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NUREG-1176

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Technical Specifications
Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Appendix "A" to
License No. NPF-49

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

January 1986

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DEFINITIONS

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 two 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 for valves that are opened under administrative control as permitted by 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 the Containment Leakage Rate Testing Program, 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 of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor 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 position.

DOSE EQUIVALENT I-131

1.10 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 NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

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* In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ 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 in the sample.

1.12 DELETED

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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

1.14 DELETED

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 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.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

OPERABLE - OPERABILITY

1.19 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

1.20 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

1.21 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

1.22 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.

PURGE - PURGING

1.23 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

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower 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

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 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 full-length 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.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

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.

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

STAGGERED TEST BASIS

1.34 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 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

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

UNRESTRICTED AREA

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

VENTING

1.39 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.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 and 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

ALLOWED POWER LEVEL

1.43 APL^{ND} is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.

1.44 APL^{BL} is the maximum allowable power level when transitioning into base load operation.

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.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

Correction letter of 10-2-97

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.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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 Figure 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.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

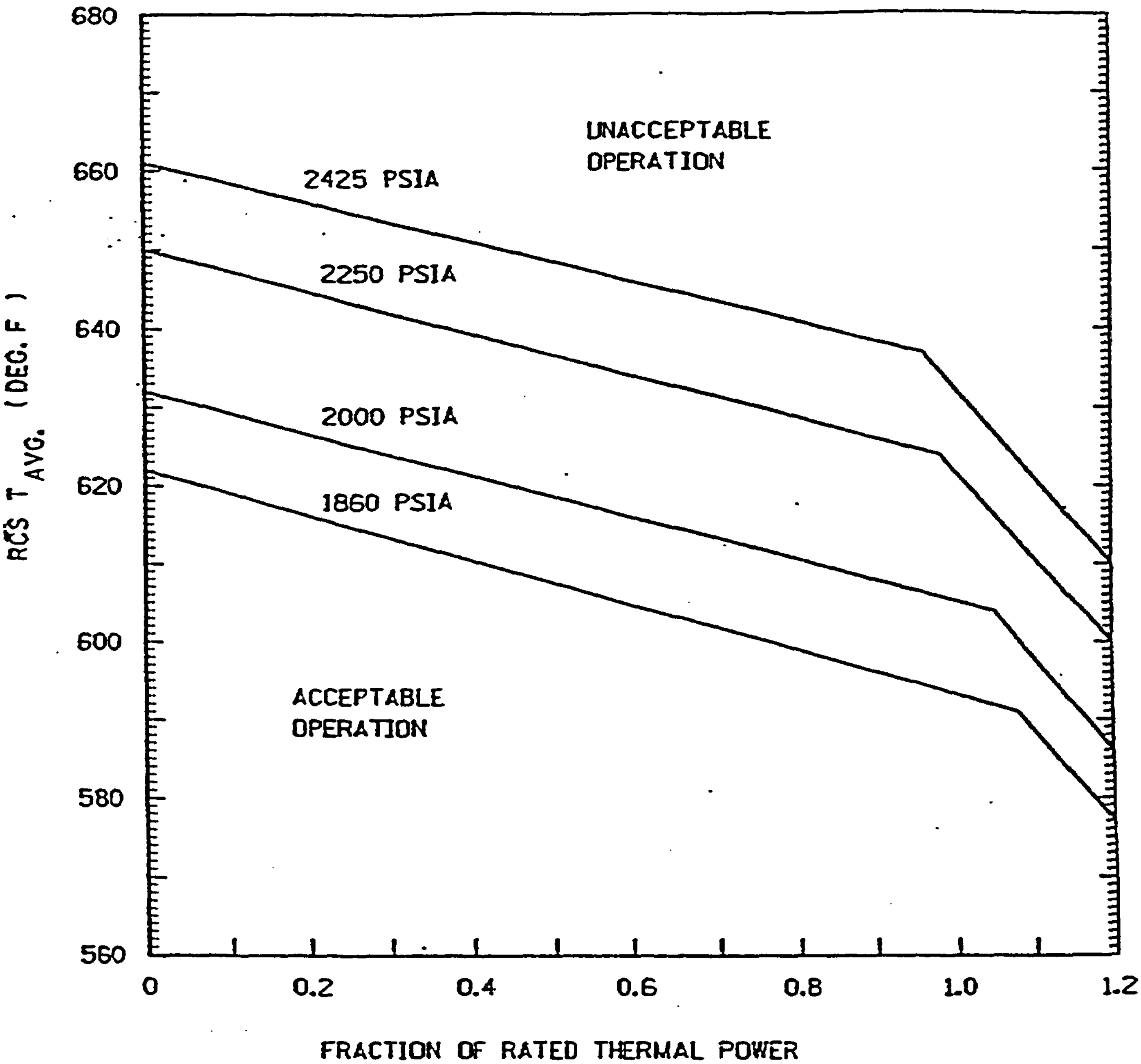

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

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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 Nominal 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 Channel or Interlock Channel Nominal Trip Setpoint inconsistent with the value shown in the Nominal Trip Setpoint column of Table 2.2-1, adjust the Setpoint consistent with the Nominal Trip Setpoint value.

b. With a Reactor Trip System Instrumentation Channel or Interlock Channel found to be inoperable, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	109% of RTP**	≤ 109.6% of RTP**
b. Low Setpoint	25% of RTP**	≤ 25.6% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	5% of RTP** with a time constant ≥ 2 seconds	≤ 5.6% of RTP** with a time constant ≥ 2 seconds
4. Deleted		
5. Intermediate Range, Neutron Flux	25% of RTP**	≤ 27.4% of RTP**
6. Source Range, Neutron Flux	$1 \times 10^{+5}$ cps	≤ $1.06 \times 10^{+5}$ cps
7. Overtemperature ΔT	See Note 1	See Note 2

**RTP = RATED THERMAL POWER

TABLE 2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
8.	Overpower ΔT	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	1900 psia	≥ 1897.6 psia
10.	Pressurizer Pressure-High	2385 psia	≤ 2387.4 psia
11.	Pressurizer Water Level-High	89% of instrument span	≤ 89.3% of instrument span
12.	Reactor Coolant Flow-Low	90% of loop design flow*	≥ 89.8% of loop design flow*
13.	Steam Generator Water Level Low-Low	18.1% of narrow range instrument span	≥ 17.8% of narrow range instrument span
14.	General Warning Alarm	N.A.	N.A.
15.	Low Shaft Speed - Reactor Coolant Pumps	92.4% of rated speed	≥ 92.2% of rated speed

*Minimum Measured Flow Per Loop = 1/4 of the RCS Flow Rate Limit as listed in Section 3.2.3.1.a

TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
16. Turbine Trip		
a. Low Fluid Oil Pressure	500 psig	≥ 450 psig
b. Turbine Stop Valve Closure	1% open	≥ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	1×10^{-10} amp	$\geq 9.0 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7		
1) Power Range Neutron Flux, P-10 input (Note 5)	11% of RTP**	≤ 11.6% of RTP**
2) Turbine Impulse Chamber Pressure, P-13 input	10% RTP** Turbine Impulse Pressure Equivalent	≤ 10.6% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	37.5% of RTP**	≤ 38.1% of RTP**

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
d. Power Range Neutron Flux, P-9	51% of RTP**	≤ 51.6% of RTP**
e. Power Range Neutron Flux, P-10 (Note 6)	9% of RTP**	≥ 8.4% of RTP**
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.
21. DELETED		

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\left(\frac{\Delta T}{\Delta T_0}\right)\frac{(1+\tau_1 s)}{(1+\tau_2 s)} \le K_1 - K_2\frac{(1+\tau_4 s)}{(1+\tau_5 s)}(T - T') + K_3(P - P') - f_1(\Delta I)$$

Where: ΔT is measured Reactor Coolant System ΔT, °F;

ΔT_0 is loop specific indicated ΔT at RATED THERMAL POWER, °F;

$\frac{(1+\tau_1 s)}{(1+\tau_2 s)}$ is the function generated by the lead-lag compensator on measured ΔT;

τ_1 and τ_2 are the time constants utilized in the lead-lag compensator for ΔT, $\tau_1 \ge$ [*] sec, $\tau_2 \le$ [*] sec;

$K_1 \le$ [*]

$K_2 \ge$ [*]/°F;

$\frac{(1+\tau_4 s)}{(1+\tau_5 s)}$ is the function generated by the lead-lag compensator for T_{avg};

τ_4 and τ_5 are the time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \ge$ [*] sec, $\tau_5 \le$ [*] sec;

T is measured Reactor Coolant System average temperature, °F;

T' is loop specific indicated T_{avg} at RATED THERMAL POWER, $\le$ [*]°F;

$K_3 \ge$ [*]/psi

P is measured pressurizer pressure, psia;

P' is nominal pressurizer pressure, $\ge$ [*] psia;

s is the Laplace transform operator, sec^{-1};

(The values denoted with [*] are specified in the COLR.)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power range neutron ion chambers; with nominal gains to be selected based on measured instrument response during plant startup tests calibrations such that:

(1) For $q_t - q_b$ between [*]% and [*]%, f_1, (ΔI) $\geq$ [*], where q_t and q_b are percent RATED THERMAL POWER in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by $\geq$ [*]% of its value at RATED THERMAL POWER.

(3) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by $\geq$ [*]% of its value at RATED THERMAL POWER.

NOTE 2: The maximum channel as left trip setpoint shall not exceed its computed trip setpoint by more than the following:

(1) 0.4% ΔT span for the ΔT channel

(2) 0.4% ΔT span for the T_{avg} channel

(3) 0.4% ΔT span for the pressurizer pressure channel

(4) 0.8% ΔT span for the f(ΔI) channel

(The values denoted with [*] are specified in the COLR.)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\left(\frac{\Delta T}{\Delta T_0}\right)\frac{(1+\tau_1 s)}{(1+\tau_2 s)} \le K_4 - K_5\frac{(\tau_7 s)}{(1+\tau_7 s)}T - K_6(T - T'')$$

Where: ΔT is measured Reactor Coolant System ΔT, °F;

ΔT_0 is loop specific indicated ΔT at RATED THERMAL POWER, °F;

$\frac{(1+\tau_1 s)}{(1+\tau_2 s)}$ is the function generated by the lead-lag compensator on measured ΔT;

τ_1 and τ_2 are the time constants utilized in the lead-lag compensator for ΔT, $\tau_1 \ge$ [*] sec, $\tau_2 \le$ [*] sec;

$K_4 \le$ [*];

$K_5 \ge$[*]/°F for increasing T_{avg} and $K_5 \le$ [*] for decreasing T_{avg};

$\frac{(\tau_7 s)}{(1+\tau_7 s)}$ is the function generated by the rate-lag compensator for T_{avg};

τ_7 is the time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 \ge$ [*] sec;

T is measured average Reactor Coolant System temperature, °F;

T" is loop specific indicated T_{avg} at RATED THERMAL POWER, $\le$ [*]°F;

$K_6 \ge$ [*]/°F when T > T" and $K_6 \le$ [*]/°F when T $\le$ T";

s is the Laplace transform operator, sec^{-1};

(The values denoted with [*] are specified in the COLR.)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 4: The maximum channel as left trip setpoint shall not exceed its computed trip setpoint by more than 0.4% ΔT span for the ΔT channel and 0.4% ΔT span for the T_{avg} channel.

NOTE 5: Setpoint is for increasing power.

NOTE 6: Setpoint is for decreasing power.

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. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. 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.

The DNB design basis is as follows: uncertainties in the WRB-1 or WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.70 (includes measurement uncertainty) 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:

$$F^N_{\Delta H} = 1.70\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming axial imbalance is within the limits of F_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T 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% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3125 psia) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Nominal Trip Setpoints specified in Table 2.2-1 are the nominal values at which the reactor trips are set for each functional unit. The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) are considered the Limiting Safety System Settings as identified in 10CFR50.36 and 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 consistent with the nominal value when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, the Nominal Trip Setpoints may be adjusted in the conservative direction provided the calibration tolerance remains unchanged.

Measurement and Test Equipment accuracy is administratively controlled by plant procedures and is included in the plant uncertainty calculations as defined in WCAP-10991. OPERABILITY determinations are based on the use of Measurement and Test Equipment that conforms with the accuracy used in the plant uncertainty calculation.

The Allowable Value specified in Table 2.2-1 defines the limit beyond which a channel is inoperable. If the process rack bistable setting is measured within the "as left" calibration tolerance, which specifies the difference between the Allowable Value and Nominal Trip Setpoint, then the channel is considered to be OPERABLE.

The methodology, as defined in WCAP-10991 to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

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 the redundant channels and trains, the design approach provides Reactor Trip System functional diversity. The

Bases Change of 8-25-2005

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

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 reactivity insertion 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 Positive Rate

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 all rod ejection accidents.

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. 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. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

Overtemperature ΔT

The Overtemperature ΔT 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 temperature 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 loop 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. Although a direction of conservatism is identified for the Overtemperature ΔT reactor trip function K_2 and K_3 gains, the gains should be set as close as possible to the values contained in Note 1 to ensure that the Overtemperature ΔT setpoint is consistent with the assumptions of the safety analyses.

Overpower ΔT

The Overpower ΔT 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 ΔT

trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT 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.

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 impulse 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 impulse 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 impulse 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 38% 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.

Low Shaft Speed - Reactor Coolant Pumps

The Low Shaft Speed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on two reactor coolant pumps in any two operating reactor coolant loops. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained greater than the design above limit following a four-pump loss of flow event.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

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 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.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power, P-6 becomes active above the Interlock Allowable Value specified on Table 2.2-1 to allow the manual block of the Source Range trip (i.e., prevents premature block of the Source Range trip during reactor startup) and deenergizes the high voltage to the detectors. On decreasing power during a reactor shutdown, Source Range Level trips are automatically reactivated and high voltage restored when P-6 deactivates. The P-6 deactivation will occur at a value below its activation value and may be calibrated to occur below the P-6 Interlock Allowable Value specified on Table 2.2-1 to prevent overlap and chatter based upon the expected bistable drift.

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

Correction letter of 10-21-97

Enclosure

Reactor Trip System Interlocks (Continued)

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 provides input to P-7 to ensure that Reactor Trips on low flow in more than one reactor coolant loop, reactor coolant pump low shaft speed, pressurizer low pressure and pressurizer high level are active when power reaches 11%. It also 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, P-10 resets to automatically reactivate the Intermediate Range trip and the Low Setpoint Power Range trip before power drops below 9%. It also provides input to reset P-7.

P-13 On increasing power, P-13 provides input to P-7 to ensure that Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump low shaft speed, pressurizer low pressure and pressurizer high level are active when power reaches 10%.

On decreasing power, P-13 resets when power drops below 10% and provides input, along with P-10, to reset P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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, except as provided in Specification 3.0.5.

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, except as provided in Specification 3.0.5. 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 Condition 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 permit 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.

3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to Specifications 3.0.1 and 3.0.2 for the system returned to service under administrative controls to perform the testing required to demonstrate OPERABILITY.

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. Failure to meet a Surveillance, whether such failure is experienced during the performance

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified surveillance interval shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable Condition(s) must be entered.

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 Part 50, Section 50.55a;

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:

LIMITING CONDITION FOR OPERATION (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the SHUTDOWN MARGIN not within the limits specified in the COLR, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 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 within the limits specified in the COLR:

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 4.1.1.1.2, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

* See Special Test Exceptions Specification 3.10.1.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least 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.

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.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 3, 4 AND 5 LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the Core Operating Limits Report (COLR).*

APPLICABILITY: MODES 3, 4 and 5

ACTION:

With the SHUTDOWN MARGIN less than the required value, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

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); 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.

4.1.1.1.2.2 Valve 3CHS-V305 shall be verified closed and locked at least once per 31 days.

* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

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SHUTDOWN MARGIN - COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to

a. the limits specified in the CORE OPERATING LIMITS REPORT (COLR) for MODE 5 with RCS loops not filled* or

b. the limits specified in the COLR for MODE 5 with RCS loops filled* with the chemical and volume control system (CVCS) aligned to preclude reactor coolant system boron concentration reduction.

APPLICABILITY: MODE 5 LOOPS NOT FILLED

ACTION:

a. With the SHUTDOWN MARGIN less than the above, immediately initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

b. With the CVCS dilution flow paths not closed and secured in position in accordance with Specification 3.1.1.2(b), immediately close and secure the paths or meet the limits specified in the COLR for MODE 5 with RCS loops not filled.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be within the limits specified in the COLR:

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,

* Additional SHUTDOWN MARGIN requirements, if required, are given in Specification 3.3.5.

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.2.2 At least once per 31 days the following valves shall be verified closed and locked. The valves may be opened on an intermittent basis under administrative controls except as noted.

Valve Number	Valve Function	Valve Position
1. V304(Z-)	Primary Grade Water to CVCS	Closed
2. V120(Z-)	Moderating Hx Outlet	Closed
3. V147(Z-)	BTRS Outlet	Closed
4. V797(Z-)	Failed Fuel Monitoring Flushing	Closed
5. V100(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
6. V571(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
7. V111(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
8. V112(Z-)	Resin Sluice, CVCS Cation Bed Demineralizer	Closed
9. V98(Z-)/V99(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
10. V569(Z-)/V570(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
11. V107(Z-)/V109(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
12. V108(Z-)/V110(Z-)	Resin Sluice, CVCS Mixed Bed Demineralizer	Closed
13. V305(Z-)*	Primary Grade Water to Charging Pumps	Closed

*This valve may not be opened under administrative controls.

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORTS (COLR). The maximum upper limit shall be less positive than $+0.5 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ for all the rods withdrawn, beginning of cycle life (BOL), condition for power levels up to 70% RATED THERMAL POWER with a linear ramp to 0 $\Delta k/k/^{\circ}F$ at 100% RATED THERMAL POWER.

APPLICABILITY: BOL - MODES 1 and 2* only**.
End of Cycle life (EOL) Limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit of Specification 3.1.1.3 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 the above limits 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 EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*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 3.1.1.3, 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 the 300 ppm surveillance limit specified in the COLR (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 the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

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.

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

**See Special Test Exceptions Specification 3.10.3.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length 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 full-length 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 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 Specification 3.1.3.6. 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.

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) 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.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ±12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its group step counter demand height by more than ±12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length 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 full-length 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 92 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

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:

   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:

   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

4.1.3.2.1 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.

4.1.3.2.2 Each of the above required digital rod position indicator(s) shall be determinded 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 24 months.

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 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 full-length 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.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length 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 24 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the core operating limits report (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limits specified in the COLR except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Restore the rod to within the limit specified in the COLR, 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 within the insertion limits specified in the COLR:

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.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the core operating limits report (COLR).

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limits specified in the COLR, 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 insertion limits specified in the COLR, 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.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. The limits specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or

b. Within the target band about the target flux difference during base load operation, specified in the COLR.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes 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.

b. For base load operation above APL^{ND} with the indicted AFD outside of the applicable target band about the target flux differences:

   1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue base load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

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* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS

4.2.1.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE:

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.

4.2.1.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.1.3 When in base load operation, 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.

4.2.1.1.4 When in base load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.1.3 or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) provided in the core operating limits report (COLR).

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

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. For RAOC operation with Specification 4.2.2.1.2.b not being satisfied or for base load operation with Specification 4.2.2.1.4.b not being satisfied:

(1) 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 ΔT Trip setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit, and

(2) Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by item (1) above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limits.

b. For RAOC operation with Specification 4.2.2.1.2.c not being satisfied, one of the following actions shall be taken:

(1) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by at least 1% AFD for each percent $F_Q(Z)$ exceeds its limits. Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

(2) Verify that the requirements of Specification 4.2.2.1.3 for base load operation are satisfied and enter base load operation.

Where it is necessary to calculate the percent that $F_Q(Z)$ exceeds the limits for item (1) above, it shall be calculated as the maximum percent over the core height (Z), consistent with Specification 4.2.2.1.2.f, that $F_Q(Z)$ exceeds its limit by the following expression:

$$\left[\left[\frac{F_Q^M(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)}\right] - 1\right] \times 100 \text{ for } P > 0.5$$

$$\left[\left[\frac{F_Q^M(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)}\right] - 1\right] \times 100 \text{ for } P \leq 0.5$$

c. For base load operation with Specification 4.2.2.1.4.c not being satisfied, one of the following actions shall be taken:

(1) Place the core in an equilibrium condition where the limit in 4.2.2.1.4.c is satisfied, and remeasure $F_Q^M(Z)$, or

(2) 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 ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds its limit shall be calculated as the maximum percent over the core height (Z), consistent with Specification 4.2.2.1.4.f, by the following expression:

$$\left[\left[\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(Z)}\right] - 1\right] \times 100 \text{ for } P \geq APL^{ND}$$

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 For RAOC operation, $F_Q(Z)$ 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. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel Factor, $F_Q^M(Z)$ by increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e. 0-100% inclusive.

Correction letter of 6-17-99

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \quad for\ P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{W(Z) \times 0.5} \quad for\ P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and W(Z) is the cycle-dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.6.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

(1) Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined,* or

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

e. With the maximum value of

$$\frac{F_Q^M(Z)}{K(Z)}$$

over the core height (Z) increasing since the previous determination of $F_Q^M(Z)$, either of the following actions shall be taken:

(1) Increase $F_Q{}^M(Z)$ by an appropriate factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.1.2.c, or

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map outlined.

(2) $F_Q^M(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that the maximum value of

$$\frac{F_Q^M(Z)}{K(Z)}$$

over the core height (Z) is not increasing.

f. The limits specified in Specifications 4.2.2.1.2c and 4.2.2.1.2e above are not applicable in the following core plane regions:

(1) Lower core region from 0% to 15%, inclusive.
(2) Upper core region from 85% to 100%, inclusive.

4.2.2.1.3 Base load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering base load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.1.2 for at least the previous 24 hours. Maintain base load operation surveillance (AFD within the target band limit about the target flux difference of Specification 3.2.1.1) during this time period. Base load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.1.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \times 100\%$$

over the core height (Z) where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is F_Q^{RTP}. $W(Z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, K(Z), and $W(Z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

SURVEILLANCE REQUIREMENTS (Continued)

b. During base load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.1.3.a shall be satisfied before reentering base load operation.

4.2.2.1.4 During base load operation $F_Q(Z)$ 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 above APL^{ND}.

b. Evaluate the computed heat flux hot channel factor by performing both of the following:

(1) Determine the computed heat flux hot channel factor, $F_Q^M(Z)$, by increasing the measured $F_Q^M(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties, and

(2) Verify that $F_Q^M(Z)$ satisfies the requirements of Specification 3.2.2.1 for all core plane regions, i.e., 0 - 100% inclusive.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{BL}} \quad for\ P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height, P is the relative THERMAL POWER, and $W(Z)_{BL}$ is the cycle-dependent function that accounts for limited power distribution transients encountered during base load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the COLR as per Specification 6.9.1.6.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

(1) Prior to entering base load operation after satisfying Section 4.2.2.1.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

(2) At least once per 31 Effective Full Power Days.

e. With the maximum value of

$$\frac{F_Q^M(Z)}{K(Z)}$$

over the core height (Z) increasing since the previous determination of $F_Q^M(Z)$, either of the following actions shall be taken:

(1) Increase $F_Q^M(Z)$ by appropriate factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.1.4.c, or

(2) $F_Q^M(Z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive maps indicate that the maximum value of

$$\frac{F_Q^M(Z)}{K(Z)}$$

over the core height (Z) is not increasing.

f. The limits specified in 4.2.2.1.4.c and 4.2.2.1.4.e are not applicable in the following core plane regions:

(1) Lower core region 0% to 15%, inclusive.
(2) Upper core region 85% to 100%, inclusive.

4.2.2.1.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1.2 or 4.2.2.1.4, 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.

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

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 371,920 gpm, and

b. $F^N_{\Delta H} \leq F^{RTP}_{\Delta H}$ [1.0 + $PF_{\Delta H}$ (1.0 - P)]

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the CORE OPERATING LIMITS REPORT (COLR),

4) $PF_{\Delta H}$ - The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ 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.

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 that $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ 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 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

4.2.3.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.1.2 RCS total flow rate and $F^N_{\Delta H}$ shall be determined to be within the acceptable range:

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.

4.2.3.1.3 The indicated RCS total flow rate shall be verified to be within the acceptable range at least once per 12 hours when the most recently obtained value of $F^N_{\Delta H}$, obtained per Specification 4.2.3.1.2, is assumed to exist.

