 3.1-1

Use same figure

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

b. The refueling water storage tank with:

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

2. Between 1720 and 2300 ppm of boron, and

3. A solution temperature between 40°F and ~~120°F.~~ 100°F.

X

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

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

3. Verifying the boric acid makeup tank solution temperature.

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

X

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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ACTION: (Continued)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, but within its above specified alignment requirements and either withdrawn to greater than or equal to 145 inches or within the Long Term Steady State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part length CEAs in its group.

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SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length and part length CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted and each part length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

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POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other at least once per 12 hours.

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POSITION INDICATOR CHANNEL - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*With the reactor trip breakers in the closed position.

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CEA DROP TIME

LIMITING CONDITION FOR OPERATION

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

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

~~a.~~ With the drop time of any full length CEA determined to exceed the above limit, ~~restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.~~ be in at least HOT STANDBY within six hours.

X

X

SURVEILLANCE REQUIREMENTS

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

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

X

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

c. At least once per 18 months.

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SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.2.

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

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REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

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

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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

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ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

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Figure 3.1-2

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Use same figure

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PART LENGTH CEA INSERTION LIMITS

change 3.1.3.7 to

LIMITING CONDITION FOR OPERATION

3.1.3.7 ~~The position of the part length CEA group shall be restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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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.

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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 $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ 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.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

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LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

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

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within two hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS), when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4, is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Linear Power Level - High trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

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SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

c. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

d. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

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3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

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

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR 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 DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 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 DNBR.

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SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)*
0-2.4	0.0
2.4-5	3.0
5-10	7.1
10-15	10.3
15-20	12.9
20-25	15.3
25-30	17.4
30-35	19.4
35-40	21.2

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches. An alternate method is to determine the penalty for each individual assembly in the core based on that assembly's burnup and apply that penalty to theat assembly's radial power peak.

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Figure 3.2-1

Use same figure

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Figure 3.2-2

Use same figure

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3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 148×10^6 lbm/hr, and less than or equal to 177.6×10^6 lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be outside the above limits, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

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REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (Tc) shall be maintained between 544°F and 558°F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit at least once per 12 hours.

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AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.50$

a. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.50$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

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PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the average pressurizer pressure exceeding its limit, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective 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 4.3-1.

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier and each optical isolator for CEA Calculator to Core Protection Calculator data transfer shall be verified at least once per 18 months during the shutdown per the following tests:

a. For the CEA position isolation amplifiers:

   1. With 120 volts AC (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not exceed 0.015 volts DC.

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SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 8 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.5 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

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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#

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TABLE 3.3-1 (Continued)

TABLE NOTATION

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

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

#The provisions of Specification 3.0.4 are not applicable.

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

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

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ 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 ~~6.5.1.6k~~. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.
6.5.1.6e.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

Process Measurement Circuit	Functional Unit Bypassed
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

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

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

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2..	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

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

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and

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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~~ 1.50 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

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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.

##Response time shall be measured from the output of the sensor. RTD response time shall be measured at least once per 18 months by means of the Loop Current Step Response (LCSR) method. The measured R_T of the slowest RTD shall be less than or equal to 6.0 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time constant shall be less than or equal to 0.7 seconds where the pressure transmitter response time is equivalent to the time interval required for the transmitter to achieve 63.2% of its total change when subjected to a step change in pressure transmitter pressure.

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

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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(10)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)(10)	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(4,5,10)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5,10)	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*

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TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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TABLE NOTATION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.*

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. ~~The provisions of specification 4.0.4 are not applicable for entry into mode 3 for items 8(2) and 9(2) of Table 3.3-5.~~

*See Special Test Exception 3.10.5

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

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

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

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

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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*

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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
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*#

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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*#

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TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to ~~500~~ 400 psia.

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

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

The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

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

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

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TABLE 3.3-3 (Continued)

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TABLE NOTATION

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

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

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

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

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

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

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

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Table 3.3-3 (Continued)

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TABLE NOTATION

ACTION 13 -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.
ACTION 14 -	With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.
ACTION 15 -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.
ACTION 16 -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.
ACTION 17 -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9. (Mode 6 only)
ACTION 17a -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)
ACTION 17b -	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.9. (At all times)

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

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

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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

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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

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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	≤ per ODCM	≤ per ODCM
ii. Particulate	≤ per ODCM	≤ per ODCM
iii. Iodine	≤ per ODCM	≤ per ODCM
c. Containment Area Radiation (Gamma)	≤ 2.4 mR/hr	≤ 2.5 mR/hr
d. Automatic Actuation Logic	Not Applicable	Not Applicable

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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 ~~500~~ 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.

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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

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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 ~~4)~~ 4a)	~~11.2~~ 21.2
(c) CCW Valves (Note 4b)	23.2*
(d) ~~(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 4a)	21.0 ~~11.0 (NOTE 4)~~
(c) CCW Valves (NOTE 4b)	23.0*
(d) ~~(c)~~ Emergency Cooling Fans	21.0*
b. CIAS	
~~(1)~~ Containment Isolation	10.9* (NOTE 2)
4. Containment Pressure - High-High	
~~a.~~ CSAS	
~~(1)~~ Containment Spray	21.0*

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Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
~~a.~~ MSIS	
(1) Main Steam Isolation (MSIV)	20.9
(2) Main Feedwater Isolation	10.9
6. Refueling Water Storage Tank - Low	
~~a.~~ RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Valves Shut	40.7*
7. 4.16 kv Emergency Bus Undervoltage	
~~a.~~ LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
~~a.~~ 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)	
~~a.~~ 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	
~~a.~~ CRIS	
~~(1)~~ Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
~~a.~~ TGIS	
~~(1)~~ Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
~~a.~~ TGIS	
~~(1)~~ Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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Table 3.3-5 (Continued)

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INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
~~a.~~ TGIS	
~~(1)~~ Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
~~a.~~ TGIS	
~~(1)~~ Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
~~a.~~ FHIS	
~~(1)~~ Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
~~a.~~ CPIS	
~~(1)~~ Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
~~a.~~ CPIS	
~~(1)~~ Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.

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

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.

4.a CCW non-critical loop isolation valves ~~2~~3HV-6212, ~~2~~3HV-6213, ~~2~~3HV-6218 and ~~2~~3HV-6219* close.

4b. Containment Emergency Cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372, and 3HV-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) is included.

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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, 4
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, 4
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
b. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
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

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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

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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.

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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 3.3-6

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
a. Containment - High Range	2	1, 2, 3	2 R/hr	$1\text{-}10^{8}$ R/hr	18
		4	2 R/hr		19
b. Containment - Purge Isolation	1	1, 2, 3, 4	$\leq$ 325 mR/hr	$10^{-1}\text{-}10^{5}$ mR/hr	19
		6	#		(a)
c. Main Steam Line	1/line	1, 2, 3	1 mR/hr (low); 1 R/hr (high)	$10^{-1}\text{-}10^{4}$ mR/hr;	18
		4	1 mR/hr (low); 1 R/hr (high)		19
2. Process Monitors					
a. Fuel Storage Pool Airborne					
i. Gaseous	1	*	#	10^{1} -10^{7} cpm	(d)
ii. Particulate/Iodine	1	*	#	10^{1} -10^{7} cpm	(d)
b. Containment Airborne					
i. Gaseous	1	All	Per ODCM	10^{1} -10^{7} cpm	(a)(b)(c)
ii. Particulate	1	All	Per ODCM	10^{1} -10^{7} cpm	(a)(b)(c)
iii. Iodine	1	All	Per ODCM	10^{1} -10^{7} cpm	(a)(c)
c. Control Room Airborne					
i. Particulate/Iodine	1	All	#	10^{1} -10^{7} cpm	(e)
ii. Gaseous	1	All	#	10^{1} -10^{7} cpm	(e)

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TABLE 3.3-6 (Continued)

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
3. Noble Gas Monitors					
a. Plant Vent Stack	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)
b. Condenser Evacuation System	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)

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ACTION STATEMENTS

ACTION 18 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.6.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

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

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

#In accordance with Engineered Safety Feature trip value specified by Table 3.3-4.

*With irradiated fuel in the storage pool.

(a) In accordance with Table 3.3-3 - ACTION 17.

(b) In accordance with Table 3.3-3 - ACTION 17a.

(c) In accordance with Table 3.3-3 - ACTION 17b.

(d) In accordance with Table 3.3-3 - ACTION 16.

(e) In accordance with Table 3.3-3 - ACTION 13.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
a. Containment - High Range	S	R	M	1, 2, 3, 4
b. Containment - Purge Isolation	S ##	R ##	M ##	1, 2, 3, 4 6
c. Main Steam Line	S	R	M	1, 2, 3, 4
2. Process Monitors				
a. Fuel Storage Pool Airborne				
i. Gaseous	#	#	#	*
ii. Particulate/Iodine	#	#	#	*
b. Containment Airborne				
i. Gaseous	@	@	@	All
ii. Particulate	@	@	@	All
iii. Iodine	@	@	@	All
c. Control Room Airborne				
i. Particulate	#	#	#	All
ii. Gaseous	#	#	#	All

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
3. Noble Gas Monitors				
a. Plant Vent Stack	@	@	@	All
b. Condenser Evacuation System	@	@	@	All

NOTES:

#In accordance with Table 4.3-2 surveillance requirements for these instrument channels.

*With irradiated fuel in the storage pool.

@In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

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USE SAME FIGURE

Figure 3.3-1
DEGRADED BUS VOLTAGE TRIP SETTING

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LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

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SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments 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 instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments which is accessible during power operation and which is actuated during a seismic event (one or more valid basemat accelerations of 0.05g or greater) shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from the accessible actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety. Each of the above seismic monitoring instruments which is actuated during a seismic event (one or more valid basemat accelerations of 0.05g or greater) but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time the plant enters MODE 3 or below. A supplemental report shall then be prepared and submitted to the Commission within 10 days pursuant to Specification 6.9.2 describing the additional data from these instruments.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

Instruments & Sensor Locations	Measurement Range	Minimum Instrument Operable
1. Triaxial Time-History Strong Motion Accelerometers		
a. Steam Generator Base Support	-2 to +2g	1
b. Pressurizer Base Support	-2 to +2g	1
c. Reactor Coolant Pump	-2 to +2g	1
d. Containment Base in Tendon Gallery	-2 to +2g	1
e. Containment Operating Level	-2 to +2g	1
f. Unit #1 Free Field	-1 to +1g	1
g. Control Building Basement	-2 to +2g	1
h. Control Building Roof	-2 to +2g	1
i. Safety Equipment Building Base Slab	-2 to +2g	1
j. Safety Equipment Building Piping Support	-2 to +2g	1
k. Radwaste Building Equipment Support	-2 to +2g	1
2. Triaxial Peak Reading Accelerographs		
a. Control Building-Control Room	-2 to +2g	1
b. Control Building Base	-2 to +2g	1
c. Top of Containment Structure	-5 to +5g	1
d. Reactor Coolant Piping	-2 to +2g	1
Seismic Triggers		
a. Containment Base in Tendon Gallery	+0.005 to +0.05g	1
b. Containment Operating Level	+0.005 to +0.05g	1
4. Seismic Switches		
a. Steam Generator Base Support	Set pt. 0.45 Horz/0.30 Vert.	1**
b. Containment Base in Tendon Gallery	Set pt. 0.40 Horz/0.50 Vert.	1**
5. Seismic Alarm Annunciator (4a & 4b are sensors)		
a. Control Room Panel L-167		
6. Peak Shock Recorder		
a. Containment Base in Tendon Gallery	2 to 25.4 Hz 1.6 to 90g	1**
7. Peak Shock Annunciator	2 to 25.4 Hz 1.6 to 90g	1
a. Control Room Panel L-167		

**With control room indication

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	FUNCTIONAL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Strong Motion Accelerometers			
a. Steam Generator Base Support	M*	R	SA
b. Pressurizer Base Support	M*	R	SA
c. Reactor Coolant Pump	M*	R	SA
d. Containment Base in Tendon Gallery	M*	R	SA
e. Containment Operating Level	M*	R	SA
f. Control Building Basement	M*	R	SA
g. Control Building Roof	M*	R	SA
h. Safety Equipment Building Base	M*	R	SA
i. Safety Equipment Building Piping Support	M*	R	SA
j. Radwaste Building Equipment Support	M*	R	SA
2. Triaxial Peak Recording Accelerographs			
a. Control Building-Control Room	N/A	R	N/A
b. Control Building Base	N/A	R	N/A
c. Top of Containment Structure	N/A	R	N/A
d. Reactor Coolant Piping	N/A	R	N/A
Seismic Triggers			
a. Containment Base in Tendon Gallery	M	R	SA
b. Containment Operating Level	M	R	S/U***
4. Seismic Switches			
a. Steam Generator Base Support	M	R**	SA**
b. Containment Base in Tendon Gallery	M	R**	SA**
5. Seismic Alarm Annunciators (4a & 4b are sensors)			
a. Control Room Panel L-167	M	R	SA
6. Peak Shock Recorder			
a. Containment Base in Tendon Gallery	N/A	R**	N/A
7. Peak Shock Annunciator			
a. Control Room Panel L-167	N/A	R**	N/A

* Except seismic trigger
** With Control Room indication
*** Need not be performed more frequently than once per 6 months.

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 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

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT		LOCATION	MINIMUM OPERABLE
1. WIND SPEED			
	a. 0-50 mph	Nominal Elev. 10 meters	1
	b. 0-50 mph	Nominal Elev. 40 meters	1
2. WIND DIRECTION			
	a. 0-360-180°	Nominal Elev. 10 meters	1
	b. 0-360-180°	Nominal Elev. 40 meters	1
3. AIR TEMPERATURE			
	a. -30 to +50°C	Nominal Elev. 10 meters	1
4. Delta Temperature			
	a. -3°C to +3°C	Nominal Elev. 10/40 meters	1
	b. -3°C to +3°C	Nominal Elev. 10/40 meters	1
5. Sigma Azimuth			
	a. -0 to 45°	Nominal Elev. 10 meters	1

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 10 meters	D	SA
b. Nominal Elev. 40 meters	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 10 meters	D	SA
b. Nominal Elev. 40 meters	D	SA
3. AIR TEMPERATURE		
a. Nominal Elev. 10 meters	D	SA
4. Delta Temperature		
a. Nominal Elev. 10/40 meters	D	SA
b. Nominal Elev. 10/40 meters	D	SA
5. Sigma Azimuth		
a. Nominal Elev. 10 meters	D	SA

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REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

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TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	CHANNEL RANGE	MINIMUM CHANNELS OPERABLE
1. Log Power Level	*	10^{-8} - 200%	1
2. Reactor Coolant Cold Leg Temperature	*	0-600°F	1/loop
3. Pressurizer Pressure	*	0-3000 psia	1
4. Pressurizer Level	*	0-100%	1
5. Steam Generator Pressure	*	0-1200 psia	1/steam generator
6. Steam Generator Level	*	0-100%	1/steam generator
7. Reactor Coolant Boron Concentration	*	0-2500 ppm	1
8. Condenser Vacuum	*	0-5" Hg	1
9. Volume Control Tank Level	*	0-100%	1
10. Letdown Heat Exchanger Pressure	*	0-600 psig	1
11. Letdown Heat Exchanger Temperature	*	0-200°F	1
12. Boric Acid Makeup Tank Level	*	0-100%	1
13. Condensate Storage Tank Level	*	0-100%	1
14. Reactor Coolant Hot Leg Temperature	#	190-625°F	1
15. Pressurizer Pressure - Low Range	#	0-1600 psia	1
16. Pressurizer Pressure - High Range	#	1500-2500 psia	1
17. Pressurizer Level	#	0-100%	1
18. Steam Generator Pressure	#	0-1050 psia	1/steam generator
19. Steam Generator Level	#	0-100%	1/steam generator

*Panel L042
#Panel L411

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TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Log Power Level	M	R
2. Reactor Coolant Cold Leg Temperature	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Level	M	R
6. Steam Generator Pressure	M	R
7. Reactor Coolant Boron Concentration	M	R
8. Condenser Vacuum	M	R
9. Volume Control Tank Level	M	R
10. Letdown Heat Exchanger Pressure	M	R
11. Letdown Heat Exchanger Temperature	M	R
12. Boric Acid Makeup Tank Level	M	R
13. Condensate Storage Tank Level	M	R
14. Reactor Coolant Hot Leg Temperature	M	R
15. Pressurizer Pressure - Low Range	M	R
16. Pressurizer Pressure - High Range	M	R
17. Pressurizer Level	M	R
18. Steam Generator Pressure	M	R
19. Steam Generator Level	M	R

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.*

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

a. ~~With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.~~

~~b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

b. ~~c.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

*See Special Test Exception 3.10.5.

