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Dear Bob:

SUBJECT: TECHNICAL SPECIFICATIONS FOR COMANCHE PEAK, UNIT 1

Enclosed is the typed version of the Comanche Peak Unit 1 Technical Specifications - Second Draft.

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Enclosure: As stated

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SECTION 1.0

DEFINITIONS

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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 Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a four section excore neutron detector.

CHANNEL CALIBRATION

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

CHANNEL CHECK

1.6 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.

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 in compliance with the requirements of 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 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS 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 ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and injecting simulated process data to verify OPERABILITY of alarm and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

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$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides with a halflife greater than ten (10) minutes in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) 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.

EXCLUSION AREA BOUNDARY

1.14 The Exclusion Area Boundary, used for establishing effluent release limits is shown in Figure 5.1-1.

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED 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 Coolant System.

MASTER RELAY TEST

1.17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.19 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.20 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

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

PHYSICS TESTS

1.22 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (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.

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY PLANT VENTILATION SYSTEM

1.24 A PRIMARY PLANT VENTILATION SYSTEM shall be 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 Features Atmospheric Cleanup Systems are not considered to be PRIMARY PLANT VENTILATION SYSTEM components.

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any 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.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper half excore detector calibrated output to the average of the upper half excore detector calibrated outputs, or the ratio of the maximum lower half excore detector calibrated output to the average of the lower half excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 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 loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

SHUTDOWN MARGIN

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

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

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

SOURCE CHECK

1.34 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.35 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

1.38 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTING

1.40 VENTING shall be 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.

WASTE GAS HOLDUP SYSTEM

1.41 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

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TABLE 1.1

FREQUENCY NOTATION

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

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TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 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**	$\leq$ 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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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, 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 2735 psig.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, 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 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock 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:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. 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 setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.7% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.2% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature N-16	6.4	4.71	1.8	See Note 1	See Note 2
8. Overpower N-16	4.0	1.91	1.3	≤112%	≤114.5%
9. Pressurizer Pressure-Low	8.8	2.81	1.5	≥1910 psig	≥1896 psig
10. Pressurizer Pressure-High	7.5	4.96	0.5	≤2385 psig	≤2399 psig

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.31	0.6	≥90% of loop design flow**	≥88.8% of loop design flow**
13. Steam Generator Water Level - Low-Low	8.8	7.08	1.5	≥43.4% of narrow range instrument span	≥42.1% of narrow range instrument span
14 Undervoltage - Reactor Coolant Pumps	7.7	0	0	≤4830 volts-each bus	≥4781 volts-each bus
15. Underfrequency - Reactor Coolant Pumps	4.4	0	0	≥57.2 Hz	≥57.1 Hz
16. Turbine Trip					
a. Low Trip System Pressure	N.A.	N.A.	N.A.	≥45 psig	≥43 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**Loop design flow = 95,700 gpm.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.2% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	$\leq$10% RTP* Turbine First Stage Pressure Equivalent	$\leq$12.2% RTP* Turbine First Stage Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	$\leq$48% of RTP*	$\leq$50.2% of RTP*
d. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	$\geq$10% of RTP*	$\leq$7.8% of RTP*
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: Overtemperature N-16

$$N = K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}(T_C - T_C^{o})\right] + K_3 (P - P^1) - f_1 (\Delta q)$$

Where:

N	=	Measured N-16 Power by ion chambers,
T_C	=	Cold leg temperature, °F,
T_C^o	=	559.6°F, Reference T_C at RATED THERMAL POWER,
K_1	=	1.069
K_2	=	0.00948/°F,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag compensator for measured T_C,
τ_1, τ_2	=	Time constants utilized in the lead-lag compensator for T_C, τ_1 = 10 s, and τ_2 = 3 s,
K_3	=	0.000494/psig,

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta q)$ is a function of the indicated difference between top and bottom halves of detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -35% and +10%, $f_1(\Delta q) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER,

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35%, the N-16 Trip Setpoint shall be automatically reduced by 1.25% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +10%, the N-16 Trip Setpoint shall be automatically reduced by 1.55% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of span.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel 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 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.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB. DNBR is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNBR through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These heat flux conditions are more limiting than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature N-16 trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature N-16 trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% 2735 psig of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the 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. The setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy and instrument drift.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the insertion of positive reactivity that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. In addition, the Source Range Neutron Flux trip provides similar protection during shutdown operations with the reactor trip breakers closed and the rod control system capable of control rod withdrawal. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature N-16

The Overtemperature N-16 trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the N-16 detectors, and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the cold leg temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower N-16

The Overpower N-16 trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature trip, and provides a backup to the High Neutron Flux trip. The Overpower N-16 trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Pressure (Continued)

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine first stage chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer Water Level-High trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), provides a backup block for Source Range Neutron Flux doubling, and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables the Reactor trips on low flow in one reactor coolant loop. On decreasing power, the P-8 automatically blocks the reactor trip on low flow in one reactor coolant loop.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Turbine first stage chamber pressure provides input to P-7.

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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 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 1 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:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. 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 when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. 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 met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

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

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

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. 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 has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

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 of 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 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i);

SURVEILLANCE REQUIREMENTS (Continued)

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:

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
Every 9 months	At least once per 276 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; and

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 1.6% Δk/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 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 1.6% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod 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 control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor Coolant System boron concentration,

2) Control rod 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 ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD 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 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 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 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

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

   1) Reactor Coolant System boron concentration,

   2) Control rod 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 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and

b. Less negative than -4.0×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{\circ}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

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*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.1×10^{-4} $\Delta k/k/^{\circ}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.1×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

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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 551°F.

APPLICABILITY: MODES 1 and 2* #.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°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 551°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 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

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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 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid storage tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. 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 flow path is greater than or equal to 65°F when a flow path from the boric acid storage tanks is used, and

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 three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage tanks via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the RCS.

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

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ 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 flow path from the boric acid storage tanks is greater than or equal to 65°F when it is a required water source;

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 by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2b. is capable of delivering at least 120 gpm to the RCS.

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*A maximum of two charging pumps shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F. An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

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CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by 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 OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least once per 92 days the above required charging pump shall be demonstrated OPERABLE by verifying that the flow path required by Specification 3.1.2.1a is capable of delivering at least 30 gpm to the RCS; or

4.1.2.3.2 At least once per 92 days by verifying that the flow path required by Specification 3.1.2.1b is capable of delivering at least 120 gpm to the RCS.

4.1.2.3.3 A maximum of two charging pumps shall be OPERABLE, one charging pump shall be demonstrated inoperable* at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

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 1% $\Delta k/k$ 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

4.1.2.4.1 The required centrifugal charging pump(s) shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.3.2.

4.1.2.4.2 The required positive displacement charging pump shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.3.1.

4.1.2.4.3 Whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 350°F, a maximum of two charging pumps shall be OPERABLE, one charging pump shall be demonstrated inoperable* at least once per 31 days by verifying that the motor circuit breakers are secured in the open position.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage tank with:

1) A minimum contained borated water volume of [6385] gallons, ([Later]% of span), when using the boric acid transfer pump.

2) A minimum contained borated water volume of 15,123 gallons ([Later]% of span), when using the gravity feed connection,

3) A minimum boron concentration of 7000 ppm and

4) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 101,120 gallons, ([Later]% of span),

2) A minimum boron concentration of 2000 ppm and

3) A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.5 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, and

3) Verifying the boric acid storage 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 and the outside air temperature is less than 40°F.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage tank with:

1) A minimum contained borated water volume of [22,870] gallons, ([Later]% of span),

2) A minimum boron concentration of 7000 ppm, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of [479,900] gallons, ([Later]% of span),

2) A boron concentration between 2000 ppm and [2200] ppm,

3) A minimum solution temperature of 40°F, and

4) A maximum solution temperature of 120°F.

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

ACTION:

a. With the boric acid storage tank inoperable and being used as one of the above required borated water sources, 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 1% $\Delta k/k$ at 200°F; restore the boric acid storage tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

4.1.2.6 Each borated water source 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 storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 40°F or greater than 120°F.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With more than one full-length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

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

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

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

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Decrease in Reactor Coolant Inventory

- Inadvertent opening of a pressurizer safety or relief valve
- Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment
- Steam generator tube rupture
- Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

Increases in Heat Removal by the Secondary System (steam system piping failure)

Spectrum of Rod Cluster Control Assembly Ejection Accidents

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POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

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

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

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POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

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

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

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SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

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FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER:

   1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. Reduce the Power Range Neutron Flux - High Trip** Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

---

*See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [(4.64)] [K(Z)] \text{ for } P \leq 0.5$$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower N-16 Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height [± 2.08 inches] about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 1.8% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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FIGURE 3.2-3

RCS TOTAL FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

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3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 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 Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. 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 QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

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*See Special Test Exceptions Specification 3.10.2.

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SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric thimble locations or

b. Using the Movable Incore Detection System to monitor the QUADRANT POWER TILT RATIO subject to the requirement of Specification 3.3.3.2.

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3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the stated limits:

a. Indicated Reactor Coolant System $T_{avg} \leq 592°F$

b. Indicated Pressurizer Pressure $\geq$ 2207 PSIG**

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter 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

4.2.5 Each of the above parameters shall be verified to be within its limits at least once per 12 hours.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of rated thermal power.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with Reactor Trip System 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 Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 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 train such that both trains are tested at least once per 36 months and 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.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3[a], 4[a], 5[a]	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2[b]
b. Low Setpoint	4	2	3	1[d], 2	2[b]
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2[b]
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2b
5. Intermediate Range, Neutron Flux	2	1	2	1[d], 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2[c]	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature N-16	4	2	3	1, 2	6b
8. Overpower N-16	4	2	3	1, 2	6[b]
9. Pressurizer Pressure--Low	4	2	3	1[e]	6[b,f]
10. Pressurizer Pressure--High	4	2	3	1, 2	6[b]

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1[e]	6[b]
12. Reactor Coolant Flow--Low					
a. Single Loop	3/loop	2/loop in any loop	2/loop	1[g]	6[b]
b. Two Loops	3/loop	2/loop in any two loops	2/loop	1[h]	6[b]
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen.	1, 2	6[b,f]
14. Undervoltage--Reactor Coolant Pumps	4-1/bus	2	3	1[e]	6[b]
15. Underfrequency--Reactor Coolant Pumps	4-1/bus	2	3	1[e]	6[b]
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1[e]	6[b]
b. Turbine Stop Valve Closure	4	4	1	1[e]	10[b]
17. Safety Injection Input from ESFAS	2	1	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2[c]	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1,2	7
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1,2	7
19. Reactor Trip Breakers	2	1	2	1, 2	8, 11
	2	1	2	3[a], 4[a], 5[a]	9
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	8
	2	1	2	3[a], 4[a], 5[a]	9

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

[a]When the Reactor Trip breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

[b]The provisions of Specification 3.0.4 are not applicable.

[c]Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

[d]Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

[e]Above the P-7 (At Power) Setpoint

[f]The applicable modes and action statements for these channels noted in Table 3.3-3 are more restrictive and therefore, applicable.

[g]Above the P-8 (3-loop flow permissive) setpoint.

[h]Above the P-7 and below P-8 setpoints.

ACTION STATEMENTS

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

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

a. The inoperable channel is placed in the tripped condition within 2 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint,

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify either valve ICS-8455 or valves ICS-8560, FCV-11113, ICS-8439, ICS-8441, and ICS-8453 are closed and secured in position, and verify this position at least once per 14 days thereafter. With no channels OPERABLE complete all the above actions within 4 hours and verify the positions of the above valves at least once per 14 days thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - 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 System breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	≤ 0.5 seconds
7. Overtemperature N-16	≤ 7 seconds*#
8. Overpower N-16	≤ 7 seconds*
9. Pressurizer Pressure--Low	≤ 2 seconds
10. Pressurizer Pressure--High	≤ 2 seconds
11. Pressurizer Water Level--High	N.A.

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

#Response time includes the thermal well response time.

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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Reactor Coolant Flow--Low	
a. Single Loop (Above P-8)	≤ [1] second
b. Two Loops (Above P-7 and below P-8)	≤ [1] second
13. Steam Generator Water Level--Low-Low	≤ [2] seconds
14. Steam Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch	N.A.
15. Undervoltage - Reactor Coolant Pumps	≤ [1.5] seconds
16. Underfrequency - Reactor Coolant Pumps	≤ [0.6] second
17. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
18. Safety Injecti Input from ESF	N.A.
19. Reactor Trip System Interlocks	N.A.
20. Reactor Trip Breakers	N.A.
21. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3[a], 4[a], 5[a]
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q(17)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1[c], 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(17)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1[c], 2
6. Source Range, Neutron Flux	S	R(4, 13)	S/U(1),Q(9,17)	R(12)	N.A.	2[b], 3, 4, 5
7. Overtemperature N-16	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	Q(17)	N.A.	N.A.	1, 2
8. Overpower N-16	S	D(2, 4) R(4, 5)	Q(17)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(8, 17)	N.A.	N.A.	1[d]
10. Pressurizer Pressure--High	S	R	Q(17)	N.A.	N.A.	1, 2

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TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Water Level--High	S	R	Q(17)	N.A.	N.A.	1[d]
12. Reactor Coolant Flow--Low	S	R	Q(17)	N.A.	N.A.	1[d]
13. Steam Generator Water Level--Low-Low	S	R	Q(8, 17)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(10, 17)	N.A.	1[d]
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q(17)	N.A.	1[d]
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1[d]
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1[d]
17. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2[b]

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7						
1) Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
2) Turbine First Stage Pressure P-13	N.A.	R	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
e. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3[a], 4[a], 5[a]
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3[a], 4[a], 5[a]
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(15),R(16)	N.A.	1, 2, 3[a], 4[a], 5[a]

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

[a]When the Reactor Trip breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

[b]Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

[c]Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

[d]Above the P-7 (at power) setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power and N-16 power indication above 15% of RATED THERMAL POWER. Adjust excore channel and/or N-16 channel gains consistent with calorimetric power if absolute difference of the respective channel is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. For the purpose of these surveillance "M" requirements is defined as at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron and N-16 detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained and evaluated. For the Intermediate Range and Power Range Neutron Flux and N-16 channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. For the purpose of these surveillance requirements "Q" is defined as at least once per 92 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) The surveillance frequency and/or modes specified for these channels in Table 4.3-2 are more restrictive and therefore applicable.

(9) Quarterly surveillance in MODES 3[a], 4[a], and 5[a] shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to (an increase of twice the count rate within a 10-minute period).

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves close and the centrifugal charging pumps suction valves from the RWST open.

(13) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service. (Or for plants that do not actuate the shunt trip attachment of the bypass breakers on a manual reactor trip): Remote manual undervoltage trip when breaker placed in service.

(16) Automatic undervoltage trip.

(17) Each channel shall be tested at least every 92 days on a staggered test basis.

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks 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 ESF RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

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

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SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 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 train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per 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.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (ECCS, Reactor Trip, Feedwater Isolation, Control Room Emergency Recirculation, Emergency Diesel Generator Operation, Containment Vent Isolation, Station Service Water, Phase A Isolation, Auxiliary Feedwater-Motor Driven Pump, Turbine Trip, Component Cooling Water, Essential Ventilation Systems, and Containment Spray Pump.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	12
c. Containment Pressure--High-1	3	2	2	1, 2, 3	13[a]
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3[b]	17[a]
e. Steam Line Pressure--Low	3/Steam Line	2/Steam Line in any Steam Line	2/Steam Line	1, 2, 3[b]	13[a]

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair operated simultaneously	2 pair	1, 2, 3, 4	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	12
c. Containment Pressure--High-3	4	2	3	1, 2, 3	14
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	12
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	See Item 2a above. Phase "B" isolation is manually initiated when containment spray function is manually initiated.			1, 2, 3, 4	16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	12
3) Containment Pressure--High-3	4	2	3	1, 2, 3	14

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
c. Containment Vent Isolation					
1) Manual Initiation	See Item 2a. and 3.a.1 above. Containment vent isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiated.			1, 2, 3, 4	15
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	15
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual Steam Line	1/steam line	1/steam line	1/operating steam line	1, 2[d], 3[d]	21
2) System	2	1	2	1, 2[d], 3[d]	20
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2[d], 3[d]	19
c. Containment Pressure--High-2	3	2	2	1, 2[d], 3[d]	13[a]

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2[d], 3[b,d]	13[a]
e. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/steam line	3[c,d]	13[a]
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Steam Generator Water Level--High-High (P-14)	3/stm. gen.[f]	2/stm. gen. in any stm. gen.	2/stm. gen.	1, 2	13[a]
c. Safety Injection	See Item 1 above for all safety injection initiating functions and requirements.				
6. Auxiliary Feedwater					
a. Manual Initiation	2	1	2	1, 2, 3	20
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	19
c. Stm. Gen. Water Level--Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	17[a]

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3[e]	17[a]
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	1/train	1/train	1/train	1, 2, 3	16
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2/pump	1/pump	1/pump	1, 2	16
7. Automatic Initiation of ECCS Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Automatic Initiation of ECCS Switchover to Containment Sump (Continued)					
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	17
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Loss of Power (6.9 kV Safeguards System Undervoltage)					
a. Preferred Offsite Source Undervoltage					
1) Undervoltage Relay	2/bus	2/bus	1/bus	1, 2, 3, 4	24[a]
2) Diesel Start Timer	1/bus	1/bus	1/bus	1, 2, 3, 4	27[a]
3) Source Bkr Trip Timer	1/bus	1/bus	1/bus	1, 2, 3, 4	27[a]
b. Bus Undervoltage					
1) Diesel Start					
a) Undervoltage Relay	2/bus	2/bus	1/bus	1, 2, 3, 4	24[a]
b) Timer	1/bus	1/bus	1/bus	1, 2, 3, 4	24[a]
2) Initiation of Solid State Safeguards System Sequencer					
a) Undervoltage Relay	4/bus	2/bus	3/bus	1, 2, 3, 4	17[a]
b) Timer	4/bus	2/bus	3/bus	1, 2, 3, 4	17[a]

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Control Room Emergency Recirculation					
a. Manual Initiation	2	1	2	All	24
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	24
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	18
b. Reactor Trip, P-4	2	2	2	1, 2, 3	20
11. Solid State Safeguards Sequencer (SSSS)					
a. Safety Injection Sequence	1/train	1/train	1/train	1, 2, 3, 4	12
b. Black Out Sequence	1/train	1/train	1/train	1, 2, 3, 4	26

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

[a]The provisions of Specification 3.0.4 are not applicable.

[b]Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

[c]Trip function automatically blocked above P-11 and may be unblocked below P-11 by blocking the Safety Injection on low steam line pressure.

[d]Not applicable if each affected main steam isolation valve and its associated upstream drain pot isolation valve per steam line is closed.

[e]The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

[f]The channel which provides a steam generator water level control signal (if one of three specific trip channels is selected to provide input into steam generator water level control) must be placed in the tripped condition within 1 hour and maintained in the tripped condition with the exception that the channel may be taken out of the tripped condition for up to 2 hours to allow testing of redundant channels.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 14 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 15 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment pressure relief valves are closed within 4 hours and maintained closed.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, 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 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 20 - 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 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 21 - 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 declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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b. The Minimum Channels OPERABLE requirement is met.

ACTION 24 - 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 initiate and maintain operation of the Control Room Emergency Recirculation System.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 26 - With the number of OPERABLE Channels on one or more trains less than the Minimum Channels OPERABLE requirement, declare the diesel generator(s) associated with the affected train(s) inoperable and apply the appropriate ACTION for Specification 3.8.1.1.