4.2.3.1.4 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.1.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.1.6 If the feedwater venturis are not inspected at least once per 18 months, an additional 0.1% will be added to the total RCS flow measurement uncertainty.

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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 but less than or equal to 1.09:

   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.

*See Special Test Exceptions Specification 3.10.2.

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   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. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the 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.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

   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.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

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 that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR):

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

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 DNB-related parameters shall be verified to be within the limits specified in the COLR at least once per 12 hours.

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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.

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 verified to be within its limit at least once per 18 months. Neutron detectors and speed sensors are exempt from response time verification. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel (to include input relays to both trains) per function such that all channels are verified 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 2	1 1	2 2	1, 2 3*, 4*, 5*	1 11
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	11
7. Overtemperature ΔT	4	2	3	1, 2	6
8. Overpower ΔT	4	2	3	1, 2	6
9. Pressurizer Pressure--Low	4	2	3	1**	6 (1)
10. Pressurizer Pressure--High	4	2	3	1, 2	6 (1)
11. Pressurizer Water Level--High	3	2	2	1**	6

TABLE 3.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12.	Reactor Coolant Flow--Low					
	a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
	b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop	1	6
13.	Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2	6 (1)
14.	Low Shaft Speed--Reactor Coolant Pumps	4-1/pump	2	3	1**	6
15.	Turbine Trip					
	a. Low Fluid Oil Pressure	3	2	2	1***	12
	b. Turbine Stop Valve Closure	4	4	4	1***	6
16.	Deleted					
17.	Reactor Trip System Interlocks					
	a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
	b. Low Power Reactor Trips Block, P-7					
	Power Range Neutron Flux, P-10 Input	4	2	3	1	8
	or					
	Turbine Impulse Chamber Pressure, P-13 Input	2	1	2	1	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
17. Reactor Trip System Interlocks (Continued)					
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1,2	8
18. Reactor Trip Breakers(2)	2 2	1 1	2 2	1, 2 3*, 4*, 5*	10, 13 11
19. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	13A 11
20. DELETED					
21. DELETED					

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

**Above the P-7 (At Power) Setpoint.

***Above the P-9 (Reactor Trip/Turbine Trip Interlock) Setpoint.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) The applicable MODES and ACTION statements for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

(2) Including any reactor trip bypass breakers that are racked in and closed for bypassing a reactor trip breaker.

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 6 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.

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, and

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 - (Not used)

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 - (Not used)

ACTION 8 - 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 STATEMENTS (Continued)

ACTION 9 - (Not used)

ACTION 10 - 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 11 - 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 12 - 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. When the Minimum Channels OPERABLE requirement is met, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of the Turbine Control Valves.

ACTION 13 - With one of the diverse trip features (undervoltage or shunt trip attachments) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. 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.

ACTION 13A - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is operable.

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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*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4, 5)	Q	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1), Q(9)	N.A.	N.A.	2**, 3*, 4*, 5*
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(18)	N.A.	N.A.	1*****
10. Pressurizer Pressure--High	S	R	Q(18)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1*****
12. Reactor Coolant Flow--Low	S	R	Q	N.A.	N.A.	1

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
13. Steam Generator Water Level--Low-Low	S	R	Q(18)	N.A.	N.A.	1, 2
14. Low Shaft Speed - Reactor Coolant Pumps	N.A.	R(13)	Q	N.A.	N.A.	1
15. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)****	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)****	N.A.	1
16. Deleted						
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	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
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1

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 Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20. DELETED						
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(7, 15) R(16)	N.A.	1, 2, 3*, 4*, 5*
22. DELETED						

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

**** Above the P-9 (Reactor Trip/Turbine Interlock) Setpoint.

***** Above the P-7 (At Power) Setpoint

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference 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%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Source Range, Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

(8) (Not used)

(9) Quarterly surveillance in MODES 3*, 4*, and 5* 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.

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) (not used)

(13) Reactor Coolant Pump Shaft Speed Sensor may be excluded from CHANNEL CALIBRATION.

(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.

(16) Automatic undervoltage trip.

(17) (not used).

(18) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 should be reviewed for applicability.

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 Nominal Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation Channel or Interlock Channel Nominal Trip Setpoint inconsistent with the value shown in the Nominal Trip Setpoint column of Table 3.3-4, adjust the Setpoint consistent with the Nominal Trip Setpoint value.

b. With an ESFAS Instrumentation Channel or Interlock Channel found to be inoperable, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status.

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 verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel (to include input relays to both trains) per function such that all channels are verified 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.

*The provisions of Specification 4.0.4 are not applicable for response time verification of steam line isolation for entry into MODE 4 and MODE 3 and turbine driven auxiliary feedwater pump for entry into MODE 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 (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3	20
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20
e. Steam Line Pressure--Low	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	1, 2, 3#	20
2. Containment Spray (CDA)					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19

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 (CDA) (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-3	4	2	3	1, 2, 3, 4	17
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

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
3. Containment Isolation (Continued)					
3. Containment Pressure--High-3	4	2	3	1, 2, 3, 4	17
c. DELETED					
4. Steam Line Isolation					
a. Manual Initiation					
1. Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3, 4	24
2. System	2	1	2	1, 2, 3, 4	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	22
c. Containment Pressure-- High-2	3	2	2	1, 2, 3, 4	20
d. Steam Line Pressure-- Low	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	1, 2, 3#	20
e. Steam Line Pressure - Negative Rate--High	3/steam line in each operating loop	2/steam line in any operating loop	2/steam line in each operating loop	3****	20

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
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuaion Relays	2	1	2	1, 2	25
b. Steam Generator Water Level -- High-High (P-14)	4/stm. gen. in each operating loop	2/stm. gen. in any operating loop	3/stm. gen. in each operating loop	1, 2, 3	20, 21
c. Safety Injection Actuation Logic	2	1	2	1, 2	22
d. T_{ave} Low Coincident with P-4	1 T_{ave}/loop	1 T_{ave} in any two loops	1 T_{ave} in any three loops	1, 2	20

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					
a. Manual Initiation	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
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	20
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	20
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	2	1	2	1, 2, 3	19

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)					
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA functions and requirements.				
7. Control Building Isolation					
a. Manual Actuation	2	1	2	*	19
b. Manual Safety Injection Actuation	2	1	2	1, 2, 3, 4	19
c. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
d. Containment Pressure--High-1	3	2	2	1, 2, 3	16
e. Control Building Inlet Ventilation Radiation	2/intake	1	2/intake	*	18
8. Loss of Power					
a. 4 kV Bus Under-voltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	27
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	27

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. Engineering Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23
10. Emergency Generator Load Sequencer	2	1	2	1, 2, 3, 4	15

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

\# The Steamline Isolation Logic and Safety Injection Logic for this trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

* MODES 1, 2, 3, 4, 5 and 6.
During fuel movement within containment or the spent fuel pool.

**** Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE

ACTION 16 - 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 17 - 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 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 7 days. After 7 days, or if no channels are OPERABLE, immediately suspend fuel movement, if applicable, and be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - 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.

ACTION STATEMENTS (Continued)

ACTION 20 - 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 4.3.2.1.

ACTION 21 - 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 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 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, 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 24 - 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 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 26 - DELETED

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - a. 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:

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

2. 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 4.3.2.1.

b. With the number of OPERABLE channels one less than the Minimum Channels required OPERABLE:

1. Place one channel in bypass and other channel in trip condition within one hour and restore one channel to OPERABLE status in 48 hours,

OR

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Containment Pressure--High 1	17.7 psia	≤ 17.9 psia
d. Pressurizer Pressure--Low		
1) Channels I and II	1892 psia	≥ 1889.6 psia
2) Channel III and IV	1892 psia	≥ 1889.6 psia
e. Steam Line Pressure--Low	658.6 psig*	≥ 654.7 psig*
2. Containment Spray (CDA)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-3	22.7 psia	≤ 22.9 psia
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.

TABLE 3.3-4
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation (Continued)		
2. Automatic Actuation Logic and Actuation Relays	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	N.A.	N.A.
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3. Containment Pressure--High-3	22.7 psia	≤ 22.9 psia
c. DELETED		
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-2	17.7 psia	≤ 17.9 psia
d. Steam Line Pressure--Low	658.6 psig*	≥ 654.7 psig*
e. Steam Line Pressure - Negative Rate--High	100 psi/s**	≤ 103.9 psi/s**

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	80.5% of narrow range instrument span.	≤ 80.8% of narrow range instrument span.
c. Safety Injection Actuation Logic	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
d. T_{ave} Low Coincident with Reactor Trip (P-4)	564°F	≥ 563.6°F
6. Auxiliary Feedwater		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level--Low-Low		
1) Start Motor-Driven Pumps	18.1% of narrow range instrument span.	≥ 17.8% of narrow range instrument span.

TABLE ‌ .-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)		
2) Start Turbine-Driven Pumps	18.1% of narrow range instrument span.	≥ 17.8% of narrow range instrument span.
d. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
e. Loss-of-Offsite Power Start Motor-Driven Pumps	2800V	≥ 2720V
f. Containment Depressurization Actuation (CDA) Start Motor-Driven Pumps	See Item 2. above for all CDA Trip Setpoints and Allowable Values.	
7. Control Building Isolation		
a. Manual Actuation	N.A	N.A.
b. Manual Safety Injection Actuation	N.A	N.A.
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
d. Containment Pressure--High 1	17.7 psia	≤ 17.9 psia
e. Control Building Inlet Ventilation Radiation	$\leq 1.5 \times 10^{-5} \mu ci/cc$	$\leq 1.5 \times 10^{-5} \mu ci/cc$

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power		
a. 4 kV Bus Undervoltage (Loss of Voltage)	2800 volts with a ≤ 2 second time delay.	≥ 2720 volts with a ≤ 2 second time delay.
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	3730 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.	≥ 3706 volts with a ≤ 8 second time delay with ESF actuation or ≤ 300 second time delay without ESF actuation.
9. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	1999.7 psia	≤ 2002.1 psia
b. Low-Low T_{avg}, P-12	553°F	≥ 552.6°F
c. Reactor Trip, P-4	N.A.	N.A.
10. Emergency Generator Load Sequencer	N.A.	N.A.

TABLE 3.3-4 (Continued)

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 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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 Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
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(4)	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	Q	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
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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
3. Containment Isolation								
a. Phase "A" Isolation								
1. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
3. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
3. Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. DELETED								
4. Steam Line Isolation								
d. Manual Initiation								
1. Individual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2. System	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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
4. Steam Line Isolation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	Q	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(4)	1, 2
b. Steam Generator Water Level-High-High	S	R	Q	N.A.	M(1)	M(1)	Q(4)	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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
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 and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	Q	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	See Item 8. below for all Loss of Power Surveillance.							
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	*
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
d. Containment Pressure--High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

Correction letter of 3-29-2002

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	Q	N.A.	N.A.	N.A.	N.A.	*
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	Q(1, 2)	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

2. This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated operable by the performance of an ACTUATION LOGIC TEST at least once per 92 days.

3. On a monthly basis, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the channel calibration required once per 18 months.

4. For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP- 13900, the surveillance interval for slave relay testing is R.

* MODES 1, 2, 3 , 4, 5 and 6.
During fuel movement within containment or the spent fuel pool.

Correction letter of 7-8-2004

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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

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. Containment					
a. Deleted					
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2. Fuel Storage Pool Area Monitors					
a. Radiation Level	1	2	*	≤ 15 mR/h	28

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* With fuel in the fuel storage pool areas.

ACTION STATEMENTS

ACTION 27 - Not used.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, fuel movement 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 29 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Deleted				
b. RCS Leakage Detection				
1) Particulate Radio-activity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Fuel Storage Pool Area Monitors				
a. Radiation Level	S	R	Q	*

TABLE NOTATIONS

* With fuel in the fuel storage pool area.

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REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown Instrumentation 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 one or more Remote Shutdown Instrumentation 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.

c. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

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

4.3.3.5.2 Each Remote Shutdown Instrumentation transfer switch, power and control circuit including the actuated components, shall be demonstrated OPERABLE at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	1/trip breaker	1/trip breaker
2. Pressurizer Pressure	Aux. Shutdown Panel	2	1
3. Pressurizer Level	Aux. Shutdown Panel	2	1
4. Steam Generator Pressure	Aux. Shutdown Panel	2/steam generator	1/steam generator
5. Steam Generator Water Level	Aux. Shutdown Panel	2/steam generator	1/steam generator
6. Auxiliary Feedwater Flow Rate	Aux. Shutdown Panel	1/steam generator	1/steam generator
7. Loop Hot Leg Temperature	Aux. Shutdown Panel	1/loop	1/loop
8. Loop Cold Leg Temperature	Aux. Shutdown Panel	1/loop	1/loop
9. Reactor Coolant System Pressure (Wide Range)	Aux. Shutdown Panel	2	1
10. DWST Level	Aux. Shutdown Panel	2	1
11. RWST Level	Aux. Shutdown Panel	2	1
12. Containment Pressure	Aux. Shutdown Panel	2	1
13. Emergency Bus Voltmeters	Aux. Shutdown Panel	1/train	1/train
14. Source Range Count Rate	Aux. Shutdown Panel	2	1
15. Intermediate Range Flux	Aux. Shutdown Panel	2	1
16. Boric Acid Tank Level	Aux. Shutdown Panel	2/tank	1/tank

TRANSFER SWITCHES	SWITCH LOCATION
1. Auxiliary Feedwater Isolation FWA*MOV35A	Transfer Switch Panel
2. Auxiliary Feedwater Isolation FWA*MOV35B	Transfer Switch Panel
3. Auxiliary Feedwater Isolation FWA*MOV35C	Transfer Switch Panel
4. Auxiliary Feedwater Isolation FWA*MOV35D	Transfer Switch Panel
5. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23A	Transfer Switch Panel
6. Auxiliary Feedwater Pump Ah. Suction FWA*AOV23B	Transfer Switch Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

TRANSFER SWITCHES	SWITCH LOCATION
7. Turbine Driven Pump Steam Supply MSS*AOV31A	Transfer Switch Panel
8. Turbine Driven Pump Steam Supply MSS*AOV31B	Transfer Switch Panel
9. Turbine Driven Pump Steam Supply MSS*AOV31D	Transfer Switch Panel
10. Reactor Vessel Head Vent Isolation RCS*SV8095A	Transfer Switch Panel
11. Reactor Vessel Head Vent Isolation RCS*SV8095B	Transfer Switch Panel
12. Reactor Vessel Head Vent Isolation RCS*SV8096A	Transfer Switch Panel
13. Reactor Vessel Head Vent Isolation RCS*SV8096B	Transfer Switch Panel
14. Reactor Vessel to Excess Letdown RCS*MV8098	Transfer Switch Panel
15. Pressurizer Level Control RCS*LCV459	Transfer Switch Panel
16. Pressurizer Level Control RCS*LCV460	Transfer Switch Panel
17. Letdown Orifice Isolation CHS*AV8149A	Transfer Switch Panel
18. Letdown Orifice Isolation CHS*AV8149B	Transfer Switch Panel
19. Letdown Orifice Isolation CHS*AV8149C	Transfer Switch Panel
20. Volume Control Tank Outlet Isolation CHS*LCV112B	Transfer Switch Panel
21. Volume Control Tank Outlet Isolation CHS*LCV112C	Transfer Switch Panel
22. RWST to CHS Pump Suction CHS*LCV112D	Transfer Switch Panel
23. RWST to CHS Pump Suction CHS*LCV112E	Transfer Switch Panel
24. Charging to RCS Isolation CHS*AV8146	Transfer Switch Panel
25. Charging to RCS Isolation CHS*AV8147	Transfer Switch Panel
26. Boric Acid Gravity Feed CHS*MV8507A	Transfer Switch Panel
27. Boric Acid Gravity Feed CHS*MV8507B	Transfer Switch Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

TRANSFER SWITCHES	SWITCH LOCATION
28. Charging Header Isolation Bypass CHS*MV8116	Transfer Switch Panel
29. Pressurizer Heater Backup RCS*H1A (Group A)	Transfer Switch Panel
30. Pressurizer Heater Backup RCS*H1B (Group B)	Transfer Switch Panel

CONTROL CIRCUITS	SWITCH LOCATION
1. Auxiliary Feedwater Flow Control FWA*HV31A	Auxiliary Shutdown Panel
2. Auxiliary Feedwater Flow Control FWA*HV31B	Auxiliary Shutdown Panel
3. Auxiliary Feedwater Flow Control FWA*HV31C	Auxiliary Shutdown Panel
4. Auxiliary Feedwater Flow Control FWA*HV31D	Auxiliary Shutdown Panel
5. Auxiliary Feedwater Flow Control FWA*HV32A	Auxiliary Shutdown Panel
6. Auxiliary Feedwater Flow Control FWA*HV32B	Auxiliary Shutdown Panel
7. Auxiliary Feedwater Flow Control FWA*HV32C	Auxiliary Shutdown Panel
8. Auxiliary Feedwater Flow Control FWA*HV32D	Auxiliary Shutdown Panel
9. Auxiliary Feedwater Flow Control FWA*HV36A	Auxiliary Shutdown Panel
10. Auxiliary Feedwater Flow Control FWA*HV36B	Auxiliary Shutdown Panel
11. Auxiliary Feedwater Flow Control FWA*HV36C	Auxiliary Shutdown Panel
12. Auxiliary Feedwater Flow Control FWA*HV36D	Auxiliary Shutdown Panel

TABLE 3.3-9 (Continued)

REMOTE SHUTDOWN INSTRUMENTATION

CONTROL CIRCUITS	SWITCH LOCATION
13. Reactor Vessel to PRT Control RCS*HCV442A	Auxiliary Shutdown Panel
14. Reactor Vessel to PRT Control RCS*HCV442B	Auxiliary Shutdown Panel
15. Charging Header Flow Control CHS*HCV190A	Auxiliary Shutdown Panel
16. Charging Header Flow Control CHS*HCV190B	Auxiliary Shutdown Panel
17. Excess Letdown Flow Control CHS*HCV123	Auxiliary Shutdown Panel
18. Charging Flow Control CHS*FCV121	Auxiliary Shutdown Panel
19. Low Pressure Letdown Control CHS*PCV131	Auxiliary Shutdown Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Water Level	M	R
6. Auxiliary Feedwater Flow Rate	M	R
7. Loop Hot Leg Temperature	M	R
8. Loop Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure (Wide Range)	M	R
10. DWST Level	M	R
11. RWST Level	M	R
12. Containment Pressure	M	R
13. Emergency Bus Voltmeters	M	R
14. Source Range Count Rate	M*	R
15. Intermediate Range Amps	M	R
16. Boric Acid Tank Level	M	R

* When below P-6 (intermediate range neutron flux interlock setpoint).

INSTRUMENTATION

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 except the containment area high range radiation monitor, and reactor vessel water level, 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 containment area-high range radiation monitor, and reactor vessel water level 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 containment area-high range radiation monitor less than required by either the total or 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. Deleted

e. With the number of OPERABLE channels for the reactor vessel water level monitor less than the Total number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the

action taken, the cause of the inoperability, and the plans and schedule for restoring the channel to OPERABLE status.

f. With the number of OPERABLE channels for the reactor vessel water level monitor less than the minimum channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the channel(s) to OPERABLE status; and

3. Restore the channel(s) to OPERABLE status at the next scheduled refueling.

g. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

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

4.3.3.6.2 Deleted

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Normal Range	2	1
b. Extended Range	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Demineralized Water Storage Tank Water Level	2	1
11. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
12. Reactor Coolant System Subcooling Margin Monitor	2	1
13. Containment Water Level (Wide Range)	2	1
14. Core Exit Thermocouples	4/core quadrant	2/core quadrant
15. DELETED		

TABLE 3.3-10 (Continued)
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
16.	Containment Area - High Range Radiation Monitor	2	1
17.	Reactor Vessel Water Level	2*	1*
18.	Deleted		
19.	Neutron Flux	2	1

* A channel consists of eight sensors in a probe. A channel is operable if four or more sensors, half or more in the upper head region and half or more in the upper plenum region, are operable.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure		
a. Normal Range	M	R
b. Extended Range	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Demineralized Water Storage Tank Water Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. Containment Water Level (Wide Range)	M	R
14. Core Exit Thermocouples	M	R
15. DELETED		

TABLE 4.3-7 (Continued)
ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
16.	Containment Area - High Range Radiation Monitor	M	R*
17.	Reactor Vessel Water Level	M	R**
18.	Deleted		
19.	Neutron Flux	M	R

* 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.

** Electronic calibration from the ICC cabinets only.

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3/4.3.5 SHUTDOWN MARGIN MONITOR

LIMITING CONDITION FOR OPERATION

3.3.5 Two channels of Shutdown Margin Monitors shall be OPERABLE

a. With a minimum count rate as designated in the CORE OPERATING LIMITS REPORT (COLR), or

b. If the minimum count rate in Specification 3.3.5.a cannot be met, then the Shutdown Margin Monitors may be made operable with a lower minimum count rate, as specified in the COLR, by borating the Reactor Coolant System above the requirements of Specification 3.1.1.1.2 or 3.1.1.2. The additional boration shall be:

1. A minimum of 150 ppm above the SHUTDOWN MARGIN requirements specified in the COLR for MODE 3, or

2. A minimum of 350 ppm above the SHUTDOWN MARGIN requirements specified in the COLR for MODE 4, MODE 5 with RCS loops filled, and MODE 5 with RCS loops not filled.

APPLICABILITY: MODES 3*, 4, and 5.

ACTION:

a. With one Shutdown Margin Monitor inoperable, restore the inoperable channel to OPERABLE status within 48 hours.

b. With both Shutdown Margin Monitors inoperable or one Shutdown Margin Monitor inoperable for greater than 48 hours, immediately suspend all operations involving positive reactivity changes via dilution and rod withdrawal. Verify the valves listed in Specification 4.1.1.2.2 are closed and secured in position within the next 4 hours and at least once per 14 days thereafter.** Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1.2 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

---

* The shutdown margin monitors may be blocked during reactor startup in accordance with approved plant procedures.

** The valves may be opened on an intermittent basis under administrative controls as noted in Surveillance 4.1.1.2.2.

3/4.3.5 SHUTDOWN MARGIN MONITOR (continued)

SURVEILLANCE REQUIREMENTS

4.3.5 a. Each of the above required shutdown margin monitoring instruments shall be demonstrated OPERABLE by an ANALOG CHANNEL OPERATIONAL TEST at least once per 92 days that shall include verification that the Shutdown Margin Monitor is set per the Core Operating Limits Report (COLR).

b. At least once per 24 hours VERIFY the minimum count rate (counts/sec) as defined within the COLR.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Four reactor coolant loops shall be OPERABLE and 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.

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE, with at least three reactor coolant loops in operation when the Control Rod Drive System is capable of rod withdrawal or with at least one reactor coolant loop in operation when the Control Rod Drive System is not capable of rod withdrawal:*

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 less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

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

SURVEILLANCE REQUIREMENTS

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

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 17% 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.

---

*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.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 Either:*,**

a. With the Control Rod Drive System capable of rod withdrawal, at least two RCS loops shall be OPERABLE and in operation, or

b. With the Control Rod Drive System not capable of rod withdrawal, at least two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and at least one of these loops shall be in operation. For RCS loop(s) to be OPERABLE, at least one reactor coolant pump (RCP) shall be in operation.

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.

b. With less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

c. 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.

---

*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.

** The first reactor coolant pump shall not be started when any RCS loop wide range cold leg temperature is ≤ 226°F unless:

a. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

b. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required 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 17% at least once per 12 hours.

4.4.1.3.3 The required loop(s) shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one 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 17%.

APPLICABILITY: MODE 5 with at least two reactor coolant loops filled***.

---

*a. 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.

b. All RHR loops may be removed from operation during a planned heatup to MODE 4 when at least one RCS loop is OPERABLE and in operation and when two additional steam generators are OPERABLE as required by LCO 3.4.1.4.1.b.

**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 first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With less than the required RHR loop(s) OPERABLE 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.

4.4.1.4.1.3 The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

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 less than two reactor coolant loops 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.

---

*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.

***The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

COLD SHUTDOWN - LOOPS NOT FILLED

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

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

REACTOR COOLANT SYSTEM

LOOP STOP VALVES

LIMITING CONDITION FOR OPERATION

3.4.1.5 Each RCS loop stop valve shall be open and the power removed from the valve operator.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With power available to one or more loop stop valve operators, remove power from the loop stop valve operators within 30 minutes or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b.[1] With one or more RCS loop stop valves closed, maintain the valve(s) closed and be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.5 Verify each RCS loop stop valve is open and the power removed from the valve operator at least once per 31 days.