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

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

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
19. Containment Area Radiation - High Range	2	1	20 ~~22~~, ~~23~~ 21
20. Main Steam Line Area Radiation	1/steam line	N.A.	20 ~~22~~
21. Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	20 ~~22~~
22. Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	~~22~~ 20, ~~23~~ 21
23. Cold Leg HPSI Flow	~~2/cold leg~~ 1/cold leg	~~1/cold leg~~ N.A.	20
24. Hot Leg HPSI Flow Lc Flow	1/hot leg	N.A.	20

NOTES:

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

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TABLE 3.3-10 (Continued)

ACTION STATEMENTS

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

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

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

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

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

~~2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R

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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.

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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. Restore the inoperable instrument(s) to OPERABLE status within 14 days 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 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

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.

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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"		~~6~~	10			
	Cable Tray Areas Elev 45'		~~4~~	9			
	Cable Tray Areas Elev 30'		~~4~~	4			
	Elevator Machinery Room						
	Combustible Oil Area			1			
	Two steam generator rooms				32		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 301-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.

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	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 30'						
	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		

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
29	Control Building Elev 30'						
	Cable Riser Gallery Rm 236			3	51		
	Cable Riser Gallery Rm 224			3	52		
30	Electrical Tunnel Elev 30'6"			13	50		
31	Control Building Elev 30'			29			
32A	Control Building Elev 30'						
	Fan Room Rm 219 & Corridor Rm 221			1	2		
32B	Control Building Elev 30'						
	Fan Room Rm 233 & Corridor Rm 234			1			
34	Radwaste Elev 9' & 24'						
	Secondary Radwaste Tank Rms 126A,B & 127A,B	None					
35	Radwaste Elev 9' & 24'						
	Spent Resin Tank Rms 125A,B	None					
36	Fuel Handling Building Elev 17'6"						
	Spent Fuel Pool Pump Rm 107			2			
37	Radwaste Elev 24'						
	Letdown Heat Exchanger Rms 209A,B	None					
38	Radwaste Elev 24'						
	Letdown Control Valve Rms 218A,B	None					
39	Radwaste Elev 24'						
	Filter Crvd Tank Rm 216	None					
40	Radwaste Elev 9' & 24'						
	Primary Radwaste Tank Rms 211A,D	None					
41	Control Building Elev 9'						
	Cable Spreading Rm 111A			17	36		
	Cable Spreading Rm 111B			14	36		
42	Control Building Elev 9'						
	Cable Riser Gallery Rm 110			6	44		
	Cable Riser Gallery Rm 112			6	39		

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TABLE 3.3-11 (Continued)

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Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
43	Control Building Elev 9'						
	Emgy. Chiller Rm 115			2			
	Emgy. Chiller Rm 117			2			
44	Intake Structure						
	Pump Rm T2-106			4			
	Pump Rm T3-106			4			
45	Penetration Area Elev 9' & 15'						
	Piping Penetration Area 15'	None					
48	Safety Equipment Building 9'						
	CCW HX and Piping Rm 022-025	None					
50	Radwaste Elev 9'						
	Charging Pump Rms 106A-F			6			
51	Radwaste Elev 9'						
	Boric Acid Makeup Tank Rms 105A-D	None					
53	Electrical Tunnel Elev 9'6", 11'6", (-) 2'6"			21	54		
54	Safety Eqpmt Bldg Elev 15'6" & 8'						
	Shutdown HX Rms 003, 004, 016, 018	None					
55	Safety Eqpmt Bldg Elev 8'						
	Chemical Storage Tank Rm 019			1			
56	Safety Eqpmt Bldg Elev 8'						
	Component Cooling Water Surge Tank Rms 020, 021	None					
57	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 005			1			
58	Radwaste Elev 37'						
	Reactor Trip System Rms 308A-D, 309-A-C			9			
59	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 001			1			

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
60	Safety Eqpmt Bldg Elev 15'6" Pump Rm 015			1			
61	Safety Eqpmt Bldg Elev 15'6" Component Cooling Water Pump Rms 006, 007, 008			3			
62	Radwaste Elev 50' Volume Control Valve Rooms	None					
63	Control Building Elev 50' Corridor			12			
64	Control Building Elev 50' Vital Power Distribution Rms 310A-H			8			
65	Control Building Elev 50' Battery Rms 306B-J			8			
66	Control Building Elev 50' Evacuation Rm 311			1			
67	Radwaste Elev 63'6" Cable Riser Gallery Rm 506A Cable Riser Gallery Rm 506B			2 2	4 4		
68	Penetration 9' - 63'6" Cable Riser Shaft			1	21		
69	Safety Eqpmt Bldg Elev 5'3" Salt Water Cooling Piping Rm 010	None					
70	Radwaste Elev 24' Duct Shaft Rms 222A,B	None					
72	Control Building Elev 70' Corridor 442	None					
75	Refueling Water Storage Tank T-005	None					
76	Refueling Water Storage Tank T-006	None					

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
78	Control Building Elev 9'						
	Corridor Rm 105			4			
79	Control Building Elev 50'						
	ESF Switchgear Rm 302A			2			
	ESF Switchgear Rm 302B			2			
80	Radwaste Elev 37' & 50'						
	Duct Shaft Rms	None					
81	Radwaste Elev 63'6"						
	Duct Shaft Rms 527A,B	None					
83	Salt Water Cooling Tunnel			6*			
84	Safety Eqpmt Bldg Elev 8'						
	HVAC Rm 017			3			

*3 in UNIT 2, 3 in UNIT 3

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.*

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain in 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

4.3.3.8.1 Each radioactive liquid 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 4.3-8.

4.3.3.8.2 At least once per 4 hours at least one circulating water pump shall be determined to be operating and providing dilution to the discharge structure whenever dilution is required to meet the site radioactive effluent concentration limits of Specification 3.11.1.1.

*See Special Test Exception 3.10.5.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - 2/3 RT - 7813	1	28
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - ~~2~~3RT - 7817	1	29
c. Turbine Building Sumps Effluent Line - ~~2~~3RT - 7821	1	30
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	1	31

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TABLE 3.3-12 (Continued)

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TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.3, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - 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 analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 30 - 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 that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 31 - 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 during actual releases. Pump curves may be used to estimate flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line - 2/3 RT - 7813	D	P	R(2)	Q(1)
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - ~~2~~3RT - 7817	D	M	R(2)	Q(1)
c. Turbine Building Sumps Effluent Line - ~~2~~3RT - 7821	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	D(3)	N.A.	R	Q

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TABLE 4.3-8 (Continued)

TABLE NOTATION

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(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 call for declaring the channel inoperable.

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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

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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 - ~~2~~3RT - 7818 or ~~2~~3RT - 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 ~~2~~3RT - 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

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TABLE 3.3-13 (Continued)

TABLE NOTATION

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* 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.

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TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
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	P	P	R(3)	Q(1)	*
b. Flow Rate Monitor	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (periodic)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (periodic)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor - 3~~2~~RT - 7818, 3~~2~~RT - 7870-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
4. PLANT VENT STACK					
a. Noble Gas Activity Monitor - 2/3 RT - 7808, or 2RT - 7865-1 and 3RT-7865-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE SYSTEM(7)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 3~~2~~ RT - 7804-1	S	P(6)	R(3)	M(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

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TABLE 4.3-9 (Continued)

TABLE NOTATION

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* 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 ~~2~~3RT-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.

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3/4.3.3 MONITORING INSTRUMENTATION

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.

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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

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 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.

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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 ~~3.4.1~~ 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 ~~4.4.1~~ 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.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required Reactor Coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no Reactor Coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Reactor Coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

3. Shutdown Cooling Train A,

4. Shutdown Cooling Train B.

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

APPLICABILITY: MODE 4~~#~~

ACTION:

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

b. With no Reactor Coolant loop or shutdown cooling train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop/ train to operation.

~~#With the Reactor Coolant System cold leg temperature less than or equal to 235°F, the SDCS isolation valves HV-9337, HV-9339, HV-9377, and HV-9378 shall be open with the SDCS relief valve PSV-9349 OPERABLE.~~

*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.

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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.

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COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling train shall be OPERABLE ~~with all suction line valves open~~ 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).

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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.

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REACTOR COOLANT SYSTEM

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COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 with Reactor Coolant loops not filled.

ACTION:

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

b. With no shutdown cooling trains in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling train to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.4.2 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
STANDBY HOT T X

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.
HOT STANDBY HOT SHUTDOWN x x

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in both sets of inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to the nominal tube wall thickness as determined in Figure 4.4-1.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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Table 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED
DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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Figure 4.4-1
TUBE WALL THINNING ACCEPTANCE CRITERIA

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3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. A containment sump inlet flow monitoring system, and

c. A containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump inlet flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Containment atmosphere gaseous monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

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REACTOR COOLANT SYSTEM

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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.

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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 returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 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.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SECTION A

3-018-A-551	HPSI Check
3-019-A-551	HPSI Check
3-020-A-551	HPSI Check
3-021-A-551	HPSI Check
3-152-A-551	Hot leg injection to loop #1
3-156-A-551	Hot leg injection to loop #2
3-157-A-551	Hot leg injection check
3-158-A-551	Hot leg injection check
3 ~~2~~ HV-9337	SDC Suction Isolation
3 ~~2~~ HV-9339	SDC Suction Isolation
3 ~~2~~ HV-9377	SDC Suction Isolation
3 ~~2~~ HV-9378	SDC Suction Isolation

SECTION B

8-072-A-552	LPSI Check
8-073-A-552	LPSI Check
8-074-A-552	LPSI Check
8-075-A-552	LPSI Check
12-027-A-551*	Cold leg injection to loop #1A
12-029-A-551*	Cold leg injection to loop #1B
12-031-A-551*	Cold leg injection to loop #2A
12-033-A-551*	Cold leg injection to loop #2B
12-040-A-551	SIT Check
12-041-A-551	SIT Check
12-042-A-551	SIT Check
12-043-A-551	SIT Check

*Redundant to LPSI and SIT checks

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REACTOR COOLANT SYSTEM

3/4.4.6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and 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 operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

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3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

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

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

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ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE

AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

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Figure 3.4-1

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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~~ plate C-6802-1 ~~C-6404-2~~ as determined by impact testing, or X

b. The predicted shift in reference temperature for ~~weld seams~~ weld seam 9-203 ~~3-203A or 3-203B~~ as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." X

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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

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Figure 3.4-2

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Figure 3.4-3

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REACTOR COOLANT SYSTEM

PRESSURIZER – HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer ~~temperature~~ shall be limited to:

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE ≤ 235°F

LIMITING CONDITION FOR OPERATION

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

Change "a" to See Next page

a. ~~The Shutdown Cooling System (SDCS) Relief Valve (PSV9349) with a lift setting of less than or equal to 402 psig~~, or,

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

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

ACTION:

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

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 3HV9337 and 3HV9339 or valve pair 3HV9377 and 3HV9378) closed, open the closed valve(s) within 7 days or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. In the event either the SDCS Relief Valve or an RCS vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that ~~at least one pair of~~ SDCS Relief Valve isolation valves ~~(valve pair~~ 3HV9337, ~~and~~ 3HV9339, ~~or valve pair~~ 3HV9377 and 3HV9378~~) is~~ are open.

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

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Change 3.4.8.3.1 to the following:

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

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

2) Relief Valve isolation valves 3HV9337, 3HV9339, 3HV9377 and 3HV9378 open, or,

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

4.4.8.3.1.2 The RCS vent shall be verified to be open at least once per 12 hours* when the vent is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE > 235°F

LIMITING CONDITION FOR OPERATION

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

Change "a" to see next page

a. ~~The Shutdown Cooling System (SDCS) Relief Valve (PSV 9349) with a setting of less than or equal to 402 psig~~, or,

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

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

Change "ACTION" to see next page

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 3HV9337, 3HV9339, 3HV9377 and 3HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.

b. ~~Testing pursuant to Specification 4.0.5 with an inservice test interval of at least once per 30 months~~. Verify relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

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

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

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

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

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Change 3.4.8.3.2 "a" to:

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

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

2) Relief Valve isolation valves 3HV9337, 3HV9339, 3HV9377 and 3HV9378 open, or,

Change "ACTION" of 3.4.8.3.2 to:

ACTION:

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

b. In the event the SDCS Relief Valve or an RCS vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve code safety valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

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3.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: ALL MODES

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure greater than or equal to 715 psia.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days by verifying the fuses removed from each safety injection tank vent valve.

d. At least once per 31 days when the RCS pressure is above 715 psia, by verifying that the isolation valve operator breakers are padlocked in the open position.

e. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. Before an actual or simulated RCS pressure signal exceeds ~~715~~ 515 psia, and

   2. Upon receipt of an SIAS test signal.

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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 ~~Sump~~ 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.

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EMERGENCY CORE COOLING SYSTEMS

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~~ HV0396	SDC Bypass Flow Control	CLOSED ~~LOCKED OPEN (THROTTLED)(MANUAL)~~
h.	~~14-153~~ HV8161	SDC Bypass Flow Isolation	~~LOCKED CLOSED (MANUAL)~~ OPEN
i.	14-081	HV-0396 Isolation	LOCKED OPEN (MANUAL)
j.	14-082	HV-0396 Isolation	LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV~~9316~~ 8160	BYPASS SDC ~~(HX)~~ Flow Control	OPEN ~~(THROTTLED)(AIR REMOVED)~~
n.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
o.	~~14-78~~ HV8162	~~HV9316~~ Isolation	~~LOCKED~~ OPEN ~~(MANUAL)~~
p.	~~14-80~~ HV8163	LPSI MINIFLOW ~~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 700 ~~psia~~ psig, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia. X

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. 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 SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

   c. Charging pump.

3. Verifying that on a Recirculation Actuation Test Signal, the containment sump isolation valves open and tha recirculation valves ot the refueling water tank close.

f. By verifying that each of the following pumps develops the indicated developed head and/or flow rate when tested pursuant to Specification 4.0.5:

1. High-Pressure Safety Injection pumps developed head, at an indicated flow rate of 650 gpm, greater than or equal to ~~2142~~ (later) feet for P017, ~~2101~~ (later) feet for P018 and ~~2103~~ (later) for P019. X

2. Low-Pressure Safety Injection pump developed head greater than or equal to ~~406.1~~ (later) feet. X

3. Charging pump flow rate greater than or equal to 40 gpm.

g. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

1. For High-Pressure Safety Injection pump cold leg injection with a single pump running:

   a. The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to (later) ~~657~~ gpm for P017 running, ~~667~~ (later) gpm for P018 running and ~~672~~ (later) gpm for P019 running, and X

   b. The total pump flow rate is greater than or equal to (later) ~~900~~ gpm for P017 running, ~~913~~ (later) gpm for P018 running and ~~918~~ (later) gpm for P019 running. X

SURVEILLANCE REQUIREMENTS (Continued)

2. For a single High-Pressure Safety Injection pump hot/cold leg injection.

   a. The sum of the cold leg injection flow rates is greater than or equal to ~~385~~ (later) gpm, and X

   b. The hot leg injection flow rate is greater than or equal to ~~385~~ (later) gpm. X

   c. The combined total hot/cold legs injection flow rate is greater than or equal to ~~895~~ (later) gpm. X

3. For the Low-Pressure Safety Injection pump with a single pump running:

   a. The flow through each injection leg shall be greater than or equal to 3000 gpm when tested individually and corrected to the same pump suction source and leg back pressure conditions. The difference between high and low flow legs shall be less than or equal to 100 gpm.

   b. The total ECCS flow through 2 cold leg injection lines shall be greater than or equal to ~~4450~~ (later) gpm when corrected for elevation head. X

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3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

b. An 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 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 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.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure less than 400 psia.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

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

b. Between 1720 and 2300 ppm of boron, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

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

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a 55.7 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.10 percent by weight of the containment air per 24 hours at P_a, 55.7 psig, or

2. Less than or equal to L_t, 0.05 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 27.9 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (55.7 psig) or at P_t (27.9 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection. Prior to the Type A tests a visual inspection shall be conducted in accordance with Specification 4.6.1.6 to demonstrate the containment structural integrity.

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SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. For the superimposed leak test, verifies that the difference between the supplemental and Type A test data is within 0.25 L_a or 0.25 L_t, has a sufficient duration to establish accurately the change in leakage rate between the Type A test and the supplemental test, and requires the quantity of gas bled from the containment during the supplemental test to be equal to at least 25 percent of the total measured leakage at P_a (55.7) psig or P_t (27.9) psig.

2. For the mass step change test, verifies that the metered mass of air bled from or injected into the containment and the change of mass in containment air as measured by the Type A test instrumentation are within 25 percent, does not remove or inject more than 25 percent of the daily allowable leakage in any one hour period, and involves a total metered mass change between 75 and 125 percent of the daily allowable leakage.

d. Type B and C tests shall be conducted with gas at P_a (55.7 psig) at intervals no greater than 24 months except for tests involving:

1. Air locks, and

2. Valves pressurized with fluid from a seal system.

3. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.7.3.

f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (61.3 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.