ACTION 27 - With less than the Minimum Channels OPERABLE, Startup and/or Power Operation may proceed provided the timer in the affected channel is bypassed and actions are taken immediately to restore the timer to OPERABLE status.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (ECCs, Reactor Trip, Feedwater Isolation, Control Room Emergency Recirculation, Emergency Diesel Generator Operation, Containment Vent Isolation, Station Service Water, Phase A Isolation, Auxiliary Feedwater-Motor Driven Pump, Turbine Trip, Component Cooling Water, Essential Ventilation Systems, and Containment Spray Pump.					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1	2.5	0.71	1.5	≤ 3.35 psig	≤ 3.9 psig
d. Pressurizer Pressure--Low	16.1	14.41	1.5	≥ 1829 psig	≥ 1823 psig
e. Steam Line Pressure--Low	17.3	14.81	1.5	≥ 605 psig*	≥ 586 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	2.5	0.71	1.5	≤ 18.35 psig	≤ 18.9 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	See Item 2.a above. Phase "B" isolation is manually initiated when containment spray function is manually initiated.				
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	2.5	0.71	1.5	≤ 18.35 psig	≤ 18.9 psig
c. Containment Vent Isolation					
1) Manual Initiation	See Items 3.a.1 and 2.a above. Containment Vent Isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiation.				
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	2.5	0.71	1.5	≤6.35 psig	≤6.9 psig
d. Steam Line Pressure--Low	17.3	14.81	1.5	≥605 psig*	≥586 psig*
e. Steam Line Pressure - Negative Rate--High	8.0	0.5	0]	≤100 psi**	≤ 111.6 psi**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	7.6	4.3	1.5	≤82.4% of narrow range instrument span.	≤84.2% of narrow range instrument span.
c. Safety Injection	See Item 1 above for all Safety Injection setpoints and allowable values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	8.8	7.08	1.5	≥ 43.4% of narrow range instrument span.	≥ 42.1% of narrow range instrument span.
d. Safety Injection - Start Motor Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power	N.A.	N.A.	N.A.	N.A.	N.A.
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
7. Automatic Initiation of ECCS Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low Coincident With Safety Injection	[2.1]	[0.71]	[1.2]	≥ 40.6% of span	≥ 34.94% of span
	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power (6.9 kV Safeguards System Undervoltage)					
a. Preferred Offsite Source Undervoltage					
1) Undervoltage Relay	N.A.	N.A	N.A	≥4800 V	≥4692 V
2) Diesel Start Timer	N.A.	N.A.	N.A.	≤0.75 s	≤0.825 s
3) Source Bkr. Trip Timer	N.A.	N.A.	N.A.	≤0.5 s	≤0.55 s
b. Bus Undervoltage					
1) Diesel Start					
a) Undervoltage Relay	N.A.	N.A.	N.A.	≥2100 V	≥1992 V
b) Timer	N.A.	N.A.	N.A.	≤0.75s	≤0.825s
2) Initiation of Solid State Safeguards System Sequence					
a) Undervoltage Relays	N.A.	N.A.	N.A.	≥4800 V	≥4692 V
b) Timer	N.A.	N.A.	N.A.	≤0.5 s	≤0.55 s
9. Control Room Emergency Recirculation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ [1985] psig	≤ [1996] psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
11. Solid State Safeguards Sequencer (SSSS)	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 3.3-4 (Continued)

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TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray (Phase "B" Isolation and Containment Ventilation Isolation)	N.A.
c. Phase "A" Isolation (Containment Ventilation Isolation)	N.A.
d. Steam Line Isolation	N.A.
e. Feedwater Isolation (SI)	N.A.
f. Auxiliary Feedwater (SI)	N.A.
g. Station Service Water (SI)	N.A.
h. Component Cooling Water (SI)	N.A.
i. Control Room Emergency Recirculation (SI)	N.A.
j. Reactor Trip	N.A.
k. Emergency Diesel Generator Operation	N.A.
l. Essential Ventilation Systems (SI)	N.A.
m. Turbine Trip	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(1,5(a))}/[12]^{(4,5b)}$
b. Reactor Trip	≤ 2
c. Feedwater Isolation	≤ 6.5
d. Phase "A" Isolation	$\leq 17^{(2)}/27^{(1)}$
e. Containment Ventilation Isolation	N.A.
f. Auxiliary Feedwater	≤ 60
g. Station Service Water	N.A.
h. Component Cooling Water	N.A.
i. Essential Ventilation Systems	N.A.
j. Emergency Diesel Generator Operation	≤ 12
k. Turbine Trip	N.A.
l. Control Room Emergency Recirculation	N.A.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1,5a)}/12^{(4,5b)}$
b. Reactor Trip	≤ 2
c. Feedwater Isolation	≤ 7
d. Phase "A" Isolation	$\leq 17^{(2)}/27^{(1)}$
e. Containment Ventilation Isolation	$\leq 5^{(6)}$
f. Auxiliary Feedwater	≤ 60
g. Station Service Water	N.A.
h. Component Cooling Water	N.A.
i. Essential Ventilation Systems	N.A.
j. Emergency Diesel Generators Operation	≤ 12
k. Turbine Trip	N.A.
l. Control Room Emergency Recirculation	N.A.
4. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq$ [illegible]1,5b)/12(4,5b)
b. Reactor Trip	≤ 2
c. Feedwater Isolation	≤ 6.5
d. Phase "A" Isolation	$\leq 17^{(2)}/27^{(1)}$
e. Containment Ventilation Isolation	N.A.
f. Auxiliary Feedwater	≤ 60
g. Station Service Water	N.A.
h. Component Cooling Water	N.A.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIG. AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure--Low (Continued)	
i. Essential Ventilation Systems	N.A.
j. Emergency Diesel Generator Operation	≤ 12
k. Turbine Trip	N.A.
l. Control Room Emergency Recirculation	N.A.
m. Steam Line Isolation	6.5
5. Containment Pressure--High-3	
a. Containment Spray Pump	≤ 22(2)/32(1)
b. Phase "B" Isolation	N.A.
6. Containment Pressure--High-2	
Steam Line Isolation	≤ 6.5
7. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	≤ 7
8. Steam Generator Water Level-High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 11
9. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
Auxiliary Feedwater	N.A.
11. Trip of All Main Feedwater Pumps	
Motor-Driven Auxiliary Feedwater Pumps	N.A.
12. RWST Level--Low-Low Coincident with Safety Injection	
a. Automatic Initiation of ECCS Switchover to Containment Sump	≤30

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
13. Loss of Power (6.3 kV Safeguards System Undervoltage)	
a. Preferred Offsite Source Undervoltage	≤70
b. Bus Undervoltage	≤10

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting delay included. RHR pumps not included.

(4) Diesel generator starting and sequence loading delays not included. RHR pumps not included.

(5) Response Time Limit includes opening of injection path valves. Following additional time is allowed for completion of the transfer of the pump suction from the VCT to the RWST.

a) 10 seconds
b) 15 seconds

(6) Includes containment pressure relief line isolation only.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Recirculation, Emergency Diesel Generator Operation, Containment Vent Isolation, Station Service Water, Phase A Isolation, Auxiliary Feedwater-Motor Driven Pump, Turbine Trip, Component Cooling Water, Essential Ventilation Systems, and Containment Spray Pump.								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	See item 2.a above. Phase "B" isolation manually initiated when containment spray function is manually initiated.							1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	See Item 2.a. above Phase "B" isolation is manually initiated when containment spray function is manually initiated.							1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
c. Containment Vent Isolation								
1) Manual Initiation	See Item 3.a.1 and 2 a above. Containment vent isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiated.							1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Turbine Trip and Feedwater Isolation (Continued)								
b. Steam Generator Water Level-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	N.A.	R	N.A.	M(3, 4)	N.A.	N.A.	N.A	1, 2, 3
f. Trip of All Main Feed water Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
7. Automatic Initiation of ECCS Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
b. RWST Level-Low-Low Coincident With	S	A	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power (6.9 kV Safeguards System Undervoltage)								
a. Preferred Offsite Source Undervoltage								
1) Undervoltage Relay	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Diesel Start Timer	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Source Bkr. Trip Timer	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Bus Undervoltage								
1) Diesel Start								
a) Undervoltage Relay	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b) Timer	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2) Initiation of Solid State Safeguards System Sequence								
a) Undervoltage Relay	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b) Timer	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Control Room Emergency Recirculation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
11. Solid State Safeguards Sequencer (SSSS)								
a. Safety Injection Sequence	N.A.	R	N.A.	N.A.	M(1,2)	N.A.	N.A.	1, 2, 3, 4
b. Blackout Sequence	N.A.	R	N.A.	N.A.	M(1,2)	N.A.	N.A.	1, 2, 3, 4

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Performed by Solid State Safeguards Sequencer Automatic Test.
(3) Setpoint verification is not applicable.
(4) Actuation of final devices not included.

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3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations 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/Trip Setpoint for plant operations 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 channels for plant operations 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 instrumentation channel for plant operations shall be demonstrated OPERABLE:

a. At least once per 12 hours by performance of a CHANNEL CHECK,

b. At least once per 18 months by performance of a CHANNEL CALIBRATION,

c. At least once per 31 days by performance of a DIGITAL CHANNEL OPERATIONAL TEST.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. RCS Leakage Detection					
a. Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	31
b. Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	31
2. Containment Ventilation Isolation					
a. Particulate Radioactivity	1	2	All	*	28
b. Gaseous Radioactivity	1	2	All	*	28
3. Fuel Storage Pool Areas					
a. Criticality-Radiation Level	1	2	**	$\leq$ 15 mR/h	30
4. Control Room Emergency Recirculation					
a. Air Intake-Radiation Level	1/intake	2/intake	All	[Later] μCi/ml	29

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* Must satisfy Specification 3.11.2.1 requirements.

** With fuel in the fuel storage pool areas or fuel building.

*** Must satisfy Specification 3.11.2.1 requirements.

ACTION STATEMENTS

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment ventilation valves are maintained closed. The containment pressure relief valves may only be opened in compliance with Specification 3.6.1.7 and 3.3.3.11.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, within 1 hour secure the Control Room makeup air supply fan from the affected intake or initiate and maintain operation of the Control Room Emergency Air Cleanup System in emergency recirculation.

ACTION 30 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 31 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

ACTION:

With the Movable 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 Movable Incore Detection System shall be demonstrated OPERABLE within 24 hours prior to use by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

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INSTRUMENTATION

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 of the above required 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 required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments which are accessible during power operations and which is actuated during a seismic event greater than or equal to 0.01g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from 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 14 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 greater than or equal to 0.01g 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 5 or below. A supplemental report shall then be prepared and submitted to the Commission within 14 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 AND SENSOR LOCATIONS	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs	
a. Accelerometer-Fuel Building	1
b. Accelerometer-Containment	1
c. Accelerometer-Electrical Manhole	1
d. Seismic Trigger-Fuel Building	1
e. Recorder Unit, SMA-3	1
f. Playback Unit, SMP-1	1
2. Triaxial Peak Accelerographs	
a. Pressurizer Lifting Trunion	1
b. Reactor Coolant Piping	1
c. Component Cooling Water Heat Exchanger	1
3. Triaxial Seismic Switch	
Fuel Building	1*
4. Triaxial Response-Spectrum Recorders	
a. Fuel Building	1
b. Reactor Bldg. Internal Structure	1
c. Safeguards Building	1
5. Response Spectrum Annunciator	1*

*With control room indication.

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TABLE 4.3-3

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG* CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerographs			
a. Accelerometer-Fuel Building	M	R	SA
b. Accelerometer-Containment	M	R	SA
c. Accelerometer-Electrical Manhole	M	R	SA
d Seismic Trigger-Fuel Building	M	R	SA
e. Recorder Unit, SMA-3	M	R	SA
f. Playback Unit, SMP-1	M	R	SA
2. Triaxial Peak Accelerographs			
a. Pressurizer Lifting Trunion	N.A.	R	N.A.
b. Reactor Coolant Piping	N.A.	R	N.A.
c. Component Cooling Water Heat Exchanger	N.A.	R	N.A.
3. Triaxial Seismic Switch			
Fuel Building**	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. Fuel Building	N.A.	R	N.A.
b. Reactor Bldg. Internal Structure	N.A.	R	N.A.
c. Safeguards Building	N.A.	R	N.A.
5. Response Spectrum Annunciator**	M	R	SA

*Setpoint verification is not applicable.
**With control room indication.

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:

a. At least once per 24 hours by performance of a CHANNEL CHECK, and

b. At least once per 184 days by performance of a CHANNEL CALIBRATION.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		1 of 3
a. X-SY-4117	Nominal Elev. 60 m.	
b. X-SY-4118	Nominal Elev. 10 m.	
c. X-SY-4128*	Nominal Elev. 10 m.	
2. WIND DIRECTION		1 of 3
a. X-ZY-4115	Nominal Elev. 60 m.	
b. X-ZY-4116	Nominal Elev. 10 m.	
c. X-ZY-4126*	Nominal Elev. 10 m.	
3. AIR TEMPERATURE - ΔT		1 of 2
a. X-TY-4119	Nominal Elev. 60 m. and Nominal Elev. 10 m.	
b. X-TY-4120	Nominal Elev. 60 m. and Nominal Elev. 10 m.	

*Mounted on backup tower.

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, pursuant to Specification 6.9.2, submit a Special Report that defines the corrective action to be taken.

c. With one or more Remote Shutdown System transfer switches, power, or control circuits inoperable, restore the inoperable switch(s)/ circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE

a. At least once per 31 days by performance of a channel check, and

b. At least once per 18 months by performance of a channel calibration.*

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit shall be demonstrated OPERABLE at least once per 18 months by verifying its capability to perform its intended function(s).

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*Neutron detectors may be excluded from channel calibration.

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TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Neutron Flux Monitors	HSP	2	1
2. Wide Range RCS Temp. - T_c	HSP	1/Loop	1/Loop
3. Wide Range RCS Temp. - T_h	HSP	1/Loop	1/Loop
4. Pressurizer Pressure	HSP	1	1
5. Pressurizer Level	HSP	2	1
6. Steam Generator Pressure	HSP	1/SG	1/SG
7. Steam Generator Level	HSP	1/SG	1/SG
8. Auxiliary Feedwater Flow Rate to Steam Generator	HSP	2/SG	1/SG
9. Condensate Storage Tank Level	HSP	2	1
10. Charging Pump to CVCS Charging and RCP Seals - Flow Indication	HSP	1	1

TRANSFER SWITCHES [Illustrational only]	SWITCH LOCATION
1. Auxiliary Feedwater Control	
2. Safe Shutdown Equipment Power	
a. Auxiliary Feedwater	
b. Charging	
c. Pressurizer Heaters	
d. Valves	
3. CVCS Makeup Flow Control	
4. Diesel Generator Control	
5. Electrical Distribution System Control	

CONTROL CIRCUITS [Illustrational only]	SWITCH LOCATION
1. Auxiliary Feedwater Flow	
2. Pressurizer Heaters	
3. CVCS Makeup Flow	
4. Diesel Generator	
5. Electrical Distribution System	

HSP = Hot Shutdown Panel
SG = Steam Generator

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels except the unit vent-high range noble gas monitor, and the steam relief-high range radiation monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE channels for the unit vent-high range noble gas monitor, or the steam relief-high range radiation monitor or the containment atmosphere-high range radiation monitor, or the reactor coolant radiation level monitor less than required by the Minimum Channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL CHECK, and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.*

*Containment Area Radiation (High Range) CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure (Wide Range)	2	1
2. Containment Pressure (Narrow Range)	2	1
3. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
4. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
5. Reactor Coolant Pressure - Wide Range	2	1
6. Pressurizer Water Level	2	1
7. Steam Generator Water Level - Wide Range and Auxiliary Feedwater Flow (Secondary Coolant Availability)	1/steam generator	1/steam generator
8. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
9. Containment Water Level (Wide Range)	2	1
10. Core Exit Temperature (Thermocouples)	4/core quadrant	2/core quadrant
11. Steam Relief Vent - Noble Gas Monitors	N.A.	4
12. Containment Area Radiation (High Range)	2	1
13. Reactor Vessel Water Level	2	1

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TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Neutron Flux (Source Range)	2	1
15. Neutron Flux (Intermediate Range)	2	1
16. Condensate Storage Tank Level	2 (2 Sensors/Channel)	1 (1 Sensor/Channel)
17. Steam Line Pressure	2/steam generator	1/steam generator
18. Refueling Water Storage Tank Water Level	2	1
19. Reactor Coolant System Subcooling Margin Monitor	2	1
20. Plant Vent Stack - Mobile Gas Monitors		
a. Intermediate Range	N.A.	1
b. High Range	N.A.	1

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CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems for each fresh air intake, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: All MODES, if chlorine gas is stored on site in quantities greater than 20 lbs.

ACTION:

a. With one Chlorine Detection System at a fresh air intake inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours isolate the affected fresh air intake and comply with the provisions of Specification 3.7.7 and either (1) operate the Control Room HVAC System from the unaffected fresh air intake or (2) initiate and maintain operation of the Control Room HVAC System in the isolation mode of operation.

b. With both Chlorine Detection Systems at a fresh air intake inoperable, within 1 hour isolate the affected fresh air intake and comply with the provisions of Specification 3.7.7 and either (1) operate the Control Room HVAC System from the unaffected fresh air intake or (2) initiate and maintain operation of the Control Room HVAC System in the isolation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE:

a. At least once per 12 hours by performance of a CHANNEL CHECK,

b. At least once per 31 days by verifying alarm and trip relay actuation when each channel is tested using installed test circuitry, and

c. At least once per 18 months by performance of a CHANNEL CALIBRATION.

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INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 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

4.3.3.8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST* at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

*Setpoint verification is not applicable.

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INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-11 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 and adjusted in accordance with the methodology and parameters in 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 3.3-11. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST or ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-5.

TABLE 3.3-11

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1	32
b. Turbine Building (Floor Drains) Sumps Effluent Line	1	33
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
Service Water System Effluent Line	1/train	34
3. Flow Rate Measurement Devices		
Liquid Radwaste Effluent Line	1	35

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ACTION STATEMENTS

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, 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 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

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

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operations may continue provided that:

a. With the component cooling water monitor OPERABLE and indicating an activity of less than [1×10^{-4}] microCurie/ml, a grab sample is collected and analyzed for radioactivity at a lower limit of detection at least every 31 days; or

b. At least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

TABLE 4.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release					
a. Liquid Radwaste Effluent Line	D	P	R(4)	Q(1)	N.A.
b. Turbine Building (Floor Drains) Sumps Effluent Line	D	M	R(4)	Q(2)	N.A.
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release					
Service Water System Effluent Line	D	M	R(4)	Q(3)	N.A.
3. Flow Rate Measurement Devices					
Liquid Radwaste Effluent Line	D(5)	N.A.	R	N.A.	Q

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TABLE 4.3-5 (Continued)

TABLE NOTATIONS

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

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Channel Out of Service - Loss of Power, Loss of Counts, Loss of Flow, or Check Source Failure).

(2) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that automatic flow diversion of this pathyway (from the Low Volume Waste Treatment System to the Co-Current Waste Treatment System) and Control Room alarm annunciation occur if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Channel Out of Service - Loss of Power, Loss of Counts, Loss of Sample Flow, or Check Source Failed).

(3) The DIGITAL CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (Channel Out of Service - Loss of Power, Loss of Counts, Loss of Flow or Check Source Failure).

(4) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) 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, reference standards certified by NBS, or standards that have been obtained from suppliers that participate in measurement assurance activities with NBS shall be used.

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TABLE 4.3-5 (Continued)

TABLE NOTATIONS (Continued)

(5) 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.