[1] All required actions of Action Statement 3.4.1.5.b shall be completed whenever this action is entered.

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops, and

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The isolated loop boron concentration shall be determined to be greater than or equal to the boron concentration required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6 within 2 hours prior to opening the hot or cold leg stop valve.

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3/4.4.2 SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2500 psia ± 3%.**

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 226°F.

ACTION:

With one pressurizer Code safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes. If the inoperable valve is not restored to OPERABLE status within 15 minutes, or if two or more pressurizer Code safety valves are inoperable, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with any RCS cold leg temperature ≤ 226°F within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

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

**The lift setting shall be within ± 1% following pressurizer Code safety valve testing.

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3/4.4.3 PRESSURIZER

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level maintained at programmed level +/-6% of full scale (Figure 3.4-5).

APPLICABILITY: MODES 1 and 2.

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.

b. With pressurizer water level outside the parameters described in Figure 3.4-5, within 2 hours restore programmed level to within +/- 6% of full scale, or be in at least HOT STANDBY within the next 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water level shall be verified to be within programmed level +/- 6% of full scale at least once per 12 hours.

4.4.3.1.2 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 each refueling interval.

PRESSURIZER LEVEL CONTROL

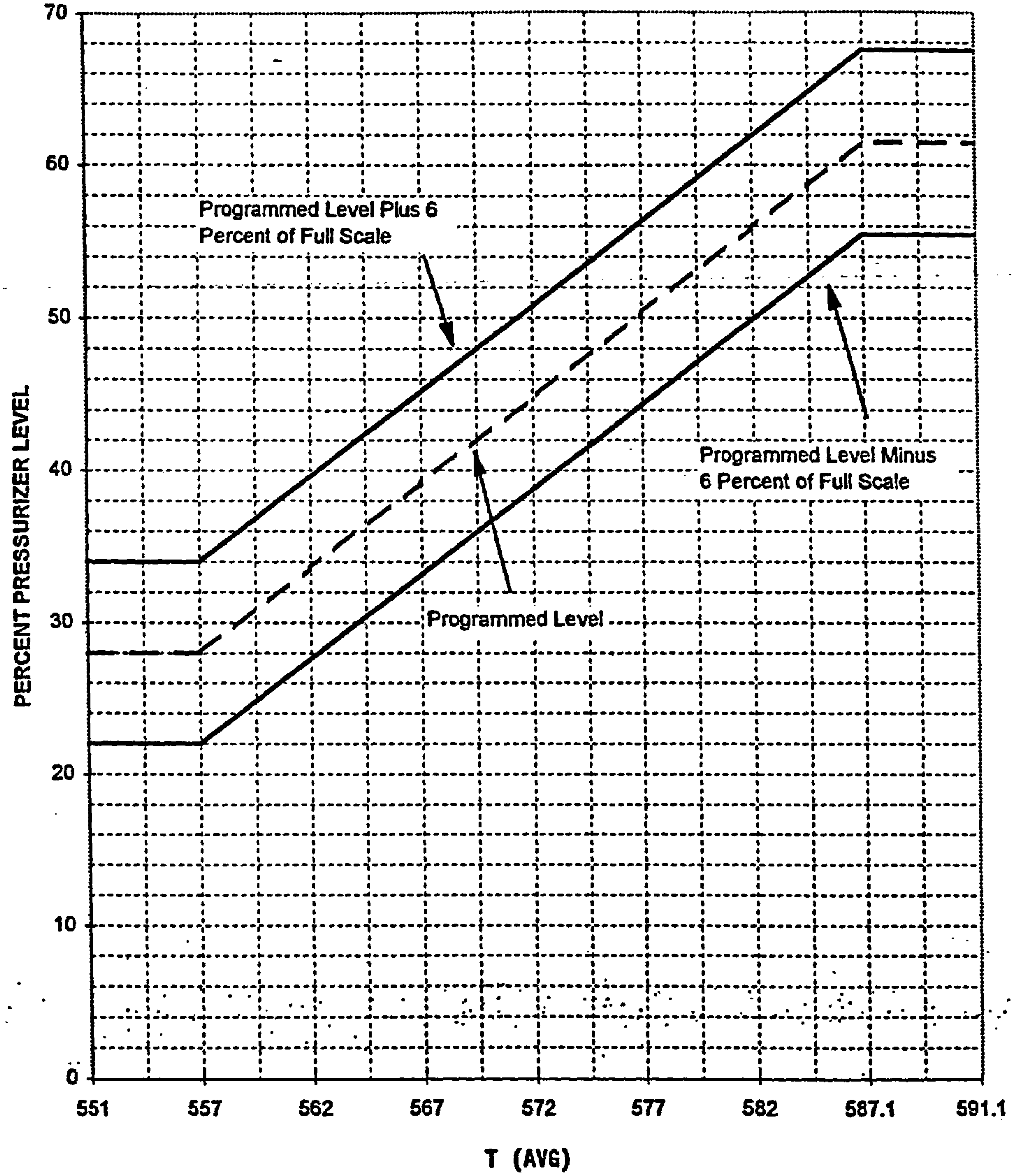

FIGURE 3.4-5

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.3.2 The pressurizer shall be OPERABLE with:

a. at least two groups of pressurizer heaters, each having a capacity of at least 175 kW; and

b. water level less than or equal to 89% of full scale.

APPLICABILITY: MODE 3

ACTION:

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

b. With the pressurizer otherwise inoperable, be in HOT SHUTDOWN within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The pressurizer water level shall be determined to be less than or equal to 89% of full scale at least once per 12 hours.

4.4.3.2.2 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 each refueling interval.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both 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) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 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 HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valve(s) inoperable, within 1 hour restore the block valve(s) to OPERABLE status, or place its associated PORV(s) control switch to "CLOSE." Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to operable status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL CALIBRATION at least once per 24 months; and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4 at least once per 24 months; and

c. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV high pressurizer pressure actuation channels, but excluding valve operation, at least once each quarter; and

d. Verify the PORV high pressure automatic opening function is enabled at least once per 12 hours.

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 with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator associated with an operating RCS loop shall be OPERABLE.

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

ACTION:

With one or more steam generators associated with an operating RCS loop 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 the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

a. 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.

*Except the surveillance related to Steam Generator Inspection, due no later than September 24, 1998, may be deferred until the next refueling outage or no later than July 1, 1999, whichever is earlier.

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; or an inspection from the point of entry (Hot Leg or Cold Leg Side) completely around the U-bend to the opposite tube end.

SURVEILLANCE REQUIREMENTS (Continued)

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 in a Special Report to the Commission pursuant to Specification 6.9.2 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	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) 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.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

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. Either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. The Containment Drain Sump Level or Pumped Capacity Monitoring System

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

ACTION:

a. With both the Containment Atmosphere Gaseous and Particulate Radioactivity Monitors INOPERABLE, operation may continue for up to 30 days provided the Containment Drain Sump Level or Pumped Capacity Monitoring System is OPERABLE and gaseous grab samples of the containment atmosphere are obtained at least once per 12 hours and analyzed for gross noble gas activity within the subsequent 2 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 Containment Drain Sump Level or Pumped Capacity Monitoring System INOPERABLE, operation may continue for up to 30 days provided either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System is OPERABLE; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN withi.. the followng 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

b. Containment Drain Sump Level and Pumped Capacity Monitoring System-performance of CHANNEL CALIBRATION at least once per 24 months.

OPERATIONAL LEAKAGE

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 2250 ± 20 psia, 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 2250 ± 20 psia 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.

---

* This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

SURVEILLANCE REQUIREMENTS

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

a. Deleted

b. Deleted

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 ± 20 psia 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;

d. Performance of a Reactor Coolant System water inventory balance within 12 hours of achieving steady state operation, and at least once per 72 hours thereafter during steady state operation. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4; and

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

4.4.6.2.2[(1)(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 24 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

---

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

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
3-SIL-V15	SI Tank 1A Discharge Isolation Valve
3-SIL-V17	SI Tank 1B Discharge Isolation Valve
3-SIL-V19	SI Tank 1C Discharge Isolation Valve
3-SIL-V21	SI Tank 1D Discharge Isolation Valve
3-SIL-V26	RHR/SI to RCS Loop 2, Hot Leg
3-SIL-V27	SIH to RCS Loop 2, Hot Leg
3-SIL-V28	RHR/SI to RCS Loop 4, Hot Leg
3-SIL-V29	SIH to RCS Loop 4, Hot Leg
3-SIL-V984	RHR/SI to RCS Loop 4, Cold Leg
3-SIL-V985	RHR/SI to RCS Loop 3, Cold Leg
3-SIL-V986	RHR/SI to RCS Loop 2, Cold Leg
3-SIL-V987	RHR/SI to RCS Loop 1, Cold Leg
3-SIH-V5	SIH to RCS Cold Legs
3-SIH-V110	SIH to RCS Loop 1, Hot Leg
3-SIH-V112	SIH to RCS Loop 3, Hot Leg
3-RCS-V26	SIH to RCS Loop 1, Hot Leg
3-RCS-V29	SIH to RCS Loop 1, Cold Leg
3-RCS-V30	SIL to RCS Loop 1, Cold Leg
3-RCS-V69	RHR/SI to RCS Loop 2, Hot Leg
3-RCS-V70	SIH to RCS Loop 2, Cold Leg
3-RCS-V71	SIL to RCS Loop 2, Cold Leg
3-RCS-V102	SIH to RCS Loop 3, Hot Leg
3-RCS-V106	SIH to RCS Loop 3, Cold Leg
3-RCS-V107	SIL to RCS Loop 3, Cold Leg
3-RCS-V142	RHR/SI to RCS Loop 4, Hot Leg
3-RCS-V145	SIH to RCS Loop 4, Cold Leg
3-RCS-V146	SIL to RCS Loop 4, Cold Leg
3-RHS-MV8701C	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702C	RCS Loop 4, Hot Leg to RHR
3-RHS-MV8701A	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702B	RCS Loop 4, Hot Leg to RHR

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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.

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

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.

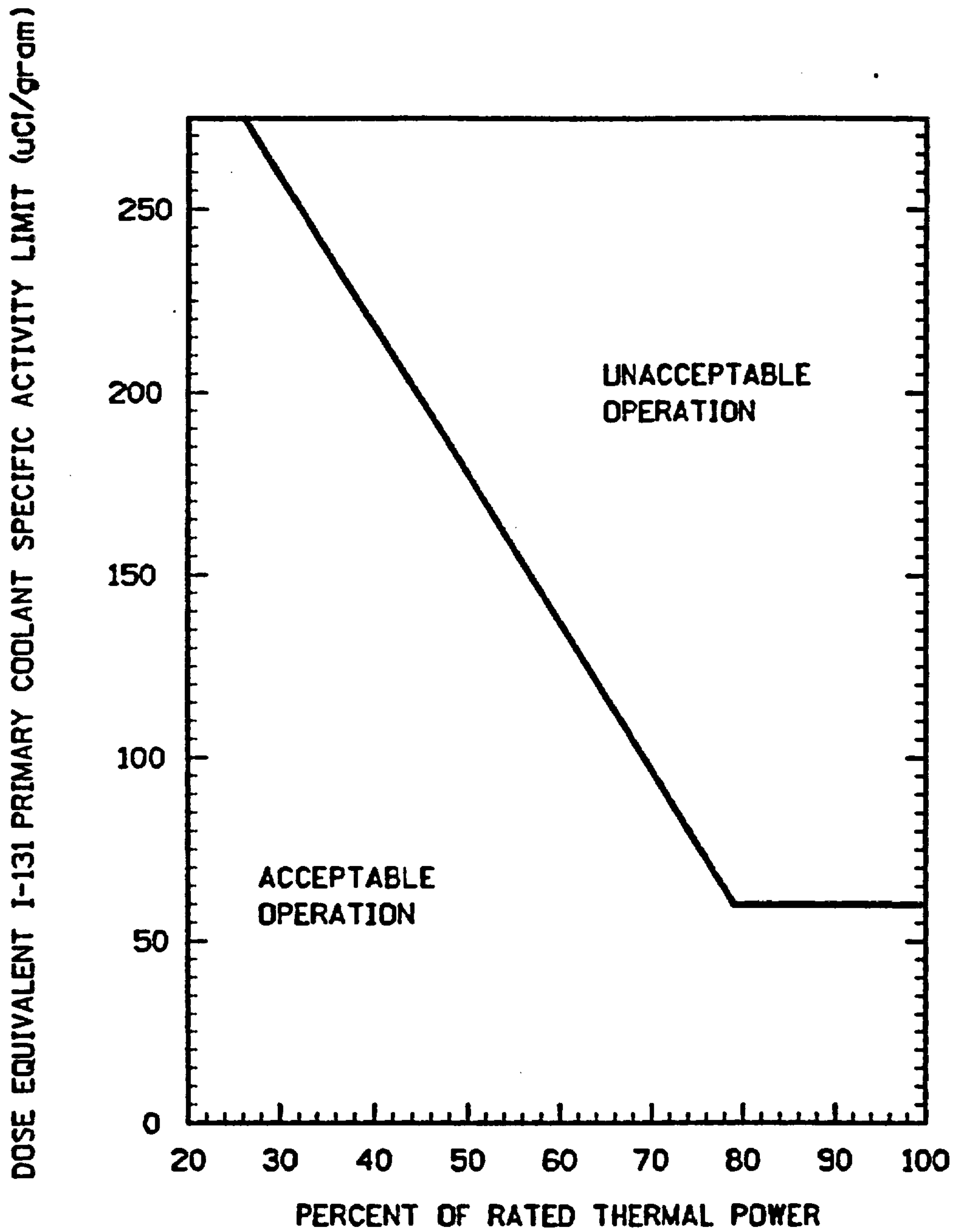

FIGURE 3.4-1

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

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. The provisions of Specification 4.0.4 are not applicable.

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates of ferritic materials shall be limited in accordance with the limits shown on Figures 3.4-2 and 3.4-3. In addition, a maximum of one reactor coolant pump can be in operation when the lowest unisolated Reactor Coolant System loop wide range cold leg temperature is $\leq$ 160°F.

APPLICABILITY: At all times.

ACTION:

a. With any of the above limits exceeded in MODES 1, 2, 3, or 4, perform the following:

   1. Restore the temperature and/or pressure to within limit within 30 minutes.

   AND

   2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation within 72 hours. Otherwise, be in at least MODE 3 within the next 6 hours and in MODE 5 with RCS pressure less than 500 psia within the following 30 hours.

b. With any of the above limits exceeded in other than MODES 1, 2, 3, or 4, perform the following:

   1. Immediately initiate action to restore the temperature and/or pressure to within limit.

   AND

   2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System is acceptable for continued operation prior to entering MODE 4.

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 and cooldown operations, and during the one-hour period prior to and during inservice leak and hydrostatic testing operations.

4.4.9.1.2 DELETED

Millstone 3 Reactor Coolant System
Heatup Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)

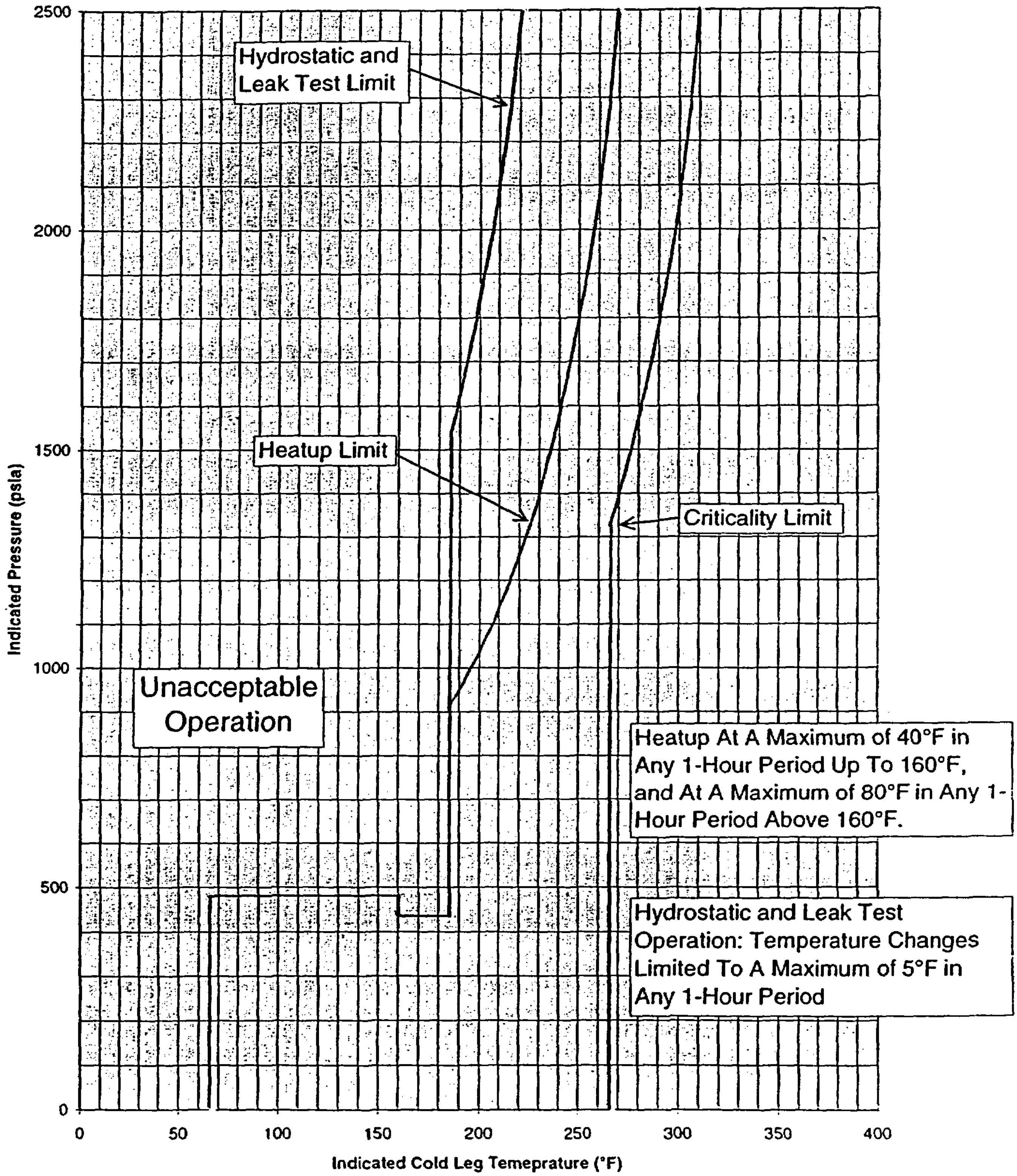

FIGURE 3.4-2

Millstone 3 Reactor Coolant System
Cooldown Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)

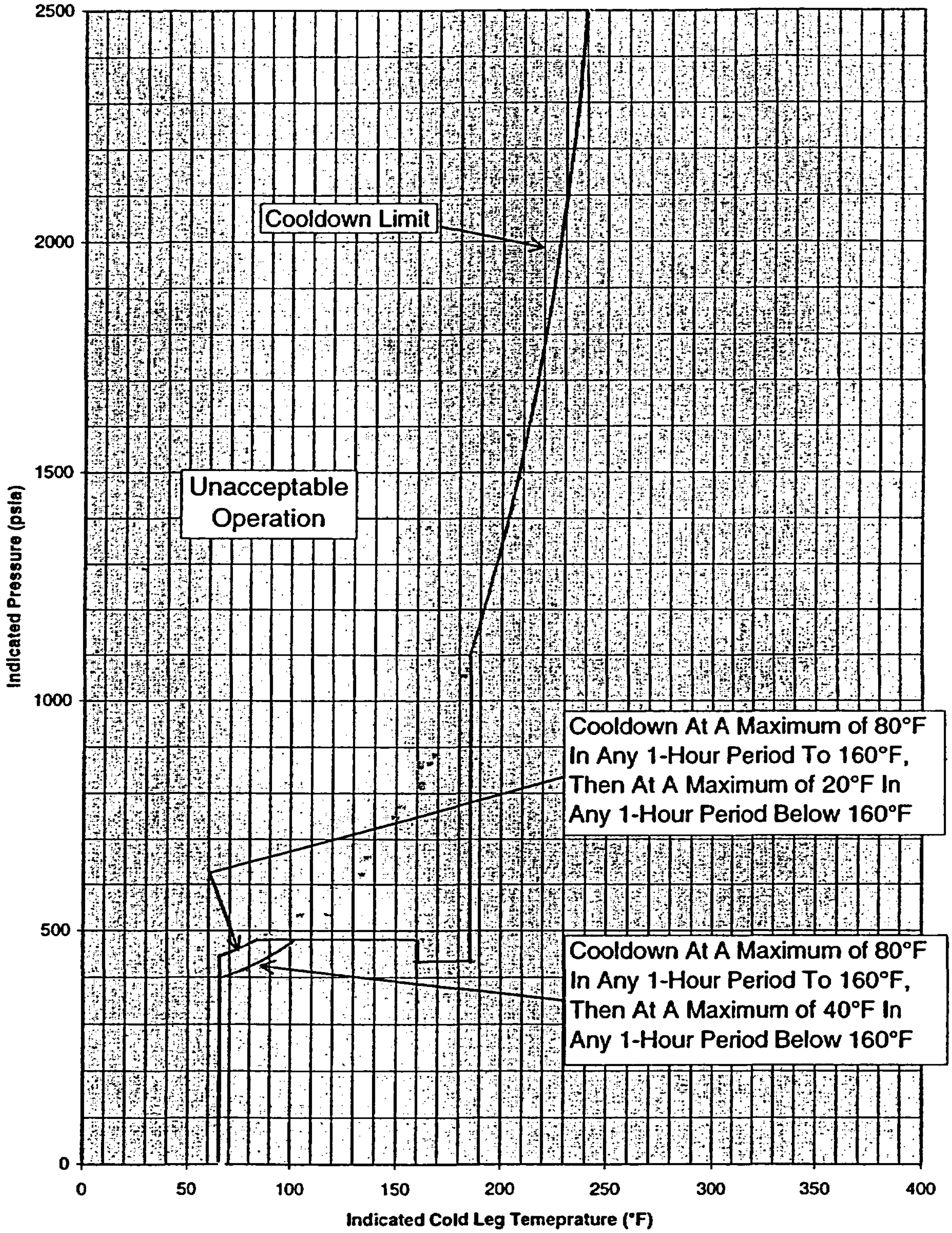

FIGURE 3.4-3

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OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Cold Overpressure Protection shall be OPERABLE with a maximum of one centrifugal charging pump* and no Safety Injection pumps capable of injecting into the Reactor Coolant System (RCS) and one of the following pressure relief capabilities:

1. One power operated relief valve (PORV) with a nominal lift setting established in Figure 3.4-4a and one PORV with a nominal lift setting established in Figure 3.4-4b with no more than one isolated RCS loop, or
2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ 426.8 psig and ≤ 453.2 psig, or
3. One PORV with a nominal lift setting established in Figure 3.4-4a or Figure 3.4-4b with no more than one isolated RCS loop and one RHR suction relief valve with a setpoint ≥ 426.8 psig and ≤ 453.2 psig, or
4. RCS depressurized with an RCS vent of ≥ 2.0 square inches.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 226°F, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With two or more centrifugal charging pumps capable of injecting into the RCS, immediately initiate action to establish that a maximum of one centrifugal charging pump is capable of injecting into the RCS.

b. With any Safety Injection pump capable of injecting into the RCS, immediately initiate action to establish that no Safety Injection pumps are capable of injecting into the RCS.

c. With one required relief valve inoperable in MODE 4, restore the required relief valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a 2.0 square inch vent within the next 12 hours.

*Two centrifugal charging pumps may be capable of injecting into the RCS for less than one hour, during pump swap operations. However, at no time will two charging pumps be simultaneously out of pull-to-lock during pump swap operations.

d. With one required relief valve inoperable in MODE 5 or 6, restore the required relief valve to OPERABLE status within 24 hours, or depressurize the RCS and establish an RCS vent of $\geq$ 2.0 square inches within the next 12 hours.

e. With two required relief valves inoperable, depressurize the RCS and establish an RCS vent of $\geq$ 2.0 square inches within 12 hours.

f. In the event the PORVs, the RHR suction relief valves, or the RCS vent 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, the RHR suction relief valves, or RCS vent on the transient, and any corrective action necessary to prevent recurrence.

g. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Demonstrate that each required PORV is 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 24 months; and

c. Verifying the PORV block valve is open and the PORV Cold Overpressure Protection System (COPPS) is armed at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Demonstrate that each required RHR suction relief valve is OPERABLE by:

a. Verifying the isolation valves between the RCS and each required RHR suction relief valve are open at least once per 12 hours; and

b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway $\geq$ 2.0 square inches at least once per 31 days for a passive vent path and at least once per 12 hours for unlocked open vent valves.