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SURVEILLANCE REQUIREMENTS (Continued)

h. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

i. The provisions of Specification 4.0.2 are not applicable.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE: Change "a" +

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 La 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.

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INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between +1.5 and -0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any four of the following locations and shall be determined at least once per 24 hours:

Location

a. Elevation 176'-0"

b. Elevation 68'-0"

c. Elevation 49'-6"

d. Elevation 34'-0"

e. Elevation 19'-6"

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CONTAINMENT SYSTEMS

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the maximum first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the minimum lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to ~~their~~ obtain a lift off force equal to +0, -5% of the prescribed upper limit. ~~observed lift off force, ± 3%.~~ During retensioning of these tendons, the change in load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a specific load differs by more than 5% from that recorded during installation of tendons, an investigation should be made to ensure

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that such difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be restored to the required level of integrity. More than one tendon ~~above or~~ below the predicted bounds out of the original sample population or the lift off force of a selected tendon lying below 90% of the prescribed lower limit is evidence of abnormal degradation of the containment structure. X

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

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

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

c. Performing a visual inspection of the following:

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

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

3. Complete ~~Minimum~~ grease coverage exists for ~~different parts of t~~he anchorage system. X

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

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TABLE 4.6-1

TENDON SURVEILLANCE

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This was in wrong place
please insure
proper placement

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TABLE 4.6-2

TENDON LIFT-OFF FORCE

U TENDONS

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

HOOP TENDONS

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

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CONTAINMENT VENTILATION SYSTEM

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.

SUREVILLANCE REQUIREMENTS

4.6.1.7.1 The 42-inch containment purge supply and exhaust isolation valves shall be verified to be:

a. Closed at least once per 24 hours.

b. 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.

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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 positioned to take suction from the RWST on a Containment Spray Actuation (CSAS) test signal.

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.

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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 up to ~~the riser vent (and overflow) valve.~~ ten feet below the lowest spray ring by connecting a precision pressure guage to the riser drain.

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.

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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.

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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.

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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 containment isolation test signal, each isolation valve actuates to its isolation position.

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SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A.	CONTAINMENT ISOLATION (CIAS)		
1	HV-0510	Pressurizer steam space sample	40
1	HV-0511	Pressurizer steam space sample	40
2	TV-9267	Letdown line to letdown heat exchanger	40
2	HV-9205	Letdown line to letdown heat exchanger	40
4	HV-0508	Reactor coolant loops hot leg sample	40
4	HV-0509	Reactor coolant loops hot leg sample	40
4	HV-0517	Reactor coolant loops hot leg sample	40
6	HV-9334	Safety injection drain to RWST	40
7	HV-9217	Reactor coolant pump seal bleed off	40
7	HV-9218	Reactor coolant pump seal bleed off	40
11	HV-7911	Demineralized water to service station and sump pump	40
12	HV-0512	Pressurizer surge line sample	40
12	HV-0513	Pressurizer surge line sample	40
13	HV-5803	Containment sump pump discharge	40
13	HV-5804	Containment sump pump discharge	40
14	HV-5686	Fire protection	40
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
18	HV-9821	Containment minipurge inlet	5
18	HV-9823	Containment minipurge inlet	5
19	HV-9824	Containment minipurge outlet	5
19	HV-9825	Containment minipurge outlet	5
22	HV-5388	Instrument air supply line	40
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	40
26	HV-7512	Reactor coolant drain tank pump discharge	40
26	HV-7513	Reactor coolant drain tank pump discharge	40
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1

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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
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

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TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-376*	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
10~~B~~C	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10~~B~~C	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-396*	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-145	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-145	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

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TABLE 3.6-1 (Continued)

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

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TABLE 3.6-1 (Continued)

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

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TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304#	Containment emergency sump recirculation	NA
54	HV-9302#	Containment emergency sump recirculation	NA
55	HV-9305#	Containment emergency sump recirculation	NA
55	HV-9303#	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C cooling water inlet	NA
61	HV-6371	Containment emergency A/C cooling water inlet	NA
62	HV-6367	Containment emergency A/C cooling water inlet	NA
63	HV-6373	Containment emergency A/C cooling water inlet	NA
67	3"-157-A-551	Hot leg injection	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
71	3"-158-A-551	Hot leg injection	NA
75	HV-4715#	Steam generator auxiliary feedwater	NA
75	HV-4731#	Steam generator auxiliary feedwater	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA
78	HV-4714#	Steam generator auxiliary feedwater	NA
78	HV-4730#	Steam generator auxiliary feedwater	NA

*May be opened on an intermittent basis under administrative control.

**Power to the valve removed in accordance with Specification 3.6.1.7.

#Not subject to Type C leakage tests.

@Shutdown cooling valves may be opened in MODE 4.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and, at least once per 92 days on a STAGGERED TEST BASIS, by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a recombiner system functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kw.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

   3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

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CONTAINMENT DOME AIR CIRCULATORS

LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent dome air circulator trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one dome air circulator train inoperable, restore the inoperable train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each dome air circulator train shall be demonstrated OPERABLE:

a. At least once per 18 months by starting each train on a CCAS signal and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least 37,000 cfm.

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3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER Line No. 1	VALVE NUMBER Line No. 2	LIFT SETTING ($\pm$ 1%)*	ORIFICE SIZE
a.	3~~2~~PSV-8401	3~~2~~PSV-8410	1100 psia	16 in^2
b.	3~~2~~PSV-8402	3~~2~~PSV-8411	1107 psia	16 in^2
c.	3~~2~~PSV-8403	3~~2~~PSV-8412	1114 psia	16 in^2
d.	3~~2~~PSV-8404	3~~2~~PSV-8413	1121 psia	16 in^2
e.	3~~2~~PSV-8405	3~~2~~PSV-8414	1128 psia	16 in^2
f.	3~~2~~PSV-8406	3~~2~~PSV-8415	1135 psia	16 in^2
g.	3~~2~~PSV-8407	3~~2~~PSV-8416	1142 psia	16 in^2
h.	3~~2~~PSV-8408	3~~2~~PSV-8417	1149 psia	16 in^2
i.	3~~2~~PSV-8409	3~~2~~PSV-8418	1155 psia	16 in^2

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

TABLE 3.7-2

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

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

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

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 upon receipt of an EFAS test signal.

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CSTs) shall be OPERABLE with a contained volume of at least 144,000 gallons in T-121 and 280,000 gallons in T-120.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either restore the CSTs to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits.

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ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

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MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

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

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SURVEILLANCE REQUIREMENTS

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

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

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

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

SURVEILLANCE REQUIREMENTS

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

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops 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.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position and each component cooling water pump starts automatically on an SIAS test signal.

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3/4.7.4 SALT WATER COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent salt water cooling loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one salt water cooling loop OPERABLE, restore at least two loops 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.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two salt water cooling loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position and each salt water cooling pump starts automatically on an SIAS test signal.

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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:

MODES 1, 2, 3 and 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.

MODES 5 and 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 1000 cfm ± 10% 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.

*Specification 3.0.4 not applicable for initial entry into MODE 6.

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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 1000 cfm ± 10% 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 1000 cfm ± 10% 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 4.3 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 1000 cfm ± 10% 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 3.2 kw ± 5% when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10% for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10% for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

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3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers listed in Tables 3.7-4a and 3.7-4b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. If less than two snubbers of any type are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

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No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Refueling Outage Inspections

During each refueling outage an inspection shall be performed of all snubbers listed in Tables 3.7-4a and 3.7-4b attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

#The provisions of Specification 4.0.2 are not applicable.

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e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of either: (1) At least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6.f. an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested, or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.1, "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.6.f. The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If any any time the point plotted falls in the "Accept" region testing of that type of snubber shall be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers of each type. The representative sample should be weighted to include more snubbers from severe service areas such as near heavy equipment. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers, may be tested to verify only that activation takes place in both directions of travel.

2. Snubber bleed, or release rate where required, is present in both tensions and compression, within the specified range.

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3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction of travel.

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubber are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers were attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptace criteria subsequent to their most recent service, and the functional test must have been performed within 12 month before being installed in the unit.

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SURVEILLANCE REQUIREMENTS (Continued)

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not fail between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information and the seals shall be replaced so that the maximum expected service life does not expire during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

j. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in Tables 3.7-4a and 3.7-4b with footnotes indicating the extent of the exemptions.

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Use Same Figure

FIGURE 4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

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TABLE 3.7-4a

SAFETY RELATED HYDRAULIC SNUBBERS*

PAUL MONROE

SYSTEM	Size (Kips)	
	543	826
RC	4	4
TOTAL	8	

*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7-4a provided that a revision to Table 3.7-4a is included with the next License Amendment request.

Table 3.7-4b

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Safety-Related Mechanical Snubbers*

PACIFIC SCIENTIFIC

System	Size (Kips) Small 1/4	1/2	Medium 1	3	6	Large 10	35	100
RC	44 ~~35~~	8 ~~12~~	13 ~~15~~	40 ~~34~~		17 ~~26~~	12 ~~8~~	2 ~~3~~
ST	33 ~~28~~	8 ~~2~~	18 ~~20~~	19 ~~12~~		4 ~~8~~	8 ~~5~~	23 ~~22~~
BM	19 ~~18~~	8 ~~11~~						
CC	17 ~~19~~	17 ~~24~~	11 ~~16~~	4		5 ~~4~~	2	
SS	39 ~~34~~	0 ~~2~~	1 ~~2~~					
SI	85 ~~86~~	47 ~~48~~	51	46		21 ~~20~~	16 ~~18~~	1
FW	1	4	2	4		15 ~~13~~	2	
FS	4 ~~2~~	22 ~~18~~	9 ~~10~~	6 ~~8~~				
VC	51 ~~39~~	18	7 ~~8~~					
CS	4 ~~5~~	17 ~~14~~	26 ~~25~~	18 ~~16~~				
SC				2 ~~4~~		2 ~~3~~	2	
CH		1						
CB			4					
GR	2 ~~4~~							
CEDM				4	212			
Subtotal-1	299 ~~277~~	150 ~~154~~	142 ~~154~~	143 ~~129~~	212	64 ~~65~~	42 ~~38~~	26
Subtotal-2	449 ~~431~~		561 ~~560~~			68 ~~59~~		
Total			1078 ~~1050~~					

*Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request.

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3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.7 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:
   1. Decontaminate and repair the sealed source, or
   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.7.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.7.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:
   1. With a half-life greater than 30 days (excluding Hydrogen 3), and
   2. In any form other than gas.

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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.

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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, 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 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to 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:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:

      a) By telephone within 24 hours,

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

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ACTION: (Continued)

c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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.

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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.

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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 one 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. Restore the system to OPERABLE status within 14 days 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 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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.

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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.

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TABLE 3.7-5

Safety Related Spray and/or Sprinkler Systems

Hazard	Location	No. of Systems	System Type
Reactor Coolant Pumps	Containment	4	Deluge-Water Spray
R.R. Tunnel	Fuel Hand. Bldg.	1	Wet Pipe
Truck Ramp	Radwaste Bldg.	1	Wet Pipe
Cable Tunnel	Section 1	1	Deluge-Water Spray
Cable Tunnel	Section 2	1	Deluge-Water Spray
Cable Tunnel	Section 3	1	Deluge-Water Spray
Cable Tunnel	Section 4	1	Deluge-Water Spray
Cable Tunnel	Section 5	1	Deluge-Water Spray
Cable Tunnel	Section 6	1	Deluge-Water Spray
Cable Tunnel	Section 7	1	Deluge-Water Spray
Cable Tunnel	Section 8	1	Deluge-Water Spray
Cable Tunnel	Section 9	1	Deluge-Water Spray
Cable Tunnel	Section 10	1	Deluge-Water Spray
Cable Tunnel Riser	Fuel Hand Bldg.	1	Deluge-Water Spray
Cable Gallery	Radwaste Bldg.	2*	Deluge-Water Spray
Cable Risers El. 9 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 30 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 50 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 70 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Spreading Room	Control Bldg.	4*	Deluge-Water Spray
~~Charcoal Filter A-353~~	~~Containment~~	~~1~~	~~Deluge-Water Spray~~
Emergency A.C. Unit - Train A	Fuel Handling Bldg.	1	Deluge-Water Spray
Emergency A.C. Unit - Train B	Fuel Handling Bldg.	1	Deluge-Water Spray
~~Charcoal Filter E-419~~	~~Control Bldg.~~	~~1~~	~~Deluge-Water Spray~~
~~Charcoal Filter A-206~~	~~Control Bldg.~~	~~1~~	~~Deluge-Water Spray~~
Diesel Generator	DG Bldg.	2	Pre-Action Sprinkler

Add these

*One half of these Systems are designated Unit ~~2~~ 3 but are required to be OPERABLE for Unit ~~2~~ 3 operation.

HVAC Room 309A; Corridor 303	Control Building 50'	1	Wet Pipe
Auxiliary Feedwater Pump	Tank Building 30'	1	Pre-action Sprinkler
Fan Room 233 and Corridor 234	Control Building 30'	1	Wet Pipe
Salt Water Cooling Pumps Salt Water Cooling Tunnel	Intake Structure	1	Wet Pipe
CCW Heat Exchangers and Piping Room; A/C Room 017	Safety Equipment Bldg.	1	Wet Pipe
Corridor 401	Control Building 70'	1	Wet Pipe
Corridor 105	Control Building 9'	1	Wet Pipe

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

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

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppresion; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days 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 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

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TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	STATION NUMBER
Containment Bldg. - Unit ~~2~~ 3	63'-6"	~~130~~ 131
Containment Bldg. - Unit ~~2~~ 3	63'-6"	~~1~~ 67
Containment Bldg. - Unit ~~2~~ 3	63'-6"	~~8~~ 74
Containment Bldg. - Unit ~~2~~ 3	45'-0"	~~2~~ 68
Containment Bldg. - Unit ~~2~~ 3	45'-0"	~~5~~ 70
Containment Bldg. - Unit ~~2~~ 3	45'-0"	~~9~~ 73
Containment Bldg. - Unit ~~2~~ 3	30'-0"	~~3~~ 64
Containment Bldg. - Unit ~~2~~ 3	30'-0"	~~6~~ 66
Containment Bldg. - Unit ~~2~~ 3	30'-0"	~~10~~ 72
Containment Bldg. - Unit ~~2~~ 3	17'-6"	~~4~~ 65
Containment Bldg. - Unit ~~2~~ 3	17'-6"	~~7~~ 69
Containment Bldg. - Unit ~~2~~ 3	17'-6"	~~11~~ 71
Electrical Penetration Area - Unit ~~2~~ 3	45'-0"	~~120~~ 124
Electrical Penetration Area - Unit ~~2~~ 3	45'-0"	~~121~~ 125
Electrical Penetration Area - Unit ~~2~~ 3	63'-6"	~~122~~ 126
Electrical Penetration Area - Unit ~~2~~ 3	63'-6"	~~123~~ 127
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~~ 3	63'-6"	~~118~~ 128
Fuel Handling Bldg.-Unit ~~2~~ 3	63'-6"	~~119~~ 129

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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 and piping penetration seals and ventilation 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 one 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. Restore the inoperable fire rated assembly and sealing device to OPERABLE status within 7 days 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 30 days outlining the action taken, the cause of the inoperable fire rated assembly and/or sealing device and the plans and schedule for restoring the fire rated assembly and sealing device to OPERABLE status.

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.

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SURVEILLANCE REQUIREMENTS

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices 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 percent each type (mechanical and electrical) of sealed penetrations. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of that particular type of sealed penetration shall be made. This inspection process shall continue until an additional complete 10 percent sample of that type of sealed penetration with no apparent changes in appearance or abnormal degradation are found.

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3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. A day fuel tank containing a minimum volume of 325 gallons of fuel,

   2. A separate fuel storage system containing a minimum volume of 47,000 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators 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.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time 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 diesel generator inoperable in addition to ACTION a or b above, verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

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ACTION (Continued)

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feed pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time 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

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availablity.

1. If transformers 2~~3~~XR2 and 2~~3~~XR1 are the second source, the following buses are required:

   4160 volt Emergency Bus #2~~3~~A04
   4160 volt Emergency Bus 2#~~3~~A06
   480 volt Emergency Bus 2#~~3~~B04
   480 volt Emergency Bus 2#~~3~~B06
   125 volt DC Bus 2#~~3~~D1
   125 volt DC Bus 2#~~3~~D2 or,

2. If transformer 3~~2~~Xu1 is the second source,* visually verify that the disconnect link to the Unit ~~2~~ 3 turbine generator is removed.

*To be used as the second source of off-site power only during initial low power PHYSICS TESTS.

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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.

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c) Verifying that all automatic diesel generator trips, except engine overspeed, generator differential and low-low lube oil pressure, are automatically bypassed.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 5170 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4700 kw. The generator voltage and frequency shall be 4360 ± 436 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained at 4360 ± 436 volts and 60 + 1.2/-0.3 Hz for the first two hours of this test and 4360 ± 436 volts and 60 ± 1.2 Hz during the remaining 22 hours of this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2.d.4b.