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous 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 Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-12

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 3.3-12. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful explain in the next Semi-annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and DIGITAL CHANNEL OPERATIONAL TEST or ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

TABLE 3.3-12

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	1/stack	***	36
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitors	1/recombiner	**	41
b. Oxygen Monitors	2/recombiner	**	39
3. Primary Plant Ventilation			
a. Noble Gas Activity Monitor	1/stack	*	38
b. Iodine Sampler	1/stack	*	40
c. Particulate Sampler	1/stack	*	40
d. Flow Rate Measuring Device	1/stack	*	37
e. Sampler Flow Rate Monitor	1/stack	*	37

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TABLE 3.3-12 (Continued)

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TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

*** During Batch Radioactive Releases via this pathway.

ACTION STATEMENTS

ACTION 36 - 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 provided that prior to initiating the release:

a. The auxiliary building vent duct monitor is confirmed OPERABLE, or

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

c. 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 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 39 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

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 provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

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TABLE 3.3-12 (Continued)

TABLE NOTATIONS (Continued)

ACTION 41 -

a. With the outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours and the oxygen concentration remains less than 1 percent.

b. With the inlet oxygen monitor inoperable, operation may continue if inlet hydrogen monitor is OPERABLE.

c. With both oxygen channels or both of the inlet oxygen and inlet hydrogen monitors inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	ANALOG CHANNEL OPERATIONAL TEST
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	N.A.
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System					
a. Hydrogen Monitors	D	N.A.	Q(4)	N.A.	M
b. Oxygen Monitors	D	N.A.	Q(4)	N.A.	M
3. Primary Plant Ventilation					
a. Noble Gas Activity Monitor	D	M#	R(3)	Q(2)	N.A.
b. Iodine Sampler	W(5)	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	W(5)	N.A.	N.A.	N.A.	N.A.
d. Flow Rate Measuring Device	D	N.A.	R	N.A.	Q
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	Q

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TABLE 4.3-6 (Continued)

TABLE NOTATIONS

#Also prior to any release from the waste gas holdup system or containment purging or vent.

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

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Channel Out of Service - Loss of Power, Loss of Counts, Loss of Sample-Flow, or Check Source Failure).

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

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure (Channel Out of Service - Loss of Power, Loss of Counts, Loss of Sample-Flow, or Check Source Failure).

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) 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, reference standards certified by NBS, or standards that have been obtained from suppliers that participate in measurement assurance activities with NBS shall be used.

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The Channel Check shall consist of visually verifying that the collection element (i.e., filter or cartridge, etc.) is in place for sampling.

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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.

APPICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) 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 overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 14 days by cycling each of the following valves through at least one complete cycle from the running position using the manual test or Automatic Turbine Tester (ATT):

   1) Four high pressure turbine stop valves,
   2) Four high pressure turbine control valves,
   3) Six low pressure turbine stop valves, and
   4) Six low pressure turbine control valves.

b. At least once per 14 days by testing of the two mechanical overspeed devices using the Automatic Turbine Tester or manual test.

c. 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.

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 (if applicable), disks and stems and verifying no unacceptable flaws. If unacceptable flaws are found, all other valves of that type shall be inspected.

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*Not applicable in MODES 2 and 3 with all main steam line isolation valves and associated bypass valves in the closed position.

3/4.4 REACTOR COOLANT SYSTEM

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All four (4) reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with at least two reactor coolant loops in operation when the reactor trip breakers are closed and at least one reactor coolant loop in operation when the reactor trip breakers are open:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

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

b. With only one reactor coolant loop in operation and the reactor trip breakers in the closed position, within 1 hour restore two loops to operation or open the reactor trip breakers.

c. With no reactor coolant loop in operation, open the reactor trip breakers and 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 OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

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*All reactor coolant pumps may be deenergized 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.

**See special test exceptions Specification 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% (narrow range) at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, or

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

*All reactor coolant pumps and RHR pumps may be deenergized 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 in Mode 4 unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

b. With no 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 loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), and/or RHR pump(s) if not in operation, shall be determined 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 secondary side water level to be greater than or equal to 10% (narrow range) at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified 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 residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

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

b. The secondary side water level of at least two steam generators shall be greater than or equal to 10% (narrow range).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR 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 RHR loop 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 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*The RHR pump may be deenergized 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.

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

***A reactor coolant pump shall not be started in Mode 5 unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

b. With no RHR 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 RHR loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

**The RHR pump may be deenergized 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

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional 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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LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 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 at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional 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

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1662 cubic feet (92% of span), and at least two groups of pressurizer heaters each having a capacity of at least [150] kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The 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.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

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3/4.4.4 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION a and b in Specification 3.4.4.

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3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 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

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 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.

4.4.5.2 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 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of all the expanded tubes and at least 3% of the remaining 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:

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

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.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, and

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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4.4.5.3 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 (EFPM) and before 12 EFPM and shall include a special inspection of all expanded tubes in all steam generators. 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, not including the preservice inspection, result in all 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 in 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

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 leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

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4.4.5.4 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 40%* of the nominal tube wall thickness;

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 Specification 4.4.5.3c., 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; and

*Value to be determined in accordance with the recommendations of Regulatory Guide 1.121, August 1976.

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 shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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.

4.4.5.5 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 the 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, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR Part 50.72 within four hours of initial discovery, and pursuant to Specification 6.3.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Four
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATIONS

1. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. For the fourth and subsequent inspections, the inservice inspection may be limited to one steam generator on a rotating schedule encompassing 12% 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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TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

	1ST SAMPLE INSPECTION		2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S = \frac{12}{n}\%$ Where n is the number of steam generators inspected during an inspection

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Sump Level and Flow Monitoring System, and

c. Either the containment air cooler condensate flow rate or the 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 Radioactive 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. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Sump Level and Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Air Cooler Condensate Flow Rate Monitoring System - performance of CHANNEL CALIBRATION at least once per 18 months.

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LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 0.5 GPM leakage per nominal inch of valve size up to a maximum of 5 GPM 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 above 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

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

a. Monitoring the Reactor Coolant System Leakage Detection System required by Specification 3.4.6.1 at least once per 12 hours;

b. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

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

d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying 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, except for valves 8701A, 8701B, 8702A, and 8702B.

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

d. Following check valve actuation due to flow through the valve.

e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

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

VALVE NUMBER	FUNCTION
8948 A, B, C, D	Accumulator Tank Discharge
8956 A, B, C, D	Accumulator Tank Discharge
8905 A, B, C, D	SI Hot Leg Injection
8949 A, B, C, D	SI Hot Leg Injection
8818 A, B, C, D	RHR Cold Leg Injection
8819 A, B, C, D	SI Cold Leg Injection
8701 A, B	RHR Suction Isolation
8702 A, B	RHR Suction Isolation
8705 A, B	RHR Suction Isolation Relief
8841 A, B	RHR Hot Leg Injection
8815	CCP Cold Leg Injection
8900 A, B, C, D	CCP Cold Leg Injection

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 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; and

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 psig, 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 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters specified in Table 3.4-2 at least once per 72 hours.*

*Sample and analysis for dissolved oxygen is not required with T_{avg} less than or equal to 250°F.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	< 0.10 ppm	≤ 1.00 ppm
Chloride	< 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

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

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3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per 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; and

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

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

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

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

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FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131

TABLE 4.4-4

REACTOR 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 Radioactivity 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 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

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TABLE 4.4-4 (Continued)

TABLE NOTATIONS

*A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

**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.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-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 operation 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 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 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.

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FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO ____ EFPY

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FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO _____ EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)

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LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 625°F.

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

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

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LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with lift settings which vary with RCS temperature and which do not exceed the limits established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.98 square inches.

APPLICABILITY: MODE 4, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.98 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.98 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are 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 PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

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4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) 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.

3/4.4.10 STRUCTURAL INTEGRITY

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LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

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(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 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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3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of two vent valves in series powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

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

ACTION:

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

b. With both Reactor Coolant System vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

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3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open with power removed,

b. A contained borated water volume of between 6253 gallons ([Later]% of span) and 6465 gallons ([Later]% span)

c. A boron concentration of between [1900] and [2100] ppm, and

d. A nitrogen cover-pressure of between 605 and 655 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each indicated solution volume increase of greater than or equal to 101 gallons ([Later]% of span) by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is removed.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days be the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

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3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq$ 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 centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically opening the containment sump suction valves during the recirculation phase of operation.

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.

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*The provisions of Specification 3.0.4 and 4.0.4 are not applicable for entry into Mode 3 for the centrifugal charging pumps and the safety injection pumps declared inoperable pursuant to Specification 3.5.3 provided the centrifugal charging pumps and the safety injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold leg exceeding 375°F, whichever comes first.

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
8802 A & B	SI Pump to Hot Legs	Closed
8808 A, B, C, D	Accum. Discharge	Open*
8809 A & B	RHR to Cold Legs	Open
8835	SI Pump to Cold Legs	Open
8840	RHR to Hot Legs	Closed
8806	SI Pump Suction from RWST	Open
8813	SI Pump Mini-Flow Valve	Open

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. 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 each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System to ensure that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to [425] psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

*Surveillance Requirements covered in Specification 4.5.1.1.

SURVEILLANCE REQUIREMENTS (Continued)

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 Safety Injection actuation and test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pumps,

b) Safety Injection pumps, and

c) RHR pumps.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ 2370 psid,

2) Safety Injection pump $\geq$ 1440 psid, and

3) RHR pump > 170 psid.

g. By verifying the correct position of each mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

CCP/SI System Valve Number	SI System Valve Number	
SI-8810A	SI-8822A	SI-8816A
SI-8810B	SI-8822B	SI-8816B
SI-8810C	SI-8822C	SI-8816C
SI-8810D	SI-8822D	SI-8816D

SURVEILLANCE REQUIREMENTS (Continued)

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

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 333 gpm, and

b) The total pump flow rate is less than or equal to 555 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the cold leg injection line flow rates, excluding the highest flow rate, is greater than or equal to 437 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For RHR pump lines, with a single pump running, the sum of the cold leg injection line flow rates is greater than or equal to 4652 gpm.

i. Prior to entering MODE 3 and following any maintenance or operations activity which drains portions of the system by venting the ECCS pump casing and accessible discharge piping high points.

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

ECCS SUBSYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. 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.

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* A maximum of two charging pumps shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.

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SURVEILLANCE REQUIREMENTS

4.5.3.1.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.1.2 A maximum of two charging pumps shall be OPERABLE; one charging pump shall be demonstrated inoperable* by verifying that the motor circuit breaker is secured in the open position within 4 hours after entering MODE 4 from 3 or prior to temperature of one or more of the RCS cold legs decreasing below 325°F, whichever occurs first and at least once per 31 days thereafter.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

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3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} < 350°F$

SAFETY INJECTION PUMPS

LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injectio pumps shall be inoperable.

APPLICABILITY: Modes 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever occurs first and at least once per 31 days thereafter.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 479,900 gallons ([Later]% of span),

b. A boron concentration of between 2000 and 2200 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 120°F.

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

ACTION:

With the RWST 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 120°F.

3/4.6 CONTAINMENT SYSTEMS

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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 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

4.6.1.1 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 3.6.4.1;

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 48.3 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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*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. The blind flange on the fuel transfer canal need not be verified closed except after each drainage of the canal.

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% by weight of the containment air per 24 hours at P_a, 48.3 psig, or

2) Less than or equal to L_t, 0.036% by weight of the containment air per 24 hours at a reduced pressure of P_t, 24.05 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 the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or 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 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations 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 Part 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 a pressure not less than either P_a, 48.3 psig, or at P_t, 24.05 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;

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 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 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 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) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the appropriate following equation:

$| L_c - (L_{am} + L_o) | \leq 0.25$ La or $| L_c - (L_{tm} + L_o) | \leq 0.25 L_t$

where L_{am} or L_{tm} is the measured Type A test leakage and L_o is the superimposed leak;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a; or 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48.3 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Containment ventilation isolation valves with resilient material seals,

3) Safety Injection Valves 1-8802A, 1-8802B and 1-8840, and

4) Containment Spray Valves 1HV-4776, 1HV-4777, 1CT-142, and 1CT-145.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Containment ventilation isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.3 or 4.6.1.7.4, as applicable;

g. Safety Injection Valves 1-8802A, 1-8802B, and 1-8840 shall be demonstrated OPERABLE by performance of a steam leakage measurement, while pressurized to a pressure not less than Pa, [48.3]psig, at intervals no greater than 92 days.

h. Containment Spray Valves 1HV-4776, 1HV-4777, 1CT-142, and 1CT-145 shall be leak tested with water, at a pressure not less than Pa, [48.3]psig, at intervals no greater than 24 months; and

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, 48.3 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; and

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:

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 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to 48.3 psig;

b. By conducting overall air lock leakage tests at not less than P_a, 48.3 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.

**This represents an exemption to 10 CFR 50 Appendix J, paragraph III.D.2(b)(ii).

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INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between [-0.5] and [1.5] 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 the following locations and shall be determined at least once per 24 hours:

Location

a. _____

b. _____

c. _____

d. _____

e. _____

Note: Minimum of three elevations required.

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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 4.6.1.6.1.

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

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1. Containment Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports. Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment and hydrogen ventilation isolation valve shall be OPERABLE and:

a. Each 48-inch and 12-inch containment and hydrogen purge supply and exhaust isolation valve shall be locked closed, and

b. The 18-inch containment pressure relief discharge isolation valve(s) may be open for up to 90 hours during a calendar year.

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

ACTION:

a. With any 48-inch or 12-inch containment or hydrogen purge supply and/or exhaust isolation valve open or not locked closed, lock close that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 18-inch containment pressure relief discharge isolation valve(s) open for more than 90 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment pressure relief discharge isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.3 or with the containment and hydrogen purge supply or exhaust isolation valve(s) having a measured leakage rate in excess of the limit of Specification 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch and 12-inch containment and hydrogen purge supply and exhaust isolation valve shall be verified to be locked closed at least once per 31 days.

4.6.1.7.2 The cumulative time that all 18-inch pressure relief discharge isolation valves have been open during a calendar year shall be determined at least once per 7 days.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.3 At least once per 184 days on a STAGGERED TEST BASIS, the inboard and outboard isolation valves with resilient material seals in each locked closed 48-inch and 12-inch containment and hydrogen purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

4.6.1.7.4 At least once per 92 days each 18-inch containment pressure relief discharge isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.06 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 and manually transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Containment Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying that in the test mode each train provides a discharge flow through the test header of greater than or equal to 5800 gpm with the pump eductor line open when tested pursuant to Specification 4.0.5;

c. 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, and

   2) Verifying that each spray pump starts automatically on a Containment Spray Actuation or Safety Injection test signal.

d. 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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SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4900 and 5167 gallons of between 28 and 30% by weight NaOH solution, and

b. Four spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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

ACTION:

With the Spray Additive 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 Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. 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.

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 Containment Spray Actuation test signal; and

d. At least once per 5 years by verifying:

1) The flow path through the Spray Additive supply line, and

2) RWST test water flow rates of between 50 GPM and 100 GPM through the eductor test loop of each of the Spray Additive System.

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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 containment 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:

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 each 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 containment 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.

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

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SURVEILLANCE REQUIREMENTS (Continued)

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

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each pressure relief discharge valve actuates to its 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

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
1. Phase "A" Isolation Valves				
1HV-2154	20	Feedwater Sample (FW to Stm Gen #1)	5	Note 1
1HV-2155	22	Sample (FW to Stm Gen #2)	5	Note 1
1HV-2399	27	Blowdown From Steam Generator #3	5	Note 1
1HV-2398	28	Blowdown From Steam Generator #2	5	Note 1
1HV-2397	29	Blowdown From Steam Generator #1	5	Note 1
1HV-2400	30	Blowdown From Steam Generator #4	5	Note 1
1-8152	32	Letdown Line to Letdown Heat Exchanger	10	C
1-8160	32	Letdown Line to Letdown Heat Exchanger	10	C
1-8890A	35	RHR to Cold Leg Loops #1 & #2 Test Line	15	Note 2
1-8890B	36	RHR to Cold Leg Loops #3 & #4 Test Line	15	Note 2
1-8047	41	Reactor Makeup Water to Pressure Relief Tank & RC Pump Stand Pipe	10	C
1-8843	42	SI to RC System Cold Leg Loops #1, #2, #3, #4 Test Line	10	Note 2
1-8881	43	SI to RC System Hot Leg Loops #2 & #3 Test Line	10	Note 2

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
1. Phase "A" Isolation Valves (Continued)				
1-8824	44	SI to RC System Hot Leg Loops #1 & #4 Test Line	10	Note 2
1-8823	45	SI to RC System Cold Leg Loops #1, #2, #3, & #4 Test Line	10	Note 2
1-8100	51	Seal Water Return and Excess Letdown	10	C
1-8112	51	Sea Water Return and Excess Letdown	10	C
1-7136	52	RCDT Heat Exchanger to Waste Hold Up Tank	10	C
LCV-1003	52	RCDT Heat Exchanger to Waste Hold Up Tank	10	C
1HV-5365	60	Demineralized Water Supply	5	C
1HV-5366	60	Demineralized Water Supply	5	C
1HV-5157	61	Containment Sump Pump Discharge	5	C
1HV-5158	61	Containment Sump Pump Discharge	5	C
1HV-3487	62	Instrument Air to Containment	5	C
1-8825	63	RHR to Hot Leg Loops #2 & #3 Test Line	15	Note 2
1HV-2405	73	Sample from Steam Generator #1	5	Note 1
1HV-4170	74	RC Sample From Hot Legs	5	C

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
1. Phase "A" Isolation Valves (Continued)				
1HV-4168	74	RC Sample From Hot Leg #1	5	C
1HV-4169	74	RC Sample From Hot Leg #4	5	C
1HV-2406	76	Sample from Steam Generator #2	5	Note 1
1HV-4167	77	Pressurizer Liquid Space Sample	5	C
1HV-4166	77	Pressurizer Liquid Space Sample	5	C
1HV-4176	78	Pressurizer Steam Space Sample	5	C
1HV-4165	78	Pressurizer Steam Space Sample	5	C
1HV-2407	79	Sample from Steam Generator #3	5	Note 1
1HV-4175	80	Accumulators	5	C
1HV-4171	80	Sample from Accumulator #1	5	C
1HV-4172	80	Sample from Accumulator #2	5	C
1HV-4173	80	Sample from Accumulator #3	5	C
1HV-4174	80	Sample from Accumulator #4	5	C
1HV-7311	81	RC PASS Sample Discharge to RCDT	5	C
1HV-7312	81	RC PASS Sample Discharge to RCDT	5	C

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
2. Phase "A" Isolation Valves (Continued)				
1HV-2408	82	Sample from Steam Generator #4	5	Note 1
1-8871	83	Accumulator Test and Fill	10	C
1-8888	83	Accumulator Test and Fill	10	C
1-8964	83	Accumulator Test and Fill	10	C
1HV-5556	84	Containment Air PASS Return	5	C
1HV-5557	84	Containment Air PASS Return	5	C
1HV-5544	94	Radiation Monitoring Sample	5	C
1HV-5545	94	Radiation Monitoring Sample	5	C
1HV-5558	97	Containment Air PASS Inlet	5	C
1HV-5559	97	Containment Air PASS Inlet	5	C
1HV-5560	100	Containment Air PASS Inlet	5	C
1HV-5561	100	Containment Air PASS Inlet	5	C
1HV-5546	102	Radiation Monitoring Sample Return	5	C
1HV-5547	102	Radiation Monitoring Sample Return	5	C
1-8880	104	N_2 Supply to Accumulators	10	C

TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

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VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
1. Phase "A" Isolation Valves (Continued)				
1-7126	105	H_2 Supply to RC Drain Tank	10	C
1-7150	105	H_2 Supply to RC Drain Tank	10	C
1HV-4710	111	CC Supply to Excess Letdown & RC Drain Tank Heat Exchanger	5	Note 1
1HV-4711	112	CC Return from Excess Letdown & RC Drain Tank Heat Exchanger	5	Note 1
1HV-3486	113	Service Air to Containment	5	C
1HV-4725	114	Containment CCW Drain Tank Pumps Discharge	5	C
1HV-4726	114	Containment CCW Drain Tank Pumps Discharge	5	C
1-8027	116	Nitrogen Supply to PRT	10	C
1-8026	116	Nitrogen Supply to PRT	10	C
1HV-6084	120	Chilled Water Supply to Containment Coolers	10	C
1HV-6082	121	Chilled Water Return From Containment Coolers	10	C
1HV-6083	121	Chilled Water Return From Containment Coolers	10	C
1HV-4075B	124	Fire Protection System Isolation	10	C
1HV-4075C	124	Fire Protection System Isolation	10	C