4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS at least once per 12 hours.

4.4.9.3.5 Verify that a maximum of one centrifugal charging pump is capable of injecting into the RCS at least once per 12 hours.

High Setpoint PORV Curve
For the Cold Overpressure Protection System

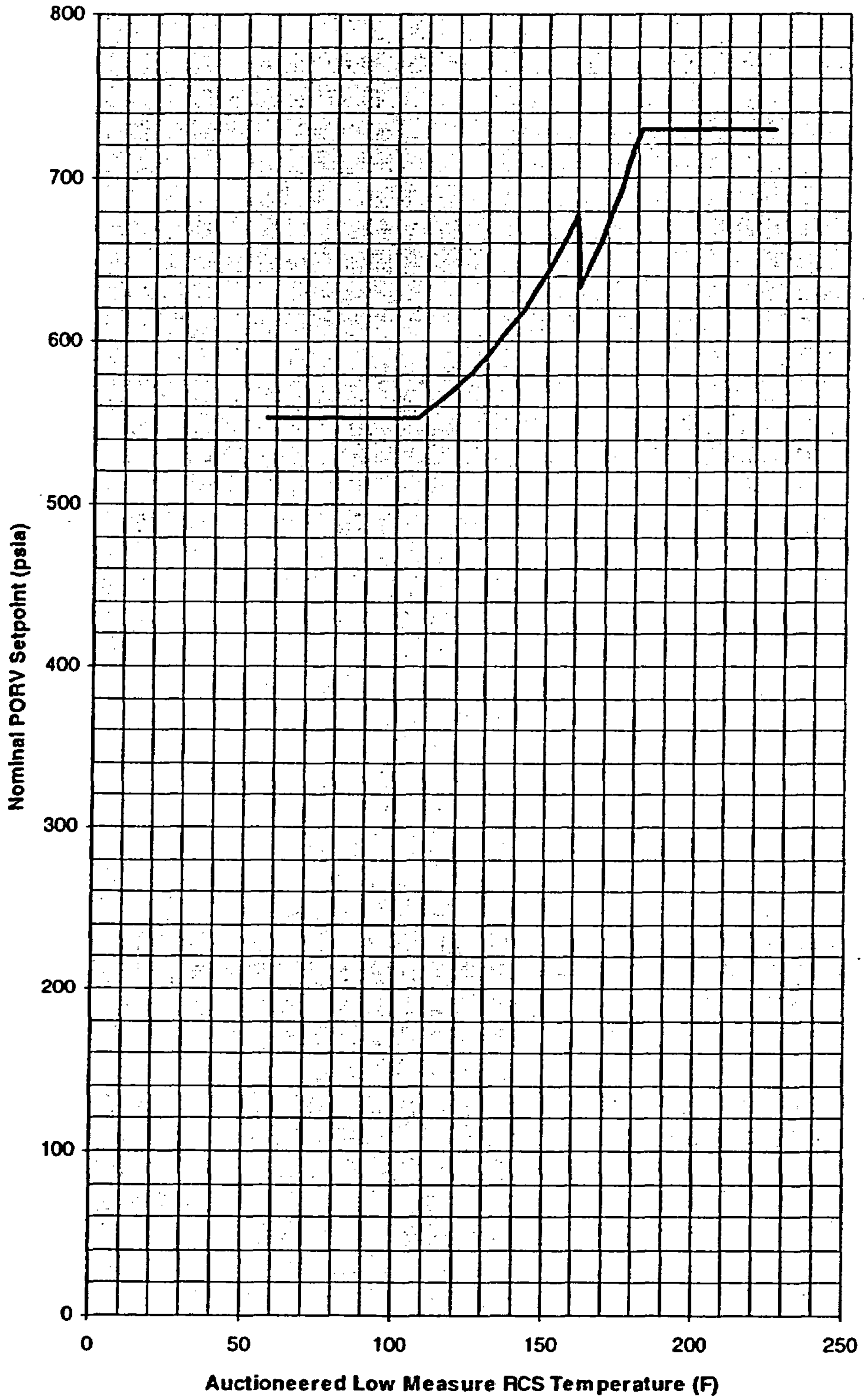

FIGURE 3.4-4a

Low Setpoint PORV Curve
For the Cold Overpressure Protection System

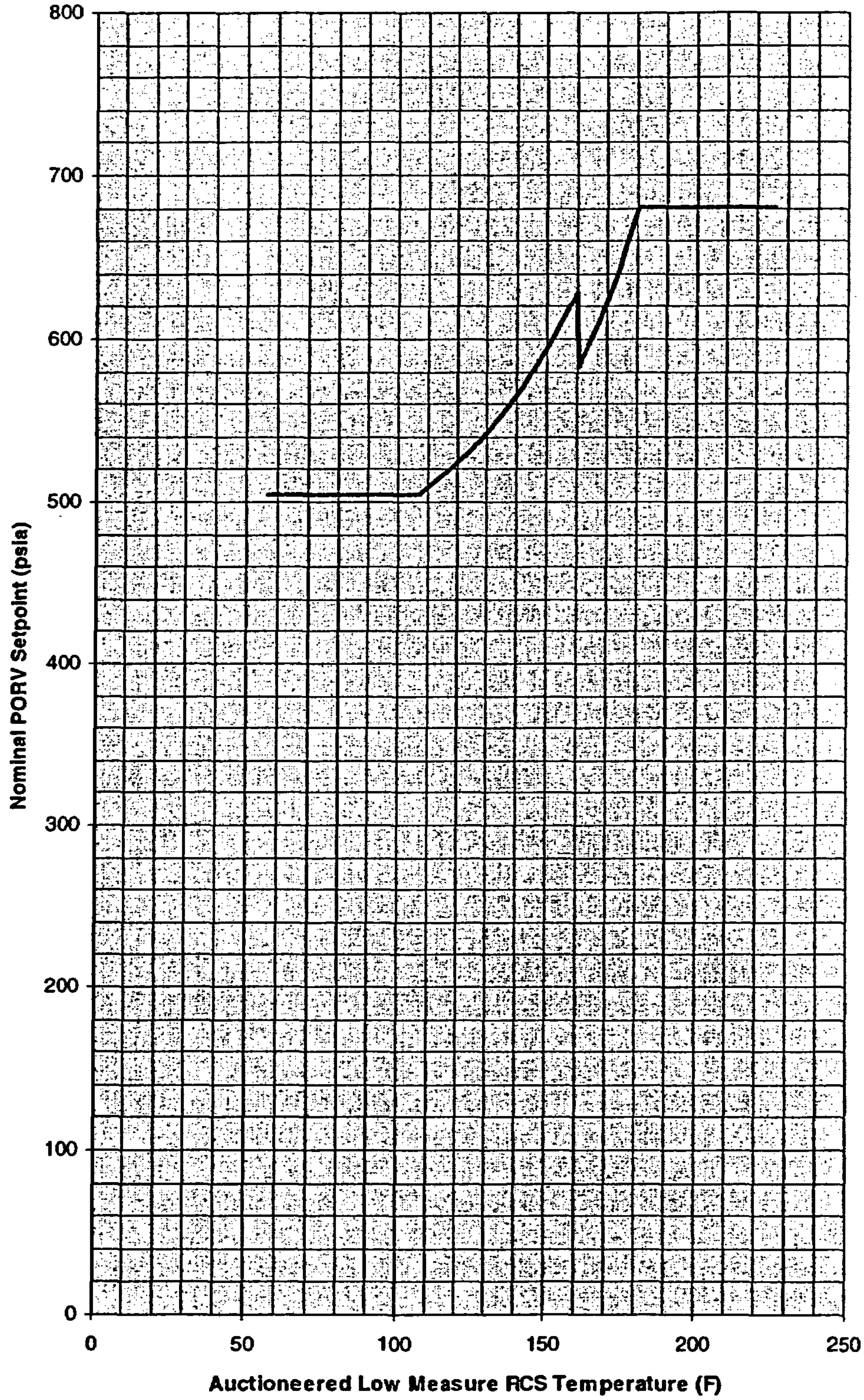

FIGURE 3.4-4b

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6618 and 7030 gallons,

c. A boron concentration of between 2600 and 2900 ppm, and

d. A nitrogen cover-pressure of between 636 and 694 psia.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 8 hours 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 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 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2) Verifying that each accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST.

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*Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the associated circuit breakers are locked in a deenergized position or removed.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,*

d. One OPERABLE RHR pump,*

e. One OPERABLE containment recirculation heat exchanger,

f. One OPERABLE containment recirculation pump, and

g. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and capable of automatically stopping the RHR pump and being manually realigned to transfer suction to the containment sump 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 allowable outage time for each RHR pump/RHR heat exchanger may be extended to 120 hours for the purpose of pump modification to change mechanical seal and other related modifications. This exception may only be used one time per RHR pump/RHR heat exchanger and is not valid after April 30, 1995.

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
3SIH*MV8806	RWST Supply to SI Pumps	OPEN
3SIH*MV8802A	SI Pump A to Hot Leg Injection	CLOSED
3SIH*MV8802B	SI Pump B to Hot Leg Injection	CLOSED
3SIH*MV8835	SI Cold Leg Master Isolation	OPEN
3SIH*MV8813	SI Pump Master Miniflow Isolation	OPEN
3SIL*MV8840	RHR to Hot Leg Injection	CLOSED
3SIL*MV8809A	RHR Pump A to Cold Leg Injection	OPEN
3SIL*MV8809B	RHR Pump B to Cold Leg Injection	OPEN

b. At least once per 31 days by:

1) Verifying that the ECCS piping, except for the operating centrifugal charging pump(s) and associated piping, the RSS pump, the RSS heat exchanger and associated piping, is full of water, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in 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) At least once daily of the areas affected (during each day) within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 24 months by:

1) Verifying automatic interlock action of the RHR System from the Reactor Coolant System by ensuring that with a simulated signal greater than or equal to 412.5 psia the interlocks prevent the valves from being opened.

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 24 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

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

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

3) Verifying that the Residual Heat Removal pumps stop automatically upon receipt of a Low-Low RWST Level test signal.

f. By verifying that each of the following pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump

2) Safety Injection pump

3) RHR pump

4) Containment recirculation pump

g. By verifying the correct position of each electrical and/or 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 24 months.

ECCS Throttle Valves

Valve Number	Valve Number
3SIH*V6	3SIH*V25
3SIH*V7	3SIH*V27

ECCS Throttle Valves Valve Number	Valve Number
3SIH*V8	3SIH*V107
3SIH*V9	3SIH*V108
3SIH*V21	3SIH*V109
3SIH*V23	3SIH*V111

h. By performing a flow balance test 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 310.5 gpm, and

b) The total pump flow rate is less than or equal to 560 gpm.

2) For Safety Injection 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 423.4 gpm, and

b) The total pump flow rate is less than or equal to 675 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3976 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump,

d. One OPERABLE containment recirculation heat exchanger,

e. One OPERABLE containment recirculation pump, and

f. An OPERABLE flow path which, with manual realignment of valves, is capable of discharging to the RCS, taking suction from the refueling water storage tank, 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 the centrifugal charging pump, the containment recirculation pump, the containment recirculation heat exchanger, the flow path from the refueling water storage tank, or the flow path capable of taking suction from the containment sump, 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.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2, with the exception that valves may be out of alignment but capable of being manually realigned.

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 contained borated water volume between 1,166,000 and 1,207,000 gallons,

b. A boron concentration between 2700 and 2900 ppm of boron,

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

d. A maximum solution temperature of 50°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.

3/4.5.5 pH TRISODIUM PHOSPHATE STORAGE BASKETS

LIMITING CONDITION FOR OPERATION

3.5.5 The trisodium phosphate (TSP) dodecahydrate Storage Baskets shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With the TSP Storage Baskets inoperable, restore the system TSP Storage Baskets to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The TSP Storage Baskets shall be demonstrated OPERABLE at least once per 24 months by verifying that a minimum total of 974 cubic feet of TSP is contained in the TSP Storage Baskets.

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[1] not capable of being closed by OPERABLE containment automatic isolation valves,[2] and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions,[3] except for valves that are open under administrative control as permitted by Specification 3.6.3; and

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

c. Deleted

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(1) 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.

(2) In MODE 4, the requirement for an OPERABLE containment isolation valve system is satisfied by use of the containment isolation actuation pushbuttons.

(3) Isolation devices in high radiation areas may be verified by use of administrative means.

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With the containment leakage rates exceeding the limits, restore the leakage rates to within limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The 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 in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

NOTE

Entry and exit through the containment air lock doors is permitted to perform repairs on the affected air lock components.

a. With only one containment air lock door inoperable:

1. Verify the OPERABLE air lock door is closed within 1 hour 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 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. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

b. With only the containment air lock interlock mechanism inoperable, verify an OPERABLE air lock door is closed within 1 hour and lock an OPERABLE air lock door closed within 24 hours. Verify an OPERABLE air lock door is locked closed at least once per 31 days thereafter. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Entry into and exit from containment is permissible under the control of a dedicated individual).

LIMITING CONDITION FOR OPERATION

Continued

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the air lock 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

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage results in accordance with the Containment Leakage Rate Testing Program. Containment air lock leakage test results shall be evaluated against the leakage limits of Technical Specification 3.6.1.2. (An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test).

b. Deleted

c. At least once per 24 months by verifying that only one door in each air lock can be opened at a time.

CONTAINMENT SYSTEMS

CONTAINMENT PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment pressure shall be maintained greater than or equal to 10.6 psia and less than or equal to 14.0 psia.

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

ACTION:

With the containment pressure less than 10.6 psia or greater than 14.0 psia, restore the containment 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 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 be maintained greater than or equal to 80°F and less than or equal to 120°F.

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

ACTION:

With the containment average air temperature less than 80°F or greater than 120°F, restore 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. 94 ft elevation, E outside crane wall

b. 86 ft elevation, NW outside crane wall

c. 75 ft elevation, W Steam Generator platform

d. 75 ft elevation, E Steam Generator platform

e. 45 ft elevation, Pressurizer cubicle, crane wall

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the structural integrity of the containment not conforming to the above requirements, 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 at the frequency specified in the Containment Leakage Rate Testing Program.

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.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and each 42-inch containment shutdown purge supply and exhaust isolation valve shall be closed and locked closed.

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

ACTION:

a. With a 42-inch containment purge supply and/or exhaust isolation valve open or not locked closed, close and/or 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.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The containment purge supply and exhaust isolation valves shall be verified to be locked closed and closed at least once per 31 days.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days, by:

1) 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; and

2) Verifying the temperature of the borated water in the refueling water storage tank is between 40°F and 50°F.

b. By verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

c. At least once per 24 months, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal, and

2) Verifying that each spray pump starts automatically on a CDA test signal.

d. By verifying each spray nozzle is unobstructed following maintenance that could cause nozzle blockage.

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation 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 each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5;

c. At least once per 24 months by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 ±20 second delay;

d. At least once per 24 months, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal; and

e. By verifying each spray nozzle is unobstructed following maintenance that could cause nozzle blockage.

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3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.[1] [2]

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier OPERABLE in the affected penetration(s), and:

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

b. Isolate the affected penetration(s) within 4 hours by use of deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of closed manual valve(s) or blind flange(s); or

d. Isolate the affected penetration that has only one containment isolation valve and a closed system within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange; or

e. 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 DELETED

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 24 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 High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

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

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[1] The provisions of this Specification are not applicable for main steam line isolation valves. However, provisions of Specification 3.7.1.5 are applicable for main steam line isolation valves.

[2] Containment isolation valves may be opened on an intermittent basis under administrative controls.

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CONTAINMENT SYSTEMS

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

3.6.5.1 The inside and outside isolation valves in the steam jet air ejector suction line shall be closed.

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

ACTION:

With the inside or outside isolation valves in the steam jet air ejector suction line not closed, restore the valve to the closed position 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

4.6.5.1.1 The steam jet air ejector suction line outside isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F and at least once per 31 days thereafter.

4.6.5.1.2 The steam jet air ejector suction line inside isolation valve shall be determined to be locked in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 200°F.

CONTAINMENT SYSTEMS

3/4.6.6 SECONDARY CONTAINMENT

SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Supplementary Leak Collection and Release Systems shall be OPERABLE with each system comprised of:

a. one OPERABLE filter and fan, and

b. one OPERABLE Auxiliary Building Filter System as defined in Specification 3.7.9.

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

ACTION:

With one Supplementary Leak Collection and Release System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Supplementary Leak Collection and Release System 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 a system flow rate of 7600 cfm to 9800 cfm and that the system operates for at least 10 continuous hours with the heaters operating.

b. At least once per 24 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 system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 7600 cfm to 9800 cfm;

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%; and

3) Verifying a system flow rate of 7600 cfm to 9800 cfm during system operation when tested in accordance with ANSI N510-1980.

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F) and a relative humidity of 70%:

d. At least once per 24 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.25 inches Water Gauge while operating the system at a flow rate of 7600 cfm to 9800 cfm,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 50 ±5 kW when tested in accordance with ANSI N510-1980.

*ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 7600 cfm to 9800 cfm; and

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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 7600 cfm to 9800 cfm.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.6.6.2 Secondary Containment shall be OPERABLE.

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

ACTION:

With Secondary Containment inoperable, restore Secondary Containment to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENT

4.6.6.2.1 OPERABILITY of Secondary Containment shall be demonstrated at least once per 31 days by verifying that each door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

4.6.6.2.2 At least once per 24 months, verify each Supplementary Leak Collection and Release System produces a negative pressure of greater than or equal to 0.4 inch water gauge in the Auxiliary Building at 24'-6" elevation within 120 seconds after a start signal.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.3 The structural integrity of the Secondary Containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.6.3.

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

ACTION:

With the structural integrity of the Secondary 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 REQUIREMENT

4.6.6.3 The structural integrity of the Secondary Containment shall be determined at the frequency specified in the Containment Leakage Rate Testing Program, by a visual inspection of the exposed accessible interior and exterior surfaces of the Secondary Containment and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the Secondary Containment 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.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. 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 HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

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	65
2	46
3	28

TABLE 3.7-2

DELETED

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING* (±3%)**	ORIFICE SIZE
LOOP 1		
RV22A	1185 psig	16.0 square inches
RV23A	1195 psig	16.0 square inches
RV24A	1205 psig	16.0 square inches
RV25A	1215 psig	16.0 square inches
RV26A	1225 psig	16.0 square inches
LOOP 2		
RV22B	1185 psig	16.0 square inches
RV23B	1195 psig	16.0 square inches
RV24B	1205 psig	16.0 square inches
RV25B	1215 psig	16.0 square inches
RV26B	1225 psig	16.0 square inches
LOOP 3		
RV22C	1185 psig	16.0 square inches
RV23C	1195 psig	16.0 square inches
RV24C	1205 psig	16.0 square inches
RV25C	1215 psig	16.0 square inches
RV26C	1225 psig	16.0 square inches
LOOP 4		
RV22D	1185 psig	16.0 square inches
RV23D	1195 psig	16.0 square inches
RV24D	1205 psig	16.0 square inches
RV25D	1215 psig	16.0 square inches
RV26D	1225 psig	16.0 square inches

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

**The lift setting shall be within ± 1% following main steam line Code safety valve testing.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. Entry into an OPERATIONAL MODE pursuant to Specification 3.0.4 is not permitted with three auxiliary feedwater pumps inoperable.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) 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

2) Verifying that each auxiliary feedwater control and isolation valve in the flow path is in the fully open position when above 10% RATED THERMAL POWER.

b. At least once per 92 days on a STAGGERED TEST BASIS, tested pursuant to Specification 4.0.5, by:

1) Verifying that on recirculation flow each motor-driven pump develops a total head of greater than or equal to 3385 feet;

2) Verifying that on recirculation flow the steam turbine-driven pump develops a total head of greater than or equal to 3780 feet when the secondary steam supply pressure is greater than 800 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

c. At least once per 24 months by verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal. For the steam turbine-driven auxiliary feedwater pump, the provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying flow to each steam generator.

DEMINERALIZED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The demineralized water storage tank (DWST) shall be OPERABLE with a water volume of at least 334,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the DWST inoperable, within 4 hours either:

a. Restore the DWST 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 condensate storage tank (CST) as a backup supply to the auxiliary feedwater pumps and restore the DWST 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 DWST shall be demonstrated OPERABLE at least once per 12 hours by verifying the water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying that the combined volume of both the DWST and CST is at least 384,000 gallons of water whenever the CST and DWST are the supply source for the auxiliary feedwater pumps.

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.

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.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Four main steam line isolation valves (MSIVs) shall be OPERABLE.

APPLICABILITY: MODE 1

MODES 2, 3, and 4, except when all MSIVs are closed and deactivated.

ACTION:

MODE 1:

With one MSIV inoperable, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 8 hours; otherwise be in MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one or more MSIVs inoperable, subsequent operation in MODE 2, or 3, or 4 may proceed provided the inoperable isolation valve(s) is (are) closed* within 8 hours and verified closed once per 7 days. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Separate condition entry is allowed for each MSIV.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 DELETED

4.7.1.5.2 Each MSIV shall be demonstrated OPERABLE, pursuant to Specification 4.0.5, by verifying full closure within 10 seconds (120 seconds for MODE 4 only) on an actual or simulated actuation signal. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4 or MODE 3.

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*The MSIVs may be opened to perform Surveillance Requirement 4.7.1.5.2 when Reactor Coolant System temperature is greater than or equal to 320°F.

PLANT SYSTEMS

STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each steam generator atmospheric relief bypass valve (SGARBV) line shall be OPERABLE, with the associated main steam atmospheric relief isolation (block) valve in the open position.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

a. With one required SGARBV line inoperable, restore required SGARBV line to OPERABLE status within 7 days or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours. LCO 3.0.4 is not applicable.

b. With two or more required SGARBV lines inoperable, restore all but one required SGARBV line to OPERABLE status within 24 hours or be in at least MODE 3 within the next 6 hours and be in MODE 4 without reliance upon steam generator for heat removal within the next 18 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 Verify one complete cycle of each SGARBV every 18 months.

4.7.1.6.2 Verify one complete cycle of each main steam atmospheric relief isolation (block) valve every 18 months.

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3/4.7.3 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With only one reactor plant component cooling water safety loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 24 months by verifying that:

   1) Each automatic valve actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

   2) Each Component Cooling Water System pump starts automatically on an SIS test signal.

3/4.7.4 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 At least two service water loops shall be demonstrated OPERABLE:

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

b. At least once per 24 months by verifying that:

   1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal, and

   2) Each Service Water System pump starts automatically on an SIS test signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with an average water temperature of less than or equal to 75°F.

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

ACTION:

If the UHS temperature is above 75°F, monitor the UHS temperature once per hour for 12 hours. If the UHS temperature does not drop below 75°F during this period, place the plant in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. During this period, if the UHS temperature increases above 77°F, place the plant in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature to be within limits.

b. At least once per 6 hours by verifying the average water temperature to be within limits when the average water temperature exceeds 70°F.

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3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Control Room Emergency Air Filtration Systems shall be OPERABLE.#

APPLICABILITY: MODES 1, 2, 3,4, 5 and 6.
During fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3 and 4:

a. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Control Room Emergency Air Filtration Systems inoperable, except as specified in ACTION c., immediately suspend the movement of fuel within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With both Control Room Emergency Air Filtration Systems inoperable due to an inoperable Control Room boundary, immediately suspend the movement of fuel within the spent fuel pool and restore the Control Room boundary to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6, and durimg fuel movement within containment or the spent fuel pool:

d. With one Control Room Emergency Air Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE Control Room Emergency Air Filtration System in the recirculation mode of operation, or immediately suspend the movement of fuel.

e. With both Control Room Emergency Air Filtration Systems inoperable, or with the OPERABLE Control Room Emergency Air Filtration System required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE emergency power source, immediately suspend the movement of fuel.