9. Verifying that the auto-connected loads to each diesel generator do not exceed 4700 kw.

10. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that each fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

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SURVEILLANCE REQUIREMENTS (Continued)

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

14. Verifying that lockout relay K23 prevents diesel generator starting when the diesel generator is actuated.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

f. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or the equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests.*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

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LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day fuel tanks containing a minimum volume of 325 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 47,000 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and movement of irradiated fuel, or operation of the fuel handling machine with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5) and 4.8.1.1.3.

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3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt battery bank A (~~2~~3B007), and its associated full capacity charger.

b. 125-volt battery bank B (~~2~~3B008), and its associated full capacity charger.

c. 125-volt battery bank C (~~2~~3B009) and its associated full capacity charger.

d. 125-volt battery bank D (~~2~~3B010) and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

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b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

   1. The parameters in Table 4.8-2 meet the Category B limits,

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3. The average electrolyte temperature of ten connected cells is above 60°F.

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

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

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

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

   4. The battery charger will supply at least 300 amperes at 125-volts for at least 12 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ 1.200(b)	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature in accordance with IEEE Std 450-1980.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

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SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two 125-volt battery banks and their associated full capacity chargers shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery banks inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery banks to OPERABLE status as soon as possible.

b. With the required full capacity chargers inoperable, demonstrate the OPERABILITY of their associated battery banks by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the batteries inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Division #1 A.C. Emergency Busses consisting of:
   1. 4160 volt Emergency Bus #3~~2~~A04
   2. 480 volt Emergency Bus #3~~2~~B04

b. Division #2 A.C. Emergency Busses consisting of:
   1. 4160 volt Emergency Bus #3~~2~~A06
   2. 480 volt Emergency Bus #3~~2~~B06

c. 120 volt A.C. Vital Bus #3~~2~~Y01 energized from its associated inverter connected to D.C. Bus #3~~2~~D1*.

d. 120 volt A.C. Vital Bus #3~~2~~Y02 energized from its associated inverter connected to D.C. Bus #3~~2~~D2*.

e. 120 volt A.C. Vital Bus #3~~2~~Y03 energized from its associated inverter connected to D.C. Bus #3~~2~~D3*.

f. 120 volt A.C. Vital Bus #3~~2~~Y04 energized from its associated inverter connected to D.C. Bus #3~~2~~D4*.

g. 125 volt D.C. Bus #3~~2~~D1 energized from Battery Bank 3~~2~~B007.

h. 125 volt D.C. Bus #3~~2~~D2 energized from Battery Bank 3~~2~~B008.

i. 125 volt D.C. Bus #3~~2~~D3 energized from Battery Bank 3~~2~~B009.

j. 125 volt D.C. Bus #3~~2~~D4 energized from Battery Bank 3~~2~~B010.

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

ACTION:

a. With one of the required divisions of A.C. Emergency busses not fully energized, re-energize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Vital Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) re-energize the A.C. Vital Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*One inverter may be disconnected from its D.C. Bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is energized, and (2) the vital busses associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. Busses.

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SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. Emergency Buses consisting of one 4160-volt and one 480-volt A.C. Emergency Bus.

b. 2 - 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. 2 - 125 volt D.C. Busses energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuits(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out,or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.~~and as specified in Table 3.8-1.~~

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(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. ~~For the lower voltage circuit breakers the nominal trip setpoint and short circuit response times are listed in Table 3.8-1~~. 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.

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~Z~~B0106	~~1000 (6)~~	~~.06~~	3~~Z~~BLP0101	~~960 (6)~~	~~.02~~	Containment Normal Cooling Fan E-~~387~~ 397
3~~Z~~B0107	~~4000 (6)~~	~~.06~~	3~~Z~~BLP0102	~~4400 (6)~~	~~.02~~	CEDM Cooling Supply Fan E-403B
3~~Z~~B0109	~~4000 (6)~~	~~.06~~	3~~Z~~BLP0103	~~4400 (6)~~	~~.02~~	CEDM Cooling Supply Fan E-403A
3~~Z~~B0111	~~1000 (6)~~	~~.06~~	3~~Z~~BLP0104	~~960 (6)~~	~~.02~~	Standby Containment Normal Cooling Fan E-333
3~~Z~~B0209	~~1000 (6)~~	~~.06~~	3~~Z~~BLP0201	~~960 (6)~~	~~.02~~	Containment Normal Cooling Fan E-334
3~~Z~~B0406	~~600 (6)~~	~~.06~~	3~~Z~~BLP0301	~~906 (6)~~	~~.02~~	Hydrogen Recombiner E-145 Power Panel L-~~160~~ 180
3~~Z~~B0409	~~1200 (6)~~	~~.06~~	3~~Z~~BLP0302	~~960 (6)~~	~~.02~~	Upper Dome Air Circulator A-701
3~~Z~~B0410	~~1500 (6)~~	~~.06~~	3~~Z~~BLP0303	~~1900 (6)~~	~~.02~~	Containment Emergency Fan E-399
3~~Z~~B0411	~~1500 (6)~~	~~.06~~	3~~Z~~BLP0304	~~1900 (6)~~	~~.02~~	Containment Emergency Fan E-401
3~~Z~~B0419	~~1200 (6)~~	~~.06~~	3~~Z~~BLP0305	~~960 (6)~~	~~.02~~	Standby Upper Dome Air Circulator A-074
3~~Z~~B0606	~~600 (6)~~	~~.06~~	3~~Z~~BLP0401	~~960 (6)~~	~~.02~~	Hydrogen Recombiner E-146 Power Panel L-~~161~~ 181
3~~Z~~B0609	~~1200 (6)~~	~~.06~~	3~~Z~~BLP0402	~~960 (6)~~	~~.02~~	Upper Dome Air Circulator A-072
3~~Z~~B0610	~~1500 (6)~~	~~.06~~	3~~Z~~BLP0403	~~1900 (6)~~	~~.02~~	Containment Emergency Fan E-400
3~~Z~~B0611	~~1500 (6)~~	~~.06~~	3~~Z~~BLP0404	~~1900 (6)~~	~~.02~~	Containment Emergency Fan E-402
3~~Z~~B0619	~~1200 (6)~~	~~.06~~	3~~Z~~BLP0405	~~960 (6)~~	~~.02~~	Standby Upper Dome Air Circulator A-073
3~~Z~~B0809	~~1000 (6)~~	~~.06~~	3~~Z~~BLP0501	~~960 (6)~~	~~.02~~	Containment Normal Cooling Fan E-396
3~~Z~~B0811	~~1000 (6)~~	~~.06~~	3~~Z~~BLP0601	~~1200 (6)~~	~~.02~~	Containment Normal Cooling Fan E-398
3~~Z~~B0903	~~1500 (6)~~	~~.06~~	3~~Z~~BLP0701	~~1200 (6)~~	~~.02~~	Containment Recirculation Unit E-333
3~~Z~~B0906	~~1600 (6)~~	~~.06~~	3~~Z~~BLP0702	~~3200 (6)~~	~~.02~~	Polar Crane (Containment) R001 (C)
3~~Z~~B0907	~~4000 (6)~~	~~.06~~	3~~Z~~BLP0703	~~3200 (6)~~	~~.02~~	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
3~~Z~~B0909	~~4000 (6)~~	~~.06~~	3~~Z~~BLP0704	~~3200 (6)~~	~~.02~~	Standby CEDM Cooling Supply Fan E-404B
3~~Z~~B0911	~~600 (6)~~	~~.06~~	3~~Z~~BLP0705	~~1200 (6)~~	~~.02~~	Containment Recirculating Unit Heater E-568
3~~Z~~BA02	~~7 (1)~~	~~.03~~	3~~Z~~BLP0812	~~15 (2)~~	~~.02~~	CCW from RCP P-001 Seal Heat Exchanger TV-9144
3~~Z~~BA03	~~6 (1)~~	~~.03~~	3~~Z~~BLP0813	~~15 (2)~~	~~.02~~	CCW from RCP P-003 Seal Heat Exchanger TV-9154
3~~Z~~BA04 (3~~Z~~BA04-A)	~~15 (2)~~	~~.03~~	3~~Z~~BLP0801	~~30 (3)~~	~~.02~~	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3 ~~Z~~BA04 3(~~Z~~BA04-B)	~~15 (2)~~	~~.03~~	3 ~~Z~~BLP0802	~~30 (3)~~	~~.02~~	CEDM Cooling Supply Fan E-403B (Enclosure Heater)
3 ~~Z~~BA04 3(~~Z~~BA04-C)	~~15 (2)~~	~~.03~~	3 ~~Z~~BLP0814	~~30 (3)~~	~~.02~~	Standby Containment Normal Cooling Fan E-393 (Enclosure Heater)
3 ~~Z~~BA04 3(~~Z~~BA04-D)	~~15 (2)~~	~~.03~~	3 ~~Z~~BLP0826	~~30 (3)~~	~~.02~~	Containmnt Normal Cooling Fan E-394 (Enclosure Heater)
3 ~~Z~~BA04 3(~~Z~~BA04-E)	~~15 (2)~~	~~.03~~	3 ~~Z~~BLP0828	~~30 (3)~~	~~.02~~	Containment Normal Cooling Fan E-397
3 ~~Z~~BA08	~~15 (2)~~	~~.03~~	3 ~~Z~~BLP0803	~~30 (3)~~	~~.02~~	Movable Incore Detector Drive Package W338A
3 ~~Z~~BA11	~~100 (3)~~	~~.02~~	3 ~~Z~~BLP0905	~~750 (6)~~	~~.02~~	Cont. Structure Electric Heater E-467
3 ~~Z~~BA25	~~10 (1)~~	~~.03~~	3 ~~Z~~BLP0910	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FB
3 ~~Z~~BA26	~~10 (1)~~	~~.02~~	3 ~~Z~~BLP0911	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-394 Circ. Water Outlet HV-~~9930EB~~ 9940EB
3 ~~Z~~BA27	~~10 (1)~~	~~.02~~	3 ~~Z~~BLP0912	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-397 Circ. Water Outlet HV-9940DB
3 ~~Z~~BA31	~~10 (1)~~	~~.02~~	3 ~~Z~~BLP0913	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FC
3 ~~Z~~BA32	~~10 (1)~~	~~.02~~	3 ~~Z~~BLP0914	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-394 Circ. Water Inlet HV-9940EC
3 ~~Z~~BA33	~~10 (1)~~	~~.02~~	3 ~~Z~~BLP0915	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-397 Circ. Water Inlet HV-9940DC
3 ~~Z~~BA36	~~150 (1)~~	~~.02~~	3 ~~Z~~BLP0808	~~70 (4)~~	~~.02~~	RCP 1A Oil Lift Pump 1A1 P-260
3 ~~Z~~BA37	~~150 (1)~~	~~.02~~	3 ~~Z~~BLP0809	~~70 (4)~~	~~.02~~	RCP 1B Oil Lift Pump 1B1 P-264
3 ~~Z~~BA38	~~150 (1)~~	~~.02~~	3 ~~Z~~BLP0810	~~70 (4)~~	~~.02~~	RCP 2B Oil Lift Pump 2B1 P-262
3 ~~Z~~BA39	~~500 (1)~~	~~.02~~	3 ~~Z~~BLP0901	~~750 (6)~~	~~.02~~	Reactor Coolant Drain Pump (W) P-023
3 ~~Z~~BA40	~~150 (1)~~	~~.02~~	3 ~~Z~~BLP0811	~~70 (4)~~	~~.02~~	RCP 2A Oil Lift Pump 2A1 P-266
3 ~~Z~~BA41	~~13 (1)~~	~~.02~~	3 ~~Z~~BLP0817	~~15 (2)~~	~~.02~~	RCP 1A Anti Rev. Rotation Device Lube Pump 1 P-399
3 ~~Z~~BA42	~~13 (1)~~	~~.02~~	3 ~~Z~~BLP0818	~~15 (2)~~	~~.02~~	RCP 2B Anti Rev. Rotation Device Lube Pump 1 P-401
3 ~~Z~~BA43	~~13 (1)~~	~~.02~~	3 ~~Z~~BLP0819	~~15 (2)~~	~~.02~~	RCP 1B Anti Rev. Rotation Device Lube Pump 1 P-403
3 ~~Z~~BA44	~~13 (1)~~	~~.02~~	3 ~~Z~~BLP0820	~~15 (2)~~	~~.02~~	RCP 2A Anti Rev. Rotation Device Lube Pump 1 P-405
3 ~~Z~~BA45	~~625 (1)~~	~~.02~~	3 ~~Z~~BLP0902	~~750 (6)~~	~~.02~~	Reactor Cavity Cooling Fan A-~~313~~ 319
3 ~~Z~~BA46	~~625 (1)~~	~~.02~~	3 ~~Z~~BLP0903	~~750 (6)~~	~~.02~~	Standby Reactor Cavity Cooling Fan A-321

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~BA47	~~7 (1)~~	~~.02~~	3~~2~~BLP0807	~~15 (2)~~	~~.02~~	Charging Line to Reactor Cooling Loop 1A HV-9203
3~~2~~BA49	~~7 (1)~~	~~.02~~	3~~2~~BLP0821	~~30 (3)~~	~~.02~~	Reactor Cavity Cooling Unit C HV-9905C
3~~2~~BA50	~~7 (1)~~	~~.02~~	3~~2~~BLP0822	~~30 (3)~~	~~.02~~	Reactor Cavity Cooling Unit A HV-9905A
3~~2~~BA51	~~7 (1)~~	~~.02~~	3~~2~~BLP0804	~~15 (2)~~	~~.02~~	Quench Tank to Reactor Drain Tank HV-9101
3~~2~~BA55	~~7 (1)~~	~~.02~~	3~~2~~BLP0805	~~15 (2)~~	~~.02~~	RCP Bleed Off to Quench Tank HV-9216
3~~2~~BA57	~~7 (1)~~	~~.02~~	3~~2~~BLP0916	~~15 (2)~~	~~.02~~	CEDM Cooling Unit E-403 CCW Outlet HV-9907AA
3~~2~~BA58	~~7 (1)~~	~~.02~~	3~~2~~BLP0917	~~15 (2)~~	~~.02~~	CEDM Cooling Unit E-403 CCW Inlet HV-9907AC
3~~2~~BA59	~~7 (1)~~	~~.02~~	3~~2~~BLP0806	~~15 (2)~~	~~.02~~	Safety Injection Tank to Reactor Drain Tank HV-9335
3~~2~~BA60	~~90 (3)~~	~~.02~~	3~~2~~BLP0904	~~750 (6)~~	~~.02~~	Welding Receptacles Containment (50 KVA)
3~~2~~BA62	~~15 (2)~~	~~.03~~	3~~2~~BLP0824	~~30 (3)~~	~~.02~~	Recept. for Portable Cont. Sump Pump (H.P.) P-005 ~~(A)~~
3~~2~~BA63	~~50 (3)~~	~~.03~~	3~~2~~BLP0906	~~750 (6)~~	~~.02~~	Containment Elevator P-~~002 (A)~~ -003
3~~2~~BA65	~~150 (1)~~	~~.02~~	3~~2~~BLP0815	~~70 (4)~~	~~.02~~	Lower Level Air Circulator A-031
3~~2~~BA66	~~150 (1)~~	~~.02~~	3~~2~~BLP0816	~~70 (4)~~	~~.02~~	Lower Level Air Circulator A-033
3~~2~~BE09	~~30 (1)~~	~~.02~~	3~~2~~BLP1001	~~15 (2)~~	~~.02~~	Saf. Inj. Tank Drain to Refueling Wtr Tank HV-9334
~~2BE11~~ 3BE10	~~100 (1)~~	~~.02~~	3~~2~~BLP1002	~~30 (3)~~	~~.02~~	Saf. Inj. Tk T-007 to Reactor Coolant Loop 1B HV-9350
3~~2~~BE11	~~100 (1)~~	~~.02~~	3~~2~~BLP1003	~~30 (3)~~	~~.02~~	Saf. Inj. Tk T-009 to Reactor Coolant Loop 1V HV-9360
3~~2~~BE17	~~13 (1)~~	~~.02~~	3~~2~~BLP1010	~~15 (2)~~	~~.02~~	Auxiliary Spray to Pressurizer HV-9201
3~~2~~BE21	~~13 (1)~~	~~.02~~	3~~2~~BLP1012	~~15 (2)~~	~~.02~~	CCW Noncritical Cont. Inlet Isolation Valve HV-6223
3~~2~~BE25	~~200 (1)~~	~~.02~~	3~~2~~BLP1005	~~70 (4)~~	~~.02~~	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9337
3~~2~~BE26	~~7 (1)~~	~~.02~~	3~~2~~BLP1015	~~15 (2)~~	~~.02~~	Reac. Coolant Drain Tk Sample Cont. Isolation HV-0516
3~~2~~BE27	~~7 (1)~~	~~.02~~	3~~2~~BLP1016	~~15 (2)~~	~~.02~~	Containment Isolation Reactor Coolant Drain to Radwaste System HV-7512
3~~2~~BE30	~~7 (1)~~	~~.02~~	3~~2~~BLP1017	~~15 (2)~~	~~.02~~	Quench Tank Vapor Sample Cont. Isol. HV-0514
3~~2~~BE31	~~7 (1)~~	~~.02~~	3~~2~~BLP1004	~~15 (2)~~	~~.02~~	Containment Sump to Radwaste Sump HV-5803
3~~2~~BE33	~~100 (1)~~	~~.02~~	3~~2~~BLP1021	~~30 (3)~~	~~.02~~	Containment Purge Inlet HV-9949
3~~2~~BE35	~~20 (1)~~	~~.02~~	3~~2~~BLP1018	~~15 (2)~~	~~.02~~	Containment Emergency Sump Oulet HV-9305