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
2. Phase "B" Isolation Valves				
1HV-4708	117	CC Return from RCP's Motors	10	C
1HV-4701	117	CC Return from RCP's Motors	10	C
1HV-4700	118	CC Supply to RCP's Motors	10	C
1HV-4709	119	CC Return From RCP's Thermal Barrier	10	C
1HV-4696	119	CC Return From RCP's Thermal Barrier	10	C
3. Containment Ventilation Isolation Valves				
1HV-5542	58	Hydrogen Purge Supply	10	C
1HV-5543	58	Hydrogen Purge Supply	10	C
1HV-5563	58	Hydrogen Purge Supply	10	C
1HV-5540	59	Hydrogen Purge Exhaust	10	C
1HV-5541	59	Hydrogen Purge Exhaust	10	C
1HV-5562	59	Hydrogen Purge Exhaust	10	C
1HV-5536	109	Containment Purge Air Supply	5	C
1HV-5537	109	Containment Purge Air Supply	5	C
1HV-5538	110	Containment Purge Air Exhaust	5	C
1HV-5539	110	Containment Purge Air Exhaust	5	C
1HV-5548	122	Containment Pressure Relief	3	C

TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

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VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
3. Containment Ventilation Isolation Valves (Continued)				
1HV-5549	122	Containment Pressure Relief	3	C
4. Manual Valves				
1MS-711	4a	TDAFW Pump Warm-up Valve	N.A.	Note 1, 11
1MS-390	5a	N_2 Supply to Steam Generator #1	N.A.	Note 1
1MS-387	9a	N_2 Supply to Steam Generator #2	N.A.	Note 1
1MS-384	13a	N_2 Supply to Steam Generator #3	N.A.	Note 1
1MS-712	17a	TDAFW Pump Warm-up Valve	N.A.	Note 1, 11
1MS-393	18a	N_2 Supply to Steam Generator #4	N.A.	Note 1
1FW-106	20b	N_2 Supply to Steam Generator #1	N.A.	Note 1
1FW-104	22b	N_2 Supply to Steam Generator #2	N.A.	Note 1
1FW-110	24	Secondary Sampling	N.A.	Note 1
1FW-102	24b	N_2 Supply to Steam Generator #3	N.A.	Note 1
1FW-119	26	Secondary Sampling	N.A.	Note 1
1FW-108	26b	N_2 Supply to Steam Generator #4	N.A.	Note 1
1-7135	52	RCDT Heat Exchanger to Waste Holdup Tank	N.A.	C
1SF-011	56	Refueling Water Purification to Refueling Cavity	N.A.	C

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
4. Manual Valves (Continued)				
1SF-012	56	Refueling Water Purification to Refueling Cavity	N.A.	C
1SF-021	67	Refueling Cavity to Refueling Water Purification Pump	N.A.	C
1SF-022	67	Refueling Cavity to Refueling Water Purification Pump	N.A.	C
1SF-053	71	Refueling Cavity Skimmer Pump Discharge	N.A.	C
1SF-054	71	Refueling Cavity Skimmer Pump Discharge	N.A.	C
1HV-2333B	2	MSIV Bypass from Steam Generator #1	N.A.	Note 1, 6
1HV-2334D	7	MSIV Bypass from Steam Generator #2	N.A.	Note 1, 6
1HV-2335B	11	MSIV Bypass from Steam Generator #3	N.A.	Note 1, 6
1HV-2336B	15	MSIV Bypass from Steam Generator #4	N.A.	Note 1, 6
5. Power-Operated Isolation Valves				
1HV-2452-1	4	Main Steam to Aux. FPT From Steam Line #1	N.A.	Note 1
1PV-2325	5	Steam Generator #1 Atmospheric Relief	N.A.	Note 1
1PV-2326	9	Steam Generator #2 Atmospheric Relief	N.A.	Note 1

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
5. Power-Operated Isolation Valves (Continued)				
1PV-2327	13	Steam Generator #3 Atmospheric Relief	N.A.	Note 1
1HV-2452-2	17	Main Steam to Aux. FPT From Steam Line #4	N.A.	Note 1
1PV-2328	18	Steam Generator #4 Atmospheric Relief	N.A.	Note 1
1HV-2491A	20a	Auxiliary Feedwater to Steam Generator #1	N.A.	Note 1
1HV-2491B	20a	Auxiliary Feedwater to Steam Generator #1	N.A.	Note 1
1HV-2492A	22a	Auxiliary Feedwater to Steam Generator #2	N.A.	Note 1
1HV-2492B	22a	Auxiliary Feedwater to Steam Generator #2	N.A.	Note 1
1HV-2493A	24a	Auxiliary Feedwater to Steam Generator #3	N.A.	Note 1
1HV-2493B	24a	Auxiliary Feedwater to Steam Generator #3	N.A.	Note 1
1HV-2494A	26a	Auxiliary Feedwater to Steam Generator #4	N.A.	Note 1
1HV-2494B	26a	Auxiliary Feedwater to Steam Generator #4	N.A.	Note 1
1-8701B	33	RHR From Hot Leg Loop #4	N.A.	Note 5
1-8701A	34	RHR From Hot Leg Loop #1	N.A.	Note 5
1-8809A	35	RHR to Cold Leg Loops #1 and #2	N.A.	Note 4
1-8809B	36	RHR to Cold Leg Loops #3 and #4	N.A.	Note 4

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
5. Power-Operated Isolation Valves (Continued)				
1-8801A	42	High Head Safety Injection to Cold Leg Loops #1, #2, #3, & #4	N.A.	Note 7
1-8801B	42	High Head Safety Injection to Cold Leg Loops #1, #2, #3, & #4	N.A.	Note 7
1-8802A	43	SI Injection to Hot Leg Loops #2 and #3	N.A.	Note 8
1-8802B	44	SI Injection to Hot Leg Loops #1 and #4	N.A.	Note 8
1-8835	45	SI Injection to Cold Leg Loops #1, #2, #3, and #4	N.A.	Note 4
1-8351A	47	Seal Injection to RC Pump (Loop #1)	N.A.	C
1-8351B	48	Seal Injection to RC Pump (Loop #2)	N.A.	C
1-8351C	49	Seal Injection to RC Pump (Loop #3)	N.A.	C
1-8351D	50	Seal Injection to RC Pump (Loop #4)	N.A.	C
1HV-4777	54	Containment Spray to Spray Header (Tr. B)	N.A.	Note 3
1HV-4776	55	Containment Spray to Spray Header (Tr. A)	N.A.	Note 3
1-8840	63	RHR to Hot Leg Loops #2 and #3	N.A.	Note 8
1-8811A	125	Containment Recirc. Sump to RHR Pumps (Train A)	N.A.	Note 1, 10

TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
5. Power-Operated Isolation Valves (Continued)				
1-8811B	126	Containment Recirc. Sump to RHR Pumps (Train B)	N.A.	Note 1, 10
1HV-4782	127	Containment Recirc. to Spray Pumps (Train A)	N.A.	Note 1, 10
1HV-4783	128	Containment Recirc. to Spray Pumps (Train B)	N.A.	Note 1, 10
6. Check Valves				
1-8818A	35	RHR to Cold Leg Loop #1	N.A.	Note 2
1-8818B	35	RHR to Cold Leg Loop #2	N.A.	Note 2
1-8818C	36	RHR to Cold Leg Loop #3	N.A.	Note 2
1-8818D	36	RHR to Cold Leg Loop #4	N.A.	Note 2
1-8046	41	Reactor Makeup Water to Pressurizer Relief Tank and RC Pump Stand Pipe	N.A.	C
1-8815	42	High Head Safety Injection to Cold Leg Loops #1, #2, #3 and #4	N.A.	Note 2
1SI-8905A	44	SI to RC System Hot Leg Loop #1	N.A.	Note 2
1SI-8905B	43	SI to RC System Hot Leg Loop #2	N.A.	Note 2
1SI-8905C	43	SI to RC System Hot Leg Loop #3	N.A.	Note 2

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
6. Check Valves (Continued)				
1SI-8905D	44	SI to RC System Hot Leg Loop #4	N.A.	Note 2
1SI-8819A	45	SI to RC System Cold Leg Loop #1	N.A.	Note 2
1SI-8819B	45	SI to RC System Cold Leg Loop #2	N.A.	Note 2
1SI-8819C	45	SI to RC System Cold Leg Loop #3	N.A.	Note 2
1SI-8819D	45	SI to RC System Cold Leg Loop #4	N.A.	Note 2
1-8381	46	Charging Line to Regenerative Heat Exchanger	N.A.	C
1CS-8368A	47	Seal Injection to RC Pump (Loop #1)	N.A.	C
1CS-8368B	48	Seal Injection to RC Pump (Loop #2)	N.A.	C
1CS-8368C	49	Seal Injection to RC Pump (Loop #3)	N.A.	C
1CS-8368D	50	Seal Injection to RC Pump (Loop #4)	N.A.	C
1CS-8180	51	Seal Water Return and Excess Letdown	N.A.	C
1CT-145	54	Containment Spray to Spray Header (Tr. B)	N.A.	Note 3
1CT-142	55	Containment Spray to Spray Header (Tr. A)	N.A.	Note 3
1CI-030	62	Instrument Air to Containment	N.A.	C
1-8841A	63	RHR to Hot Leg Loop #2	N.A.	Note 2

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
6. Check Valves (Continued)				
1-8841B	63	RHR to Hot Leg Loop #3	N.A.	Note 2
1SI-8968	104	N_2 Supply To Accumulators	N.A.	C
1CA-016	113	Service Air to Containment	N.A.	C
1CC-629	117	CC Return from RCP's Motors	N.A.	C
1CC-713	118	CC Supply to RCP's Motors	N.A.	C
1CC-831	119	CC Return from RCP's Thermal Barrier	N.A.	C
1CH-024	120	Chilled Water Supply to Containment Coolers	N.A.	C
7. Steam Line Isolation Signal				
1HV-2333A	1	MSIV #1	5	Note 1, 9, 12
1HV-2409	3	Drain From Main Steam Line #1	5	Note 1
1HV-2334A	6	MSIV #2	5	Note 1, 9, 12
1HV-2410	8	Drain From Main Steam Line #2	5	Note 1
1HV-2335A	10	MSIV #3	5	Note 1, 9, 12
1HV-2411	12	Drain From Main Steam Line #3	5	Note 1
1HV-2336A	14	MSIV #4	5	Note 1, 9, 12
1HV-2412	16	Drain From Main Steam Line #4	5	Note 1

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
8. Feedwater Line Isolation Signal				
1HV-2134	19	Feedwater Isolation Steam Generator #1	5	Note 1, 12
1FV-2193	20c	Feedwater Bypass Line	5	Note 1, 12
1HV-2185	20d	Feedwater Isolation Bypass Line	5	Note 1, 12
1HV-2135	21	Feedwater Isolation Steam Generator #2	5	Note 1, 12
1FV-2194	22c	Feedwater Bypass Line	5	Note 1, 12
1HV-2186	22d	Feedwater Isolation Bypass Line	5	Note 1, 12
1HV-2136	23	Feedwater Isolation Steam Generator #3	5	Note 1, 12
1FV-2195	24c	Feedwater Bypass Line	5	Note 1, 12
1HV-2187	24d	Feedwater Isolation Bypass Line	5	Note 1, 12
1HV-2137	25	Feedwater Isolation Steam Generator #4	5	Note 1, 12
1FV-2196	26d	Feedwater Bypass Line	5	Note 1, 12
1HV-2188	26e	Feedwater Isolation Bypass Line	5	Note 1, 12
9. Safety Injection Actuation Isolation				
1-8105	46	Charging Line to Regenerative Heat Exchanger	10	C

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
10. Relief Valves				
1-8708B	33	RHR From Hot Leg Loop #4	N.A.	Note 5
1-8708A	34	RHR From Hot Leg Loop #1	N.A.	Note 5
1MS-021	5b	Main Steam Safety Valve S.G. #1	[N.A.]	Note 1, 12
1MS-022	5b	Main Steam Safety Valve S.G. #1	[N.A.]	Note 1, 12
1MS-023	5b	Main Steam Safety Valve S.G. #1	[N.A.]	Note 1, 12
1MS-024	5b	Main Steam Safety Valve S.G. #1	[N.A.]	Note 1, 12
1MS-025	5b	Main Steam Safety Valve S.G. #1	[N.A.]	Note 1, 12
1MS-058	9b	Main Steam Safety Valve S.G. #2	[N.A.]	Note 1, 12
1MS-059	9b	Main Steam Safety Valve S.G. #2	[N.A.]	Note 1, 12
1MS-060	9b	Main Steam Safety Valve S.G. #2	[N.A.]	Note 1, 12
1MS-061	9b	Main Steam Safety Valve S.G. #2	[N.A.]	Note 1, 12
1MS-062	9b	Main Steam Safety Valve S.G. #2	[N.A.]	Note 1, 12
1MS-093	13b	Main Steam Safety Valve S.G. #3	[N.A.]	Note 1, 12
1MS-094	13b	Main Steam Safety Valve S.G. #3	[N.A.]	Note 1, 12
1MS-095	13b	Main Steam Safety Valve S.G. #3	[N.A.]	Note 1, 12

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TABLE 3.6.1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NO.	FSAR TABLE REFERENCE NO.*	LINE OR SERVICE	MAXIMUM ISOLATION TIME (Seconds)	TYPE LEAK TEST REQUIREMENTS
10. Relief Valves (Continued)				
1MS-096	13b	Main Steam Safety Valve S.G. #3	[N.A.]	Note 1, 12
1MS-097	13b	Main Steam Safety Valve S.G. #3	[N.A.]	Note 1, 12
1MS-129	18b	Main Steam Safety Valve S.G. #4	[N.A.]	Note 1, 12
1MS-130	18b	Main Steam Safety Valve S.G. #4	[N.A.]	Note 1, 12
1MS-131	18b	Main Steam Safety Valve S.G. #4	[N.A.]	Note 1, 12
1MS-132	18b	Main Steam Safety Valve S.G. #4	[N.A.]	Note 1, 12
1MS-133	18b	Main Steam Safety Valve S.G. #4	[N.A.]	Note 1, 12
1RC-036	41	RMUW to PRT & RCP Standpipe	N.A.	C
1WP-7176	52	RCDT HX to Waste Holdup Tank	N.A.	C
1DD-430	60	Demineralized Water Supply	N.A.	C
1VD-907	61	Cont. Sump Pump Discharge	N.A.	C
1PS-193	80	Sample from Accumulators	N.A.	C
1CC-1067	114	Containment CCW Drain Tank Pump/Discharge	N.A.	C
1CH-271	120	Chilled Water Supply to Cont. Coolers	N.A.	C
1CH-272	121	Chilled Water Supply from Cont. Coolers	N.A.	C

TABLE 3.6.1 (Continued)

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TABLE NOTATIONS

*Identification code for containment penetration and associated isolation valves in FSAR Tables 6.2.4-1, 6.2.4-2, and 6.2.4-3.

Note 1: These are closed systems which meet the requirements of NUREG-0800 Section 6.2.4, II.6, paragraph o. These valves are therefore not required to be leak tested.

Note 2: These valves inside containment are part of closed systems outside containment which are in service post accident at a pressure in excess of containment design pressure and satisfy single failure criterion. These valves are therefore not required to be leak tested.

Note 3: These are closed systems outside containment which are in service post accident and have a water-filled loop seal on the containment side of the valves for a period greater than 30 days following the accident. These valves are therefore leak rate tested with water at a pressure of P_a.

Note 4: These ESF valves are normally open and remain open during post-accident conditions. Postaccident they are continually pressurized in excess of containment pressure from an ESF source which meets the single failure criterion. These valves are therefore not required to be leak tested.

Note 5: An effective fluid seal on these penetrations is provided by the suction sources to the residual heat removal pumps during and following an accident. In addition, these containment isolation valves are non-automatic, are not required to operate postaccident and are located inside containment. These valves are therefore not required to be leak tested.

Note 6: All four MSIV bypass valves are locked closed in Mode 1. During Modes 2, 3, and 4, one MSIV bypass valve may be opened provided the other three MSIV bypass valves are locked closed and their associated MSIVs are closed.

Note 7: These are parallel ESF valves that are normally closed, but are designed to open during post-accident conditions. Failure of one valve to open will not prevent system pressurization on both sides of both valves in excess of containment pressure. These valves are therefore not required to be leak tested.

Note 8: These valves located outside containment are normally closed and see a pressure in excess of containment pressure in post-accident conditions. A valve stem leakage check will be performed on a quarterly basis to assure no significant stem leakage would occur in post-accident conditions.

Note 9: These valves require steam to be tested are are thus not required to be tested until the plant is in MODE 3.

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TABLE 3.6.1 (Continued)

TABLE NOTATIONS

Note 10: These valves will have water against them during post-accident conditions to preclude any release of containment atmosphere to the environment.

Note 11: These valves are normally locked closed and are open only to warm-up the steam supply lines prior to normal surveillance testing.

Note 12: These valves are included for table completeness, the requirements of Specification 3.6.3 do not apply. Instead, the requirements of Specification 3.7.1.1, 3.7.1.5 and 3.7.1.6 apply for main steam safety valves, main steam isolation valves and feedwater isolation valved, respectively.

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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 monitor trains (with at least one channel per train) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor train inoperable, restore the inoperable monitor train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours. The provisions of 3.0.4 are not applicable.

b. With both hydrogen monitor trains inoperable, restore at least one monitor train to OPERABLE status within 72 hours 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:

a. At least once per 31 days by performing a channel check, and

b. At least once per 92 days on a staggered test basis by performing a channel sequence using sample gas in accordance with the manufacturer's recommendations and by verifying that the current calibration constants are contained in the microprocessor data base.

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ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent 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 Hydrogen 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, and

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 all 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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3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four 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 Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; 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 requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (± 1%)*	ORIFICE SIZE
LOOP 1	LOOP 2	LOOP 3	LOOP 4		
1MS-021,	058,	093,	029	1185 psig	16 in^2
1MS-022,	059,	094,	130	1195 psig	16 in^2
1MS-023,	060,	095,	131	1205 psig	16 in^2
1MS-024,	061,	096,	132	1215 psig	16 in^2
1MS-025,	062,	097,	133	1235 psig	16 in^2

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

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 or associated flow path inoperable, restore the required auxiliary feedwater pumps or associated flow path 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 feedwater pumps or associated flow paths inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps or associated flow paths 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 and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to [Later] psig at a flow of greater than or equal to 430 gpm;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to [Later] psig at a flow of greater than or equal to 860 gpm when the secondary steam supply pressure is greater than 532 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is in standby for auxiliary feedwater automatic initiation or when above 10% RATED THERMAL POWER.

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 Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal. The provisions of Specification 4.0.4 are not applicable to the turbine driven auxiliary feedwater pump for entry into Mode 3.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least 282,540 gallons (_____% of span) of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the Station Service Water (SSW) system as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The SSW system shall be demonstrated OPERABLE at least once per 12 hours whenever the SSW system is being used as an alternate supply source to the auxiliary feedwater pumps by verifying the SSW system operable and each motor operated valve between the SSW system and each operable auxiliary feedwater pump is operable.

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SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/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 Radioactivity Determination*	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines.
	b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 10 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

DRAFT

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

DRAFT

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 generator 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 Each component cooling water loop 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; and

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

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated engineering safety feature actuation signal, and

   2) Each Component Cooling Water System pump starts automatically on a safety injection test signal.

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3/4.7.4 STATION SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent service water loops shall be OPERABLE.

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

ACTION:

With only one service 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

4.7.4 Each service water loop 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; and

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

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

   2) Each station service water system pump starts automatically on a Safety Injection test signal.

DRAFT

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

a. A minimum water level at or above elevation 770 Mean Sea Level, USGS datum, and

b. A station service water intake temperature of less than or equal to 102°F, and

c. A maximum average sediment depth of less than or equal to 1.5 feet in the service water intake channel.

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

ACTION:

a. With the requirements for water level and intake temperature not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average sediment depth in the service water channel greater than 1.5 feet, the channel shall be cleaned within [30] days to reduce the average sediment depth to less than 0.5 feet.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the station service water intake temperature and UHS water level to be within their limits.

b. At least once per 12 months by visually inspecting the dam and verifying no abnormal degradation or erosion, and

c. At least once per 12 months by verifying that the average sediment depth in the service water intake channel is less than or equal to 1.5 feet.