The Control Room boundary may be opened intermittently under administrative control.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Control Room Emergency Air Filtration System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 95°F;

b. 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 a system flow rate of 1,120 cfm ±20% and that the system operates for at least 10 continuous hours with the heaters operating;

c. At least once per 24 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 system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses 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 system flow rate is 1,120 cfm ±20%;

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 54 ft/min; and

3) Verifying a system flow rate of 1,120 cfm ±20% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), and a relative humidity of 70%, and a face velocity of 54 ft/min.

e. At least once per 24 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.75 inches Water Gauge while operating the system at a flow rate of 1,120 cfm ±20%;

SURVEILLANCE REQUIREMENTS (Continued)

(2) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas and outside atmosphere during the filtered pressurization mode of operation; and

(3) Verifying that the heaters dissipate 9.4 ±1 kW when tested in accordance with ANSI N510-1980.

f. 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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1120 cfm ±20%; and

g. 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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1120 cfm ±20%.

* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent Control Room Envelope Pressurization Systems shall be OPERABLE.#

APPLICABILITY: MODES 1, 2, 3,4, 5 and 6.
During fuel movement within containment or the spent fuel pool.

ACTION:

MODES 1, 2, 3, and 4:

a. With one Control Room Envelope Pressurization System inoperable, restore the system to OPERABLE status within 7 days, or be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With both Control Room Envelope Pressurization Systems inoperable, except as specified in ACTION c. or ACTION d., immediately suspend the movement of fuel within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With both Control Room Envelope Pressurization Systems inoperable due to an inoperable Control Room boundary, immediately suspend the movement of fuel within the spent fuel pool. Restore the Control Room boundary to OPERABLE status within 24 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With both Control Room Envelope Pressurization Systems inoperable during the performance of Surveillance Requirement 4.7.8.c and the system not being tested under administrative control, immediately suspend the movement of fuel within the spent fuel pool. Restore at least one inoperable system to OPERABLE status within 4 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool:

e. With one Control Room Envelope Pressurization System inoperable, restore the inoperable system to OPERABLE status within 7 days. After 7 days, immediately suspend the movement of fuel.

f. With both Control Room Envelope Pressurization Systems inoperable, immediately suspend the movement of fuel.

The Control Room boundary may be opened intermittently under administrative control.

SURVEILLANCE REQUIREMENTS

4.7.8 Each Control Room Envelope Pressurization System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the storage air bottles are pressurized to greater than or equal to 2200 psig,

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

c. At least once per 24 months or following a major alteration of the control room envelope pressure boundary by:

   1. Verifying that the control room envelope is isolated in response to a Control Building Isolation test signal,

   2. Verifying that after a 60 second time delay following a Control Building Isolation test signal, the control room envelope pressurizes to greater than or equal to 1/8 inch W.G. relative to adjacent areas and outside atmosphere, and

   3. Verifying that the positive pressure of Specification 4.7.8.c.2 is maintained for greater than or equal to 60 minutes.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 Two independent Auxiliary Building Filter Systems shall be OPERABLE.

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

ACTION:

With one Auxiliary Building Filter System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. In addition, comply with the ACTION requirements of Specification 3.6.6.1.

SURVEILLANCE REQUIREMENTS

4.7.9 Each Auxiliar Building Filter System 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 a system flow rate of 30,000 cfm ±10% and that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 24 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 cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% 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 30,000 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,* shows the methyl

iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 52 ft/min; and

3) Verifying a system flow rate of 30,000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

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,* shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D3803-89 at a temperature of 30°C (86°F), a relative humidity of 70%, and a face velocity of 52 ft/min;

d. At least once per 24 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.8 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ±10%,

2) Verifying that the system starts on a Safety Injection test signal, and

3) Verifying that the heaters dissipate 180 ±18 kW when tested in accordance with ANSI N510-1980.

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 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 30,000 cfm ±10%; and

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 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ±10%.

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* ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2, March 1978.

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 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 or 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 4.7.10g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and 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 categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-2.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable 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 4.7.10.f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

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 24 months thereafter,* 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 4.7.10f., an additional 5% 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

*Except the surveillance related to snubber functional testing due no later than March 10, 1999 may be deferred until the end of the next refueling outage or no later than September 10, 1999, whichever is earlier.

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 4.7.10f. The cumulative number of snubbers of a type tested is denoted by "N". Test results shall be plotted sequentially in the order of sample assignment (i.e. each snubber shall be plotted by its assigned order in the random sample, not by the order of testing). 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 4.7.10e. 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 Quality Assurance Program Topical Report.

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer included a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but no greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

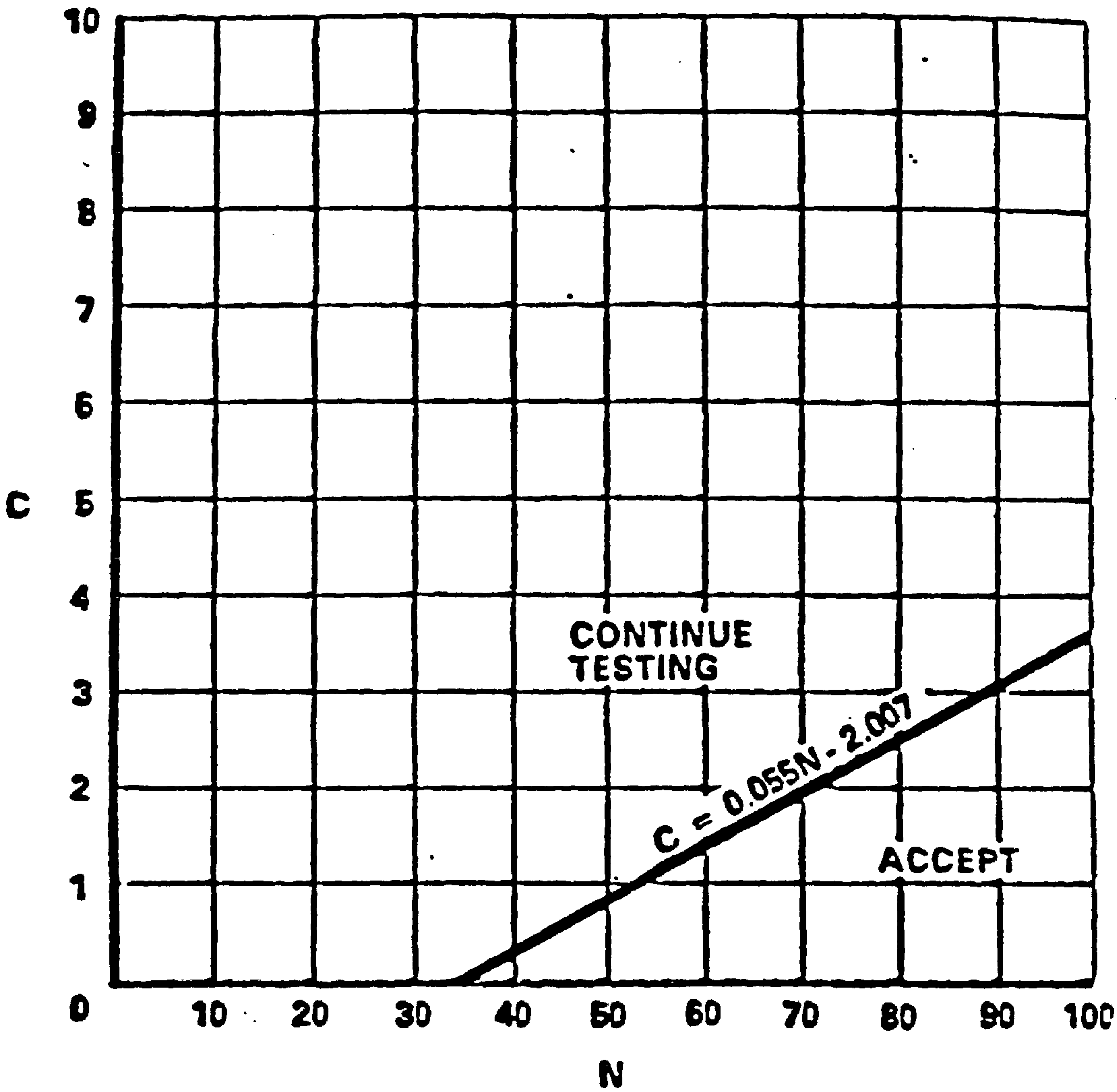

FIGURE 4.7-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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3/4.7.14 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.14 The temperature limit of each area shown in Table 3.7-6 shall not be exceeded.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6:

a. By less than 20°F and for less than 8 hours, record the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s).

b. By less than 20°F and for greater than or equal to 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 Specification 3.0.3 are not applicable.

c. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by greater than or equal to 20°F, prepare and submit a Special Report as required by ACTION b. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.14 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limits:

a. At least once per seven days when the alarm is OPERABLE, and;

b. At least once per 12 hours when the alarm is inoperable.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA		TEMPERATURE LIMIT (°F)
1. AUXILIARY BUILDING		
AB-02,	VCT and Boric Acid Transfer Pump Area, El 43'6"	≤ 120
AB-03,	Charging Pump Area, El 24'6"	≤ 110
AB-04,	General Area, El 66'6"	≤ 120
AB-06,	General Area, El 43'6"	≤ 120
AB-07,	General Area, El 4'6"	≤ 120
AB-08,	General Area (East), El 4'6"	≤ 120
AB-09,	General Area (South), El 4'6"	≤ 120
AB-10,	General Area, El 4'6"	≤ 120
AB-11,	General Area, El 43'6"	≤ 120
AB-13,	General Area (North), El 4'6"	≤ 120
AB-16,	Supplemental Leak Collection Filter Area, El 66'6"	≤ 120
AB-19,	MCC/Rod Drive Area, El 24'6"	≤ 120
AB-21,	MCC Air Conditioning Room, El 66'6"	≤ 120
AB-22,	Rod Drive Area, El 43'6"	≤ 120
AB-25,	Charging Pump Area, El 24'6"	≤ 110
AB-26,	RPCCW Pump Area, El 24'6"	≤ 110
AB-29,	General Area (Southeast), El 24'6"	≤ 120
AB-33,	Boric Acid Tank Area, El 43'6"	≤ 120
AB-35,	Boric Acid Tank Area, El 43'6"	≤ 120
AB-39,	Fuel Building and Auxiliary Building Filter Area, El 66'6"	≤ 120

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA		TEMPERATURE LIMIT (°F)
2.	CONTROL BUILDING	
CB-01,	Switchgear and Battery Rooms, El 4'6"	≤ 104
CB-02,	Cable Spreading Room, El 24'6"	≤ 110
CB-03,	Control and Computer Rooms, El 47'6"	≤ 95
CB-04,	Chiller Room, El 64'6"	≤ 104
CB-05,	Mechanical Equipment Room, El 64-6"	≤ 104
3.	CONTAINMENT	
CS-01,	Inside Crane Wall, El all except CS-03 and CS-04	≤ 120
CS-02,	Outside Crane Wall, El all	≤ 120
CS-03,	Pressurizer Cubicle, El all	≤ 130
CS-04,	Inside Crane wall, El 51'4" except CS-03 and steam generator enclosures	≤ 120
4.	INTAKE STRUCTURE	
CW-01,	Entire Building	≤ 110
5.	DIESEL GENERATOR BUILDING	
DG-01,	Entire Building	≤ 120
6.	ESF BUILDING	
ES-01,	HVAC and MCC Area, El 36'6"	≤ 110
ES-02,	SIH Pump Area, El 21'6"	≤ 110
ES-03,	Pipe Tunnel Area, El 4'6"	≤ 110
ES-04,	RHS Cubicles, El all	≤ 110
ES-05,	RSS Cubicles, El all	≤ 110
ES-06,	Motor Driven Auxiliary Feedwater Pump Area, El 24'-6"	≤ 110
ES-07,	Turbine Driven Auxiliary Feedwater Pump Area, El 24'6"	≤ 110

TABLE 3.7-6 (Continued)

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
7. FUEL BUILDING	
FB-02, Fuel Pool Pump Cubicles, El 24'6"	≤ 119
FB-03, General Area, El 52'4"	≤ 108
8. FUEL OIL VAULT	
FV-01, Diesel Fuel Oil Vault	≤ 95
9. HYDROGEN RECOMBINER BUILDING	
HR-01, Recombiner Skid Area, El 24'6"	≤ 125
HR-02, Controls Area, El 24'6"	≤ 110
HR-03, Sampling Area, El 24'6"	≤ 110
HR-04, HVAC Area, El 37'6"	≤ 110
10. MAIN STEAM VALVE BUILDING	
MS-01, Areas above El. 58'0"	≤ 140
MS-02, Areas below El. 58'0"	≤ 140
11. TURBINE BUILDING	
TB-01, Entire Building	≤ 115
12. TUNNEL	
TN-02, Pipe Tunnel-Auxiliary, Fuel and ESF Building	≤ 112
13. YARD	
YD-01, Yard	≤ 115

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 tank containing a minimum volume of 278 gallons of fuel,
2) A separate Fuel Storage System containing a minimum volume of 32,760 gallons of fuel,
3) A separate fuel transfer pump,
4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and
5) Capability to transfer lubricating oil from storage to the diesel generator unit.

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

ACTION:

Inoperable Equipment	Required Action
a. One offsite circuit	a.1 Perform Surveillance Requirement 4.8.1.1.1.a for remaining offsite circuit within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter. AND a.2 Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
b. One diesel generator	b.1 Perform Surveillance Requirement 4.8.1.1.1.a for the offsite circuits within 1 hour prior to or after entering this condition, and at least once per 8 hours thereafter. AND b.2 Demonstrate OPERABLE diesel generator is not inoperable due to common cause failure within 24 hours or perform Surveillance Requirement 4.8.1.1.2.a.5 for the OPERABLE diesel generator within 24 hours. AND

ACTION (continued)

Inoperable Equipment	Required Action
b. One diesel generator	b.3 Verify all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are OPERABLE, and the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). 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. AND b.4 (Applicable only if the 14 day allowed outage time specified in Action Statement b.5 is to be used). Verify the required Millstone Unit No. 2 diesel generator(s) is/are OPERABLE and the Millstone Unit No. 3 SBO diesel generator is available within 1 hour prior to or after entering this condition, and at least once per 24 hours thereafter. Restore any inoperable required Millstone Unit No. 2 diesel generator to OPERABLE status and/or Millstone Unit No. 3 SBO diesel generator to available status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND b.5 Restore the inoperable diesel generator to OPERABLE status within 72 hours (within 14 days if Action Statement b.4 is met) or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
c. One offsite circuit AND One diesel generator	c.1 Perform Surveillance Requirement 4.8.1.1.1.a for remaining offsite circuit within 1 hour and at least once per 8 hours thereafter. AND c.2 Demonstrate OPERABLE diesel generator is not inoperable due to common cause failure within 8 hours or perform Surveillance Requirement 4.8.1.1.2.a.5 for the OPERABLE diesel generator within 8 hours. AND

LIMITING CONDITION FOR OPERATION

ACTION (continued)

Inoperable Equipment	Required Action
c. One offsite circuit AND One diesel generator	c.3 Verify all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are OPERABLE, and the steam-driven auxiliary feedwater pump is OPERABLE (MODES 1, 2, and 3 only). 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. AND c.4 Restore one inoperable A.C. source to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. AND c.5 Restore remaining inoperable A.C. source to OPERABLE status following the time requirements of Action Statements a. or b. above based on the initial loss of the remaining inoperable A.C. source.
d. Two offsite circuits	d.1 Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. AND d.2 Following restoration of one offsite source, restore remaining inoperable offsite source to OPERABLE status following the time requirements of Action Statement a. above based on the initial loss of the remaining inoperable offsite source.
e. Two diesel generators	e.1 Perform Surveillance Requirement 4.8.1.1.1.a for the offsite circuits within 1 hour and at least once per 8 hours thereafter. AND

LIMITING CONDITION FOR OPERATION

ACTION (continued)

Inoperable Equipment	Required Action
e. Two diesel generators	e.2 Restore 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 COLD SHUTDOWN within the following 30 hours. AND e.3 Following restoration of one diesel generator, restore remaining inoperable diesel generator to OPERABLE status following the time requirements of Action Statement b. above based on the initial loss of the remaining inoperable diesel generator.

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) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:*

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day 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 tank,
4) Verifying the lubricating oil inventory in storage,
5) Verifying the diesel starts from standby conditions and achieves generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

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*All planned starts for the purpose of these surveillances may be preceded by an engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

6) Verifying the generator is synchronized and gradually loaded in accordance with the manufacturer's recommendations between 4800-5000 kW* and operates with a load between 4800-5000 kW* for at least 60 minutes, and

7) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 184 days by:

1) Verifying that the diesel generator starts from standby conditions and attains generator voltage and frequency of 4160 ± 420 volts and 60 ± 0.8 Hz within 11 seconds after the start signal.

2) Verifying the generator is synchronized to the associated emergency bus, loaded between 4800-5000 kW* in accordance with the manufacturer's recommendations, and operate with a load between 4800-5000 kW* for at least 60 minutes.

The diesel generator shall be started for this test using one of the signals in Surveillance Requirement 4.8.1.1.2.a.5. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

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

d. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

e. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

1) By verifying in accordance with the tests specified in ASTM-D975-81 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 27 degrees but less than or equal to 39 degrees;

*The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

b) A kinematic viscosity at 40°C 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; and

d) Water and sediment less than 0.05 percent by volume when tested in accordance with ASTM-D1796-83.

2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that: (1) the cetane index shall be determined in accordance with ASTM-D976 (this test is an appropriate approximation for cetane number as stated in ASTM-D975-81 [Note E]), and (2) the analysis for sulfur may be performed in accordance with ASTM-D1552-79, ASTM-D2622-82 or ASTM-D4294-83.

f. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

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

1) DELETED

2) Verifying the generator capability to reject a load of greater than or equal to 595 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz;

3) Verifying the generator capability to reject a load of 4986 kW without tripping. The generator voltage shall not exceed 5000 volts during and 4784 volts 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 from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 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 4160 ± 420 volts and 60 ± 0.8 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts from standby conditions on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 0.8 Hz within 11 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 an ESF Actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 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 4160 ± 420 volts and 60 ± 0.8 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, lube oil pressure low (2 of 3 logic) and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) DELETED

8) Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 5335 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.

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) DELETED

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval; and

13) DELETED

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby conditions, during shutdown, and verifying that both diesel generators achieve generator voltage and frequency at 4160 ± 420 volts and 60 ± 0.8 Hz in less than or equal to 11 seconds; and

i. At least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution.

j. At least once per 18 months by verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded between 5400-5500 kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded between 4800-5000 kW*. The generator voltage and frequency shall be 4160 $\pm$ 420 volts and 60 $\pm$ 0.8 Hz within 11 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 4.8.1.1.2.a.5) excluding the requirement to start the diesel from standy conditions.***

k. At least once per 18 months by verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines.

l. At least once per 18 months by verifying that the following diesel generator lockout features prevent diesel generator starting:

1) Engine overspeed,

2) Lube oil pressure low (2 of 3 logic),

3) Generator differential, and

4) Emergency stop.

* The operating band is meant as guidance to avoid routine overloading of the diesel. Momentary transients outside the load range shall not invalidate the test.

** Diesel generator loadings may include gradual loading as recommended by the manufacturer.

*** If Surveillance Requirement 4.8.1.1.2.a.5) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated between 4800-5000 kW for 2 hours or until operating temperature has stabilized.

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A. C. SOURCES

SHUTDOWN

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) A day tank containing a minimum volume of 278 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 32,760 gallons of fuel,

   3) A fuel transfer pump,

   4) Lubricating oil storage containing a minimum total volume of 280 gallons of lubricating oil, and

   5) Capability to transfer lubricating oil from storage to the diesel generator unit.

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, crane operation with loads over the fuel storage pool, or operation with a potential for draining the reactor vessel; initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENT

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 Specifications 4.8.1.1.2.a.6 and 4.8.1.1.2.b.2).

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank 301A-1, and an associated full capacity charger,

b. 125-volt Battery Bank 301A-2, and an associated full capacity charger,

c. 125-volt Battery Bank 301B-1 and an associated full capacity charger, and

d. 125-volt Battery Bank 301B-2 and an associated full capacity charger.

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

ACTION:

a. With either Battery Bank 301A-1 or 301B-1, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or 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.

b. With either Battery Bank 301A-2 or 301B-2 inoperable, and/or one of the required full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

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

1) The parameters in Table 4.8-2a meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 129 volts on float charge.

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 4.8-2a 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 six connected cells is above 60°F.

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

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

2) The cell-to-cell and terminal connections are clean, tight, and coated with 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) Each battery charger will supply at least the amperage indicated in Table 4.8-2b at greater than or equal to 132 volts for at least 24 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.

TABLE 4.8-2a

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	$\geq$ 2.13 volts	$\geq$ 2.13 volts(6)	> 2.07 volts
Specific Gravity(4)	$\geq$ 1.200(5)	$\geq$ 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 $\geq$ 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 and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

TABLE 4.8-2b

BATTERY CHARGER CAPACITY

CHARGER	AMPERAGE
301A-1	200
301A-2	50
301A-3	200
301B-1	200
301B-2	50
301B-3	200

ELECTRICAL POWER SYSTEMS

D. C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one train (A or B) of batteries and their associated full capacity chargers shall be OPERABLE:

a. Train - "A" consisting of:

1) Battery Bank 301A-1 and a full capacity battery charger, and

2) Battery Bank 301A-2 and a full capacity battery charger.

OR

b. Train - "B" consisting of:

1) Battery Bank 301B-1 and a full capacity battery charger, and

2) Battery Bank 301B-2 and a full capacity battery charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With the required train inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel; crane operation with loads over the fuel storage pool, or operation with a potential for draining the reactor vessel; initiate corrective action to restore the required train to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required train shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be OPERABLE in the specified manner:

a. Train A A.C. Emergency Busses consisting of:

   1) 4160-Volt Emergency Bus #34C, and

   2) 480-Volt Emergency Bus #32R, 32S, 32T, and 32Y.

b. Train B A.C. Emergency Busses consisting of:

   1) 4160-Volt Emergency Bus #34D, and

   2) 480-Volt Emergency Bus #32U, 32V, 32W, and 32X.

c. 120-Volt A.C. Vital Bus #VIAC-1 energized from its associated inverter connected to D.C. Bus #301A-1*,

d. 120-Volt A.C. Vital Bus #VIAC-2 energized from its associated inverter connected to D.C. Bus #301B-1*,

e. 120-Volt A.C. Vital Bus #VIAC-3 energized from its associated inverter connected to D.C. Bus #301A-2*,

f. 120-Volt A.C. Vital Bus #VIAC-4 energized from its associated inverter connected to D.C. Bus #301B-2*,

g. 125-Volt D.C. Bus #301A-1 energized from Battery Bank #301A-1,

h. 125-Volt D.C. Bus #301A-2 energized from Battery Bank #301A-2,

i. 125-Volt D.C. Bus #301B-1 energized from Battery Bank #301B-1, and

j. 125-Volt D.C. Bus #301B-2 energized from Battery Bank #301B-2.

---

* Two 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 bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

LIMITING CONDITION FOR OPERATION

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

ACTION:

a. With one of the required trains of A.C. emergency busses not OPERABLE, restore the inoperable train to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined OPERABLE in the specified manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, one train (A or B) of the following electrical busses shall be OPERABLE:

a. Train - "A" consisting of:

1) One 4160 volt AC Emergency Bus #34C, and
2) Four 480 volt AC Emergency Busses #32R, #32S, #32T, #32Y, and
3) Two 120 volt AC Vital Busses consisting of:

a) Bus #VIAC-1 energized from Inverter #INV-1 connected to DC Bus #301A-1, and
b) Bus #VIAC-3 energized from Inverter #INV-3 connected to DC Bus #301A-2, and

4) Two 125 volt DC Busses consisting of:

a) Bus #301A-1 energized from Battery Bank #301A-1, and
b) Bus #301A-2 energized from Battery Bank #301A-2.

OR

b. Train - "B" consisting of

1) One 4160 volt AC Emergency Bus #34D, and^[
2) Four 480 volt AC Emergency Busses #32U, #32V, #32W, #32X, and
3) Two 120 volt AC Vital Busses consisting of:

a) Bus #VIAC-2 energized from Inverter #INV-2 connected to DC Bus #301B-1, and
b) Bus #VIAC-4 energized from Inverter #INV-4 connected to DC Bus #301B-2, and

4) Two 125 volt DC Busses consisting of:

a) Bus #301B-1 energized from Battery Bank #301B-1, and |

b) Bus #301B-2 energized from Battery Bank #301B-2.

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, movement of irradiated fuel, crane operation with loads over the fuel storage pool, or operations with a potential for draining the reactor vessel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

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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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained 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 the limit specified in the CORE OPERATING LIMITS REPORT (COLR).

Additionally, the CVCS valves of Specification 4.1.1.2.2 shall be closed and secured in position.