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~BE46	~~13 (1)~~	~~.02~~	3~~2~~BLP1011	~~15 (2)~~	~~.02~~	CCW Noncritical Containment Isolation Valve HV-6336
3~~2~~BF08	~~55 (1)~~	~~.02~~	3~~2~~BLP0823	~~30 (3)~~	~~.02~~	Containment Sump Pump P-008
3~~2~~BF09	~~55 (1)~~	~~.02~~	3~~2~~BLP1220	~~30 (3)~~	~~.02~~	Containment Sump Pump P-007
3~~2~~BJ05	~~200 (1)~~	~~.02~~	3~~2~~BLP1101	~~70 (4)~~	~~.02~~	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9339
3~~2~~BJ06	~~100 (1)~~	~~.02~~	3~~2~~BLP1104	~~30 (3)~~	~~.02~~	Saf. Inj. Tk T-008 to Reactor Coolant Loop 1A HV-9340
3~~2~~BJ07	~~100 (1)~~	~~.02~~	3~~2~~BLP1105	~~30 (3)~~	~~.02~~	Saf. Inj. Tk T-010 to Reactor Coolant Loop 2B HV-9370
3~~2~~BJ17	~~7 (1)~~	~~.02~~	3~~2~~BLP1123	~~15 (2)~~	~~.02~~	RCP Bleed off to Volume Control Tank HV-9217
3~~2~~BJ21	~~13 (1)~~	~~.02~~	3~~2~~BLP1106	~~15 (2)~~	~~.02~~	Cont. Isol. Safety Injection Tank Vent Header HV-7258
3~~2~~BJ22	~~7 (1)~~	~~.02~~	3~~2~~BLP1115	~~15 (2)~~	~~.02~~	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0508
3~~2~~BJ23	~~7 (1)~~	~~.02~~	3~~2~~BLP1116	~~15 (2)~~	~~.02~~	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0517
3~~2~~BJ26	~~7 (1)~~	~~.02~~	3~~2~~BLP1117	~~15 (2)~~	~~.02~~	Pressurizer Vapor Sample Containment Isol. HV-0510
3~~2~~BJ27	~~7 (1)~~	~~.02~~	3~~2~~BLP1121	~~15 (2)~~	~~.02~~	Pressur. Surge Line Liquid Smpl. Cont. Isol. HV-0512
3~~2~~BJ29	~~100 (1)~~	~~.02~~	3~~2~~BLP1110	~~30 (3)~~	~~.02~~	Containment Purge Outlet HV-9950
3~~2~~BJ30	~~7 (1)~~	~~.02~~	3~~2~~BLP1102	~~15 (2)~~	~~.02~~	Hydrogen Purge Exhaust Unit Inlet HV-9917
3~~2~~BJ31	~~7 (1)~~	~~.02~~	3~~2~~BLP1103	~~15 (2)~~	~~.02~~	Hydrogen Purge Supply Unit Discharge HV-9946
3~~2~~BJ34	~~20 (1)~~	~~.02~~	3~~2~~BLP1118	~~15 (2)~~	~~.02~~	Containment Emergency Sump Outlet HV-9304
3~~2~~BJ47	~~13 (1)~~	~~.02~~	3~~2~~BLP1124	~~15 (2)~~	~~.02~~	Containment Normal Cooling Supply Isol. Valve HV-~~9400~~ 9900
3~~2~~BJ48	~~13 (1)~~	~~.02~~	3~~2~~BLP1125	~~15 (2)~~	~~.02~~	Containment Normal Cooling Return Isol. Valve HV-9971
3~~2~~BN04	~~15 (2)~~	~~.03~~	3~~2~~BLP1201	~~30 (3)~~	~~.02~~	Movable Incore Detector Drive Pack W-~~3383~~ 3388
3~~2~~BN07	~~100 (3)~~	~~.02~~	3~~2~~BLP1304	~~750 (6)~~	~~.02~~	Containment Structure Electric Heater E-466
3~~2~~BN21	~~7 (1)~~	~~.02~~	3~~2~~BLP1206	~~15 (2)~~	~~.02~~	Charging Line to Reactor Coolant Loop 2A HV-9202
3~~2~~BN24	~~625 (1)~~	~~.02~~	3~~2~~BLP1301	~~750 (6)~~	~~.02~~	Reactor Cavity Cooling Fan A-320
3~~2~~BN25	~~625 (1)~~	~~.02~~	3~~2~~BLP1302	~~750 (6)~~	~~.02~~	Standby Reactor Cavity Cooling Fan A-322
3~~2~~BN26	~~7 (1)~~	~~.02~~	3~~2~~BLP1226	~~15 (2)~~	~~.02~~	CCW from RCP P-004 Seal Heat Exchanger TV-9164
3~~2~~BN27	~~7 (1)~~	~~.02~~	3~~2~~BLP1227	~~15 (2)~~	~~.02~~	CCW from RCP P-002 Seal Heat Exchanger TV-9174

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~BN28	~~7 (1)~~	~~.02~~	3~~2~~BLP1207	~~30 (3)~~	~~.02~~	Reactor Cavity Cooling Unit D HV-9905D
3~~2~~BN29	~~7 (1)~~	~~.02~~	3~~2~~BLP1208	~~30 (3)~~	~~.02~~	Reactor Cavity Cooling Unit B HV-9905B
3~~2~~BN30	~~150 (1)~~	~~.02~~	3~~2~~BLP1209	~~70 (4)~~	~~.02~~	RCP 1A Oil Lift Pump 1A2 P-261
3~~2~~BN31	~~150 (1)~~	~~.02~~	3~~2~~BLP1210	~~70 (4)~~	~~.02~~	RCP 1B Oil Lift Pump 1B2 P-265
3~~2~~BN32	~~150 (1)~~	~~.02~~	3~~2~~BLP1211	~~70 (4)~~	~~.02~~	RCP 2B Oil Lift Pump 2B2-263
3~~2~~BN33	~~150 (1)~~	~~.02~~	3~~2~~BLP1212	~~70 (4)~~	~~.02~~	RCP 2A Oil Lift Pump 2A2-267
3~~2~~BN34	~~500 (1)~~	~~.02~~	3~~2~~BLP1303	~~750 (6)~~	~~.02~~	Reactor Coolant Drain Tank Pump (E) P-022
3~~2~~BN37	~~13 (1)~~	~~.02~~	3~~2~~BLP1213	~~15 (2)~~	~~.02~~	RCP 1A Anti Rev. Rotation Device Lube Pump 2 P-402
3~~2~~BN38	~~13 (1)~~	~~.02~~	3~~2~~BLP1214	~~15 (2)~~	~~.02~~	RCP 2B Anti Rev. Rotation Device Lube Pump 2 P-402
3~~2~~BN39	~~13 (1)~~	~~.02~~	3~~2~~BLP1215	~~15 (2)~~	~~.02~~	RCP 1B Anti Rev. Rotation Device Lube Pump 2 P-404
3~~2~~BN40	~~13 (1)~~	~~.02~~	3~~2~~BLP1216	~~15 (2)~~	~~.02~~	RCP 2A Anti Rev. Rotation Device Lube Pump 2 P-406
3~~2~~BN42	~~90 (3)~~	~~.02~~	3~~2~~BLP1305	~~750 (6)~~	~~.02~~	Welding Recpt. Cont. (50KVA) 2R005A, 2R005b, 2R005C
3~~2~~BN43	~~15 (2)~~	~~.03~~	3~~2~~BLP1217	~~30 (3)~~	~~.02~~	CEA Change Mechanism Transfer Machine Control Console (8 KVA) L-023
3~~2~~BN44	~~90 (3)~~	~~.02~~	3~~2~~BLP1306	~~750 (6)~~	~~.02~~	Welding Recpt. Cont. (50 KVA) 2R007A, 2R007B, 2R007C
3~~2~~BN45	~~15 (2)~~	~~.03~~	3~~2~~BLP1218	~~30 (3)~~	~~.02~~	Refueling Pool End Junction Box (8KVA) L-371
3~~2~~BN46	~~90 (3)~~	~~.02~~	3~~2~~BLP1308	~~750 (6)~~	~~.02~~	Welding Recpt. Cont. (50KVA) 2R013A, 2R013B, 2R013C
3~~2~~BN47	~~15 (2)~~	~~.03~~	3~~2~~BLP1219	~~30 (3)~~	~~.02~~	Receptable for Portable Cont. Sump Pump (1hp) P-005
3~~2~~BN48	~~30 (2)~~	~~.03~~	3~~2~~BLP1319	~~20 (4)~~	~~.02~~	Equipment Hatch 200R, Electrical Hoist Z-028, Z-029
3~~2~~BN52	~~150 (1)~~	~~.02~~	3~~2~~BLP1221	~~70 (4)~~	~~.02~~	Lower Level Air Circulator A-032
3~~2~~BN53	~~150 (1)~~	~~.02~~	3~~2~~BLP1222	~~70 (4)~~	~~.02~~	Lower Level Air Circulator A-~~024~~ 034
3~~2~~BN56	~~10 (1)~~	~~.02~~	3~~2~~BLP1310	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-~~346~~ 396 Circ. Water Outlet HV-9940BB
3~~2~~BN57	~~10 (1)~~	~~.02~~	3~~2~~BLP1311	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-396 Circ. Water Inlet HV-9940BC
3~~2~~BN58	~~10 (1)~~	~~.02~~	3~~2~~BLP1312	~~15 (2)~~	~~.02~~	Cont. Coolng Unit E-~~348~~ 398 Circ. Water Outlet HV-9940CB
3~~2~~BN59	~~17 (1)~~	~~.02~~	3~~2~~BLP1313	~~15 (2)~~	~~.02~~	Cont. Cooling Unit E-398 Circ. Water Inlet HV-9940CC
3~~2~~BN60	~~7 (1)~~	~~.02~~	3~~2~~BLP1314	~~15 (2)~~	~~.02~~	CEDM Cooling Unit E-404 CCW Outlet HV-9907BA

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~Z~~BN61	~~7 (1)~~	~~.02~~	3~~Z~~BLP1315	~~15 (2)~~	~~.02~~	CEDM Cooling Unit E-404 CCW Inlet HV-9907BC
3~~Z~~BN62 3(~~Z~~BN62-A)	~~15 (2)~~	~~.03~~	3~~Z~~BLP1223	~~30 (3)~~	~~.02~~	Containment Recirculation Unit A-353 (Motor Enclosure Heater)
3~~Z~~BN62 3(~~Z~~BN62-B)	~~15 (2)~~	~~.03~~	3~~Z~~BLP1224	~~30 (3)~~	~~.02~~	CEDM Cooling Supply Fan E-404A (Motor Enclosure Heater)
3~~Z~~BN62 3(~~Z~~BN62-C)	~~15 (2)~~	~~.03~~	3~~Z~~BLP1225	~~30 (3)~~	~~.02~~	CEDM Cooling Supply Fan E-404B (Motor Enclosure Heater)
3~~Z~~BN62 3(~~Z~~BN62-H)	~~15 (2)~~	~~.03~~	3~~Z~~BLP1202	~~30 (3)~~	~~.02~~	Containment Normal Cooling Fan A-398 (Motor Enclosure Heater)
3~~Z~~BN62 3(~~Z~~BN62-G)	~~15 (2)~~	~~.03~~	3~~Z~~BLP1228	~~30 (3)~~	~~.02~~	Containment Normal Cooling Fan E-~~398~~ 396 (Motor Enclosure Heater)
L0~~108~~ 208	~~900 (6)~~	~~.02~~	L0~~101~~ 201	~~8000 (6)~~	~~.05~~	Panel 3~~Z~~LP4 Emergency Lighting
L0~~118~~ 218	~~900 (6)~~	~~.02~~	L0~~101~~ 201	~~8000 (6)~~	~~.05~~	Panel 3~~Z~~LP11 Emergency Lighting
L0~~120~~ 220	~~900 (6)~~	~~.02~~	L0~~101~~ 201	~~8000 (6)~~	~~.05~~	Panel 3~~Z~~LP16 Emergency Lighting
3~~Z~~BHP0201	~~70 (3)~~	~~.03~~	3~~Z~~B0205	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-607
3~~Z~~BHP0202	~~70 (3)~~	~~.03~~	3~~Z~~B0205	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-608
3~~Z~~BHP0203	~~70 (3)~~	~~.03~~	3~~Z~~B0205	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-609
3~~Z~~BHP0204	~~70 (3)~~	~~.03~~	3~~Z~~B0205	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-610
3~~Z~~BHP0301	~~70 (3)~~	~~.03~~	3~~Z~~B0206	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-611
3~~Z~~BHP0302	~~70 (3)~~	~~.03~~	3~~Z~~B0206	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-612
3~~Z~~BHP0303	~~70 (3)~~	~~.03~~	3~~Z~~B0206	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-613
3~~Z~~BHP0304	~~70 (3)~~	~~.03~~	3~~Z~~B0206	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-614
3~~Z~~BHP0101	~~70 (3)~~	~~.03~~	3~~Z~~B0210	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-601
3~~Z~~BHP0102	~~70 (3)~~	~~.03~~	3~~Z~~B0210	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-602
3~~Z~~BHP0103	~~70 (3)~~	~~.03~~	3~~Z~~B0210	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-603
3~~Z~~BHP0401	~~70 (3)~~	~~.03~~	3~~Z~~B0402	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-615

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~BHP0402	~~70 (3)~~	~~.03~~	3~~2~~B0402	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-616
3~~2~~BHP0403	~~70 (3)~~	~~.03~~	3~~2~~B0402	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-617
3~~2~~BHP0404	~~70 (3)~~	~~.03~~	3~~2~~B0402	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-618
3~~2~~BHP0601	~~70 (3)~~	~~.03~~	3~~2~~B0805	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-619
3~~2~~BHP0602	~~70 (3)~~	~~.03~~	3~~2~~B0805	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-~~602~~ 620
3~~2~~BHP0603	~~70 (3)~~	~~.03~~	3~~2~~B0805	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-621
3~~2~~BHP0604	~~70 (3)~~	~~.03~~	3~~2~~B0805	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-622
3~~2~~BHP0701	~~70 (3)~~	~~.03~~	3~~2~~B0806	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-623
3~~2~~BHP0702	~~70 (3)~~	~~.03~~	3~~2~~B0806	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-624
3~~2~~BHP0703	~~70 (3)~~	~~.03~~	3~~2~~B0806	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-625
3~~2~~BHP0704	~~70 (3)~~	~~.03~~	3~~2~~B0806	~~1600 (6)~~	~~.06~~	Backup Pressurizer Heater E-626
3~~2~~BHP0501	~~70 (3)~~	~~.03~~	3~~2~~B0810	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-604
3~~2~~BHP0502	~~70 (3)~~	~~.03~~	3~~2~~B0810	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-605
3~~2~~BHP0503	~~70 (3)~~	~~.03~~	3~~2~~B0810	~~900 (6)~~	~~.06~~	Proportional Pressurizer Heater E-606
3~~2~~BHP0801	~~70 (3)~~	~~.03~~	3~~2~~B0602	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-627
3~~2~~BHP0802	~~70 (3)~~	~~.03~~	3~~2~~B0602	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-628
3~~2~~BHP0803	~~70 (3)~~	~~.03~~	3~~2~~B0602	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-639
3~~2~~BHP0804	~~70 (3)~~	~~.03~~	3~~2~~B0602	~~1000 (6)~~	~~.06~~	Backup Pressurizer Heater E-630
3~~2~~BY40	~~20 (3)~~	~~.02~~	3~~2~~BLP1013	~~15 (2)~~	~~.02~~	Cont. Bldg. Emer. A/C Unit E-399 (Motor Enclos. Htr.)
3~~2~~BY40	~~20 (3)~~	~~.02~~	3~~2~~BCP1014	~~15 (2)~~	~~.02~~	Cont. Bldg. Emer. A/C Unit E-401 (Motor Enclos. Htr.)
3~~2~~BZ32	~~35 (1)~~	~~.02~~	3~~2~~BLP1111	~~15 (2)~~	~~.02~~	Reactor Coolant Regen. Heat Exch. Isol. Valve TV-9267
3~~2~~BZ38	~~20 (3)~~	~~.02~~	3~~2~~BLP1112	~~15 (2)~~	~~.02~~	Containment Bldg. Emergency A/C Unit E-400
3~~2~~BZ38	~~20 (3)~~	~~.02~~	3~~2~~BLP1126	~~15 (2)~~	~~.02~~	Containment Bldg. Emergency A/C Unit E-403
3~~2~~Q01704	~~15 (2)~~	~~.03~~	3~~2~~Q017 (Main Breaker)	~~100 (3)~~	~~.02~~	Containment Reactor Cavity Cooling Fan ~~E-319~~ A-319 (Motor Enclosure Heater)
3~~2~~Q01706	~~15 (2)~~	~~.03~~	3~~2~~Q017 (Main Breaker)	~~100 (3)~~	~~.02~~	Containment Reactor Cavity Cooling Fan A-321 (Motor Enclosure Heater)