3/4.7.6 FLOOD PROTECTION

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LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for all safety-related systems, components, and structures when the water level of the Squaw Creek Reservoir (SCR) exceeds 777.5 Mean Sea Level, USGS datum.

APPLICABILITY: At all times.

ACTION:

With the water level of SCR above elevation 777.5 Mean Sea Level, USGS datum, initiate and complete within 2 hours, the flood protection measures verifying that any equipment which is to be opened or is opened for maintenance is isolated from the SCR by isolation valves, or stop gates, or is at an elevation above 790 feet.

SURVEILLANCE REQUIREMENTS

4.7.6 The water level of SCR shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 776 Mean Sea Level, USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 776 Mean Sea Level, USGS datum.

c. With the water level of SCR above 777.0 Mean Sea Level, USGS datum, verify flood protection measures are in effect by verifying once per 12 hours that flow paths from the SCR which are open for maintenance are isolated from the SCR by isolation valves, or stop gates, or are at an elevation above 790 feet.

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3/4.7.7 CONTROL ROOM HVAC SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room HVAC trains shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

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

b. With both control room HVAC trains inoperable, or with the OPERABLE control room HVAC trains required to be in the emergency 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.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room HVAC train 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 absorbers and verifying that the train operates for at least 10 continuous hours with the emergency pressurization unit heaters operating;

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:

1) Verifying that the filtration unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% by using the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revisions 2, March 1978, and the emergency filtration unit flow rate is 8000 cfm ± 10%, and the emergency pressurization unit flow rate is 800 cfm ± 10%;

2) 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, for a methyl iodide penetration of less than 0.2%; and

3) Verifying an emergency filtration unit flow rate of 8000 cfm ± 10% and an emergency pressurization unit flow rate of 800 cfm ± 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, for a methyl iodide penetration of less than 0.2%;

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 7.7 inches Water Gauge while operating the emergency filtration unit at a flow rate of 8000 cfm ± 10%, and is less than 9.25 inches Water Gauge while operating the emergency pressurization unit at a flow rate of 800 cfm ± 10%;

2) Verifying that on a Safety Injection, Loss-of-Offsite Power, Intake Vent-High Radiation, or Plant Vent-High Radiation test signal, the train automatically switches into the emergency recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;

3) Verifying that the emergency pressurization unit maintains the control room at a positive pressure of greater than or equal

to 1/8 inch Water Gauge relative to the adjacent areas, including the outside atmosphere, at a flow rate of less than or equal to 800 cfm during system operation;

4) Verifying that the heaters in the emergency pressurization units dissipate 10 ± 1 kW when tested in accordance with ANSI N510-1975; and

5) Verifying that on a High Chlorine test signal, the train automatically switches into the isolation mode of operation with flow through the emergency filtration HEPA filters and charcoal adsorber banks within 10 seconds.

e. After each complete or partial replacement of a HEPA filter bank in the emergency filtration unit(s), by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the unit at a flow rate of 8000 cfm ± 10%;

f. After each complete or partial replacement of a charcoal adsorber bank in the emergency filtration unit(s), by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 8000 cfm ± 10%;

g. After each complete or partial replacement of a HEPA filter bank in the emergency pressurization unit(s), by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the unit at a flow rate of 800 cfm ± 10%; and

h. After each complete or partial replacement of a charcoal absorber bank in the emergency pressurization unit(s), by verifying that the unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the unit at a flow rate of 800 cfm ± 10%.

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3/4.7.8 PRIMARY PLANT VENTILATION SYSTEM - ESF FILTRATION UNITS

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ESF Filtration Units shall be OPERABLE.

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

ACTION:

With one ESF Filtration Unit inoperable, restore the inoperable ESF Filtration Unit 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.

SURVEILLANCE REQUIREMENTS

4.7.8 Each ESF Filtration Unit 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 each ESF Filtration Unit operates for at least 10 continuous hours with the heaters operating;

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:

   1) Verifying that each ESF Filtration Unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1.0% by using the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and verifying the flow rate is 15,000 cfm ± 10% per ESF Filtration Unit when tested in accordance with ANSI N510-1975; and

   2) 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, for a methyl iodide penetration of less than 1.0%.

SURVEILLANCE REQUIREMENTS (Continued)

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, for a methyl iodide penetration of less than 1.0%;

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 8.25 inches Water Gauge while operating each ESF Filtration Unit at a flow rate of 15,000 cfm ± 10%,

   2) Verifying that each ESF Filtration Unit starts on a Safety Injection test signal, and

   3) Verifying that the heaters dissipate 100 ± 5 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the associated ESF Filtration Unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1.0% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the associated ESF Filtration Unit at a flow rate of 15,000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the associated ESF Filtration Unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1.0% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the associated ESF filtration unit system at a flow rate of 15,000 cfm ± 10%.

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed would have no adverse effect on any safety-related system.

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 on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

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

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. 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. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of 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%

*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.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. 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 4.7.9f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan for each type shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber 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 4.7.9f., 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

e. Functional Tests (Continued)

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 4.7.9f. 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 at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may 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; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

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

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 snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are 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 lock up 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 4.7.9e. 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 results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

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FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 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 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately 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

4.7.10.1 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 microCurie per test sample.

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

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

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 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

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.

4.7.10.3 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 microCurie of removable contamination.

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3/4.7.11 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.11 The maximum temperature limit for normal conditions of each area shown in Table 3.7-3 shall not be exceeded for more than 8 hours and the maximum temperature for abnormal conditions of each area given in Table 3.7-3 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the maximum temperature limit(s) for abnormal conditions shown in Table 3.7-3 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the maximum temperature limit(s) for abnormal conditions shown in Table 3.7-3, prepare and submit a Special Report as required by ACTION a. above and within 4 hours either restore the area(s) to within the maximum temperature limit(s) for abnormal conditions, or

   1) Declare equipment in the affected area(s) INOPERABLE; or,

   2) Verify that the qualification envelope for the affected equipment has not been exceeded, or declare the affected equipment which exceeded the qualification envelope INOPERABLE; or,

   3) Perform an analysis that justifies continued operation.

SURVEILLANCE REQUIREMENTS

4.7.11 The temperature in each of the areas shown in Table 3.7-3 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7-3

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F)	
	Normal Conditions	Abnormal Conditions
1. Electrical and Control Building		
Normal Areas	104	[122]
Control Rooms	80	[80]
UPS/Battery Rooms	104	[104]
2. Fuel Building		
Normal Areas	104	[122]
Spent Fuel Pool Cooling Pump Rooms	122	[122]
3. Safegaurds Building		
Normal Areas	104	[122]
AF, RHR, SI, Containment Spray Pump Rooms	122	[122]
Diesel Generator Area	129	[129]
Day Tank Room	122	[122]
4. Auxiliary Building		
Normal Areas	104	[122]
CCW, CCP Pump Rooms	122	[122]
5. Service Water Intake Structure	127	[127]
6. Containment Building		
General Areas	120	[120]
CRDM Platform	140	[140]
Detector Well/Reactor Cavity Exhaust	135	[175]
R.C. Pipe Penetrations	200	[200]
CRDM Shroud Exhaust	163	[163]

3/4.8 ELECTRICAL POWER SYSTEMS

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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 separate day fuel tank containing a minimum volume of 1440 gallons of fuel,

   2) A separate Fuel Storage System containing a minimum volume of 88,175 gallons of fuel, and

   3) A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either diesel generator has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator, separately, within 24 hours.# Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With either diesel generator inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours unless the diesel is already operating and loaded.# Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

#During performance of surveillance activities as a requirement for ACTION statements, the air-roll test shall not be performed.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater 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 separately by performing Surveillance Requirements 4.8.1.1.2a.4 and 4.8.1.1.2.a.5) within 8 hours 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 4.8.1.1.1a. within 1 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 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, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) the 6.9 kV safeguards bus power supply from the preferred offsite source to the alternate offsite source.

#During performance of surveillance activities as a requirement for ACTION statements, the air-roll test shall not be performed.

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 and engine-mounted fuel tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 441 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be [6900 ± 690] volts and 60 ± 1.2 Hz within 10 seconds* after the start signal**. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Start-up transformer secondary winding undervoltage, or

c) Simulated loss of preferred offsite power by itself, or

d) Simulated safeguards bus undervoltage, or

e) Safety Injection Actuation test signal in conjunction with loss of preferred offsite power, or

f) Safety Injection Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded to between [6,800] and 7,000 kW# in less than or equal to 60 seconds*, and operates at this load condition for at least 60 minutes**, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

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*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall include the acceleration to rated speed in less than or equal to 10 seconds and be preceded by an engine pre-lube period and/or other warmup procedures such as gradual loading (>80 sec) recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

**During performance of surveillance activities as a requirement for ACTION statements, the air-roll test shall not be performed.

#This band is meant as guidance to avoid routine overloading of diesel generator. Loads in excess of the band or momentary variations due to changing bus loads shall not invalidate the test.

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 fuel tanks;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil in accordance with ASTM-D4057-1981 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-1981 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to [26] degrees but less than or equal to [38] degrees;

b) A kinematic viscosity at 40°F of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

c) A flash point equal to or greater than 125°F;

d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-1982;

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-1981 are met when tested in accordance with ASTM-D975-1981 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-1979 or ASTM-D2622-1982.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-1978, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-1978, Method A;

f. 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 [783] kW while maintaining voltage at 6900 ± 690 volts and frequency at 60 ± 1.2 Hz;

3) Verifying the generator capability to reject a load of 7000 kW without tripping. The generator voltage shall not exceed 7590 volts during and following the load rejection;

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

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

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 shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on a Safety Injection Actuation 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 generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection Actuation test signal, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

*For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

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 sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz during this test; and

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) 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 an indicated 7600 - 7700 kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 6800 - 7000 kW#. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2e.5)b);*

8) Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 7,000 kW;

9) 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.

#This band is meant as guidance to avoid routine overloading of the diesel generator. Loads in excess of the band or momentary variations due to changing bus loads shall not invalidate the test.

*If Specification 4.8.1.1.2e.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated between 6800 - 7000 kW for 1 hour or until operating temperature has stabilized.

10) 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 energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from fuel storage tank to the day tank of its associated diesel via the installed lines;

12) Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within ± 10% of its design interval;

13) Verifying that the following diesel generator lockout features prevent diesel generator starting:

a) Barring device engaged, or

b) Maintenance Lockout Mode.

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 440 rpm in less than or equal to 10 seconds; and

h. At least once per 10 years by:

1) Pumping out each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or 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, when tested pursuant to Specification 4.0.5.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. 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 20 VALID TESTS*	NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 days

TABLE 4.8-2

ADDITIONAL RELIABILITY ACTIONS

NO. OF FAILURES IN LAST 20 VALID TESTS	NO. OF FAILURES IN LAST 100 VALID TESTS	ACTION
3	6	Within 14 days prepare and maintain a report for NRC audit describing the diesel generator reliability improvement program implemented at the site. Minimum requirements for the report are indicated in Attachment 1 to this table.
5	11	Declare the diesel generator inoperable. Perform a re-qualification test program for the affected diesel generator. Requalification test program requirements are indicated in Attachment 2 to this table.

*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, but determined on a per diesel generator basis.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

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ATTACHMENT 1 TO TABLE 4.8-2

REPORTING REQUIREMENT

As a minimum the Reliability Improvement Program report for NRC audit shall include:

a) a summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed

b) analysis of failures and determination of root causes of failures

c) evaluation of each of the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability in Operating Reactors," with respect to their application to the plant

d) identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and (2) achieve a general improvement of diesel generator reliability

e) the schedule for implementation of each action from d) above

f) an assessment of the existing reliability of electric power to engineered-safety-feature equipment

Once a licensee has prepared and maintain an initial report detailing the diesel generator reliability improvement program at his site, as defined above, the licensee need prepare only a supplemental report within 14 days after each failure during a valid demand for so long as the affected diesel generator unit continues to violate the criteria (3/20 or 6/100) for the reliability improvement program remedial action. The supplemental report need only update the failure/demand history for the affected diesel generator unit since the last report for that diesel generator. The supplemental report shall also present an analysis of the failure(s) with a root cause determination, if possible, and shall delineate any further procedural, hardware or operational changes to be incorporated into the site diesel generator improvement program and the schedule for implementation of those changes.

In addition to the above, submit a yearly data report on the diesel generator reliability.

ATTACHMENT 2 TO TABLE 4.8-2

DIESEL GENERATOR REQUALIFICATION PROGRAM

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1. Perform seven consecutive successful demands without a failure within 30 days of diesel generator being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of diesel generator of being restored to operable status.

2. If a failure occurs during the first seven tests in the requalification test program, perform seven successful demands without an additional failure within 30 days of diesel generator of being restored to operable status and fourteen consecutive successful demands without a failure within 75 days of being restored to operable status.

3. If a failure occurs during the second seven tests (tests 8 through 14) of (1) above, perform fourteen consecutive successful demands without an additional failure within 75 days of the failure which occurred during the requalification testing.

4. Following the second failure during the requalification test program, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

5. During requalification testing the diesel generator should not be tested more frequently than at 24-hour intervals.

After a diesel generator has been successfully requalified, subsequent repeated requalification tests will not be required for that diesel generator under the following conditions:

(a) The number of failures in the last 20 valid demands is less than 5.

(b) The number of failures in the last 100 valid demands is less than 11.

(c) In the event that following successful requalification of a diesel generator, the number of failures is still in excess of the remedial action criteria (a and/or b above) the following exception will be allowed until the diesel generator is no longer in violation of the remedial action criteria (a and/or b above).

Requalification testing will not be required provided that after each valid demand the number of failures in the last 20 and/or 100 valid demands has not increased. Once the diesel generator is no longer in violation of the remedial action criteria above the provisions of those criteria alone will prevail.

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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 tank containing a minimum volume of 1440 gallons of fuel,

2) A fuel storage system containing a minimum volume of [88,175] 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, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 2.98 square inch vent. 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 requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

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OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. Train A - 125 volt D.C. Station Batteries BT1ED1 and BT1ED3 and at least one full capacity charger associated with each battery and

b. Train B - 125 volt D.C. Station Batteries BT1ED2 and BT1ED4 and at least one full capacity charger associated with each battery.

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

ACTION:

With one of the required battery trains and/or required full-capacity chargers inoperable, restore the inoperable battery train and/or required full-capacity charger 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.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125 V D.C. station battery 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 128 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

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} ohm, and

3) The average electrolyte temperature of 12 of connected cells is above 70°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 anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, 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 Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity 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 4.8-3

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(6)	> 2.07 volts
Specific Gravity(4)	≥ 1.200(5)	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

TABLE NOTATIONS

(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.

(4) Corrected for electrolyte temperature (reference temperature of 77°F) and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

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SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two 125V D.C. station batteries of one train and at least one associated full-capacity charger for each required battery shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required battery trains and/or required full-capacity chargers inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery train and full-capacity charger to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 2.98 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125V D.C. station batteries and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

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OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Train A A.C. Emergency Busses consisting of:

1) 6900-Volt Emergency Bus 1EA1,
2) 480-Volt Emergency Bus 1EB1 from transformer T1EB1, and
3) 480-Volt Emergency Bus 1EB3 from transformer T1EB3.

b. Train B A.C. Emergency Busses consisting of:

1) 6900-Volt Emergency Bus 1EA2,
2) 480-Volt Emergency Bus 1EB2 from transformer T1EB2, and
3) 480-Volt Emergency Bus 1EB4 from transformer T1EB4.

c. 118-Volt A.C. Instrument Bus 1PC1, 1PC3, and 1EC1 energized from its associated inverter connected to D.C. Bus 1ED1*;

d. 118-Volt A.C. Instrument Bus 1PC2, 1PC4, and 1EC2 energized from its associated inverter connected to D.C. Bus 1ED2*;

e. 118-Volt A.C. Instrument Bus 1EC5 energized from its associated inverter connected to D.C. Bus 1ED3*;

f. 118-Volt A.C. Instrument Bus 1EC6 energized from its associated inverter connected to D.C. Bus 1ED4*;

g. Train A 125-Volt D.C. Busses 1ED1 and 1ED3 energized from Station Batteries BT1ED1 and BT1ED3, respectively; and

h. Train B 125-Volt D.C. Busses 1ED2 and 1ED4 energized from Station Batteries BT1ED2 and BT1ED4, respectively.

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

ACTION:

a. With one of the required trains of A.C. emergency busses not fully energized, reenergize the trains within 8 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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*The inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery train provided: (1) their instrument busses are energized, and (2) the instrument busses associated with the other battery train are energized from their associated inverters and connected to their associated D.C. bus.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With one A.C. instrument bus or two A.C. instrument busses (consisting of one 7.5 KVA protection channel and one 10KVA vital bus of the same train) de-energized, re-energize the A.C. instrument bus(ses) within 2 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 A.C. instrument bus or two A.C. instrument busses (consisting of one 7.5 KVA protection channel and one 10 KVA vital bus of the same train) operating with the associated inverter(s) not connected with the D.C. source(s), or operating with the inverter not supplying the A.C. instrument bus (but with the instrument bus energized from its associated bypass distribution source), energize the A.C. instrument bus(ses) from 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.

d. With one D.C. bus not energized from its associated station battery, reenergize the D.C. bus from its associated station battery within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 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 train of A.C. emergency busses consisting of one 6900-volt and two 480-volt A.C. emergency bus;

b. Two 118-volt A.C. instrument busses (channel-oriented) energized from their associated inverters connected to their respective D.C. busses;

c. One train of A.C. instrument busses consisting of two 118-volt A.C. instrument busses energized from their associated inverters connected to their respective D.C. busses. Busses shall be of the same train as Specifications 3.8.3.2a. and d.; and

d. One train of D.C. busses consisting of two 125-volt D.C. busses energized from their associated battery banks. Busses shall be of the same train as Specifications 3.8.3.2a. and c.

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, and within 8 hours, depressurize and vent the RCS through at least a 2.98 square inch vent.

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

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 The following A.C. feeding the Fuel Transfer System circuits inside primary containment shall be deenergized:

a. Circuit number 4BL in panel MCC 1EB2-3.

b. Circuit number 5D in panel MCC 1EB1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 31 days by verifying that the associated circuit breakers are locked in the open position.

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ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be OPERABLE.