APPLICABILITY: MODE 6.*

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 33 gpm of a solution containing greater than or equal to 6600 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 the limit specified in the COLR, whichever is the more restrictive.

b. With any of the CVCS valves of Specification 4.1.1.2.2 not closed** and secured in position, immediately close and secure the valves.

SURVEILLANCE REQUIREMENTS

4.9.1.1.1 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.

4.9.1.1.2 The boron concentration of the Reactor Coolant System and the refueling cavity shall be determined by chemical analysis at least once per 72 hours.

4.9.1.1.3 The CVCS valves of Specification 4.1.1.2.2 shall be verified closed and locked at least once per 31 days.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** Except those opened under administrative control.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be greater than or equal to 800 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

a. With the boron concentration less than 800 ppm, initiate action to bring the boron concentration in the fuel pool to at least 800 ppm within 72 hours, and

b. With the boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 800 ppm every 7 days.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 Two Source Range Neutron Flux Monitors shall be OPERABLE 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 immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable determine the boron concentration of the Reactor Coolant System within 4 hours and at least once per 12 hours thereafter.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK and verification of audible counts at least once per 12 hours,

b. A CHANNEL CALIBRATION at least once per 18 months.*

---

* Neutron detectors are excluded from CHANNEL CALIBRATION.

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.

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 access hatch shall be either:

1. closed and held in place by a minimum of four bolts, or

2. open under administrative control * and capable of being closed and held in place by a minimum of four bolts,

b. A personnel access hatch shall be either:

1. closed by one personnel access hatch door, or

2. capable of being closed by an OPERABLE personnel access hatch door, under administrative control,* 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 under administrative control.*

APPLICABILITY: During movement of fuel within the containment building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4.a Verify each required containment penetrations is in the required status at least once per 7 days.

4.9.4.b DELETED

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* Administrative controls shall ensure that appropriate personnel are aware that the equipment access hatch penetration, personnel access hatch doors and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations within 30 minutes if a fuel handling accident occurs. Any obstructions (e.g. cables and hoses) that could prevent closure of the equipment access hatch penetration, a personnel access hatch door and/or other containment penetrations must be capable of being quickly removed.

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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 or in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and suspend loading irradiated fuel assemblies in the core 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 2800 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause dilution of the RCS boron concentration.

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 2800 gpm at least once per 12 hours.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period, provided no operations are permitted that could cause dilution of the RCS boron concentration.

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3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth at least once per 24 hours.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 The Reactivity Condition of the Spent Fuel Pool shall be such that k_{eff} is less than or equal to 0.95 at all times.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3 or 3.9-4, to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment and burnup limits of Figure 3.9-3 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.3. Ensure that all fuel assemblies to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel asssembly's design, decay time, and burn-up documentation.

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.14 Each STORAGE PATTERN of the Region 1 spent fuel pool racks shall require that:

a. Prior to storing fuel assemblies in the STORAGE PATTERN per Figure 3.9-2, the cell blocking device for the cell location must be installed.

b. Prior to removal of a cell blocking device from the cell location per Figure 3.9-2, the STORAGE PATTERN must be vacant of all stored fuel assemblies

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: Take immediate action to comply with 3.9.14(a), (b).

SURVEILLANCE REQUIREMENTS

4.9.14 Verify that 3.9.14 is satisfied with no fuel assemblies stored in the STORAGE PATTERN prior to installing and removing a cell blocking device in the spent fuel racks.

FIGURE 3.9-1 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 1 4-OUT-OF-4 Fuel Storage Configuration

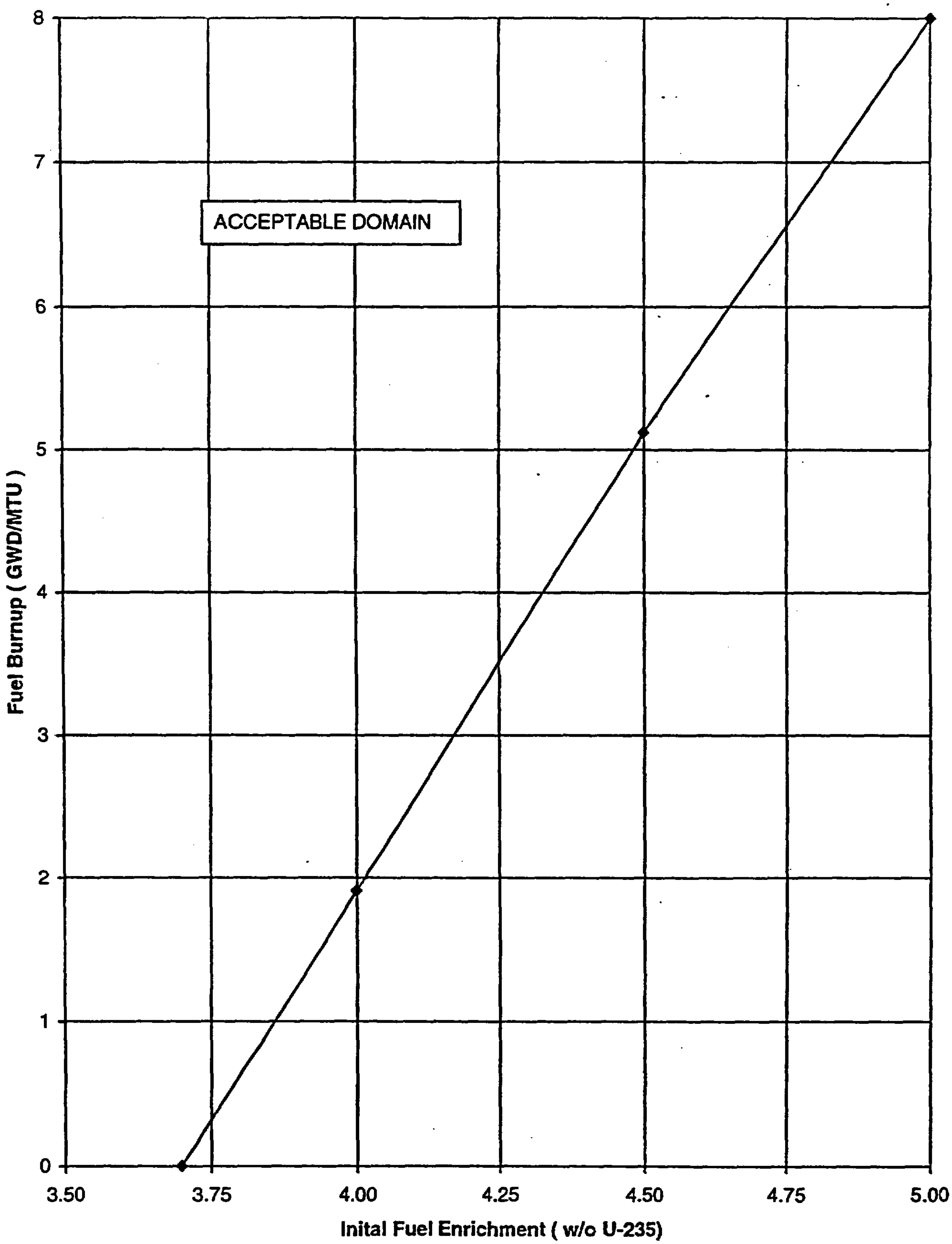
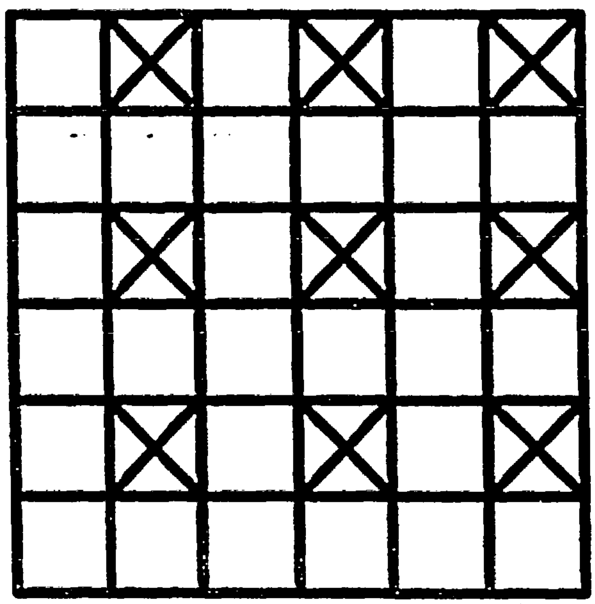


FIGURE 3.9-2 Region 1 3-OUT-OF-4 Storage Fuel Assembly Loading Schematic

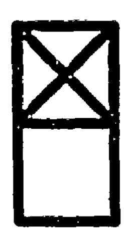
Region 2 or Region 1 4-OUT-OF-4
may be placed along this face

This face must be along
the wall of the spent fuel
pool, or other Region 1
3-OUT-OF-4 storage



Region 2 or Region 1 4-OUT-OF-4
may be placed along this face

This face must be along the wall of the
spent fuel pool, or other Region 1
3-OUT-OF-4 storage



Cell Blocker location

Fuel Assembly Storage
location

FIGURE 3.9-3 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration

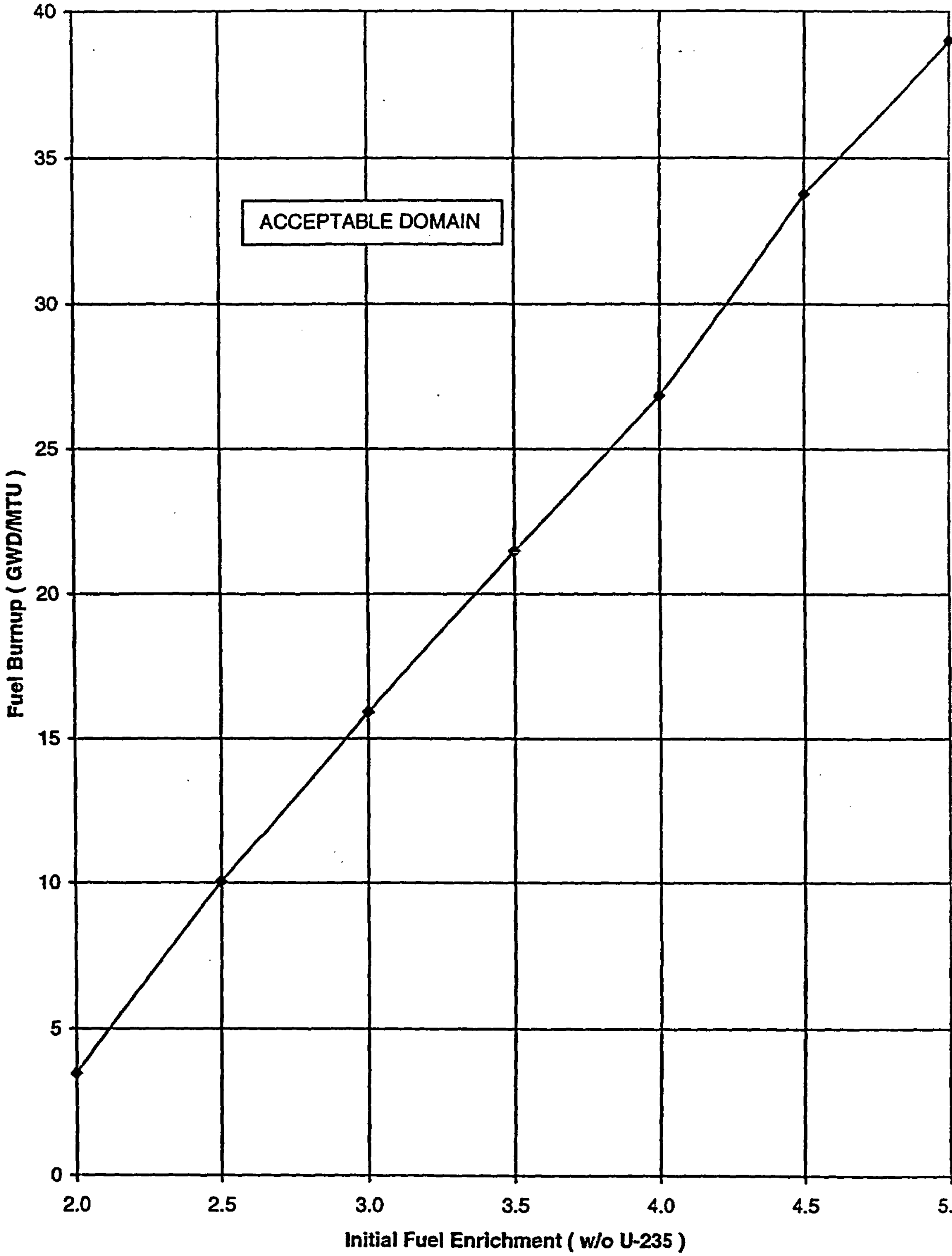
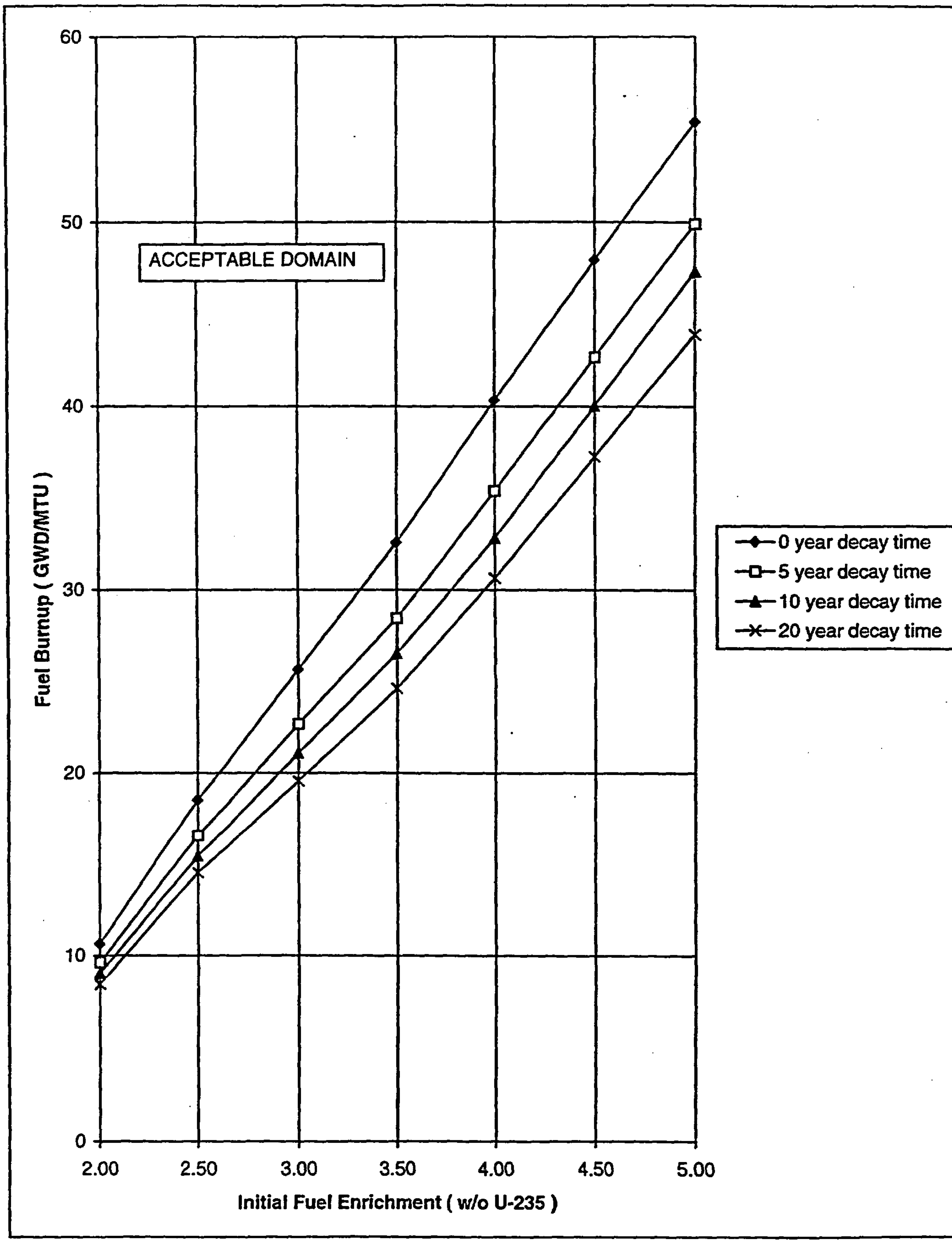


FIGURE 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration



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 full-length 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 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length 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 33 gpm of a solution containing greater than or equal to 6600 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length 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.

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

LIMITING CONDITION FOR OPERATION

3.10.2.1 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, 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 3.2.2.1 and 3.2.3.1 are maintained and determined at the frequencies specified in Specification 4.10.2.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2.1 or 3.2.3.1 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.1 and 3.2.3.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1.1 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.

4.10.2.1.2 The Surveillance Requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.1.2 and 4.2.2.1.3, and

b. Specification 4.2.3.1.2.

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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 at less than or equal to 25% of RATED THERMAL POWER, 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.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of STARTUP and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

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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.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

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 the Code of Federal Regulations, 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 provides 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 of 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. 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 it is identified that a Limiting Condition for Operation is not met. 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 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 high 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 provision of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

Specification 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to Specifications 3.0.1 and 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate either:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

BASES

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

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 met 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 OPERABILITY 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. Failure to meet a Surveillance within the specified surveillance interval, in accordance with Specification 4.0.2, constitutes a failure to meet a Limiting Condition for Operation.

Systems and components are assumed to be OPERABLE when the associated Surveillance Requirements have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the Surveillance Requirements or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillance requirements do not have to be performed when the facility is in an OPERATIONAL MODE or other specified conditions 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.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given Surveillance Requirement. In this case, the unplanned event may be credited as fulfilling the performance of the Surveillance Requirement. This allowance includes those Surveillance Requirement(s) whose performance is normally precluded in a given MODE or other specified condition.

Surveillance Requirements, including Surveillances invoked by ACTION requirements, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specification 4.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Auxiliary feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressure > 800 psi. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

b. High pressure safety injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

Specification 4.0.2 This specification establishes the limit for which the specified time interval for surveillance requirements may be extended. It 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. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified typically with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outage. The limitation of 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the surveillance requirements. This provision is 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 flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified surveillance interval was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with ACTION requirements or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

3/4.0 APPLICABILITY

BASES

When a Surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations, (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified surveillance interval to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by ACTION requirements.

Failure to comply with specified surveillance intervals for the Surveillance Requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified surveillance interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the entry into the ACTION requirements for the applicable Limiting Condition for Operation begins immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and entry into the ACTION requirements for the applicable Limiting Conditions for Operation begins immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Allowed Outage Time of the applicable ACTIONS, restores compliance with Specification 4.0.1.

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 shutdown 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 10CFR50.55a. 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 of 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}. In MODES 1 and 2, the most restrictive condition occurs at EOL with T_{avg} at no load operating emperature, 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 as defined in Specification 3/4.1.1.1.1 is required to control the reactivity transient. Accordingly, the SHDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4 and 5, the most restrictive condition occurs at BOL, associated with a boron dilution accident. In the analysis of this accident, a minimum SHUTDOWN MARGIN as defined in Specification 3/4.1.1.1.2 is required to allow the operator 15 minutes from the initiation of the Shutdown Margin Monitor alarm to total loss of SHUTDOWN MARGIN. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting requirement and is consistent with the accident analysis assumption.

The locking closed of the required valves in MODE 5 (with the loops not filled) will preclude the possibility of uncontrolled boron dilution of the Reactor Coolant System by preventing flow of unborated water to the RCS.

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.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

These corrections involved: (1) a conversion of the MDC used in the FSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR safety analyses into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making corrections for burnup and soluble boron to the limiting EOL MTC value.

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. 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 P-12 interlock is above its setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 DELETED

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

MOVABLE CONTROL ASSEMBLIES (Continued)

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 fully withdrawn position for the Control Banks and 18, 210, and fully withdrawn position for the Shutdown Banks provides assururances 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.

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.

The required rod drop time of $\leq$ 2.7 seconds specified in Technical Specification 3.1.3.4 is used in the FSAR accident analysis. A rod drop time was calculated to validate the Technical Specification limit. This calculation accounted for all uncertainties, including a plant specific seismic allowance of 0.51 seconds. Since the seismic allowance should be removed when verifying the actual rod drop time, the acceptance criteria for surveillance testing is 2.19 seconds (References 4 and 5).

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.

The Digital Rod Position Indication (DRPI) System is defined as follows:

- Rod position indication as displayed on DRPI display panel (MB4), or
- Rod position indication as displayed by the Plant Process Computer System

With the above definition, LCO, 3.1.3.2, "ACTION a." is <u>not</u> applicable with either DRPI display panel or the plant process computer points OPERABLE.

The plant process computer may be utilized to satisfy DRPI System requirements which meets LCO 3.1.3.2, in requiring diversity for determining digital rod position indication.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable,

MOVABLE CONTROL ASSEMBLIES (Continued)

then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

Additional surveillance is required to ensure the plant process computer indications are in agreement with those displayed on the DRPI. This additional SURVEILLANCE REQUIREMENT is as follows:

> Each rod position indication as displayed by the plant process computer shall be determined to be OPERABLE by verifying the rod position indication as displayed on the DRPI display panel agrees with the rod position indication as displayed by the plant process computer at least once per 12 hours.

The rod position indication, as displayed by DRPI display panel (MB4), is a non-QA system, calibrated on a refueling interval, and used to implement T/S 3.1.3.2. Because the plant process computer receives field data from the same source as the DRPI System (MB4), and is also calibrated on a refueling interval, it fully meets all requirements specified in T/S 3.1.3.2 for rod position. Additionally, the plant process computer provides the same type and level of accuracy as the DRPI System (MB4). The plant process computer does not provide any alarm or rod position deviation monitoring as does DRPI display panel (MB4).

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

For LCO 3.1.3.6 the control rods shall be limited in insertion as defined in the Core Operating Limits Report (COLR). The BASES for the Rod Insertion Limit (RIL) is located in the COLR (Reference 3.) and the current cycle reload 50.59 evaluation.

MOVABLE CONTROL ASSEMBLIES (Continued)

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

For LCO 3.1.3.6 the control bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). These insertion limits are the initial assumptions in safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions, assumptions of available SHUTDOWN MARGIN, and initial reactivity insertion rate.

The applicable I&C calibration procedure (Reference 1.) being current indicates the associated circuitry is OPERABLE.

There are conditions when the Lo-Lo and Lo alarms of the RIL Monitor are limited below the RIL specified in the COLR. The RIL Monitor remains OPERABLE because the lead control rod bank still has the Lo and Lo-Lo alarms greater than or equal to the RIL.

When rods are at the top of the core, the Lo-Lo alarm is limited below the RIL to prevent spurious alarms. The RIL is equal to the Lo-Lo alarm until the adjustable upper limit setpoint on the RIL Monitor is reached, then the alarm remains at the adjustable upper limit setpoint. When the RIL is in the region above the adjustable upper limit setpoint, the Lo-Lo alarm is below the RIL.

References:

1. IC 3469N08, Rod Control Speed, Insertion Limit, and Control TAVE Auctioneered/Deviation Alarms.

2. Letter NS-OPLS-OPL-1-91-226, (Westinghouse Letter NEU-91-563), dated April 24, 1991.

3. Millstone Unit 3 Technical Requirements Manual, Appendix 8.1, "CORE OPERATING LIMITS REPORT".

4. Westinghouse Letter NEU-97-298, "Millstone Unit 3 - RCCA Drop Time," dated November 13, 1997.

5. Westinghouse Letter 98NEU-G-0060, "Millstone Unit 3 - Robust Fuel Assembly (Design Report) and Generic SECL," dated October 2, 1998.

Bases Change of 8-25-2005

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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 the design limit 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; and

$F^N_{\Delta H}$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) 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.

Bases Change of 8-25-2005

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined in the COLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible: (1) RAOC, the AFD limit of which are defined in the COLR, and (2) base load operation, which is defined as the maintenance of the AFD within COLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible power level, the base load operating procedure restricts the indicated AFD to relatively small target band (as specified in the COLR) and power swings ($APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% RATED THERMAL POWER, whichever is lower). For base load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the base load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the base load procedure. After the waiting period, extended base load operation is permissible.