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~Q01724	~~20 (2)~~	~~.03~~	3~~2~~Q017 (Main Breaker)	~~100 (3)~~	~~.02~~	Containment Sump Inlet Flow 3~~2~~FT5799A/B, 3~~2~~FT5802A/B
3~~2~~Q02801	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	RCP P-001 (Motor Enclosure Heater)
3~~2~~Q02802	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	RCP P-004 (Motor Enclosure Heater)
3~~2~~Q02803	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	RCP P-002 (Motor Enclosure Heater)
3~~2~~Q02804	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	Containment Reactor Cavity Cooling Fan A-320 (Motor Enclosure Heater)
3~~2~~Q02805	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	RCP P-003 (Motor Enclosure Heater)
3~~2~~Q02808	~~15 (2)~~	~~.03~~	3~~2~~Q028 (Main Breaker)	~~100 (3)~~	~~.02~~	Containment Reactor Cavity Cooling Fan (Motor Enclosure Heater)
3~~2~~Q03904	~~15 (2)~~	~~.02~~	3~~2~~Q039 (Main Breaker)	~~100 (3)~~	~~.02~~	Dome Circulating Fan A-071 (Motor Enclosure Heater)
3~~2~~Q03906	~~15 (2)~~	~~.02~~	3~~2~~Q039 (Main Breaker)	~~100 (3)~~	~~.02~~	Dome Circulating Fan A-074 (Motor Enclosure Heater)
3~~2~~Q04104	~~15 (2)~~	~~.02~~	3~~2~~Q041 (Main Breaker)	~~100 (3)~~	~~.02~~	Standby Dome Circulating Fan A-072 (Motor Enclosure Heater)
3~~2~~Q04106	~~15 (2)~~	~~.02~~	3~~2~~Q041 (Main Breaker)	~~100 (3)~~	~~.02~~	Standby Dome Circulating Fan A-073
3~~2~~D5P108	~~70 (2)~~	~~.02~~	3~~2~~D503	~~1500 (6)~~	~~.03~~	Panel 3~~2~~LP4 Emergency Lighting
3~~2~~D5P109	~~70 (2)~~	~~.02~~	3~~2~~D503	~~1500 (6)~~	~~.03~~	Panel 3~~2~~LP11 Emergency Lighting
3~~2~~D5P118	~~100 (2)~~	~~.02~~	3~~2~~D503	~~1500 (6)~~	~~.03~~	Panel 3LP16 Emergency Lighting
3~~2~~A0101	~~(7)~~	~~(7)~~	3~~2~~A0102	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-001
			3~~2~~A0104	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-001
			3~~2~~A0105	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-001

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
3~~2~~A0103	~~(7)~~	~~(7)~~	3~~2~~A0102	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-004
			3~~2~~A0104	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-004
			3~~2~~A0105	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-004
3~~2~~A0201	~~(7)~~	~~(7)~~	3~~2~~A0202	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-002
			3~~2~~A0204	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-002
			3~~2~~A0205	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-002
3~~2~~A0203	~~(7)~~	~~(7)~~	3~~2~~A0202	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-003
			3~~2~~A0204	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-003
			3~~2~~A0205	~~24,800~~	~~1.5~~	Reactor Coolant Pump P-003
CEA04	~~100 (8)~~	~~.03~~	CB3001	~~400 (8)~~	~~.03~~	CEA4
CEA05	~~100 (8)~~	~~.03~~	CB3001	~~400 (8)~~	~~.03~~	CEA5
CEA06	~~100 (8)~~	~~.03~~	CB3001	~~400 (8)~~	~~.03~~	CEA6
CEA07	~~100 (8)~~	~~.03~~	CB3001	~~400 (8)~~	~~.03~~	CEA7
CEA08	~~100 (8)~~	~~.03~~	CB3002	~~400 (8)~~	~~.03~~	CEA8
CEA09	~~100 (8)~~	~~.03~~	CB3002	~~400 (8)~~	~~.03~~	CEA9
CEA10	~~100 (8)~~	~~.03~~	CB3002	~~400 (8)~~	~~.03~~	CEA10
CEA11	~~100 (8)~~	~~.03~~	CB3002	~~400 (8)~~	~~.03~~	CEA11
CEA12	~~100 (8)~~	~~.03~~	CB3003	~~400 (8)~~	~~.03~~	CEA12
CEA14	~~100 (8)~~	~~.03~~	CB3003	~~400 (8)~~	~~.03~~	CEA14
CEA16	~~100 (8)~~	~~.03~~	CB3003	~~400 (8)~~	~~.03~~	CEA16
CEA18	~~100 (8)~~	~~.03~~	CB3003	~~400 (8)~~	~~.03~~	CEA16
CEA13	~~100 (8)~~	~~.03~~	CB3004	~~400 (8)~~	~~.03~~	CEA13
CEA15	~~100 (8)~~	~~.03~~	CB3004	~~400 (8)~~	~~.03~~	CEA15
CEA17	~~100 (8)~~	~~.03~~	CB3004	~~400 (8)~~	~~.03~~	CEA17
CEA19	~~100 (8)~~	~~.03~~	CB3004	~~400 (8)~~	~~.03~~	CEA19

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
CEA20	~~100 (8)~~	~~.03~~	CB3005	~~400 (8)~~	~~.03~~	CEA20
CEA21	~~100 (8)~~	~~.03~~	CB3005	~~400 (8)~~	~~.03~~	CEA21
CEA22	~~100 (8)~~	~~.03~~	CB3005	~~400 (8)~~	~~.03~~	CEA22
CEA23	~~100 (8)~~	~~.03~~	CB3005	~~400 (8)~~	~~.03~~	CEA23
CEA24	~~100 (8)~~	~~.03~~	CB3006	~~400 (8)~~	~~.03~~	CEA24
CEA25	~~100 (8)~~	~~.03~~	CB3006	~~400 (8)~~	~~.03~~	CEA25
CEA26	~~100 (8)~~	~~.03~~	CB3006	~~400 (8)~~	~~.03~~	CEA26
CEA27	~~100 (8)~~	~~.03~~	CB3006	~~400 (8)~~	~~.03~~	CEA27
CEA28	~~100 (8)~~	~~.03~~	CB3007	~~400 (8)~~	~~.03~~	CEA28
CEA30	~~100 (8)~~	~~.03~~	CB3007	~~400 (8)~~	~~.03~~	CEA30
CEA32	~~100 (8)~~	~~.03~~	CB3007	~~400 (8)~~	.03	CEA32
CEA34	~~100 (8)~~	~~.03~~	CB3007	~~400 (8)~~	~~.03~~	CEA34
CEA29	~~100 (8)~~	~~.03~~	CB3008	~~400 (8)~~	~~.03~~	CEA29
CEA31	~~100 (8)~~	~~.03~~	CB3008	~~400 (8)~~	~~.03~~	CEA31
CEA33	~~100 (8)~~	~~.03~~	CB3008	~~400 (8)~~	~~.03~~	CEA33
CEA35	~~100 (8)~~	~~.03~~	CB3008	~~400 (8)~~	~~.03~~	CEA35
CEA36	~~100 (8)~~	~~.03~~	CB3009	~~400 (8)~~	~~.03~~	CEA36
CEA38	~~100 (8)~~	~~.03~~	CB3009	~~400 (8)~~	~~.03~~	CEA38
CEA40	~~100 (8)~~	~~.03~~	CB3009	~~400 (8)~~	~~.03~~	CEA40
CEA42	~~100 (8)~~	~~.03~~	CB3009	~~400 (8)~~	~~.03~~	CEA42
CEA37	~~100 (8)~~	~~.03~~	CB3010	~~400 (8)~~	~~.03~~	CEA37
CEA39	~~100 (8)~~	~~.03~~	CB3010	~~400 (8)~~	~~.03~~	CEA39
CEA41	~~100 (8)~~	~~.03~~	CB3010	~~400 (8)~~	~~.03~~	CEA41
CEA43	~~100 (8)~~	~~.03~~	CB3010	~~400 (8)~~	.03	CEA43

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
CEA44	~~100 (8)~~	~~.03~~	CB3023	~~400 (8)~~	~~.03~~	CEA44
CEA45	~~100 (8)~~	~~.03~~	CB3023	~~400 (8)~~	~~.03~~	CEA45
CEA46	~~100 (8)~~	~~.03~~	CB3023	~~400 (8)~~	~~.03~~	CEA46
CEA47	~~100 (8)~~	~~.03~~	CB3023	~~400 (8)~~	~~.03~~	CEA47
CEA48	~~100 (8)~~	~~.03~~	CB3024	~~400 (8)~~	~~.03~~	CEA48
CEA50	~~100 (8)~~	~~.03~~	CB3024	~~400 (8)~~	~~.03~~	CEA50
CEA52	~~100 (8)~~	~~.03~~	CB3024	~~400 (8)~~	~~.03~~	CEA52
CEA54	~~100 (8)~~	~~.03~~	CB3024	~~400 (8)~~	~~.03~~	CEA54
CEA49	~~100 (8)~~	~~.03~~	CB3011	~~400 (8)~~	~~.03~~	CEA49
CEA51	~~100 (8)~~	~~.03~~	CB3011	~~400 (8)~~	~~.03~~	CEA51
CEA53	~~100 (8)~~	~~.03~~	CB3011	~~400 (8)~~	~~.03~~	CEA53
CEA55	~~100 (8)~~	~~.03~~	CB3011	~~400 (8)~~	~~.03~~	CEA55
CEA56	~~100 (8)~~	~~.03~~	CB3012	~~400 (8)~~	~~.03~~	CEA56
CEA57	~~100 (8)~~	~~.03~~	CB3012	~~400 (8)~~	~~.03~~	CEA57
CEA58	~~100 (8)~~	~~.03~~	CB3012	~~400 (8)~~	~~.03~~	CEA58
CEA59	~~100 (8)~~	~~.03~~	CB3012	~~400 (8)~~	~~.03~~	CEA59
CEA60	~~100 (8)~~	~~.03~~	CB3013	~~400 (8)~~	~~.03~~	CEA60
CEA62	~~100 (8)~~	~~.03~~	CB3013	~~400 (8)~~	~~.03~~	CEA62
CEA64	~~100 (8)~~	~~.03~~	CB3013	~~400 (8)~~	~~.03~~	CEA64
CEA66	~~100 (8)~~	~~.03~~	CB3013	~~400 (8)~~	~~.03~~	CEA66
CEA61	~~100 (8)~~	~~.03~~	CB3014	~~400 (8)~~	~~.03~~	CEA61
CEA63	~~100 (8)~~	~~.03~~	CB3014	~~400 (8)~~	~~.03~~	CEA63
CEA65	~~100 (8)~~	~~.03~~	CB3014	~~400 (8)~~	~~.03~~	CEA65
CEA67	~~100 (8)~~	~~.03~~	CB3014	~~400 (8)~~	~~.03~~	CEA67

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CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
CEA68	~~100 (8)~~	~~.03~~	CB3015	~~400 (8)~~	~~.03~~	CEA68
CEA71	~~100 (8)~~	~~.03~~	CB3015	~~400 (8)~~	~~.03~~	CEA71
CEA74	~~100 (8)~~	~~.03~~	CB3015	~~400 (8)~~	~~.03~~	CEA74
CEA77	~~100 (8)~~	~~.03~~	CB3015	~~400 (8)~~	~~.03~~	CEA77
CEA69	~~100 (8)~~	~~.03~~	CB3016	~~400 (8)~~	~~.03~~	CEA69
CEA72	~~100 (8)~~	~~.03~~	CB3016	~~400 (8)~~	~~.03~~	CEA72
CEA75	~~100 (8)~~	~~.03~~	CB3016	~~400 (8)~~	~~.03~~	CEA75
CEA78	~~100 (8)~~	~~.03~~	CB3016	~~400 (8)~~	~~.03~~	CEA78
CEA70	~~100 (8)~~	~~.03~~	CB3017	~~400 (8)~~	~~.03~~	CEA70
CEA73	~~100 (8)~~	~~.03~~	CB3017	~~400 (8)~~	~~.03~~	CEA73
CEA76	~~100 (8)~~	~~.03~~	CB3017	~~400 (8)~~	~~.03~~	CEA76
CEA79	~~100 (8)~~	~~.03~~	CB3017	~~400 (8)~~	~~.03~~	CEA79
CEA80	~~100 (8)~~	~~.03~~	CB3018	~~400 (8)~~	~~.03~~	CEA80
CEA82	~~100 (8)~~	~~.03~~	CB3018	~~400 (8)~~	~~.03~~	CEA82
CEA84	~~100 (8)~~	~~.03~~	CB3018	~~400 (8)~~	~~.03~~	CEA84
CEA86	~~100 (8)~~	~~.03~~	CB3018	~~400 (8)~~	~~.03~~	CEA86
CEA81	~~100 (8)~~	~~.03~~	CB3019	~~400 (8)~~	~~.03~~	CEA81
CEA83	~~100 (8)~~	~~.03~~	CB3019	~~400 (8)~~	~~.03~~	CEA83
CEA85	~~100 (8)~~	~~.03~~	CB3019	~~400 (8)~~	~~.03~~	CEA85
CEA87	~~100 (8)~~	~~.03~~	CB3019	~~400 (8)~~	~~.03~~	CEA87
CEA88	~~100 (8)~~	~~.03~~	CB3020	~~400 (8)~~	~~.03~~	CEA88
CEA89	~~100 (8)~~	~~.03~~	CB3020	~~400 (8)~~	~~.03~~	CEA89
CEA90	~~100 (8)~~	~~.03~~	CB3020	~~400 (8)~~	~~.03~~	CEA90
CEA91	~~100 (8)~~	~~.03~~	CB3020	~~400 (8)~~	~~.03~~	CEA91

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device			Backup Device			
Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Number	~~Trip Setpoint (amperes)~~	~~Resp. Time (sec)~~	Service Description
CEA02	~~100 (8)~~	~~.03~~	CB3025	~~400 (8)~~	~~.03~~	CEA2
CEA03	~~100 (8)~~	~~.03~~	CB3025	~~400 (8)~~	~~.03~~	CEA3
CEA01	~~100 (8)~~	~~.03~~	CB3026	~~400 (8)~~	~~.03~~	CEA1

~~1. Instantaneous setpoint~~

~~2. Thermal magnetic breaker, response time is for 50 times of rated current.~~

~~3. Thermal magnetic breaker, response time is for 25 times of rated current.~~

~~4. Thermal magnetic breaker, response time is for 15 times of rated current.~~

~~5. Thermal magnetic breaker, response time is for 10 times of rated current.~~

~~6. Instantaneous setpoint, response time is for 2 times of instantaneous setpoint.~~

~~7. Use differential relay, will cause breaker to trip in 0.03 second.~~

~~8. No instantaneous setpoint, response time is for 10 times of rated current.~~

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MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 18 months,

b. Following maintenance on the valve motor starter, and

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

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TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION BYPASS DEVICES

Permanently Bypassed

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

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TABLE 3.8-2 (Continued)