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

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) given in Table 3.8-1 inoperable:

a. Restore the protective device or feeder breaker to OPERABLE status or:

   1. Deenergize the circuit(s) by racking out, locking open, or removing the inoperable circuit breaker or protective device and tripping the associated backup circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the inoperable circuit breaker or protective device racked out, locked open, or removed at least once per 31 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices or feeder breakers in circuits which have their backup circuit breakers tripped and their inoperable circuit breakers racked out, locked open, or removed; or

   2. Deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out, locking open, 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, locked open, 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 or their inoperable circuit breakers racked out, locked open, 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

4.8.4.2 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1) By verifying that the medium voltage 6.9 kV and low voltage 480V switchgear circuit breakers are OPERABLE by selecting, on a rotating basis, at least one or 10% of the circuit breakers whichever is greater of each current rating and performing the following:

      a) A CHANNEL CALIBRATION of the associated protective relays,

      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

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SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, one or 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; and

2) By selecting and functionally testing a representative sample of at least 10% of each type 480 V molded case circuit breakers and of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. 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

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
1. 6.9 KVAC from Switchgears	
a. Switchgear Bus 1A1	RCP #11
1) Primary Breaker 1PCPX1	
a) Relay 50M1-51	
b) Relay 26	
c) Relay 86M	
2) Backup Breakers 1A1-1 or 1A1-2	
a) Relay 51M2	
b) Relay 51 for 1A1-1	
c) Relay 51 for 1A1-2	
d) Relay 86/1A1	
b. Switchgear Bus 1A2	RCP #12
1) Primary Breaker 1PCPX2	
a) Relay 50M1-51	
b) Relay 26	
c) Relay 86M	
2) Backup Breakers 1A2-1 or 1A2-2	
a) Relay 51M2	
b) Relay 51 for 1A2-1	
c) Relay 51 for 1A2-2	
d) Relay 86/1A2	
c. Switchgear Bus 1A3	RCP #13
1) Primary Breaker 1PCPX3	
a) Relay 50M1-51	
b) Relay 26	
c) Relay 86M	
2) Backup Breaker 1A3-1 or 1A3-2	
a) Relay 51M2	
b) Relay 51 for 1A3-1	
c) Relay 51 for 1A3-2	
d) Relay 86/1A3	

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
1. 6.9 KVAC from Switchgears (Continued)	
a. Switchgear Bus 1A4	RCP #14
1) Primary Breaker 1PCPX4	
a) Relay 50M1-51	
b) Relay 26	
c) Relay 86M	
2) Backup Breakers 1A4-1 or 1A4-2	
a) Relay 51M2	
b) Relay 51 for 1A4-1	
c) Relay 51 for 1A4-2	
d) Relay 86/1A4	
2. 480 VAC from Switchgears	
2.1 Device Location - 480V Switchgears 1EB1, 1EB2, 1EB3 and 1EB4	Containment Recirc. Fans and CRDM Vent Fans
a. Primary Breakers - 1FNAV1, 1FNAV2, 1FNAV3, 1FNAV4, 1FNCB1 and 1FNCB2	
b. Backup Breakers - 1EB1-1, 1EB2-1, 1EB3-1 and 1EB4-1	
1) Long Time & Instantaneous Relays*	
$\frac{50/51}{1FNAV1}$ (1EB1-1) $\frac{50/51}{1FNAV2}$ (1EB2-1)	
$\frac{50/51}{1FNAV3}$ (1EB3-1) $\frac{50/51}{1FNAV4}$ (1EB4-1)	
$\frac{50/51}{1FNCB1}$ (1EB3-1) $\frac{50/51}{1FNCB2}$ 1EB4-1)	

*Associated circuit breaker shown in parentheses; e.g., 1EB3-1, is backup to 1FNAV3 and 1FNCB1.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
2. 480 VAC from Switchgears (Continued)	
2) Time Delay Relays	
62-1/1FNAV1 (1EB1-1) 62-1/1FNAV2 (1EB2-1)	
62-1/1FNAV3 (1EB3-1) 62-1/1FNAV4 (1EB4-1)	
62-1/1FNCB1 (1EB3-1) 62 1/1FNCB2 (1EB4-1)	
2.2 Device Location - 480V Switchgear 1EB4	Containment Polar Crane
a. Primary Breaker - 1SCCP1	
b. Backup Breaker - 1EB4-1	
1) 51/1SCCP1	
2) 62/1SCCP1	
3. 480 VAC from Motor Control Centers	
3.1 Device Location	- MCC 1EB1-2 Containment Numbers listed below.
Primary and Backup Breakers	- Both primary and backup breakers have identical trip ratings and are in the same MCC Compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements.

MCC 1EB1-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
4G	THED	Motor Operated Valve 1-TV-4691
4M	THED	Motor Operated Valve 1-TV-4693

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB1-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
3F	THED	Containment Drain Tank Pump-03
9H	THED	Reactor Cavity Sump Pump-01
9M	THED	Reactor Cavity Sump Pump-02
7H	THED	Containment Sump #1 Pump-01
7M	THED	Containment Sump #1 Pump-02
6H	THED	RCP #11 Motor Space Heater-01
6M	THED	RCP #13 Motor Space Heater-03
8B	THED	Incore Detector Drive "A"
8D	THED	Incore Detector Drive "B"
7B	THED	Incore Detector Drive "F"
5D	THED	Fuel Transfer System Reactor Side Cont. PNL FDR-01
3B	THED	Stud Tensioner Hoist Outlet-01
7D	THED	Hydraulic Deck Lift-01
4B	THED	Reactor Coolant Pump Motor Hoist Receptacle-42
8H	THED	RC Pipe Penetration Cooling Unit-01
8M	THED	RC Pipe Penetration Cooling Unit-02
5H	THED	RCP #11 Oil Lift Pump-01
5M	THED	RCP #13 Oil Lift Pump-03
10B	THED	Preaccess Filter Train Package Receptacle - 17

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB1-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
5B	THED	Containment L7g XFMR-14 (PML-C3)
10F	THED	S.G. Wet Layup Circ. Pump 01 (CPI-CFAPRP-01)
12M	THED	S.G. Wet Layup Circ. Pump 03 (CPI-CFAPRP-03)
12H	THFK	Cont. Ltg. Transf. CPI-ELTRNT-28 (AULC-11AMC-12)
6D	THED	Refueling Machine (Manipulator Crane-01)
2M	THED	RC Drain Tank Pump No. 1
2F	THED	Containment Ltg XFMR-16 (PNL C7 & C9)
1M	THED	Containment Ltg XFMR-12 (PNL C1 & C5)
3M	THED	Preaccess Fan No. 11

3.2 Device Location

- MCC 1EB2-2 Containment Numbers listed below.

Primary and Backup Breakers

- Both primary and backup breakers have identical trip ratings and are located in the same MCC compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements.

MCC 1EB2-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
4G	THED	Motor Operated Valve 1-TV-4692
4M	THED	Motor Operated Valve 1-TV-4694

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB2-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
3F	THED	Containment Drain Tank Pump-04
7H	THED	Containment Sump No. 2 Pump-03
7M	THED	Containment Sump No. 2 Pump-04
6H	THED	RCP No. 12 Motor Space Heater-02
6M	THED	RCP No. 14 Motor Space Heater-04
5B	THED	Incore Detector Drive "C"
2B	THED	Incore Detector Drive "D"
7B	THED	Incore Detector Drive "E"
5D	THED	Containment Fuel Storage Crane-01
3B	THED	Stud Tensioner Hoist Outlet-02
4B	THED	Containment Solid Rad Waste Compactor-01
10B	THED	RCC Change Fixture Hoist Drive-01
10F	THED	Refueling Cavity Skimmer Pump-01
12B	THED	Power Receptacles (Cont. El. 841')
1M	THED	S.G. Wet Layup Circ. Pump 02 (CPI-CFAPRP-02)
12M	THED	S.G. Wet Layup Circ. Pump 04 (CPI-CFAPRP-04)
8H	THED	RC Pipe Penetration Fan-03
8M	THED	RC Pipe Penetration Fan-04
5H	THED	RCP #12 Oil Lift Pump-02

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER
AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB2-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
5M	THED	RCP #14 Oil Lift Pump-04
12H	THED	Preaccess Filter Train Package Receptables - 18
6D	THED	Containment Auxiliary Upper Crane-01
2F	THED	Containment Ltg. XFMR-13 (PNL C-2)
7D	THED	Containment Elevator-01
2D	THED	Containment Access Rotating Platform-01
2M	THED	Reactor Coolant Drain Tank Pump-02
9F	THED	Containment Ltg. XFMER-17 (PNL C8 & C10)
9M	THED	Containment Ltg. XFMR-15 (PNL C4 & C6)
3M	THED	Preaccess Fan-12
1G	THFK	Containment Welding Machine Power Supply Unit

3.3 Device Location - MCC 1EB3-2 Containment numbers listed below.

Primary and Backup Breakers - Unless noted otherwise, both primary and backup breakers have identical trip ratings and are located in the same MCC compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements.

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER
AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB3-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
8RF	THED	JB-1S-1005 for Altern. Feed to Motor Operated Valve 1-8702A
1G	THED	Motor Operated Valve 1-8112
9G	THED	Motor Operated Valve 1-8701A
9M	THED	Motor Operated Valve 1-8701B
5M	THED	Motor Operated Valve 1-8000A
5G	THED	Motor Operated Valve 1-HV-6074
4G	THED	Motor Operated Valve 1-HV-6076
4M	THED*	Motor Operated Valve 1-HV-6078
2G	THED	Motor Operated Valve 1-HV-4696
2M	THED	Motor Operated Valve 1-HV-4701
3G	THED	Motor Operated Valve 1-HV-5541
3M	THED	Motor Operated Valve 1-HV-5543
1M	THED	Motor Operated Valve 1-HV-6083
6F	THED	Motor Operated Valve 1-HV-8808A
6M	THED	Motor Operated Valve 1-HV-8808C
7M	THED	Containment Ltg. XFMR-18 (PNL SC1 & SC3)

*Primary protection is provided by Gould Tronic TR5 fusible switch with 3.2A fuse.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB3-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
8M	THED	Neutron Detector Well Fan-09
7F	THFK	Electric H_2 Recombiner Power Supply PNL-01
8RM	THED	Fire Protection Containment Isolation MOV1-HV-4075C

3.4 Device Location - MCC 1EB4-2 Containment numbers listed below.

Primary and Backup Breakers - Unless noted otherwise, both primary and backup breakers have identical trip ratings and are located in the same MCC compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements.

MCC 1EB4-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
1M	THED	JB-1S-1230G, Altern. Power Supply Feed to Mov 1-8701B
8G	THED	Motor Operated Valve 1-8702A
8M	THED	Motor Operated Valve 1-8702B
4M	THED	Motor Operated Valve 1-8000B
4G	THED	Motor Operated Valve 1-HV-6075
3G	THED	Motor Operated Valve 1-HV-6077
3M	THED*	Motor Operated Valve 1-HV-6079

*Primary protection is provided by Gould Tronic TR5 fusible switch with 3.2A fuse.

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

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DEVICE NUMBER
AND LOCATION

3. 480 VAC from Motor Control Centers (Continued)

MCC 1EB4-2 COMPT. NO.	G.E. BKR. TYPE	SYSTEM POWERED
2G	THED	Motor Opreated Valve 1-HV-5562
2M	THED	Motor Operated Valve 1-HV-5563
5F	THED	Accumulator Iso. VLV, Mov-1-8808B
5M	THED	Accumulator Iso. VLV, Mov-1-8808D
6M	THED	Containment Ltg. XFMR-19 (PNL SC2 & SC4)
7M	THED	Neutron Detector Well Fan-10
6F	THFK	Elect. H_2 Recombiner Power Supply PNL-02

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
4. 480 VAC From Panelboards For Pressurizer Heaters	Pressurizer Heaters

a. Primary Breakers - General Electric Type TJJ Thermal Magnetic breaker.

Breaker No. & Location - Ckt. Nos. 2 thru 4 of Panelboards 1EB1-1, 1EB1-2, 1EB2-2, 1EB3-2, 1EB4-1, 1EB4-2 and Ckt. Nos. 2 thru 5 of Panelboards 1EB2-1 and 1EB3-1.

b. Backup Breakers - General Electric Type THJS with longtime and insts solid state trip device with 400 Amp. sensor.

Breaker No. & Location - Ckt. No. 1 of Panelboards 1EB1-1, 1EB1-2, 1EB2-1, 1EB2-2, 1EB3-1, 1EB3-2, 1EB4-1 and 1EB4-2.

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
5. 120V Space Heater Circuits from 480V Switchgears	Containment Recirc. Fan and CRDM Vent. Fan Motor Space Heaters

a. Primary Breakers

BKR. LOCATION & NUMBER	WESTINGHOUSE BKR. TYPE
Swgr. 1EB1, Cubicle 3A CP1-VAFNAV-01 Space Heater Bkr.	EB1010
Swgr. 1EB2, Cubicle 3A CP1-VAFNAV-02 Space Heater Bkr.	EB1010
Swgr. 1EB3, Cubicle 9A CP1-VAFNAV-03 Space Heater Bkr.	EB1010
Swgr. 1EB4, Cubicle 9A CP1-VAFNAV-04 Space Heater Bkr.	EB1010
Swgr. 1EB3, Cubicle 8A, CP1-VAFNCB-01 Space Heater Bkr.	EB1010
Swgr. 1EB4, Cubicle 8A CP1-VAFNAV-02 Space Heater Bkr.	EB1010

b. Backup Breakers

BKR. LOCATION & NUMBER	GENERAL ELECTRIC BKR. TYPE
Panel 1EC3-2 Ckt. No. 3	TED
Panel 1EC3-2 Ckt. No. 4	TED

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
5. 120V Space Heater Circuits from 480V Switchgears (Continued)	

BKR. LOCATION & NUMBER	GENERAL ELECTRIC BKR. TYPE
Panel 1EC4-2 Ckt. No. 3	TED
Panel 1EC4-2 Ckt. No. 4	TED

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
6. 125V DC Lighting	Emergency DC Lighting

a. Primary Breaker

BREAKER LOCATION AND NUMBER	G. E. BKR. TYPE
DC Panelboard 1D2-1, Ckt #6	TFJ

b. Backup Device - N/A (Fuse)

7. 125V DC Control Power	Various

a. Primary Devices - N/A Fuses

b. Backup Breakers

CAB. NO.	PANELBOARD NO.	CKT. NO.	GENERAL ELECTRIC BREAKER TYPE
01	XED1-1	1,6,7,8,9,10	TED
02	XED2-1	1,3,6,7,8,9,10	TED
03	XD2-3	8,9,12,14,17	TED
04	XED1-1	1,6,7,8,9,10	TED
05	1ED2-1	7,10,12,15,16,17	TED
06	XD2-3	8,9,12,14,17	TED
07	1ED1-1	7,10,14,17	TED
08	XED2-1	1,3,6,7,8,9,10	TED
09	1D2-3	7,10,11,14,17	TED
10	1ED1-1	7,10,14,17	TED
11	1ED2-1	7,10,12,15,16,17	TED
13	1ED1-1	7,10,14,17	TED
39A	XD2-1	11	TED

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER
AND LOCATION

8. 118V AC Instrument Distribution Panel Board 1C3-3

   a. Primary Device

   b. Backup Breaker - GE Type TED
      located in instrument
      Distribution Panel Board
      1C3-CK #11

9. 120V AC Power for Personnel and Emergency Airlocks

   a. Primary Devices

   b. Backup Breakers

PANELBOARD NO.	CKT. NO.	GENERAL ELECTRIC BREAKER TYPE
XEC2	34	TED
XEC1-2	2	TED

10. 118V AC Control Power

   a. Primary Devices

   b. Backup Breakers

PANELBOARD NO.	CKT. NO.	GENERAL ELECTRIC BREAKER TYPE
XEC12-1	3,5,7,9,10,12	TED
XEC2-1	3,5,7,9,10,12	TED
1C2	12,22	TED
1C3	12,14	TED
1PC1	10,13	TED
1PC2	10	TED
1PC4	6,10	TED
1EC1	3,4,8,9	TED
1EC2	3,4,7,9	TED
1EC5	3,8	TED
1EC6	3,8	TED

TABLE 3.8-1 (Continued)

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CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER
AND LOCATION

11. Emergency Evacuation System Warning Lights Power

a. Primary Devices

b. Backup Breakers

PANELBOARD NO.	CKT. NO.	SQUARE D SINGLE POLE BREAKER TYPE
XEC3	3	FAL-12020
XEC4	3	FAL-12020

12 ~~13~~. DRPI Data Cabinet Power Supplies

a. Primary Devices

b. Backup Breakers

PANELBOARD NO.	CKT. NO.	GENERAL ELECTRIC BREAKER TYPE
1C14	1,2	TED

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 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 the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.* Additionally, either valve 1CS-8455 or valves 1CS-8560, FCV-111B, 1CS-8439, 1CS-8441 and 1CS-8453 shall be closed and secured in position.

APPLICABILITY: MODE 6.

ACTION:

a. With the requirements a or b of the above not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 2000 ppm, whichever is the more restrictive.

b. If either valve 1CS-8455 or valves 1CS-8560, FCV-111B, 1CS-8439, 1CS-8441 and 1CS-8453 are not closed and secured in position, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and take action to isolate the dilution paths. Within 1 hour, verify the more restrictive of 3.9.1.a or 3.9.1.b or carry out Action a. above.

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 control rod in excess of 3 feet from its fully inserted position within the reactor 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.

4.9.1.3 Either valve 1CS-8455 or valves 1CS-8560, FCV-111B, 1CS-8439, 1CS-8441 and 1CS-8453 shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days to verify that dilution paths are isolated.

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*During initial fuel load, the boron concentration limitation for the refueling canal is not applicable provided the refueling canal level is verified to be below the reactor flange elevation at least once per 12 hours.

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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, 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. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

DRAFT

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

DRAFT

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 hatch 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 ventilation isolation 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 ventilation isolation valve within 100 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 that,

1. Containment ventilation isolation occurs on a high radiation test signal from a containment atmosphere gaseous monitoring instrumentation channel and the containment ventilation isolation valve(s) can be closed remotely from the control room, or

2. the containment ventilation isolation valve(s) are closed/isolated.

b. Verifying the remaining penetrations of 3.9.4, not covered by a. above, are in their closed/isolated condition.

DRAFT

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 refueiing 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 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

DRAFT

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine main hoist and auxiliary monorail hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine main hoist used for movement of fuel assemblies having:

   1) A minimum capacity of 2850 pounds, and

   2) An overload cutoff limit less than or equal to 2800 pounds.

b. The auxiliary monorail hoist used for latching, unlatching and movement of control rod drive shafts having:

   1) A minimum capacity of 610 pounds, and

   2) A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of fuel assemblies and/or latching, unlatching or movement of control rod drive shafts within the reactor vessel.

ACTION:

With the requirements for refueling machine main hoist and/or auxiliary monorail hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine main hoist and/or auxiliary monorail hoist from operations involving the movement of fuel assemblies and/or latching, unlatching, and movement of control rod drive shafts within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine main hoist used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2850 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2800 pounds.

4.9.6.2 The auxiliary monorail hoist and associated load indicator used for latching, unlatching, movement of control rod drive shafts within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 610 pounds.

DRAFT

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2150 pounds shall be prohibited from travel over fuel assemblies in a storage pool.

APPLICABILITY: With fuel assemblies in a storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

SURVEILLANCE REQUIREMENTS

4.9.7 Each hoist load indicator used for loads over spent fuel storage pools shall be demonstrated OPERABLE within 7 days prior to the start of such operations and at least once per 7 days thereafter during operation by performing a load test of at least 2200 pounds.

DRAFT

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop 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 RHR loop 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 RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3800 gpm at least once per 12 hours.

*The RHR loop 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 vessel hot legs.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR 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 RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3800 gpm at least once per 12 hours.

*Prior to initial criticality, the RHR loop 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 vessel hot legs.

DRAFT

3/4.9.9 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.9.1 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the containment.

SURVEILLANCE REQUIREMENTS

4.9.9.1 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 within the containment.

DRAFT

3/4.9.9 WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.9.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor vessel.

APPLICABILITY: During movement of control rods within the reactor vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.9.2 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 control rods.

DRAFT

3/4.9.10 WATER LEVEL - IRRADIATED FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.10 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 racks.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

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

SURVEILLANCE REQUIREMENTS

4.9.10 The water level above the storage racks 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 racks.

DRAFT

3/4.9.11 FUEL STORAGE POOL AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 Two independent Fuel Storage Pool Air Cleanup Systems shall be OPERABLE.

APPLICABILITY Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one Fuel Storage Pool Air Cleanup System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Storage Pool Air Cleanup System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Storage Pool Air Cleanup System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Storage Pool Air Cleanup System is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.11 The above required Fuel Storage Pool Air Cleanup 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 continuous hours with the heaters operating;

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:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in 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 ____ cfm ± 10%;

2) 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, for a methyl iodide penetration of less than [**]%; and

3) Verifying a system flow rate of ____ cfm ± 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, for a methyl iodide penetration of less than [**]%.

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 [6] inches Water Gauge while operating the system at a flow rate of ____ cfm ± 10%,

2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to [1/4] inch Water Gauge relative to the outside atmosphere during system operation,

4) Verifying that the filter cooling bypass valves can be manually opened, and

5) Verifying that the heaters dissipate _____ ± _____ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _____ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% of less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).

**Value applicable will be determined by the following equation:
$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any control rod 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 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each control rod 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 group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2

b. Specification 4.2.2.3, and

c. Specification 4.2.3.2.

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3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

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

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set in accordance with Table 2.2-1 Functional Units 5 and 2b, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

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

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

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 PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.2 may be suspended during the performance of hot rod drop time measurements in MODE 3 provided at least two reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During performance of hot rod drop time measurements.