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 at least 2 of 4 or 2 of 3 OPERABLE excore channels are: (1) outside the allowed delta-I power operating space (for RAOC operation), or (2) outside the allowed delta-I target band (for base load operation). These alarms are active when power is greater than (1) 50% of RATED THERMAL POWER (for RAOC operation), or

AXIAL FLUX DIFFERENCE (Continued)

(2) APL^{ND} (for base load operation). Penalty deviation minutes for base load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

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;

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. 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.

The $F^N_{\Delta H}$ as calculated in Specification 3.2.3.1 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.

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 (Continued)

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

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 heat flux hot channel factor, $F_Q(Z)$, is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions. Using the measured three dimensional power distributions, it is possible to derive $F_Q^M(Z)$, a computed value of $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations.

To account for these possible variations, the steady state limit of $F_Q(Z)$ is adjusted by an elevation dependent factor appropriate to either RAOC or base load operation, $W(Z)$ or $W(Z)_{BL}$, that accounts for the calculated worst case transient conditions. The $W(Z)$ and $W(Z)_{BL}$, factors described above for normal operation are specified in the COLR per Specification 6.9.1.6. Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion. Evaluation of the steady state $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.b and 4.2.2.1.4.b while evaluation nonequilibrium limits are performed in Specification 4.2.2.1.2.c and 4.2.2.1.4.c.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors of 2.4% for RCS total flow rate and 4% for $F^N_{\Delta H}$ 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 will be added if venturis are not inspected and cleaned at least once for 18 months. 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.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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 defined in Specifications 3.2.3.1.

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 limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such 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. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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 greater than the design limit throughout each analyzed transient. The indicated T_{avg} values

DNB PARAMETERS (Continued)

and the indicated pressurizer pressure values are specified in the CORE OPERATING LIMITS REPORT. The calculated values of the DNB related parameters will be an average of the indicated values for the OPERABLE channels.

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. Measurement uncertainties have been accounted for in determining the parameter limits.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION 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 is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the 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.

The Engineered Safety Features Actuation System Nominal Trip Setpoints specified in Table 3.3-4 are the nominal values of which the bistables are set for each functional unit. The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) are considered the Limiting Safety System Settings as identified in 10CFR50.36 and have been selected to mitigate the consequences of accidents. A Setpoint is considered to be consistent with the nominal value when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, the Nominal Trip Setpoints may be adjusted in the conservative direction provided the calibration tolerance remains unchanged.

Measurement and Test Equipment accuracy is administratively controlled by plant procedures and is included in the plant uncertainty calculations as defined in WCAP-10991. OPERABILITY determinations are based on the use of Measurement and Test Equipment that conforms with the accuracy used in the plant uncertainty calculation.

The Allowable Value specified in Table 3.3-4 defines the limit beyond which a channel is inoperable. If the process rack bistable setting is measured within the "as left" calibration tolerance, which specifies the difference between the Allowable Value and Nominal Trip Setpoint, then the channel is considered to be OPERABLE.

Bases Change of 8-25-2005

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology, as defined in WCAP-10991 to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

The above Bases does not apply to the Control Building Inlet Ventilation radiation monitors ESF Table (Item 7E). For these radiation monitors the allowable values are essentially nominal values. Due to the uncertainties involved in radiological parameters, the methodologies of WCAP-10991 were not applied. Actual trip setpoints will be reestablished below the allowable value based on calibration accuracies and good practices.

The OPERABILITY requirements for Table 3.3-3, Functional Units 7.a, "Control Building Isolation, Manual Actuation," and 7.e, "Control Building Isolation, Control Building Inlet Ventilation Radiation," are defined by table notation "*". These functional units are required to be OPERABLE at all times during plant operation in MODES 1, 2, 3, and 4. These functional units are also required to be OPERABLE during fuel movement within containment or the spent fuel pool, as specified by table notation "*". This table notation is also applicable during fuel movement within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and functional units 7.a and 7.e are required to be OPERABLE whenever new or irradiated fuel is moved within the containment or the storage pool. Table notation "*" of Table 4.3-2 has the same applicability.

The verification 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 analysis. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.).

Bases Change of 8-25-2005

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test. Detector response times may be measured by the in-situ online noise analysis-response time degradation method described in the Westinghouse Topical Report, "The Use of Process Noise Measurements to Determine Response Characteristics of Protection Sensors in U.S. Plants," dated August 1983.

WCAP-14036, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

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) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feed-water isolation, (4) startup of the emergency diesel generators, (5) quench spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start, (10) service water pumps start and automatic valves position, and (11) Control Room isolates.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

For slave relays, or any auxiliary relays in ESFAS circuits that are of the type Potter & Brumfield MDR series relays, the SLAVE RELAY TEST is performed at an "R" frequency (at least once every 18 months) provided the relays meet the reliability assessment criteria presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR series relays," and WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals." The reliability assessments performed as part of the aforementioned WCAPs are relay specific and apply only to Potter and Brumfield MDR series relays. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.

REACTOR TRIP BREAKER

This trip function applies to the reactor trip breakers (RTBs) exclusive of individual trip mechanisms. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the control rod drive (CRD) system. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip functions must be OPERABLE when the RTBs or associated bypass breakers are closed, and the CRD system is capable of rod withdrawal.

BYPASSED CHANNEL* - Technical Specifications 3.3.1 and 3.3.2 often allow the bypassing of instrument channels in the case of an inoperable instrument or for surveillance testing.

A BYPASSED CHANNEL shall be a channel which is:

- Required to be in its accident or tripped condition, but is <u>not</u> presently in its accident or tripped condition using a method described below; or

- Prevented from tripping.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

A channel may be bypassed by:

- Insertion of a simulated signal to the bistable; or
- Failing the transmitter or input device to the bypassed condition; or
- Returning a channel to service in a untripped condition; or
- An equivalent method, as determined by Engineering and I&C

*Bypass switches exist only for NIS source range, NIS intermediate range, and containment pressure Hi-3.

TRIPPED CHANNEL - Technical Specifications 3.3.1 and 3.3.2 often require the tripping of instrument channels in the case of an inoperable instrument or for surveillance testing.

A TRIPPED CHANNEL shall be a channel which is in its required accident or tripped condition.

A channel may be placed in trip by:

- The Bistable Trip Switches; or
- Insertion of a simulated signal to the bistable; or
- Failing the transmitter or input device to the tripped condition; or
- An equivalent method, as determined by Engineering and I&C

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.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

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.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, main feed pumps and main turbine, and inhibits feedwater control valve modulation.

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 or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations 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 and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms.

3/4.3.3.2 DELETED

3/4.3.3.3 DELETED

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Remote Shutdown Instrumentation 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 Part 50.

Calibration of the Intermediate Range Neutron Amps channel from Table 4.3-6 applies to the signal that originates from the output of the isolation amplifier within the intermediate range neutron flux processor drawers in the control room and terminates at the displays within the Auxiliary Shutdown Panel.

The OPERABILITY of the Remote Shutdown Instrumentation ensures that a fire will not preclude achieving safe shutdown. The remote shutdown monitoring 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 and Appendix R to 10 CFR Part 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 to monitor and assess these variables following an accident. The instrumentation included in this specification are those instruments provided to monitor key variables, designated as Category 1 instruments following the guidance for classification contained in Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

ACTION Statement "a":

The use of one main control board indicator and one computer point, total of two indicators per steam generator, meets the requirements for the total number of channels for Auxiliary Feedwater flow rate. The two channels used to satisfy this Technical Specification for each steam generator are as follows:

Steam Generator	Instrument	(MB5)	Instrument	(Computer)
S/G 1	FWA*FI51A1	(Orange)	FWA - F33A3	(Purple)
S/G 2	FWA*FI33B1	(Purple)	FWA - F51B3	(Orange)
S/G 3	FWA*FI33C1	(Purple)	FWA - F51C3	(Orange)
S/G 4	FWA*FI51D1	(Orange)	FWA - F33D3	(Purple)

The SPDS computer point for auxiliary feedwater flow will be lost 30 minutes following an LOP when the power supply for the plant computer is lost. However, this design configuration - one continuous main control board indicator and one indication via the SPDS/plant computer, total of two per steam generator - was submitted to the NRC via "Response to question 420.6" dated January 13, 1984, B11002. NRC review and approval was obtained with the acceptance of MP3, SSER 4 Appendix L, "Conformance to Regulatory Guide 1.97," Revision 2. (dated November 1985).

LCO 3.3.3.6, Table 3.3-10, Item (17), requires 2 OPERABLE reactor vessel water level (heated junction thermocouples - HJTC) channels. An OPERABLE reactor vessel water level channel shall be defined as:

1. Four or more total sensors operating.
2. At least one of two operating sensors in the upper head.
3. At least three of six operating sensors in the upper plenum.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

A channel is OPERABLE if four or more sensors, half or more in the upper head region and half or more in the upper plenum region, are OPERABLE.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, at least one channel shall be restored to OPERABLE status in the nearest refueling outage.

The Reactor Coolant System Subcooling Margin Monitor, Core Exit Thermocouples, and Reactor Vessel Water Level instruments are processed by two separate trains of ICC (Inadequate Core Cooling) and HJTC (Heated Junction ThermoCouple) processors. The preferred indication for these parameters is the Safety Parameter Display System (SPDS) via the non-qualified PPC (Plant Process Computer) but qualified indication is provided in the instrument rack room. When the PPC data links cease to transmit data, the processors must be reset in order to restore the flow of data to the PPC. During reset, the qualified indication in the instrument rack room is lost. These instruments are OPERABLE during this reset since the indication is only briefly interrupted while the processors reset and the indication is promptly restored. The sensors are not removed from service during this reset. The train should be considered inoperable only if the qualified indication fails to be restored following reset. Except for the non-qualified PPC display, the instruments operate as required.

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. Containment hydrogen concentration is also important in verifying the adequacy of mitigating actions. The requirement to perform a hydrogen sensor calibration at least once every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two point calibration, utilizing gas containing approximately one percent hydrogen gas for one of the calibration points, and gas containing approximately four percent hydrogen gas for the other calibration point.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

3/4.3.3.9 DELETED

3/4.3.3.10 DELETED

3/4.3.4 DELETED

3/4.3.5 SHUTDOWN MARGIN MONITOR

The Shutdown Margin Monitors provide an alarm that a Boron Dilution Event may be in progress. The minimum count rate of Specification 3/4.3.5 and the SHUTDOWN MARGIN requirements specified in the CORE OPERATING LIMITS REPORT for MODE 3, MODE 4 and MODE 5 ensure that at least 15 minutes are available for operator action from the time of the Shutdown Margin Monitor alarm to total loss of SHUTDOWN MARGIN. By borating an additional 150 ppm above the SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT for MODE 3 or 350 ppm above the SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT for MODE 4, MODE 5 with RCS loops filled, or MODE 5 with RCS loops not filled, lower values of minimum count rate are accepted.

Shutdown Margin Monitors

Background:

The purpose of the Shutdown Margin Monitors (SMM) is to annunciate an increase in core subcritical multiplication allowing the operator at least 15 minutes response time to mitigate the consequences of the inadvertent addition of unborated primary grade water (boron dilution event) into the Reactor Coolant System (RCS) when the reactor is shut down (MODES 3, 4, and 5).

The SMMs utilizes two channels of source range instrumentation (GM detectors). Each channel provides a signal to its applicable train of SMM. The SMM channel uses the last 600 or more counts to calculate the count rate and updates the measurement after 30 new counts or 1 second, whichever is longer. Each channel has 20 registers that hold the counts (20 registers X 30 count = 600 counts) for averaging the rate. As the count rate decreases, the longer it takes to fill the registers (fill the 30 count minimum). As the instrument's measured count rate decreases, the delay time in the instrument's response increases. This delay time leads to the requirement of a minimum count rate for OPERABILITY.

During the dilution event, count rate will increase to a level above the normal steady state count rate. When this new count rate level increases above the instrument's setpoint, the channel will alarm alerting the operator of the event.

Applicable Safety Analysis

The SMM senses abnormal increases in the source range count per second and alarms the operator of an inadvertent dilution event. This alarm will occur at least 15 minutes prior to the reactor achieving criticality. This 15 minute window allows adequate operator response time to terminate the dilution, FSAR Section 15.4.6.

LCO

LCO 3.3.5 provides the requirements for OPERABILITY of the instrumentation of the SMMs that are used to mitigate the boron dilution event. Two trains are required to be OPERABLE to provide protection against single failure.

Applicability

The SMM must be OPERABLE in MODES 3, 4, and 5 because the safety analysis identifies this system as the primary means to alert the operator and mitigate the event. The SMMs are allowed to be blocked during start up activities in MODE 3 in accordance with approved plant procedures. The alarm is blocked to allow the SMM channels to be used to monitor the 1/M approach to criticality.

The SMM are not required to be OPERABLE in MODES 1 and 2 as other RPS is credited with accident mitigation, over temperature delta temperature and power range neutron flux high (low setpoint of 25 percent RTP) respectively. The SMMs are not required to be OPERABLE in MODE 6 as the dilution event is precluded by administrative controls over all dilution flow paths (Technical Specification 4.1.1.2.2).

ACTIONS

Channel inoperability of the SMMs can be caused by failure of the channel's electronics, failure of the channel to pass its calibration procedure, or by the channel's count rate falling below the minimum count rate for OPERABILITY. This can occur when the count rate is so low that the channel's delay time is in excess of that assumed in the safety analysis. In any of the above conditions, the channel must be declared inoperable and the appropriate ACTION statement entered. If the SMMs are declared inoperable due to low count rates, an RCS heatup will cause the SMM channel count rate to increase to above the minimum count rate for OPERABILITY. Allowing the plant to increase modes will actually return the SMMs to OPERABLE status. Once the SMM channels are above the minimum count rate for OPERABILITY, the channels can be declared OPERABLE and the LCO ACTION statements can be exited.

LCO 3.3.5, ACTION a. - With one train of SMM inoperable, ACTION a. requires the inoperable train to be returned to OPERABLE status within 48 hours. In this condition, the remaining SMM train is adequate to provide protection. If the above required ACTION cannot be met, alternate compensatory actions must be performed to provide adequate protection from the boron dilution event. All operations involving positive reactivity changes associated with RCS dilutions and rod withdrawal must be suspended, and all dilution flowpaths must be closed and secured in position (locked closed per Technical Specification 4.1.1.2.2) within the following 4 hours.

LCO 3.3.5, ACTION b. -With both trains of SMM inoperable, alternate protection must be provided:

1. Positive reactivity operations via dilutions and rod withdrawal are suspended. The intent of this ACTION is to stop any planned dilutions of the RCS. The SMMs are not intended to monitor core reactivity during RCS temperature changes. The alarm setpoint is routinely reset during the plant heatup due to the increasing count rate. During cooldowns as the count rate decreases, baseline count rates are continually lowered automatically by the SMMs. The Millstone Unit No. 3 boron dilution analysis assumes steady state RCS temperature conditions.

2. All dilution flowpaths are isolated and placed under administrative control (locked closed). This action provides redundant protection and defense in depth (safety overlap) to the SMMs. In this configuration, a boron dilution event (BDE) cannot occur. This is the basis for not having to analyze for BDE in MODE 6. Since the BDE cannot occur with the dilution flow paths isolated, the SMMs are not required to be OPERABLE as the event cannot occur and OPERABLE SMMs provide no benefit.

3. Increase the SHUTDOWN MARGIN surveillance frequency from every 24 hours to every 12 hours. This action in combination with the above, provide defense in depth and overlap to the loss of the SMMs.

Surveillance Requirements

The SMMs are subject to an ACOT every 92 days to ensure each train of SMM is fully operational. This test shall include verification that the SMMs are set per the CORE OPERATING LIMITS REPORT.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The purpose of Specification 3.4.1.1 is to require adequate forced flow rate for core heat removal in MODES 1 and 2 during all normal operations and anticipated transients. Flow is represented by the number of reactor coolant pumps in operation for removal of heat by the steam generators. To meet safety analysis acceptance criteria for DNB, four reactor coolant pumps are required at rated power. An OPERABLE reactor coolant loop consists of an OPERABLE reactor coolant pump in operation providing forced flow for heat transport and an OPERABLE steam generator in accordance with Specification 3.4.5. With less than the required reactor coolant loops in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops, and in MODE 4, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, in MODE 3 a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., the Control Rod Drive System is not capable of rod withdrawal.

In MODE 4, if a bank withdrawal accident can be prevented, 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 (any combination of RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two RHR loops or at least one RHR loop and two steam generators 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.

In MODE 5, during a planned heatup to MODE 4 with all RHR loops removed from operation, an RCS loop, OPERABLE and in operation, meets the requirements of an OPERABLE and operating RHR loop to circulate reactor coolant. During the heatup there is no requirement for heat removal capability so the OPERABLE and operating RCS loop meets all of the required functions for the heatup condition. Since failure of the RCS loop, which is OPERABLE and operating, could also cause the associated steam generator to be inoperable, the associated steam generator cannot be used as one of the steam generators used to meet the requirement of LCO 3.4.1.4.1.b.

Basis Change of 8-25-2005

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 the first RCP in MODE 4 below the cold overpressure protection enable temperature (226°F), and in MODE 5 are provided to prevent RCS pressure transients. These transients, energy additions due to the differential temperature between the steam generator secondary side and the RCS, can result in pressure excursions which could challenge the P/T limits. The RCS will be protected against overpressure transients and will not exceed the reactor vessel isothermal beltline P/T limit by restricting RCP starts based on the differential water temperature between the secondary side of each steam generator and the RCS cold legs. The restrictions on starting the first RCP only apply to RCPs in RCS loops that are not isolated. The restoration of isolated RCS loops is normally accomplished with all RCPs secured. If an isolated RCS loop is to be restored when an RCP is operating, the appropriate temperature differential limit between the secondary side of the isolated loop steam generator and the in service RCS cold legs is applicable, and shall be met prior to opening the loop isolation valves.

The temperature differential limit between the secondary side of the steam generators and the RCS cold legs is based on the equipment providing cold overpressure protection as required by Technical Specification 3.4.9.3. If the pressurizer PORVs are providing cold overpressure protection, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50°F. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is above 150°F, the steam generator secondary water temperature must be at or below RCS cold leg water temperature. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is at or below 150°F, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50°F.

Specification 3.4.1.5

The reactor coolant loops are equipped with loop stop valves that permit any loop to be isolated from the reactor vessel. One valve is installed on each hot leg and one on each cold leg. The loop stop valves are used to perform maintenance on an isolated loop. Operation in MODES 1-4 with a RCS loop stop valve closed is not permitted except for the mitigation of emergency or abnormal events. If a loop stop valve is closed for any reason, the required ACTIONS of this specification must be completed. To ensure that inadvertent closure of a loop stop valve does not occur, the valves must be open with power to the valve operators removed in MODES 1, 2, 3 and 4.

Bases Change of 8-25-2005

The safety analyses performed for the reactor at power assume that all reactor coolant loops are initially in operation and the loop stop valves are open. This LCO places controls on the loop stop valves to ensure that the valves are not inadvertently closed in MODES 1, 2, 3 and 4. The inadvertent closure of a loop stop valve when the Reactor Coolant Pumps (RCPs) are operating will result in a partial loss of forced reactor coolant flow. If the reactor is at rated power at the time of the event, the effect of the partial loss of forced coolant flow is a rapid increase in the coolant temperature which could result in DNB with subsequent fuel damage if the reactor is not tripped by the Low Flow reactor trip. If the reactor is shutdown and a RCS loop is in operation removing decay heat, closure of the loop stop valve associated with the operating loop could also result in increasing coolant temperature and the possibility of fuel damage.

The loop stop valves have motor operators. If power is inadvertently restored to one or more loop stop valve operators, the potential exists for accidental closure of the affected loop stop valve(s) and the partial loss of forced reactor coolant flow. With power applied to a valve operator, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop stop valve operators. The time period of 30 minutes to remove power from the loop stop valve operators is sufficient considering the complexity of the task.

Should a loop stop valve be closed in MODES 1 through 4, the affected valve must be maintained closed and the plant placed in MODE 5. Once in MODE 5, the isolated loop may be started in a controlled manner in accordance with LCO 3.4.1.6, "Reactor Coolant System Isolated Loop Startup." Opening the closed loop stop valve in MODES 1 through 4 could result in colder water or water at a lower boron concentration being mixed with the operating RCS loops resulting in positive reactivity insertion. The time period provided in ACTION 3.4.1.5.b allows time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 30 hours. The allowed ACTION times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirement 4.4.1.5 is performed at least once per 31 days to ensure that the RCS loop stop valves are open, with power removed from the loop stop valve operators. The primary function of this Surveillance is to ensure that power is removed from the valve operators, since Surveillance Requirement 4.4.1.1 requires verification every 12 hours that all loops are operating and circulating reactor coolant, thereby ensuring that the loop stop valves are open. The frequency of 31 days ensures that the required flow is available, is based on engineering judgement, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified.

Specification 3.4.1.6

The requirement to maintain the isolated loop stop valves shut with power removed ensures that no reactivity addition to the core could occur due to the startup of an isolated loop. Verification of the boron concentration in an isolated loop prior to opening the first stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop.

RCS Loops Filled/Not Filled:

In MODE 5, any RHR train with only one cold leg injection path is sufficient to provide adequate core cooling and prevent stratification of boron in the Reactor Coolant System.

The definition of OPERABILITY states that the system or subsystem must be capable of performing its specified function(s). The reason for the operation of one reactor coolant pump (RCP) or one RHR pump is to:

- Provide sufficient decay heat removal capability
- Provide adequate flow to ensure mixing to:
  - Prevent stratification
  - Produce gradual reactivity changes due to boron concentration changes in the RCS

The definition of "Reactor coolant loops filled" includes a loop that is filled, swept, and vented, and capable of supporting natural circulation heat transfer. This allows the non-operating RHR loop to be removed from service while filling and unisolating loops as long as steam generators on the OPERABLE reactor coolant loops are available to support decay heat removal. Any loop being unisolated is not OPERABLE until the loop has been swept and vented. The process of sweep and vent will make the previously OPERABLE loops inoperable and the requirements of LCO 3.4.1.4.2, "Reactor Coolant System, COLD SHUTDOWN - Loops Not Filled," are applicable. When the RCS has been filled, swept and vented using an approved procedure, all unisolated loops may be declared OPERABLE.

One cold leg injection isolation valve on an RHR train may be closed without considering the train to be inoperable, as long as the following conditions exist:

- CCP temperature is at or below 95°F
- Initial RHR temperature is below 184°F

- The single RHR cold leg injection flow path is not utilized until a minimum of 24 hours after reactor shutdown
- CCP flow is at least 6,600 gpm
- RHR flow is at least 2,000 gpm

In the above system lineup, total flow to the core is decreased compared to the flow when two cold legs are in service. This is acceptable due to the substantial margin between the flow required for cooling and the flow available, even through a slightly restricted RHR train.

The review concerning boron stratification with the utilization of the single injection point line, indicates there will not be a significant change in the flow rate or distribution through the core, so there is not an increased concern due to stratification.

Flow velocity, which is high, is not a concern from a flow erosion or pipe loading standpoint. There are no loads imposed on the piping system which would exceed those experienced in a seismic event. The temperature of the fluid is low and is not significant from a flow erosion standpoint.

The boron dilution accident analysis, for Millstone Unit 3 in MODE 5, assumes a full RHR System flow of approximately 4,000 gpm. Westinghouse analysis, Reference (1), for RHR flows down to 1,000 gpm, determined adequate mixing results. As the configuration will result in a RHR flow rate only slightly less then 4,000 gpm there is no concern in regards to a boron dilution accident.

The basis for the requirement of two RCS loops OPERABLE is to provide natural circulation heat sink in the event the operating RHR loop is lost. If the RHR loop were lost, with two loops swept and vented and two loops air bound, natural circulation would be established in the two swept loops.

Natural circulation would not be established in the air bound loops. Since there would be no circulation in the air bound loops, there would be no mechanism for the air in those loops to be carried to the vessel, and subsequently into the swept loops rendering them inoperable for heat sink requirements.

The LCO is met as long as at least two reactor coolant loops are OPERABLE and the following conditions are satisfied:

- One RHR loop is OPERABLE and in operation, with exceptions as allowed in Technical Specifications; and

Basis Change of 8-25-2005

The LCO is met as long as at least two reactor coolant loops are OPERABLE and the following conditions are satisfied:

- One RHR loop is OPERABLE and in operation, with exceptions as allowed in Technical Specifications; and
  Either of the following:
- An additional RHR loop OPERABLE, with exceptions as allowed in Technical Specifications; or
- The secondary side water level of at least two steam generators shall be greater than 17% (These are assumed to be on OPERABLE reactor coolant loops)

When the reactor coolant loops are swept, the mechanism exists for air to be carried into previously OPERABLE loops. All previously OPERABLE loops are declared inoperable and an additional RHR loop is required OPERABLE as specified by LCO 3.4.1.4.2 for loops not filled. When the RCS has been filled, swept, and vented using an approved procedure, all unisolated loops may be declared OPERABLE.