.LVE NUMBER	FUNCTION	
HV-9302	Containment emergency sump outlet	~~Permanently Bypassed~~
HV-9304	Containment emergency sump outlet	~~Permanently Bypassed~~
HV-6211	CCW NON-CRITICAL Loop to containment - isolation valve	~~Permanently Bypassed~~
HV-6368	CCW LOOP B to emergency cooling unit E 400	~~Permanently Bypassed~~
HV-6369	CCW from emergency cooling unit E400 to loop B	~~Permanently Bypassed~~
HV-6216	CCW NON-CRITICAL LOOP from containment - isolation valve	~~Permanently Bypassed~~
HV-6372	CCW Loop B to emergency cooling unit E402	~~Permanently Bypassed~~
HV-6373	CCW Loop B from emergency cooling unit E402	~~Permanently Bypassed~~
HV-9900	Containment normal cooling supply isolation	~~Permanently Bypassed~~
HV-9971	Containment normal cooling return isolation	~~Permanently Bypassed~~
LV-0227C	Boric Acid makeup control	~~Permanently Bypassed~~
-4713	Aux. F.W. P141 Discharge to steam generator E089 control valve	~~Permanently Bypassed~~
HV-9334	SI tank drain header to refueling water tank - cont. isol.	~~Permanently Bypassed~~
HV-9350	SI tank T007 to reactor coolant loop 1B	~~Permanently Bypassed~~
HV-9360	SI tank T009 to reactor coolant loop 2A	~~Permanently Bypassed~~
HV-9325	LPSI Header to reactor coolant loop 1B	~~Permanently Bypassed~~
HV-9328	LPSI Header to reactor coolant loop ~~2B~~ 2A	~~Permanently Bypassed~~
HV-9201	Aux. spray to pressurize	~~Permanently Bypassed~~
HV-9327	HPSI Header #1 to reactor coolant loop 1B	~~Permanently Bypassed~~
HV-9330	HPSI Header #1 to reactor coolant loop 2A	~~Permanently Bypassed~~
HV-6223	CCW Non-Crit Loop Containment inlet isolation	~~Permanently Bypassed~~
HV-9324	HPSI header #1 to reactor coolant loop 1A	~~Permanently Bypassed~~
HV-9333	HPSI header #1 to reactor coolant loop 2B	~~Permanently Bypassed~~
HV-9337	Shutdown coolant flow from reactor coolant loop 2	~~Permanently Bypassed~~
V-9377	" " " " " " " "	
HV-9378	" " " " " " " "	

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TABLE 3.8-2 (Continued)

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

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TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	
HV-5686	Firewater to containment isolation	~~Permanently Bypassed~~
HV-0227B	Volume control tank (T077) drain return	~~Permanently Bypassed~~
HV-9240	Boric acid makeup tank ~~T072~~ (T071) to charging pump suction suction	~~Permanently Bypassed~~
HV-9235	Boric acid makeup tank (T072) to charging pump suction	
HV-9336	Shutdown cooling flow to LPSI pump suction	~~Permanently Bypassed~~
HV-9359	Shutdown cooling warm up valve	~~Permanently Bypassed~~
HV-9301	Refueling water tank west (T006) outlet	~~Permanently Bypassed~~
HV-6495	Saltwater from CCW HX E002	~~Permanently Bypassed~~
TV-9267	~~Reactor coolant regenerative HX~~ Letdown Line Containment isolation valve	~~Permanently Bypassed~~
HV-9434	HPSI header #2 to reactor coolant loop 1 hot leg	~~Permanently Bypassed~~
HV-8151	~~Reactor aux.~~ shutdown cooling HX ~~outlet~~ inlet isolation valve	~~Permanently Bypassed~~
HV-8153	~~Reactor aux.~~ shutdown cooling HX ~~outlet~~ inlet isolation valve	~~Permanently Bypassed~~
HV-4712	Aux F.W. Pump P504 Discharge to Steam gen. control	~~Permanently Bypassed~~

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less,

b. A boron concentration of greater than or equal to 1720 ppm,

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and 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 K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 1720 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

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3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification 4.6.3.2.

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except four finger CEAs.

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3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the fuel handling machine in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Fuel handling machine interlocks and physical stops which prevent fuel handling machine travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to fuel handling machine use and at least once per 7 days thereafter during fuel handling machine operation.

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3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling train shall be OPERABLE and in operation.#

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling train OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling train to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

#The shutdown cooling train may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling trains shall be OPERABLE and at least one shutdown cooling train shall be in operation.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required shutdown cooling trains to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange 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. Close all containment penetrations providing direct access from the containment atmophere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

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3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on a high radiation test signal.

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3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

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3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling building post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel handling building post-accident cleanup filter system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building post-accident cleanup filter systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, 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.

b. 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:

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SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm $\pm$ 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter 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 12925 cfm $\pm$ 10%.

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 12925 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. 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.

d. 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 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm $\pm$ 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.4 $\pm$ 1.5 kw for E464, 32.3 $\pm$ 1.7 kw for E465, and 3.8 $\pm$ 0.2 kw for E652 when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

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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: MODE 2.

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.

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3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 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, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 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, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 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, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

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3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification ~~3.4.1~~ 3.4.1.1 and the noted requirements of Table 2.2-1 and Table 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

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

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

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.10.4 CENTER CEA MISALIGNMENT

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LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

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~~SPECIAL TEST EXCEPTIONS~~

~~3/4.10.5 RADIATION MONITORING/SAMPLING~~

~~LIMITING CONDITION FOR OPERATION~~

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

APPLICABILITY: As shown in Table 3.10-1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.10-1

RADIATION MONITORING/SAMPLING EXCEPTIONS

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

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

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

a.	Main Steam Line Area Monitors	3~~2~~RT-7874A1 ~~2RT-7847B1~~ 3RT-7874B1 3~~2~~RT-7875A1 3~~2~~RT-7875B1
b.	Condenser Evacuation System - Wide Range Monitor	3~~2~~RT-7870-1
c.	Purge/Vent Stack Monitors - Wide Range	2RT-7865-1 3RT-7865-1
d.	Plant Vent Stack	Flow Rate Monitor
e.	Containment Purge	Flow Rate Monitor
f.	Condenser Evacuation System	Iodine Sampler Particulate Sampler Flow Rate Monitor

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

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TABLE 3.10-1 (Continued)

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

   a. Radwaste Discharge Line Monitor 2/3 RT-7813

   b. Blowdown Neutralization Sump Monitor 3~~2~~RT-7817

   c. Turbine Building Sump Monitor 3~~2~~RT-7821

5. Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitors 3~~2~~RT-7804-1 or ~~and~~ 3~~2~~RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3~~2~~RT-7804-1 and associated sampling media shall perform the above required functions.

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~~SPECIAL TEST EXCEPTIONS~~

~~MINIMUM TEMPERATURE FOR CRITICALITY~~

~~LIMITING CONDITION FOR OPERATION~~

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

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

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

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

~~APPLICABILITY: MODE 2.*~~

~~ACTION:~~

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

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

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.6.1 The Reactor Coolant System temperature and pressure relationship and the minimum temperature for criticality shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.~~

~~4.10.6.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.~~

~~4.10.6.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 320°F at least once per hour.~~

~~4.10.6.4 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.~~

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

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	$5x10^{-7}$
1. Primary Plant Makeup Storage Tanks			I-131	$1x10^{-6}$
2. Radwaste Primary Tanks	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	$1x10^{-5}$
3. Radwaste Secondary Tanks	P Each Batch	M Composite[b]	H-3	$1x10^{-5}$
4. Miscellaneous Waste Condensate Monitor Tanks			Gross Alpha	$1x10^{-7}$
. Neutralization Sump	P Each Batch	Q Composite[b]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[e,#]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	$5x10^{-7}$
1. Steam Generator Blowdown			I-131	$1x10^{-6}$
2. Turbine Building Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
3. Miscellaneous Waste Evaporator Condensate*	D Grab Sample	M Composite[c]	H-3	$1x10^{-5}$
4. Salt Water Discharge From Component Cooling Heat Exchanger			Gross Alpha	$1x10^{-7}$
	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

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TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

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

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 determined by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typcial 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).

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TABLE 4.11-1 (Continued)

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be throughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ODCM, to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

* Sampling of this flow is not required if, at least once per 31 days, condensate monitor tank bypass valve, SA 1415-2½"-200, is verified locked shut.

\# Administrative controls shall provide for composite sampling of the continuous releases per note b vice note c until January 1, 1983. Continuous proportional sampling shall be in accordance with note c from January 1, 1983 and all times subsequent as required by Table 4.11-1.

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DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, 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 cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.1.2

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

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LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (see Figure 5.1-4) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, 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 includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

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

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 15 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

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*Per reactor unit

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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, 3.0.4 and 6.9.1.13b 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.

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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.

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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]	1×10^{-4}
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters	1×10^{-4}
			H-3	1×10^{-6}
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emmitters	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
2. Plant Vent Stack	W[b,e]	W[b]	H-3	1×10^{-6}
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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TABLE NOTATION

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

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

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

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 <u>a priori</u> (before the fact) limit representing the capability of the measurement system and not as <u>a posteriori</u> (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, <u>HASL-300</u> (revised annually).

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

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

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

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3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.1.13. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With fresh leafy vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 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 cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

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SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
1. AIRBORNE Radioiodine and Particulates	Samples from at least 5 locations 3 samples from offsite locations (in different sectors) of the highest calculated annual average groundlevel D/Q. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q. 1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction[c]	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.[d]	Radioiodine cartridge. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION[e]	At least 30 locations including an inner ring of stations in the general area of the site boundary and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	At least once per 92 days.	Gamma dose. At least once per 92 days.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
3. WATERBORNE			
a. Ocean	4 Locations	At least once per month and composited[f] quarterly	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days.
b. Drinking	2 Locations	Monthly at each location.	Gamma isotopic and tritium analyses of each sample.
c. Sediment from Shoreline	4 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
d. Ocean Bottom Sediments	5 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
b. Local Crops	2 Locations	Representative vegetables, normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semiannually and I-131 analysis for leafy crops.
~~5. Local Vegetation~~	~~3 Locations[a]~~	~~Monthly~~	~~Monthly gamma isotopic analysis.~~

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TABLE 3.12-1 (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

b. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

c. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

d. Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

e. Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposed of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

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

~~g. 2 samples should be from the nearest offsite locations of highest calculated annual average ground-level D/Q. The third sample should be of similar vegetation characteristics and grown 15-30 km distant in the least prevalent wind direction.~~

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TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals (pCi/Kg, wet)	Local Crops (pCi/Kg, wet)
H-3	2×10^4 (a)			
Mn-54	1×10^3		3×10^4	
Fe-59	4×10^2		1×10^4	
Co-58	1×10^3		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^4	
Zr-Nb-95	4×10^2			
I-131	2	0.9		1×10^2
Cs-134	30	10	1×10^3	1×10^3
Cs-137	50	20	2×10^3	2×10^3
Ba-La-140	2×10^2			

(a) For drinking water samples. This is 40 CFR Part 141 value.

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TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Marine Animals (pCi/kg, wet)	Local Crops (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}			
H-3	2000				
Mn-54	15		130		
Fe-59	30		260		
Co-58, 60	15		130		
Zn-65	30		260		
Zr-95	30				
Nb-95	15				
I-131	1[b]	7×10^{-2}		60	
Cs-134	15	5×10^{-2}	130	60	150
Cs-137	18	6×10^{-2}	150	80	180
Ba-140	60				
La-140	15				

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

2.22 is the number of transformation per minute per picocurie,

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 sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculation.

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TABLE NOTATION

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

b. LLD for drinking water.

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

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*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).

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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.

*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.

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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.

*Interlaboratory comparison program not required prior to first exceeding 5% RATED THERMAL POWER or July 1, 1982, whichever occurs first.

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BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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NOTE

The BASES in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not considered a part of these Technical Specifications.

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BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

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

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

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

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

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

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

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4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 5.15% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a 2% delta k/k shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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BASES

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 53,500 gallons of 1720 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5465 gallons of 1720 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

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BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the limiting physical characteristics of the tanks and includes allowances for water not available because of discharge line location and other physical characteristics (RWST above the ECCS suction connection in lieu of the CVCS suction connection).

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.

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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.

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MOVABLE CONTROL ASSEMBLIES (Continued)

The establishment of LSSS and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accomodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accomodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that 1) the minimum SHUTDOWN MARGIN is maintained, and 2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The Part Length CEA Insertion Limits of Specification 3.1.3.7 ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and DNB considerations do not occur as a result of a part length CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation.

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BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of 13.9 kw/ft are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate includes appropriate penalty factors which provide, with a 95/95 probability/ confidence level, that the maximum linear heat rate calculated by COLSS is conservative with respect to the actual maximum linear heat rate existing in the core. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, axial densification, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs assuming minimum core power of 20% RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPCs.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos (\theta - \theta_o)$$

where:

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

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3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation System instrumentation and bypasses ensure that 1) the associated Engineered Safety Features Actuation System action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the reactor protective and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

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3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING ALARM INSTRUMENTATION

The OPERABILITY of the radiation monitoring alarm channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels; 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation in Panel L042 ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

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

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

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within two hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at two and three hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

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3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

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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 ~~above 1.19~~ greater than or equal to 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.

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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.

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STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of Figure 4.4-1. Figure 4.4-1 was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. These results are controlling items in allowable tube wall thinning for tube rows 92 to 147. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENIIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 GPM leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

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CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

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SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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 or cooldown rates of up to 60°F per hour. 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 at 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.

TAB[illegible] 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	Temperature of Charpy V-Notch @ 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

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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 pump and its injection 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.

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3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The NaOH added to the Containment Spray, via the Spray Chemical Addition pumps, minimizes the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The NaOH additive results in post-LOCA sump pH of between 8.0 and 10.0 at the end of the NaOH injection period.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. The limit on maximum boron concentration is to ensure that boron does not precipitable in the core following LOCA. The limit on RWST solution temperature is to ensure that the assumptions used in the LOCA analyses remain valid.

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the limiting physical characteristics of the tanks and includes allowances for water not available because of discharge line location and other physical characteristics (RWST above the ECCS suction connection in lieu of the CVCS suction connection).

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.

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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 and 2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA or steam line break conditions.

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.

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3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the 8-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration 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 contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

The 5 year Surveillance testing is intended to verify that no crystallization of the NaOH or other obstruction has occurred in the piping from the spray additive tank ot the suction of the containment spray pumps.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The containment cooling system and the containment spray system are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

The containment dome air circulators are provided to ensure adequate mixing of the containment atmosphere following a LOCA. The mixing action of the containment dome air circulators combined with the containment spray system and the containment emergency fan coolers will prevent localized accumulations of hydrogen from exceeding the flammable limit.

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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).

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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 tanks with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 24 hours with steam discharge to atmosphere with concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the control room emergency air cleanup system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix A, 10 CFR 50.

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.

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3/4.7.6 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

Snubbers are classified and grouped by design and manufacturer, but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip, and 100 kip capacity manufactured by company "A" are of the same type. The same design mechanical snubber manufactured by company "B", for purposes of this Specification, would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability, one of two sampling and acceptance criteria methods are used:

1) Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure or,

2) Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

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SNUBBERS (Continued)

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

3/4.7.7 SEALED SOURCE CONTAMINATION

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

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

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

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

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FIRE SUPPRESSION SYSTEMS (Continued)

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

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3/4.7.9 FIRE RATED ASSEMBLIES

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

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

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BASES

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

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

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

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

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

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

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

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

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

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal electrolyte limits for each designated pilot cell and each connected cell for ~~electrolyte~~ level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

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

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The thermal overload protection contact integral with the motor starter of each valve listed in Table 3.8-2 is permanently bypassed in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", November, 1975.

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BASES

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) refueling machine will be used for movement of CEAs and fuel assemblies, 2) each machine has sufficient load capacity to lift a CEA or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

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.

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BASES

3/4.10.1 SHUTDOWN MARGIN

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

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

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

3/4.10.3 REACTOR COOLANT LOOPS

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

3/4.10.4 CENTER CEA MISALIGNMENT

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

~~3/4.10.5 RADIATION MONITORING/SAMPLING~~

delete

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

The containment airborne monitors and associated sampling media test exception is required to allow for operation prior to and during installation of upgraded monitors/media. Adequate monitoring is provided until and subsequent to the completion of the upgraded installation. Extensive containment air mixing during high volume purge (MODES 5 and 6) occurs as a result of containment HVAC and fans resulting in representative air monitoring via 2RT-7804-1. During low volume purge operations (MODES 1, 2, 3 and 4) 32RT-7804-1 provides representative indication of purged air due to its location in the immediate vicinity of ~~the low volume purge exhaust.~~

either or 3RT-7807-2.

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~~BASES~~

~~3/4.10.6 MINIMUM TEMPERATURE FOR CRITICALITY~~

delete

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

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

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BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

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

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

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BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

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

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 OUTSIDE TEMPORARY TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

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3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109,

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"Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive materials in particulate form and tritium are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include injection of dilutants to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected. An individual is not considered a member of the public during any period in which he/she is engaged in carring out any operation which is part of the nuclear fuel cycle.

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BASES

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (25 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

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SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 150 feet.

b. Nominal inside height = 172 feet.

c. Minimum thickness of concrete walls = 4 1/3 feet.

d. Minimum thickness of concrete roof = 3 3/4 feet.

e. Minimum thickness of concrete floor pad = 9 feet.

f. Nominal thickness of steel liner = 1/4 inches.

g. Net free volume = 2,305,000 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 300°F.