ACTION:

With less than the above required reactor coolant loops OPERABLE during the performance of hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provision of the action statements of Specification 3.4.1.2.

SURVEILLANCE REQUIREMENTS

4.10.4 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to the initiation of hot rod drop time measurements by verifying current breaker alignments and indicated power availability and by verifying the indicated secondary side water level to be greater than or equal to 10% narrow range.

DRAFT

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The digital rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the required digital rod position indicator(s) inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required digital rod position indicator(s) shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

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3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR 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 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters 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)(1) (μCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	$5x10^{-7}$
			I-131	$1x10^{-6}$
a. Waste Monitor Tanks	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
b. Laundry Holdup and Monitor Tanks	P Each Batch	M Composite(4)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
c. Waste Water Holdup Tanks	P Each Batch	Q Composite(4)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
d. Condensate Polisher Backwash Recovery Tanks (6)(7)	P Each Batch	P Each Batch	Principal Gamma Emitters (3)	$5x10^{-7}$
			I-131	$1x10^{-6}$
e. Component Cooling Water Drain Tank(7)			H-3	$1x10^{-5}$
2. Continuous Releases(5)	W Grab Sample	W	Principal Gamma Emitters(3)	$5x10^{-7}$
			I-131	$1x10^{-6}$
a. Turbine Bldg Sumps No. 1 & 2 Effluent(6)(7)			H-3	$1x10^{-5}$

TABLE 4.11-1 (Continued)

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TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

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

(2) 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.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3) The principal gamma emmiters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) 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 that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) These waste streams shall be required to be sampled and analyzed in accordance with this table if either of the following conditions exist:

(a) Activity is present in the secondary system as indicated by either Steam Generator Blowdown monitors or secondary sampling and analysis; or

(b) Activity was present in the respective tanks or sumps during the previous four (4) weeks.

If neither of the above situations exists, then sampling and analysis of these tanks and sumps need not be performed.

(7) All flow from this waste stream shall be diverted to the Wastewater Holdup Tanks when results of sample analyses show radioactivity present in the waste stream at concentrations greater than or equal to the LLD values given in this table. Sampling and analysis of the respective tanks or sumps are not required when flow is diverted to the Wastewater Holdup Tanks.

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DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems 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, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-4) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the 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 from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when liquid radwaste treatment systems are not being fully utilized.

4.11.1.3.2 The installed liquid radwaste treatment system shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

DRAFT

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each unprotected outdoor tanks 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 unprotected outdoor tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the unprotected outdoor tanks 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.

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

DRAFT

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the EXCLUSION AREA BOUNDARY (see Figure 5.1-1) shall be limited to the following:

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

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore 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 methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	$1x10^{-4}$
2. Containment Purge or Vent	P Each Release(3) Grab Sample	P Each Release(3)	Principal Gamma Emitters(2)	$1x10^{-4}$
		M	H-3 (oxide)	$1x10^{-6}$
3. Plant Vent	M(3),(4),(5) Grab Sample	M(3)	Principal Gamma Emitters(2)	$1x10^{-4}$
			H-3 (oxide)	$1x10^{-4}$
	Continuous(6)	W(7) Radioiodine Absorber	I-131	$1x10^{-12}$
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	$1x10^{-11}$
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

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(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

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

TABLE 4.11-2 (Continued)

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TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in Iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period. This requirement does not apply if 1) analysis of primary coolant activity, performed pursuant to Specification 4.4.8, shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3, 2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

DRAFT

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the EXCLUSION AREA BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

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

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the EXCLUSION AREA BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DRAFT

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The PRIMARY PLANT VENTILATION SYSTEM and the GASEOUS WASTE PROCESSING SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the EXCLUSION AREA BOUNDARY (see Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

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

   1. Identification of any inoperable equipment or subsystems, and the reason for the 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

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the EXCLUSION AREA BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed PRIMARY PLANT VENTILATION SYSTEM and GASEOUS WASTE PROCESSING SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

DRAFT

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 3% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 3% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

DRAFT

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 200,000 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

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

DRAFT

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

DRAFT

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units (including outside storage tanks etc.) to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units (including outside storage tanks etc.) shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

DRAFT

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, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.3, 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 as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

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 a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 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 required by Specification 6.9.1.3.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Forty routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows An inner ring of stations, one in each meteorological sector in the general area of the EXCLUSION AREA BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly.
	Three samples (A1-A3) from close to the three EXCLUSION AREA BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q;		Particulate Sampler: Gross beta radioactivity analysis following filter change;(4) and gamma isotopic analysis(5) of composite (by location) quarterly.
	One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q: and		
	One sample from a control location, as for example 15 to 30 km distant and in the least prevalent wind direction.(3)		
3. Waterborne			
a. Surface	Squawk Creek Reservoir(6)	Monthly	Gamma isotopic analysis(5) monthly. Composite for tritium analysis quarterly.
	Lake Granbury	Monthly composite of weekly grab samples when Lake Granbury is receiving letdown from SCR. Otherwise, monthly grab sample.(8)	

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
	Control-Brazos River upstream of Lake Granbury	Monthly	
b. Ground	Samples from one or two sources only if likely to be affected.[9]	Quarterly.	Gamma isotopic[5] and tritium analysis quarterly.
c. Drinking	One sample of each of one to three of the nearest water water supplies that could be affected by its discharge. One sample from a control location.	Grab sample at least once per 2-week period when I-131 analysis is performed; monthly grab sample otherwise.	I-131 analysis on each grab sample when the dose calculated for the consumption of the water is greater than 1 mrem per year[10]. Composite for gross beta and gamma isotopic analyses[5] monthly. Composite for tritium analysis quarterly.
d. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[5] semiannually.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion			
a. Milk	Samples from available milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then one sample from available milking animals in each of three areas between 5 to 8 km distant where dosesare calculated to be greater than 1 mrem per yr.(10) One sample from milking animals at a control location 15 to 30 km distant and in the least prevalent wind direction.(3)	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic(5) and I-131 analysis semi-monthly when animals are on pasture; monthly at other times.
b. Fish and Inverte-brates	One sample of each commercially and recreationally important species in vicinity of plant discharge area. One sample of same species in areas not influenced by plant discharge.	Sample semiannually .	Gamma isotopic analysis(5) on edible portions.
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest(11).	Gamma isotopic analyses(5) on edible portion.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
c. Food Products (Continued)	Samples of three different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly during growing season.	Gamma isotopic[5] and I-131 analysis.
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km distant in the least prevalent wind direction[3] if milk sampling is not performed.	Monthly during growing season.	Gamma isotopic[5] and I-131 analysis.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for the pathway and justifying the selection of the new location(s) for obtaining samples.

(2) 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 purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. (The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information within minimal fading.)

(3) 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 that provide valid background data may be substituted.

(4) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12-1 (Continued)

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TABLE NOTATIONS (Continued)

(5) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(6) Squaw Creek Reservoir is a closed cooling water basin which receives plant effluents at the circulating water discharge. The reservoir shall be sampled in an area at or beyond but not near the mixing zone. Also the reservoir shall be sampled at a distance beyond significant influence of the discharge.

(7) Squaw Creek Reservoir is a closed cooling water basin which is composited naturally.

(8) Lake Granbury may receive letdown from Squaw Creek Reservoir to control buildup of solids. This is the only pathway for plant effluents to Lake Granbury. The lake shall be sampled near the letdown discharge and at a distance beyond significant influence of the discharge.

(9) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(10) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(11) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

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)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

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TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1) (2)

LOWER LIMIT OF DETECTION (LLD)(3)

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.

**If no drinking water pathway exists, a value of 15 pCi/l may be used.

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TABLE 4.12-1 (Continued)

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TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.9.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), 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. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

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

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the EXCLUSION AREA BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1, Part 4.c., shall be followed, including analysis of control samples.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

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 pursuant to Specification 6.9.1.3.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

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BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

Specification 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provide an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours.

Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation are not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the conditions under which the specified time interval for Surveillance Requirements may be extended. Item a. permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. Item b. limits the use of the provisions of item a. to ensure that it is not used repeatedly to extend the surveillance interval beyond that specified. The limits of Specification 4.0.2 are based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verificaion of conformance with the Surveillance Requirements. These provisions are sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval

was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shut is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement 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 shall 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. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI for the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the 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. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, 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 1.6% $\Delta k/k$ 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 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ 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 value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

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

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.0×10^{-4} $\Delta k/k/^{\circ}F$. The MTC value of -3.1×10^{-4} $\Delta k/k/^{\circ}F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.0×10^{-4} $\Delta k/k/^{\circ}F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

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 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip 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 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 transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% $\Delta k/k$ 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 [22,870] gallons of 7000 ppm borated water from the boric acid storage tanks or [479,900] gallons of 2000 ppm borated water from the refueling water storage tank (RWST).

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boron 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 Boron Injection System becomes inoperable.

The limitation for a maximum of two charging pumps to be OPERABLE and the requirement to verify one charging pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The limitation for minimum solution temperature of the borated water sources are sufficient to prevent boric acid crystallization with the highest allowable boron concentration.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either [6,385] gallons of 7000 ppm borated water from the boric acid storage tanks or [101,120] gallons of 2000 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST are also consistent with Specification 3.5.4.

The OPERABILITY of one Boron Injection System during REFUELING ensures that the system is available for reactivity control while in MODE 6.

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 rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

MOVABLE CONTROL ASSEMBLIES (Continued)

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. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°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.

Control rod positions and OPERABILITY of the rod 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 LCOs are satisfied.

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The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The ~~full-length~~ rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

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FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_s) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.051,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of [2.1]% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

QUADRANT POWER TILT RATIO (Continued)

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 592.7°F (conservatively rounded to 592°F) and the indicated pressurizer pressure value of 2207 psig correspond to analytical limits of 594.7°F and 2193 psig respectively, with allowance for measurement uncertainty. The indicated uncertainties assume that the reading from four channels will be averaged before comparing with the required limit.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation

3/4 3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated 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 and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the reactor protection and engineered safety features instrumentation, and (3) 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 safety analyses. 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. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report as approved by the NRC and documented in the SER (letter to J. J. Sheppard from Cecil O. Thomas, dated February 21, 1985).

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor draft factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety 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 time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) ECCS pumps start and automatic valves position, (2) Reactor trip, (3) feed water isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) station service water pumps start and automatic valves position, (11) Control Room Emergency Recirculation starts, and (12) essential ventilation systems (safety chilled water, electrical area fans, primary plant ventilation ESF exhaust fans, battery room exhaust fans, and UPS ventilation) start.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

To satisfy the recommendations set forth in Section 4.7 of IEEE 279-1971, in the event that one of the three channels of high steam generator level protection is used for level control that channel shall be placed in the tripped condition until level control is returned to its normal channel.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steam line pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure and low steam line pressure.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel reaches its Setpoint, and (2) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS (Continued)

radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the system sends actuation signals to initiate alarms or actuate Control Room Emergency Recirculation or actuate Containment Ventilation Isolation.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable 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 core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

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 10 CFR 100 Appendix A. 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 are 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 proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in support of Nuclear Power Plants," September 1980.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown 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 Criterion 19 of 10 CFR 50.

REMOTE SHUTDOWN SYSTEM (Continued)

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 of 10 CFR 50.

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 for which pre-planned manually controlled operator actions are required to accomplish safety functions for recovery from Design Basis Accidents, as defined by the plant safety analysis. This capability meets the intent of the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and those requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 that apply to CPSES.

The provision that allows the number of Steam Generator Water Level-Wide Range or Auxiliary Feedwater Flow Rate Channels to be reduced by combining them into a Secondary Coolant Availability function is consistent with Action Plan requirement II.E.1.2 of NUREG-0737 for Westinghouse Pressurized Water Reactors.

The specific calibration provisions for the Containment Radiation (High Range) Monitor are in accordance with the provisions of NUREG-0737, Item II.F.1.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Power Plant Control room Operators Against an Accidental Chlorine Release," January 1977.

This capability will not be required if the quantity of chlorine gas stored onsite is small ($\leq$ 20 lbs.) and utilized for laboratory and calibration purposes. This applicability is consistent with the exclusions and recommendations of Regulatory Guide 1.95, Revision 1, "Protection of Nuclear Plant Control Room Operators Against an Accidental Chlorine Release," January 1977.

3/4.3.3.8 LOOSE PART DETECTION SYSTEM

The OPERABILITY of the Loose-Part Detection System ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor 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.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING 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 and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of 10 CFR 50 Appendix A.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING 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 and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 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 10 CFR 50 Appendix A. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} μCi/ml are measurable.

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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. 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 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODES 3, 4, a 5, the operability of the required steam generators is based on maintaining sufficient level to guarantee tube coverage to assure heat transfer capability.

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

In MODE 5 with reactor coolant loops not filled, a single RHR loop 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 RHR loops be OPERABLE.

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

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of 10 CFR 50 Appendix G. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50° 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 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)

condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or 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 Code.

3/4.4.3 PRESSURIZER

The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. 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. Pressurizer heater groups are powered from sources that meet the requirements of Item II.E.3.1 of NUREG-0737.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 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. 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.

Selected tubes in the preheater section of each D4 and D5 steam generator have been modified to correct the tube vibration degradation phenomenon experienced by certain Westinghouse steam generators. The modification consisted of expanding these tubes in the vicinity of the support plates and is designed to limit the amplitude of vibration. These expanded tubes are subject to a special inspection whenever the steam generators are opened for inservice eddy current testing.

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 Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator and a total leakage of 1 GPM to all steam generators). Cracks having a reactor-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 reactor-to-secondary leakage of 500 gallons per day 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.

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STEAM GENERATORS (Continued)

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 40% of the tube nominal wall thickness. 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 reported to the Commission in a Special Report pursuant to 10 CFR 50.72 within 4 hours from initial discovery and pursuant to Specification 6.9.2 within 30 days and prior to 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 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.

If one of the required systems becomes inoperable, 30 days are permitted for restoration since two diverse and redundant RCS leakage detection systems remain OPERABLE. If, however, the inoperable system is the required containment gaseous or particulate monitoring system, grab samples are also performed as a backup to the single remaining atmospheric monitoring system.

3/4.4.6.2 OPERATIONAL LEAKAGE

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.

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 total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values 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 500 gpd 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.

OPERATIONAL LEAKAGE (Continued)

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance 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 CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

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 valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 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 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.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the EXCLUSION AREA BOUNDARY (EAB) will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady- state reactor-to-secondary steam generator leakage rate of 1 gpm. 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 CPSES site, such as EAB location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 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.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the EAB, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the EAB under any accident condition.

The activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the EAB by a factor of up to 20 following a postulated steam generator tube rupture. Therefore, operation with specific activity levels exceeding the limits of Specification 3.4.8 requires additional sampling per Table 4.4-4 and reporting of operational and sample information in the Annual Report pursuant to Specification 6.9.1.4. This is in conformance with Generic Letter 85-19 to allow NRC evaluation .

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 reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and 10 CFR 50 Appendix G.

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The new 10 CFR 50, Appendix G rule addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure (621 psig for Westinghouse plants). For Comanche Peak Unit 1, the minimum temperature of the closure flange and the vessel flange regions is 160°F since the limiting RT_{NDT} is 40°F (see Table B 3/4.4-1). The Comanche Peak Unit 1 cooldown curves shown in Figure 3.4-3 are impacted by this new rule.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUCHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU %	P %	NDTT °F	50 FT-LB/35 MIL TEMP °F LONG	50 FT-LB/35 MIL TEMP °F TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS

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FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT}
FOR REACTOR VESSELS EXPOSED TO 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50 Appendix G, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the Linear Elastic Fracture Mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement

PRESSURE/TEMPERATURE LIMITS (Continued)

effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

PRESSURE/TEMPERATURE LIMITS (Continued)

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates

PRESSURE/TEMPERATURE LIMITS (Continued)

do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.98 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of 10 CFR 50 Appendix G when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV 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 50°F above the RCS cold leg temperatures, or (2) the start of charging pumps and their injection into a water-solid RCS.

The maximum Nominal Allowed PORV Setpoint curve is derived from analyses which model the performance of the overpressure protection system for a range of mass input and heat input transients. Figure 3.4-4 is based upon this analysis including consideration of the maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. For the transients noted, the resulting pressure will not exceed the nominal 10 Effective Full Power Years (EFPY) Appendix G reactor vessel NDT limits and the forces generated due to PORV cycling do not exceed PORV piping and structural limitations.

To ensure that mass and heat input transitents more severe than those assemed cannot occur, Technical Specifications require the lockout of all safety injection pumps and one charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed, and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

Operation below 350°F but greater than 325°F with charging and safety injection pumps OPERABLE is allowed for up to 4 hours. Given the short time duration that this condition is allowed initiation of both trains of safety injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

The Maximum Allowed PORV Setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 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 edition and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 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, ____ Edition and Addenda through _____.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

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

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

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator 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 accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator 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 accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required by BTB ICSB 18. This is accomplished via key-lock control board cut-off switches.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator 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 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 accumulators 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.

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 limitation for a maximum of two charging pumps to be OPERABLE and the requirement to verify one charging pump and all safety injection pumps

ECCS SUBSYSTEMS (Continued)

to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The requirement to remove power from certain valve operators is in accordance with Branch Technical Position ICSB-18 for valves that fail to meet single failure considerations. Power is removed via key-lock switches on the control board.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and 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

The OPERABILITY of the refueling water storage tank (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, (2) for small break LOCA and steam line breaks, 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, and (3) for large break LOCAs, the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all shutdown and control rods fully withdrawn, and (4) sufficient time is available for the operator to take manual action and complete switchover of ECCS and containment spray suction to the containment sump without emptying the RWST or losing suction.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 10.5 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.

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 safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the EXCLUSION AREA BOUNDARY radiation doses to within the dose guideline values 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 safety 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 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

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

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 of 5 psig with respect to the outside atmosphere, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during LOCA.

The maximum peak pressure expected to be obtained from a LOCA event is 46.3 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 48.3 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

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 48.3 psig in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch and 12-inch containment and hydrogen purge supply and exhaust isolation valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Ventilation System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 5.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Containment Ventilation System during operations is restricted to the 18-inch pressure relief discharge isolation valves since, unlike the 48-inch and 12-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR 100 would not be exceeded in the event of an accident during containment venting operation. Operation with one pair of these valves open will be limited to 90 hours during a calendar year. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons, e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4 in any calendar year regardless of the allowable hours.

CONTAINMENT VENTILATION SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for containment ventilation valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. 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 or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System which is composed of redundant trains, provides 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 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive 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 a pH value of between 8.5 and 10.5 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 contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

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 General Design Criteria 54 through 57 of 10 CFR 50 Appendix A. 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.

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% (1305 psig) of its design pressure of 1185 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 Code, 1974 Edition. The total rated relieving capacity for all valves on all of the steam lines is 16,190,884 lbs/h which is 120% of the total secondary steam flow of 15,140,106 lbs/h at 100% RATED THERMAL POWER.

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 Coolant System steam flow and THERMAL POWER required by the reduced Reactor Trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

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-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 430 gpm to two steam generators at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 860 gpm to four steam generators at a pressure of 1221 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The Auxiliary Feedwater System is capable of delivering a total feedwater flow of 430 gpm at a pressure of 1221 psig to the entrance of at least two steam generators while allowing for: (1) any possible spillage through the design worst case break of the main feedwater line; (2) the design worst case single failure; and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed in operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 18 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power or 4 hours at HOT STANDBY followed by a cooldown to 350°F at a rate of 50°F/HR for 5 hours. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

NUREG-0737, Item II.E.1.1 requires a backup source to the CST which is the CPSES Station Service Water System, which can be manually aligned, if required.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down 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 safety 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 60°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 safety analyses.

3/4.7.4 STATION SERVICE WATER SYSTEM

The OPERABILITY of the Station Service 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 safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

ULTIMATE HEAT SINK (Continued)

The limitations on minimum water level is based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," rev. 2 (January 1976). The limitation on maximum temperature is based on the maximum allowable component temperatures in the Service Water and Component Cooling Water Systems, and the requirements for cooldown. The limitation on average sediment depth is based on the possible excessive sediment buildup in the service water intake channel.

3/4.7.6 FLOOD PROTECTION

The limitation of flood protection ensures that facility protective actions will be taken in the event of flood conditions. The only credible food condition that endangers safety related equipment is from water entry into the turbine building via the circulating water system from Squaw Creek Reservoir and then only if the level is above 778 feet Mean Sea Level. This corresponds to the elevation at which water could enter the electrical and control building endangering the safety chilled water system. The surveillance requirements are designed to implement level monitoring of Squaw Creek Reservoir should it reach an abnormally high level above 776 feet. The Limiting Condition for Operation is designed to implement flood protection, by ensuring no open flow path via the Circulating Water System exists, prior to reaching the postulated flood level.

3/4.7.7 CONTROL ROOM HVAC SYSTEM

The OPERABILITY of the Control Room HVAC System ensures that: (1) the control room ambient air temperature does not exceed the allowable temperature per 3/4 7.11 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. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. 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 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of 10 CFR 50 Appendix A. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.8 PRIMARY PLANT VENTILATION SYSTEM - ESF FILTRATION UNITS

The OPERABILITY of the ESF Filtration Units ensures that radioactive materials leaking from the ECCS equipment within the safeguards and auxiliary buildings following a LOCA are filtered prior to reaching the environment. These filtration units also ensure that radioactive materials leakage from within the fuel building are filtered prior to reaching the environment. Operation of the ESF filtration units with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of the ESF filtration units and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.9 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 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 snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with 10 CFR 50 50.71(c). The accessibility of each snubber shall be determined and approved by the Station Operation Review Committee (SORC). The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with 10 CFR 50.59.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that

SNUBBERS (Continued)

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.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

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, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

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.

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 snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, 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.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) 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 shielded mechanism.

3/4.7.11 AREA TEMPERATURE MONITORING

The limitations on nominal area temperatures ensure that safety-related equipment will not be subjected to temperatures that would impact their environmental qualification temperatures. Exposure to temperatures in excess of the maximum temperature for normal conditions for extended periods of time could reduce the qualified life or design life of that equipment. Exposure to temperatures in excess of the maximum abnormal temperature could degrade the operability of that equipment.

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

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 Criterion 17 of 10 CFR 50 Appendix A.

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 and Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability." 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 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; 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, Generic Letter 84-15, and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

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

The Diesel Generator Test schedule, Table 4.8-1, is based on the recommendations of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and NRC Technical Report A-3230, "Evaluation of Diesel Unavailability and Risk Effective Surveillance Test Intervals," May 1986, and Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability."

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, Regulatory Guide 1.32 "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," Revision 2, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

The operational requirement to energize the instrument busses from their associated inverters connected to its associated D.C. bus is satisfied only when the inverter's output is from the regulated portion of the inverter and not from the unregulated bypass source via the internal static switch.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 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 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

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 0.020 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 0.040 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. This is based on the recommendations of regulatory guide 1.63 Revision 2 "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants."

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker reliability by testing at least 10% of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

All Class 1E motor-operated valves motor starters are provided with thermal overload protection which is permanently bypassed and provides an alarm function only at Comanche Peak Steam Electric Station. Therefore, there are no OPERABILITY or Surveillance Requirements for these devices, since they will not prevent safety-related valves from performing their function (refer to Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977).

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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 safety 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 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

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 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 safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building 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 conditions during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine main hoist and auxiliary monorail hoist ensure that: (1) the main hoist will be used for movement of fuel assemblies, (2) the auxiliary monorail hoist will be used for latching, unlatching and movement of control rod drive shafts, (3) the main hoist has sufficient load capacity to lift a fuel assembly (with control rods), (4) the auxiliary monorail hoist has sufficient capacity to latch, unlatch and move the control rod drive shafts, and (5) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in a 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 safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

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

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 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 RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and IRRADIATED FUEL STORAGE

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 safety analysis.

3/4.9.11 STORAGE POOL VENTILATION SYSTEM

The limitations on the Storage Pool Ventilation 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. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod 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 control rods 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 control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 and the RCS T_{avg} may fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception is required to perform certain STARTUP and PHYSICS TESTS under no flow conditions.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Digital Rod Position Indicator(s) to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Digital Rod Position Indicator(s) remain OPERABLE. The exception to the requirement for the Digital Rod Position Indicator to be OPERABLE during the withdrawal of the rods for the initial calibration of the position indication system is required because the OPERABILITY of the Digital Rod Position Indication System can only be determined by withdrawing the control rod. The limitation on Keff during this evolution provides the necessary assurance that inadvertent critically will be avoided.

3/4.11 RADIOACTIVE EFFLUENTS

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 to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of 10 CFR 50 Appendix I, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 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.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4077 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of 10 CFR 50 Appendix I. 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 to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. 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

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DOSE (Continued)

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 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 radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Systems, the liquid effluents from the shared system are to be proportional among the units sharing that system.

3.4.11.1.3 LIQUID RADIOWASTE TREATMENT SYSTEM

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 50.36a, General Design Criterion 60 of 10 CFR 50 Appendix A and the design objective given in Section II.D of 10 CFR 50 Appendix I. 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 10 CFR 50 Appendix I for liquid effluents.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor tanks both permanent and temporary that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR 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 and beyond the Exclusion Area BOUNDARY (EAB) from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the EAB, to annual average concentrations exceeding the limits specified in Table II of 10 CFR Part 20 Appendix B (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the EAB, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the EAB. The methodology of calculating doses for such MEMBERS OF THE PUBLIC, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the EAB to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/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 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

Activities unrelated to plant operation which may be permitted within the Exclusion Area include the exercising of mineral rights and the maintenance of pipelines. The Applicants will have the necessary control to determine these activities and will require that all persons involved in them report to the CPSES Manager, Plant Operations or his designated representative prior to engaging in the activities.

Publication recreational activities within the Exclusion Area are limited to Squaw Creek Reservoir and Squaw Creek Park. Appropriate and effective arrangements have been made (in coordination with the appropriate agencies) to control access to, activities on, and the removal of persons and property from the reservoir in case of emergency. Arrangements for recreational use and emergency procedures governing such use have been completed. The Applicants have the authority to exclude or remove any person from this area at any time.

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4077 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

DOSE-NOBLE GASES (Continued)

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 10 CFR 50 Appendix I. The Limiting Condition for Operation implements the guides set forth in Section I.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 to UNRESTRICTED AREAS 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters 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 50, Appendix I, "Revision I, 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 and beyond the EAB are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. Since both units share the radwaste treatment systems, the gaseous effluents are proportioned among the units.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of 10 CFR Part 50 Appendix I. 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 to UNRESTRICTED AREAS 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters 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, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

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 Iodine-131 Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the EAB. The pathways that were examined in the development of the 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.

This specification applies to the release of radioactive materials in gaseous effluents from each unit 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.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the GASEOUS WASTE PROCESSING SYSTEM and the PRIMARY PLANT VENTILATION 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 50.36a, General Design Criterion 60 of 10 CFR Part 50 Appendix A and the design objectives given in Section II.D of 10 CFR Part 50 Appendix I. 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 10 CFR Part 50 Appendix I, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit 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.

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 isolation of the source of hydrogen and/or oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of 10 CFR Part 50 Appendix A.

3/4 11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. 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 whole body exposure to a MEMBER OF THE PUBLIC at the nearest EAB will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of 10 CFR 50 Appendix A. 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, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units (including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC 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

TOTAL DOSE (Continued)

radius of 8 km 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 (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and 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 the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum 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 an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the EXCLUSION AREA BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of 10 CFR Part 50 Appendix I. Restricting the census to gardens of greater than 50 m^2 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 (26 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 made: (1) 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/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of 10 CFR Part 50 Appendix I.

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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.

MAP DEFINING UNRESTRICTED AREAS AND EXCLUSION AREA BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the EXCLUSION AREA BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the EXCLUSION AREA BOUNDARY, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the EXCLUSION AREA BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The 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 = 135 feet.

b. Nominal inside height = 192.5 feet. (Done 67.5 feet; total = 260 feet)

c. Nominal thickness of concrete walls = 4.5 feet.

d. Nominal thickness of concrete roof = 2.5 feet.

e. Nominal thickness of concrete mat = 12.0 feet.

f. Nominal thickness of steel liner wall = 3/8 inch. (Dome = 1/2 inch, Base Mat = 1/4 inch), and

g. Net free volume = 2,985,000 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 280°F.

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FIGURE 5.1-1

EXCLUSION AREA

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FIGURE 5.1-2

LOW POPULATION ZONE

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FIGURE 5.1-3

UNRESTRICTED AREA AND EXCLUSION AREA BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods (consisting of Zircaloy-4 or stainless steel) or by vacancies may be made if justified by a cycle specific reload analysis. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment not to exceed 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment not to exceed 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 95.5% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

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 to the applicable Surveillance Requirements,

b. For a pressure of 2,485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,500 ± 100 cubic feet at a nominal T_{avg} of 589.5°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The primary meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 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 for uncertainties as described in Section 4.3 of the FSAR, and

b. A nominal 16 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 854 feet.

CAPACITY

5.6.3 The two spent fuel storage pools are designed and shall be maintained with a storage capacity limited to no more than 1116 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to > 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	> 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F, but ≤ 625°F.
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1481 psig.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Vice President, Nuclear Operations shall be responsible for overall operation of the site, while the Manager, Plant Operations shall be responsible for operation of the unit. The Vice President, Nuclear Operations and Manager, Plant Operations shall each delegate in writing the succession to this responsibility during their absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President, Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The corporate 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 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 Operator shall be in the control room;

c. A Radiation Protection 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 Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Protection 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 unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, Radiation Protection Technicians, auxiliary operators, and key maintenance personnel).

The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

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FIGURE 6.2-1

OFFSITE ORGANIZATION

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FIGURE 6.2-2

UNIT ORGANIZATION

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1*	None

SS - Shift Supervisor with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

*The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Vice President, Nuclear Operations.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 3 years professional level experience in his field.

RESPONSIBILITIES

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

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to Vice President, Nuclear Operations.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

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 Radiation Protection Manager** who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, for a Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

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*Not responsible for sign-off function.

**Until the Radiation Protection Manager meets all qualification per R.G.1.8, September 1975, an individual who meets all those qualifications shall support the Radiation Protection Manager.

UNIT STAFF QUALIFICATIONS (Continued)

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Vice President, Nuclear Operations and shall meet or exceed the requirements and recommendations of ANSI-N18.1-1971 and Appendix A of 10 CFR 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The SORC shall function to advise the Vice President, Nuclear Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SORC shall be composed of the:

Chairman:	Vice President, Nuclear Operations	
Member:	[Operations Supervisor]	Members to be equivalent to these positions, but of high management position.
Member:	[Technical Supervisor]	
Member:	[Maintenance Supervisor]	
Member:	[Plant Instrument and Control Engineer]	
Member:	[Plant Nuclear Engineer]	
Member:	[Health Physicist]	

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SORC activities at any one time.

**Until the Radiation Protection Manager meets all qualification per R.G.1.8, September 1975, and individual who meets all those qualifications shall support the Radiation Protection Manager.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the SORC necessary for the performance of the SORC 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 SORC shall be responsible for:

a. Review of all Station Administrative Procedures;

b. Review of the safety evaluations for: (1) procedures, (2) change to procedures, equipment, systems or facilities, and (3) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

c. Review of proposed procedures and changes to procedures, equipment, systems or facilities which may involve an unreviewed safety question as defined in 10 CFR 50.59 or involves a change in Technical Specifications;

d. Review of proposed test or experiments which may involve an unreviewed safety question as defined in 10 CFR 50.59 or requires a change in Technical Specifications;

e. Review of proposed changes to Technical Specifications or the Operating License;

f. Investigation of all violations of the Technical Specifications including the forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President, Nuclear Operations and to the ORC;

g. Review of reports of operating abnormalities, deviations from expected performance of plant equipment and of unanticipated deficiencies in the design or operation of structures, systems or components that affect nuclear safety;

h. Review of all REPORTABLE EVENTS;

i. Review of the Security Plan and shall submit recommended changes to the ORC;

RESPONSIBILITIES (Continued)

j. Review of the Emergency Plan and shall submit recommended changes to the ORC;

k. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and Radwaste Treatment Systems;

l. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President, Nuclear Operations, and to the ORC;

m. Review of Unit operations to detect potential hazards to nuclear safety; and

n. Investigations or analysis of special subjects as requested by the Chairman of the ORC or the Vice President, Nuclear Operations.

o Review of the Fire Protection Program and revisions thereto.

6.5.1.7 The SORC shall:

a. Recommend in writing to the Vice President, Nuclear Operations approval or disapproval of items considered under Specification 6.5.1.6a. through e, i, j, k, and l above, prior to their implementation;

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. and m. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Executive Vice President-Nuclear Engineering and Operations and the Operations Review Committee of disagreement between the SORC and the Vice President, Nuclear Operations; however, the Vice President, Nuclear Operations shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The SORC shall maintain written minutes of each SORC meeting that, at a minimum, document the results of all SORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President-Nuclear Operations and the Operations Review Committee.

DRAFT

6.5.2 OPERATIONS REVIEW COMMITTEE (ORC)

FUNCTION

6.5.2.1 The ORC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The ORC shall report to and advise the Executive Vice President, Nuclear Engineering and Operations on those areas of responsibility specified in Specifications 6.5.2 7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The ORC shall be composed of at least five individuals of whom no more than minority are members having line responsibility for operations at CPSES. The Chairman and all members will be appointed by the Executive Vice President, Nuclear Engineering and Operations.

The ORC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of 5 years technical experience. It is the responsibility of the Chairman to ensure experience and competence is available to review problems in areas listed in Specification 6.5.2.1a. through h. To a large measure, this experience and competence rests with the membership of the ORC. In specialized areas, this experience may be provided by personnel who act as consultants to the ORC.

ALTERNATES

6.5.2.3 The Alterante for the Chairman and all alternate members shall be appointed in writing by the Executive Vice President, Nuclear Engineering and Operations to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the Chairman, ORC to provide expert advice to the ORC.

MEETING FREQUENCY

6.5.2.5 The ORC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the ORC necessary for the performance of the ORC review and audit functions of these Technical Specifications shall consist of not less than a majority of the appointed individuals (or their alternates) and the Chairman or his designated alternate. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The ORC shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, 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 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

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. All REPORTABLE EVENTS;

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; and

i. Reports and meeting minutes of the SORC.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the ORC. The audits shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the specified interval provided the combined time interval for any three consecutive intervals shall not exceed 3.25 times the specified interval. 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 fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

k. Any other area of unit operation considered appropriate by the ORC or the Executive Vice President Nuclear Engineering and Operations.

RECORDS

6.5.2.9 Records of ORC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each ORC meeting shall be prepared, approved, and forwarded to the Vice President, Nuclear Operations and Executive Vice President, Nuclear Engineering and Operations within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Vice President Nuclear Operations and Executive Vice President, Nuclear Engineering and Operations within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Vice President, Nuclear Operations and Executive Vice President, Nuclear Engineering and Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROLS

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by a qualified individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. The Vice President, Nuclear Operations, shall approve Station Administrative Procedures, Security Plan Implementing Procedures, and Emergency Plan Implementing Procedures. Other procedures shall be approved by the appropriate approval authority, as designated by the Vice President, Nuclear Operations, in writing. Individuals responsible for procedure reviews shall be members of the Nuclear Operations Staff previously designated by the Vice President, Nuclear Operations. Changes to procedures which do not change the intent of approved procedures may be approved for implementation by two members of the Nuclear Operations Staff, at least one of whom holds a Senior Operator License, provided such approval is prior to implementation and is documented. Such changes shall be approved by the original approval authority within 14 days of implementation;

b. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report or Technical Specifications shall be prepared, reviewed, and

TECHNICAL REVIEW AND CONTROLS (Continued)

approved. Each such test or experiment shall be reviewed by a qualified individual/group other than the individual/group which prepared the proposed test or experiment. Proposed test and experiments shall be approved before implementation by the Manager, Plant Operations. Individuals responsible for conducting such reviews shall be members of the Nuclear Operations Staff previously designated by the Vice President, Nuclear Operations;

c. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Vice President, Engineering and Construction. Each such modification shall be reviewed by a qualified 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 modifications. Individuals/groups responsible for conducting such reviews shall be previously designated by the Vice President, Engineering and Construction. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Manager, Plant Operations prior to implementation;

d. Each review conducted in accordance with the requirements of Specifications 6.5.3.1a, 6.5.3.1b, and 6.5.3.1c, shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be done in accordance with the appropriate qualification requirements;

e. Each review shall include a determination of whether or not an unreviewed safety question is involved. Pursuant to NRC approval of items involving unreviewed safety questions shall be obtained prior to the Manager, Plant Operations, approval for implementation; and

f. The Security Plan and Emergency Plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes to the implementing procedures shall be approved by the Manager, Plant Operations. Recommended changes to the Plans shall be reviewed pursuant to the requirements of Specifications 6.5.1.6 and 6.5.2.8 and approved by the Manager, Plant Operations. NRC approval shall be obtained as appropriate.

6.5.3.2 Records of the above activities described in 6.5.3.1 shall be provided to the Vice President, Nuclear Operations, SORC, and/or ORC as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73 and

b. Each REPORTABLE EVENT shall be reviewed by the SORC, and the results of this review shall be submitted to the ORC and the Vice President Nuclear Operations.

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6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center, shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Vice President, Nuclear Operations and the Operations Review Committee (ORC) shall be notified within 24 hours.

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The License Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, Vice President, Nuclear Operations and the Operations Review Committee (ORC) within 30 days after discovery of the event.

d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory 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. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Assurance for effluent and environmental monitoring.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the SORC and shall be approved by the Vice President, Nuclear Operations prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 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 Operator license on the unit affected; and

PROCEDURES AND PROGRAMS (Continued)

c. The change is documented, reviewed by the SORC, and approved by the Vice President, Nuclear Operations within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the post accident recirculation portions of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, RHR System, and RCS Sampling System (Post Accident Sampling System portion only). The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

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

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) 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 and low pressure turbine disc stress corrosion cracking. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

PROCEDURES AND PROGRAMS (Continued)

6) 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 following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The initial Startup Report shall address each of the startup tests identified in Chapter 14 of the Final Safety Analysis Report 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. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

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 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 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.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures

*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.

ANNUAL REPORTS (Continued)

greater than 100 mrem/yr and their associated man-rem 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, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.3 Routine Annual 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 and shall include copies of reports of the preoperational Radiological Environmental Monitoring Program of the unit for at least two years prior to initial criticality.

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,

*This tabulation supplements the requirements of §20.407 of 10 CFR 20.

**A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

including a comparison with preoperational studies, with operational controls, as appropriate, and with 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 the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual 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; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.4 Routine Semiannual 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.

The Semiannual 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

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

**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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 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 on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid 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 (Figure [5.1-3]) 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 methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 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, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP)

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*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

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

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be established for at least each reload core and shall be maintained available in the Control Room. The limits shall be established and implemented on a time scale consistent with normal procedural changes.

The analytical methods used to generate the F_{xy} limits shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the F_{xy} limits for all core planes containing Bank "D" control rods and all unrodded core planes along with the plot of predicted $F_q^T \cdot P_{rel}$ axial core height (with the limit envelope for comparison) shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.

*WCAP 8385 "Power Distribution Control and Load Following Procedures" and WCAP 9272.A "Westinghouse Reload Safety Evaluation Methodology."

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 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 EVENTS;

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; and

h. Records of annual physical inventory of all sealed source material of record.

RECORD RETENTION (Continued)

6.10.3 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;

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the Quality Assurance 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 SORC and the ORC;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 10 CFR 20.203(c)(5), in lieu of the "control device" or "alarm signal" required by paragraph 10 CFR 20.203(c), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. 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; or

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 levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures 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 RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

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:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

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:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) 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, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

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

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

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

4) 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;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) 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 change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

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*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.