Use same figure

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FIGURE 5.1-1
EXCLUSION AREA
SAN ONOFRE - UNIT 3

LOW POPULATION ZONE
FIGURE 5.1-2
SAN ONOFRE - UNIT 3

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SITE BOUNDARY FOR GASEOUS EFFLUENTS

FIGURE 5.1-3

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SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1807 grams uranium. The initial core loading shall have a maximum enrichment of 2. 91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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5.3 REACTOR CORE

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a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

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VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 + 600/-0 cubic feet at a nominal T_{avg} of 582.1°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 0.014 delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 12.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in alternate rows and columns in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 60'6".

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 800 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates $\leq$ 100°F/hr.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 545°F; cooldown cycle - T_{avg} from $\geq$ 545°F to $\leq$ 200°F.
	500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/hr.	Heatup cycle - Pressurizer temperature from < 200°F to > 653°F; cooldown $\geq$ 653°F to $\leq$ 200°F
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature in accordance with Specification 3.4.8.
	200 leak testing cycles.	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for critically.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	480 cycles (in any combination) of reactor trip, turbine trip with delayed reactor trip, or complete loss of forced reactor coolant flow.	Trip from 100% of RATED THERMAL power; turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip; simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	2 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
150 - 200	50,000		
201 - 300	7,000		
301 - 400	2,000		
401 - 500	1,000		
501 - 600	800		
		$\Sigma N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles in ΔT range indicated.

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TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculational Method:

1. The spray cycle is defined as the opening and closing of a spray valve, either by main spray or auxiliary spray.

2. If the difference between the pressurizer water temperature and the spray water temperature exceeds 150°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor is calculated as follows:

   A. Fill in Column "N" above from plant records.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.

4. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 150°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that that the nozzle remains acceptable for additional service prior to removing this restriction.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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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 ~~Watch Engineer~~ (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. X

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 ~~Watch Engineer~~ 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.

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UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of individuals of the nuclear power plant operating staff who are responsible for manipulating plant controls or for adjusting on-line systems and equipment affecting plant safety which would have an immediate impact on public health and safety.

Enough plant operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime. The objective is to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a) An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period (all excluding shift turnover time).

c) A break of at least eight hours should be allowed between work periods (including shift turnover time).

d) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Station Manager, his deputy, the Manager, Operations or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime will be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not be assigned. Routine deviation from the above guidelines is not authorized.

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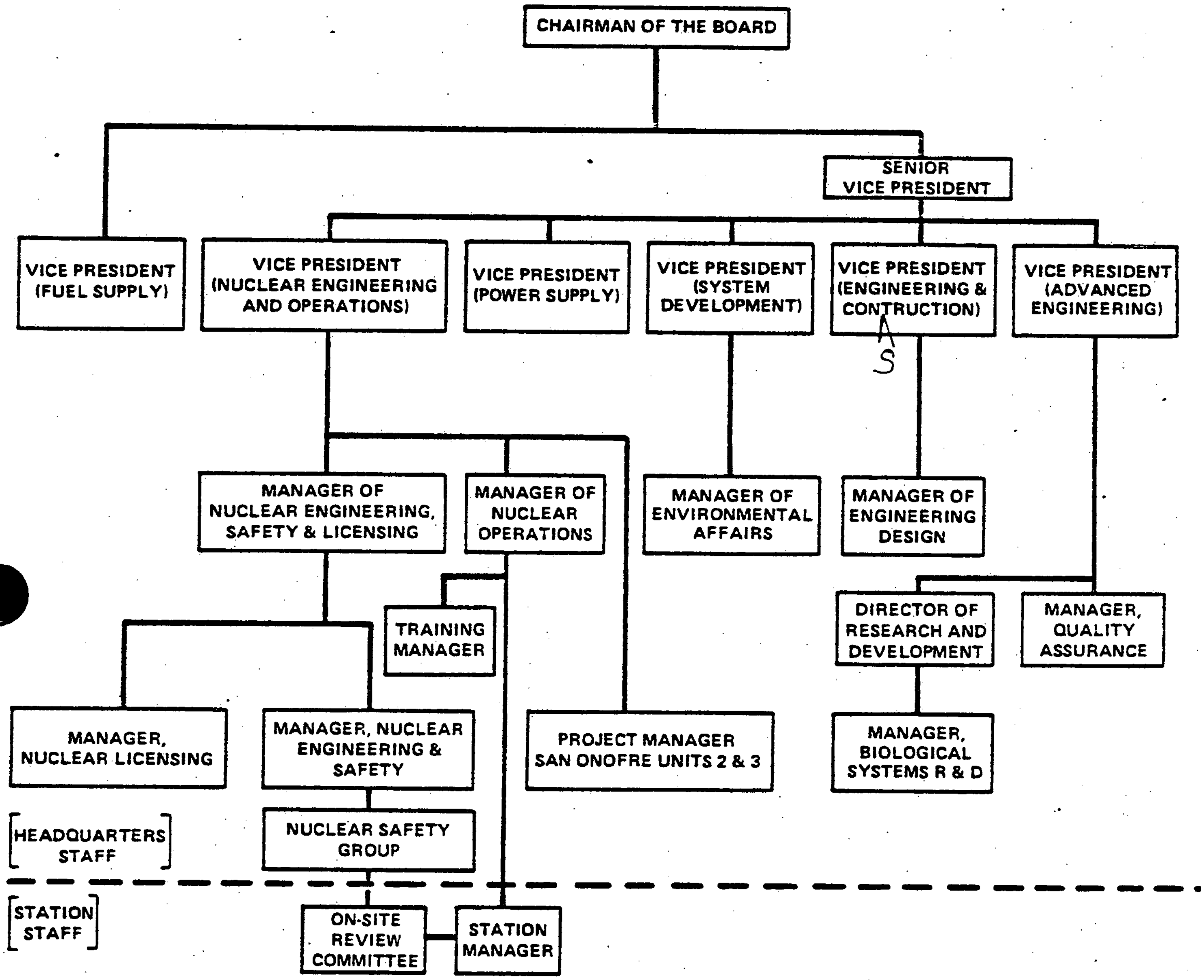

Figure 6.2-1
OFFSITE ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION – UNIT ~~2~~ 3

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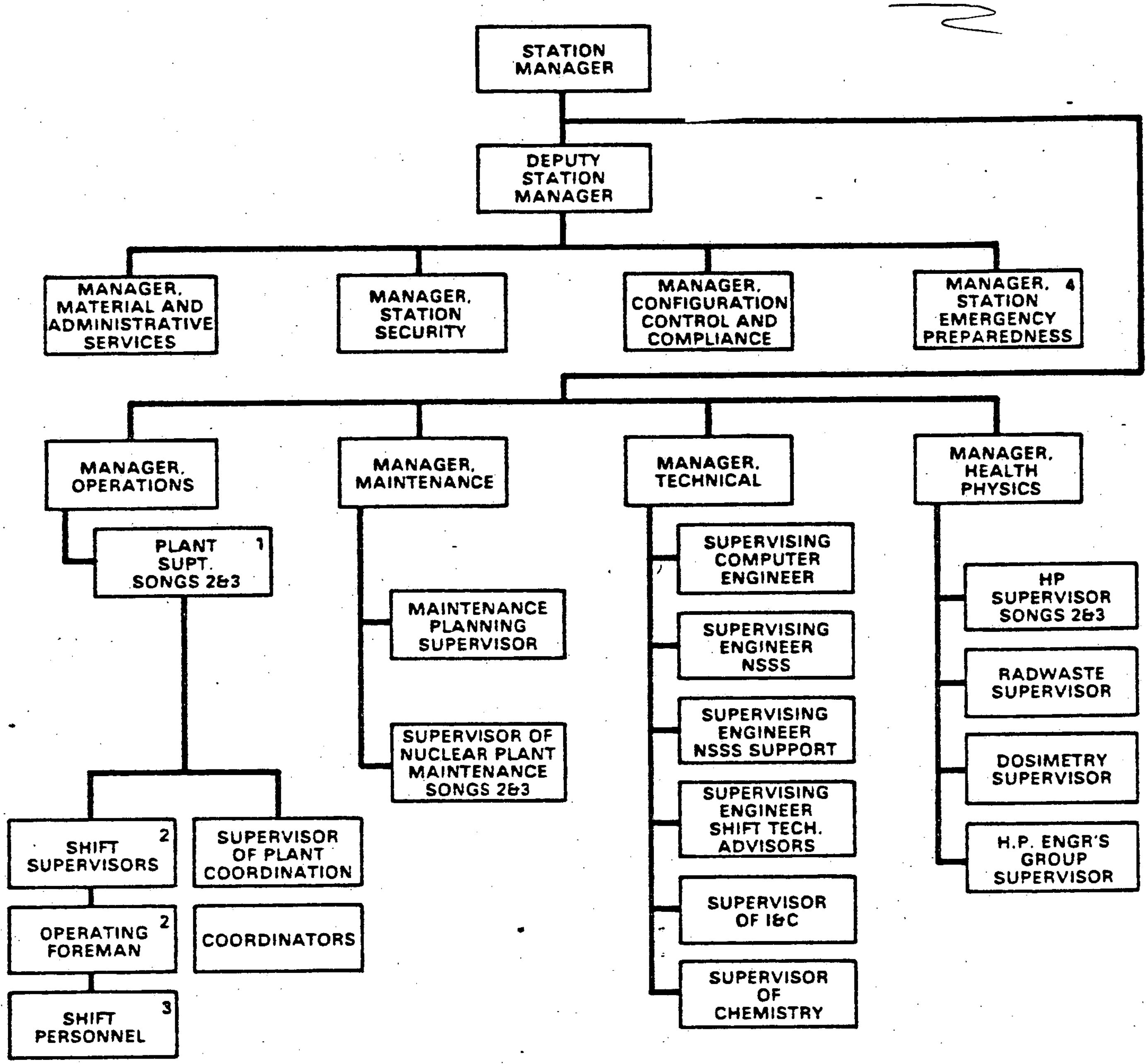

1. AT TIME OF APPOINTMENT TO THE POSITION, SENIOR REACTOR OPERATOR LICENSE REQUIRED.
2. SENIOR REACTOR LICENSE REQUIRED.
3. CONTROL AND ASSISTANT CONTROL OPERATORS ARE HOLDERS OF REACTOR OPERATOR LICENSES.
4. INCLUDES FIRE PROTECTION.

Figure 6.2-2 Unit Organization, San Onofre Nuclear Generting Station-Unit ~~2~~ 3

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Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
~~WE~~ SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

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

Except for the Shift Supervisor ~~Watch Engineer~~, 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 ~~Watch Engineer~~ 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 ~~Watch Engineer~~ 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.

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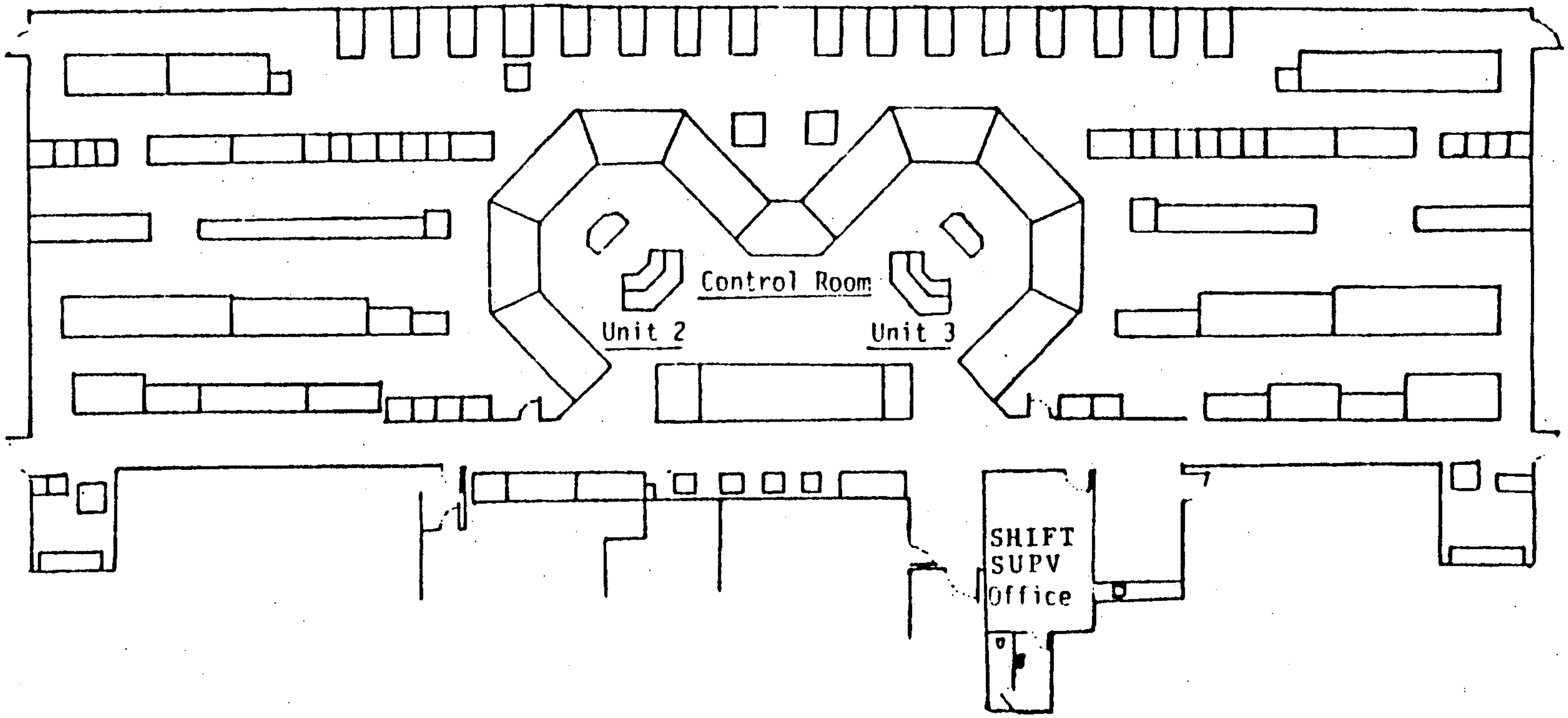

figure 6.2-3

CONTROL ROOM AREA

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT ~~2~~ 3

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

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

COMPOSITION

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

RESPONSIBILITIES

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

AUTHORITY

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

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the ~~Watch~~ Shift ~~Engineer~~ 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.

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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.

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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:	~~Station Operations~~ Manager, OPERATIONS	X
Member:	~~Station Technical~~ Manager, Technical	X
Member:	Plant Superintendent SONGS Unit 2	
Member:	Supervisor of I&C	
Member:	~~Health Physics~~ Manager, Health Physics	X
Member:	Supervisor of Chemistry	
Member:	~~Station Maintenance~~ Manager, maintenance	X
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.

[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.

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MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Onsite Review Committee shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

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

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AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Manager of Nuclear Operations and NSG of disagreement between the OSRC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Onsite Review Committee shall maintain written minutes of each OSRC meeting that, at a minimum, document the results of all OSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Nuclear Safety Group.

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Station Manager shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Proposed changes to the Appendix "A" Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the Station Manager; or by the ~~Station Technical Manager~~ as previously designated by the Station Manager. Manager, Technical

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ACTIVITIES (Continued)

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

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager, ~~the Station Technical Manager, the Station Operations Manager, the Station Maintenance Manager~~, the Deputy Station Manager or the ~~Health Physics Manager~~ as previously designated by the Station Manager. Manager, Health Physics

the Manager, Technical, the Manager, Operations, the Manager, Maintenance

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

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

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

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

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

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.3.1 The Nuclear Safety Group shall function to provide independent review and audit of designated activities in the areas of:

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FUNCTION (Continued)

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.3.2 NSG shall consist of a Supervisor and at least three staff specialists. The Supervisor shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of six years of professional leve managerial experience in the power field. Each staff specialist shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of five years of professional level experience in the field of his specialty.

The NSG shall use specialists from other technical organizations to augment its expertise in the disciplines of 6.5.2.1. Such specialists shall meet the same qualification requirements as the NSG members.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

REVIEW

6.5.3.4 The NSG shall review:

(required by Specification 6.8)

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes 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. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

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

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

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i. 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.

j. 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.

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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.

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g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

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

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

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

Change 6.8.2 to

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be 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; 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.

change "c" to

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. <u>Primary Coolant Sources Outside Containment</u>

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

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b. In-Plant Radiation Monitoring

A program* which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program* shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-Accident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program* shall include the training of personnel, the procedures for sampling and analysis and the provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC Regional Administrator unless otherwise noted.

*Not required to be implemented prior to first exceeding 5% RATED THERMAL POWER.

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STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

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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 Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, 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 Director of the Regional Office, 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.

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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.

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THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written 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. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

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

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

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

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

l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following approved plant radiation protection procedures for entry into high radiation areas.

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c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

Shift Supervisor

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the ~~Watch Engineer~~ on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. X

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.#

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

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

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

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

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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 ~~detemrination~~ 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;

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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.

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