ISOLATED LOOP STARTUP

The below requirements are for unisolating a loop with all four loops isolated while decay heat is being removed by RHR and to clarify prerequisites to meet T/S requirements for unisolating a loop at any time.

With no RCS loops operating, the two RHR loops referenced in Specification 3.4.1.4.2 are the operating loops. Starting in MODE 4 as referenced in Specification 3.4.1.3, the RHR loops are allowed to be used in place of an operating RCS loop. Specification 3.4.1.4.2 requires two RHR loops OPERABLE and at least one in operation. Ensuring the isolated cold leg temperature is within 20°F of the highest RHR outlet temperature for the operating RHR loops within 30 minutes prior to opening the cold leg stop valve is a conservative approach since the major concern is a positive reactivity addition.

SR 4.4.1.6.1: When in MODE 5 with all RCS loops isolated, the two RHR loops referenced in LCO 3.4.1.4.2 shall be considered the OPERABLE RCS loops.

ISOLATED LOOP STARTUP (Continued)

The isolated loop cold leg temperature shall be determined to be within 20°F of the highest RHR outlet temperature for the operating RHR loops within 30 minutes prior to opening the cold leg stop valve.

Surveillance requirement 4.4.1.6.2 is met when the following actions occur within 2 hours prior to opening the cold leg or hot leg stop valve:

- An RCS boron sample has been taken and analyzed to determine current boron concentration
- The SHUTDOWN MARGIN has been determined using OP 3209B, "Shutdown Margin" using the current boron concentration determined above
- For the isolated loop being restored, the power to both loop stop valves has been restored

- For the isolated loop being restored, the power to both loop stop valves has been restored

Surveillance 4.4.1.6.2 indicates that the reactor shall be determined subcritical by at least the amount required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6 within 2 hours of opening the cold leg or hot leg stop valve.

The SHUTDOWN MARGIN requirement in Specification 3.1.1.1.2 is specified in the Core Operating Limits Report for MODE 5 with RCS loops filled. Specification 3.1.1.1.2 cannot be used to determine the required SHUTDOWN MARGIN for MODE 5 loops isolated condition.

Specification 3.1.1.2 requires the SHUTDOWN MARGIN to be greater than or equal to the limits specified in the Core Operating Limits Report for MODE 5 with RCS loops not filled provided CVCS is aligned to preclude boron dilution. This specification is for loops not filled and therefore is applicable to an all loops isolated condition.

Specification 3.9.1.1 requires K_{eff} of 0.95 or less, or a boron concentration of greater than or equal to the limit specified in the COLR in MODE 6.

Specification 3.1.1.1.2 or 3.1.1.2 for MODE 5, both require boron concentration to be determined at least once each 24 hours. SR 4.1.1.1.2.1.b.2 and 4.1.1.2.1.b.1 satisfy the requirements of Specifications 3.1.1.1.2 and 3.1.1.2 respectfully. Specification 3.9.1.1 for MODE 6 requires boron concentration to be determined at least once each 72 hours. S.R. 4.9.1.1.2 satisfy the requirements of Specification 3.9.1.1.

References:

1. Letter NEU-94-623, dated July 13, 1994; Mixing Evaluation for Boron Dilution Accident in Modes 4 and 5, Westinghouse HR-59782.

2. Memo No. MP3-E-93-821, dated October 7, 1993.

Bases Change of 8-25-2005

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 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 condition which could occur during shutdown. If any pressurizer Code safety valve is inoperable, and cannot be restored to OPERABLE status, the ACTION statement requires the plant to be shut down and cooled down such that Technical Specification 3.4.9.3 will become applicable and require cold overpressure protection to be placed in service.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of 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 pressurizer provides a point in the RCS when liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during load transients.

MODES 1 AND 2

The requirement for the pressurizer to be OPERABLE, with pressurizer level maintained at programmed level within ± 6% of full scale is consistent with the accident analysis in Chapter 15 of the FSAR. The accident analysis assumes that pressurizer level is being maintained at the programmed level by the automatic control system, and when in manual control, similar limits are established. The programmed level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure and pressurizer overfill transients. A pressurizer level control error based upon automatic level control has been taken into account for those transients where pressurizer overfill is a concern (e.g., loss of feedwater, feedwater line break, and inadvertent ECCS actuation at power). When in manual control, the goal is to maintain pressurizer level at the program level value. The ± 6 % of full scale acceptance criterion in the Technical Specification establishes a band for operation to accommodate variations between level measurements. This value is bounded by the margin applied to the pressurizer overfill events.

Bases Change of 8-25-2005

3/4.4.3 PRESSURIZER (cont'd.)

The 12-hour periodic surveillances require that pressurizer level be maintained at programmed level within ± 6% of full scale. The surveillance is performed by observing the indicated level. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and to ensure that the appropriate level exists in the pressurizer. During transitory conditions, i.e., power changes, the operators will maintain programmed level, and deviations greater than 6% will be corrected within 2 hours. Two hours has been selected for pressurizer level restoration after a transient to avoid an unnecessary downpower with pressurizer level outside the operating band. Normally, alarms are also available for early detection of abnormal level indications.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of the reactor coolant. Unless adequate heater capacity is available, the hot high-pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single-phase natural circulation and decreased capability to remove core decay heat.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity of at least 175 kW. The heaters are capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The requirement for two groups of pressurizer heaters, each having a capacity of 175 kW, is met by verifying the capacity of the pressurizer heater groups A and B. Since the pressurizer heater groups A and B are supplied from the emergency 480V electrical buses, there is reasonable assurance that these heaters can be energized during a loss of offsite power to maintain natural circulation at HOT STANDBY. Providing an emergency (Class 1E) power source for the required pressurizer heaters meets the requirement of NUREG-0737, "A Clarification of TMI Action Plan Requirements," II.E.3.1, "Emergency Power Requirements for Pressurizer Heaters."

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that a demand caused by loss of offsite power would be unlikely in this time period. Pressure control may be maintained during this time using normal station powered heaters.

MODE 3

The requirement for the pressurizer to be OPERABLE, with a level less than or equal to 89%, ensures that a steam bubble exists. The 89% level preserves the steam space for pressure control. The 89% level has been established to ensure the capability to establish and maintain pressure control for MODE 3 and to ensure a bubble is present in the pressurizer. Initial pressurizer level is not significant for those events analyzed for MODE 3 in Chapter 15 of the FSAR.

3/4.4.3 PRESSURIZER (cont'd.)

The 12-hour periodic surveillance requires that during MODE 3 operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The surveillance is performed by observing the indicated level. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and to ensure that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

The basis for the pressurizer heater requirements is identical to MODES 1 and 2.

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. Requiring the PORVs to be OPERABLE ensures that the capability for depressurization during safety grade cold shutdown is met.

ACTION statements a, b, and c distinguishes the inoperability of the power operated relief valves (PORV). Specifically, a PORV may be designated inoperable but it may be able to automatically and manually open and close and therefore, able to perform its function. PORV inoperability may be due to seat leakage which does not prevent automatic or manual use and does not create the possibility for a small-break LOCA. For these reasons, the block valve may be closed but the action requires power to be maintained to the valve. This allows quick access to the PORV for pressure control. On the other hand if a PORV is inoperable and not capable of being automatically and manually cycled, it must be either restored or isolated by closing the associated block valve and removing power.

Note: PORV position indication does not affect the ability of the PORV to perform any of its safety functions. Therefore, the failure of PORV position indication does not cause the PORV to be inoperable. However, failed position indication of these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3.

Automatic operation of the PORVs is created to allow more time for operators to terminate an Inadvertent ECCS Actuation at Power. The PORVs and associated piping have been demonstrated to be qualified for water relief. Operation of the PORVs will prevent water relief from the pressurizer safety valves for which qualification for water relief has not been demonstrated. If the PORVs are capable of automatic operation but have been declared inoperable, closure of the PORV block valve is acceptable since the Emergency Operating Procedures provide guidance to assure that the PORVs would be available to mitigate the event. OPERABILITY and setpoint controls for the safety grade PORV opening logic are maintained in the Technical Requirements Manual.

Basis Change of 8-25-2005

RELIEF VALVES (Continued)

The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve(s) cannot be restored to OPERABLE status within 1 hour, the remedial action is to place the PORV in manual control (i.e. the control switch in the "CLOSE" position) to preclude its automatic opening for an overpressure event and to avoid the potential of a stuck-open PORV at a time that the block valve is inoperable. The time allowed to restore the block valve(s) to OPERABLE status is based upon the remedial action time limits for inoperable PORV per ACTION requirements b. and c. ACTION statement d. does not specify closure of the block valves because such action would not likely be possible when the block valve is inoperable. For the same reasons, reference is not made to ACTION statements b. and c. for the required remedial actions.

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.

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). 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.

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 promptly reported to the Commission in a Special Report 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.

LCO 3.4.6.1.b. Containment Sump Drain Level or Pumped Capacity Monitoring System

The intent of LCO 3.4.6.1.b is to have a system able to monitor and detect leakage from the reactor coolant pressure boundary (RCPB). The system can use sump level, pump capacity or both as the LCO implies. It does not have to have two separate systems. The "Containment Drain Sump Level or Pumped Capacity Monitoring" System is defined as any one of the following three Systems:

A. 3DAS-P10, Unidentified Leakage Sump Pump, and associated local and main board annunciation.

B. 3DAS-P10, Unidentified Leakage Sump Pump, and computer point 3DAS-L39 and CVLKR2.

C. 3DAS-P2A or 3DAS-P2B, Containment Drains Sump Pump, and computer points 3DAS-L22 and CVLKR2 or CVLKR3I.

To meet Regulatory Guide 1.45 recommendations, the Containment Drain Sump Level or Pumped Capacity Monitoring System must meet the following five criteria:

1. Must monitor changes in sump water level, changes in flow rate or changes in the operating frequency of pumps.

2. Be able to detect an UNIDENTIFIED LEAKAGE rate of 1 gpm in less than one hour.

3. Remain OPERABLE following an Operating Basis Earthquake (OBE). |

4. Provide indication and alarm in the Control Room.

5. Procedures for converting various indications to a common leakage equivalent must be available to the Operators.

The three Containment Drain Sump Level or Pumped Capacity Monitoring Systems identified above meet these five requirements as follows:

A. 3DAS-P10, Unidentified Leakage Sump Pump, and associated main board annunciation.

   1. Sump level is monitored at two locations by the starting and stopping of 3DAS-P10, Unidentified Leakage Sump Pump. Flow is measured as a function of time between pump starts/stops and the known sump levels at which these occur.

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3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

2. Two timer relays in the control circuitry of 3DAS-P10 are set to identify a 1 gpm leak rate within 1 hour.

3. This monitoring system is not seismic Category I, but is expected to remain OPERABLE during an OBE. If the monitoring system is not OPERABLE following a seismic event, the appropriate ACTION according to Technical Specifications will be taken. This position has been reviewed by the NRC and documented as acceptable in the Safety Evaluation Report.

4. If the control circuitry of 3DAS-P10 identifies a 1 gpm leak rate within 1 hour, Liquid Radwaste Panel Annunciator LWS 4-5, CTMT UNIDENT LEAKAGE TROUBLE, and Main Board Annunciator MB1 B 4-3, RAD LIQUID WASTE SYS TROUBLE, will alarm. These control circuits and alarms operate independently from the plant process computer.

   If the computer is inoperable, these control circuits and alarms meet the Technical Specification requirements for the Containment Drain Sump Level or Pumped Capacity Monitoring System.

5. To convert the unidentified leakage sump pump run times to a leakage rate, use the following formula:

$$\frac{(\text{3DAS-P10 run times in minutes} - [\text{number of 3DAS-P10 starts} \times .5 \text{ minutes}]) \times 20 \text{ gpm}}{\text{Elapsed monitored Time in minutes}}$$

B. 3DAS-P10, Unidentified Leakage Sump Pump, and computer points 3DAS-L39 and CVLKR2.

1. Sump level is monitored by 3DAS-LI39, the Unidentified Leakage Sump Level indicator. This level indicator provides an input to computer point 3DAS-L39.

2. The plant process computer calculates a leakage rate every 30 seconds when 3DASP10 indicates stop. This leakage rate is displayed via computer point CVLKR2. When pump P10 does run, the leakage rate calculation is stopped and resumes 10 minutes after pump P10 stops. If it cannot provide a value of the leakage rate within any 54 minute interval, CVDASP10NC (UNDNT LKG RT NOT CALC) alarms which alerts the Operator that UNIDENTIFIED LEAKAGE cannot be determined.

3. This monitoring system is not seismic Category I, but is expected to remain OPERABLE during an OBE. If the monitoring system is not OPERABLE following a seismic event, the appropriate ACTION according to Technical Specifications will be taken.

4. A priority computer alarm (CVLKR2) is generated if the calculated leakage rate is greater than a value specified on the Priority Alarm Point Log. This alarm value should be set to alert the Operators to a possible RCS leak rate in excess of the Technical Specification maximum allowed UNIDENTIFIED LEAKAGE. The alarm

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3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

value may be set at one gallon per minute or less above the rate of IDENTIFIED LEAKAGE, from the reactor coolant or auxiliary systems, into the unidentified leakage sump. The rate of IDENTIFIED LEAKAGE may be determined by either measurement or analysis. If the Priority Alarm Point Log is adjusted, the high leakage rate alarm will be bounded by the IDENTIFIED LEAKAGE rate and the low leakage rate alarm will be set to notify the operator that a decrease in leakage may require the high leakage rate alarm to be reset. The priority alarm setpoint shall be no greater than 2 gallons per minute. This ensures that the IDENTIFIED LEAKAGE will not mask a small increase in UNIDENTIFIED LEAKAGE that is of concern. The 2 gallons per minute limit is also within the identified leakage sump level monitoring system alarm operating range which has a maximum setpoint of 2.3 gallons per minute.

To convert unidentified leakage sump level changes to leakage rate, use the following formula:

Note: Wait 10 minutes after 3DAS-P10 stops before taking level readings.

$$\frac{1.08315\ \text{gallons}}{1\%} \times \frac{\%\ \text{change in level from 3DAS-L39}}{\text{time between level readings in minutes}}$$

C. 3DAS-P2A or 3DAS-P2B, Containment Drains Sump Pump, and computer points 3DAS-L22 and CVLKR2 or CVLKR3I.

1. Sump level is monitored by 3DAS-LI22, the Containment Drains Sump Level Indicator. This level indicator provides an input to computer point 3DAS-L22.

   This method can be used to monitor UNIDENTIFIED LEAKAGE when Pump P10 and its associated equipment is inoperable provided Pump P10 is out of service and 3DAS-L139 indicates that the unidentified leakage sump is overflowing to the containment drains sump (approximately 36% level on 3DAS-LI39). In this case, CVLKR2 and CVLKR3I monitor flow rate by comparing level indications on the containment drains sump when Pumps P10, P2A, P2B and P1 are not running.

2. The plant process computer calculates a leakage rate every 30 seconds when 3DAS-P10, 3DAS-P1, 3DAS-P2A and 3DAS-P2B indicate stop. This leakage rate is displayed via computer points CVLKR3I and CVLKR2 when 3DAS-P10 is off and when the unidentified leakage sump is overflowing to the containment drains sump. When one of these pumps does run, the leakage rate calculation is stopped and resumes 10 minutes after all pumps stop. If it cannot provide a value of the leakage rate within any 54 minute interval, two computer point alarms (CVDASP2NC, UNDNT LKG RT NOT CALC and CVDASP2NC, SMP 3 LKG RT NT CALC) are generated which alerts the Operator that UNIDENTIFIED LEAKAGE cannot be determined.

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3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

3. This monitoring system is not seismic Category I, but is expected to remain OPERABLE during an OBE. If the monitoring system is not OPERABLE following a seismic event, the appropriate ACTION according to Technical Specifications will be taken.

4. Two priority computer alarms (CVLKR2 and CVLKR3I) are generated if the calculated leakage rate is greater than a value specified on the Priority Alarm Point Log. This alarm value should be set to alert the Operators to a possible RCS leak rate in excess of the Technical Specification maximum allowed UNIDENTIFIED LEAKAGE. The alarm value may be set at one gallon per minute or less above the rate of IDENTIFIED LEAKAGE, from the reactor coolant or auxiliary systems, into the containment drains sump. The rate of IDENTIFIED LEAKAGE may be determined by either measurement or by analysis. If the Priority Alarm Point Log is adjusted, the high leakage rate alarm will be bounded by the IDENTIFIED LEAKAGE rate and the low leakage rate alarm will be set to notify the operator that a decrease in leakage may require the high leakage rate alarm to be reset. The priority alarm setpoint shall be no greater than 2 gallons per minute. This ensures that the IDENTIFIED LEAKAGE will not mask a small increase in UNIDENTIFIED LEAKAGE that is of concern. The 2 gallons per minute limit is also within the containment drains sump level monitoring system alarm operating range which has a maximum setpoint of 2.5 gallons per minute.

5. To convert containment drains sump run times to a leakage rate, refer to procedure SP3670.1 for guidance on the conversion method.

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.

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3/4.4.6.2 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 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

A Limit of 40 gpm is placed on CONTROLLED LEAKAGE. CONTROLLED LEAKAGE is determined under a set of reference conditions, listed below:

a. One Charging Pump in operation.

b. RCS pressure at 2250 +/- 20 psia.

By limiting CONTROLLED LEAKAGE to 40 gpm during normal operation, we can be assured that during an SI with only one charging pump injecting, RCP seal injection flow will continue to remain less than 80 gpm as assumed in accident analysis. When the seal injection throttle valves are set with a normal charging line up, the throttle valve position bounds conditions where higher charging header pressures could exist. Therefore, conditions which create higher charging header pressures such as an isolated charging line, or two pumps in service are bounded by the single pump-normal system lineup surveillance configuration. Basic accident analysis assumptions are that 80 gpm flow is provided to the seals by a single pump in a runout condition.

The specified allowable 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.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS Operational Leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The Surveillance Requirements for RCS pressure isolation valves provide 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of Surveillance Requirement 4.4.6.2.2 (including Surveillance Requirement 4.4.6.2.2.d) for RCS pressure isolation valves which can only be leak-tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2.d to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressure has been met.

In MODES 1 and 2, the plant is at normal operating pressure and Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual action or flow through the valve. In MODES 3 and 4, Surveillance Requirement 4.4.6.2.2.d shall be performed within 24 hours of valve actuation due to automatic or manual action or flow through the valve if and when RCS pressure is sufficiently high for performance of this surveillance.

References:

1. Letter FSD/SS-NEU-3713, dated March 25, 1985.
2. Letter NEU-89-639, dated December 4, 1989.

OPERATIONAL LEAKAGE (Continued)

The specified allowable 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 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 DELETED

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 SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 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

SPECIFIC ACTIVITY (Continued)

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 Millstone site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the 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 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 SITE BOUNDARY, 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 SITE BOUNDARY under any accident condition.

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SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

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

REACTOR COOLANT SYSTEM (EXCEPT THE PRESSURIZER)

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4-2 and 3.4-3 contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational requirements during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. A heatup or cooldown is defined as a temperature increase or decrease of greater than or equal to 10°F in any one hour period. This definition of heatup and cooldown is based upon the ASME definition of isothermal conditions described in ASME, Section XI, Appendix E.

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (continued)

Steady state thermal conditions exist when temperature increases or decreases are <10°F in any one hour period and when the plant is not performing a planned heatup or cooldown in accordance with a procedure.

The LCO establishes operating limits that provide a margin to brittle failure, applicable to the ferritic material of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the Pressurizer.

The P/T limits have been established for the ferritic materials of the RCS considering ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2), and the additional requirements of 10 CFR 50 Appendix G (Reference 3). Implementation of the specific requirements provide adequate margin to brittle fracture of ferritic materials during normal operation, anticipated operational occurrences, and system leak and hydrostatic tests.

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations may be more restrictive, and thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The P/T limits include uncertainty margins to ensure that the calculated limits are not inadvertently exceeded. These margins include gauge and system loop uncertainties, elevation differences, containment pressure conditions and system pressure drops between the beltline region of the vessel and the pressure gauge or relief valve location.

PRESSURE/TEMPERATURE LIMITS (continued)

The criticality limit curve includes the Reference 1 requirement that it be $\geq$ 40°F above the heatup curve or the cooldown curve, and not less than 160°F above the minimum permissible temperature for ISLH testing. This limit provides the required margin relative to brittle fracture. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the ferritic RCPB materials, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSIS

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1, as modified by Reference 2, combined with the additional requirements of Reference 3 provide the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The LCO limits apply to the ferritic components of the RCS, except the Pressurizer. These limits define allowable operating regions while providing margin against nonductile failure for the controlling ferritic component.

The limitations imposed on the rate of change of temperature have been established to ensure consistency with the resultant heatup, cooldown, and ISLH testing P/T limit curves. These limits control the thermal gradients (stresses) within the reactor vessel beltline (the limiting component). Note that while these limits are to provide protection to ferritic components within the reactor coolant pressure boundary, a limit of 100°F/hr applies to the reactor coolant pressure boundary (except the pressurizer) to ensure that operation is maintained within the ASME Section III design loadings, stresses, and fatigue analyses for heatup and cooldown.

PRESSURE/TEMPERATURE LIMITS (continued)

Violating the LCO limits places the reactor vessel outside of the bounds of the analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure of ferritic RCS components using ASME Section XI Appendix G as modified by Code Case N-640 and the additional requirements of 10CFR50, Appendix G (Ref. 1). The P/T limits were developed to provide requirements for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, in keeping with the concern for nonductile failure. The limits do not apply to the Pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.2.5, "DNB Parameters"; LCO 3.2.3.1, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor"; LCO 3.1.1.4, "Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Allowed Outage Times (AOTs) reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

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PRESSURE/TEMPERATURE LIMITS (continued)

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour AOT when operating in MODES 1 through 4 is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

This evaluation must be completed whenever a limit is exceeded. Restoration within the AOT alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE or not allowed to enter MODE 4 because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required evaluation for continued operation in MODES 1 through 4 cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in the ACTION statement. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psia within the next 30 hours.

Completion of the required evaluation following limit violation in other than MODES 1 through 4 is required before plant startup to MODE 4 can proceed.

The AOTs are reasonable, based on operating experience to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Verification that operation is within the LCO limits as well as the limits of Figures 3.4-2 and 3.4-3 is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This frequency is considered reasonable in view of the control room indication available to monitor RCS status.

PRESSURE/TEMPERATURE LIMITS (continued)

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This Surveillance Requirement is only required to be performed during system heatup, cooldown, and ISLH testing. No Surveillance Requirement is given for criticality operations because LCO 3.1.1.4 contains a more restrictive requirement.

It is not necessary to perform Surveillance Requirement 4.4.9.1.1 to verify compliance with Figures 3.4-2 and 3.4-3 when the reactor vessel is fully detensioned. During REFUELING, with the head fully detensioned or off the reactor vessel, the RCS is not capable of being pressurized to any significant value. The limiting thermal stresses which could be encountered during this time would be limited to flood-up using RWST water as low as 40°F. It is not possible to cause crack growth of postulated flaws in the reactor vessel at normal REFUELING temperatures even injecting 40°F Water.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.
2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.
3. 10 CFR 50 Appendix G, "Fracture Toughness Requirements."
4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706."
5. 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."
6. Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.
7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events," 1995 Edition.

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OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The Cold Overpressure Protection System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the isothermal beltline pressure and temperature (P/T) limits developed using the guidance of ASME Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2). The reactor vessel is the limiting RCPB component for demonstrating such protection.

Cold Overpressure Protection consists of two PORVs with nominal lift setting as specified in Figures 3.4-4a and 3.4-4b, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two relief valves are required for redundancy. One relief valve has adequate relieving capability to prevent overpressurization of the RCS for the required mass input capability.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The use of a PORV for Cold Overpressure Protection is limited to those conditions when no more than one RCS loop is isolated from the reactor vessel. When two or more loops are isolated, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized and vented RCS.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to stress at low temperatures (Ref. 3). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause nonductile cracking of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the limits provided in Figures 3.4-2 and 3.4-3.

This LCO provides RCS overpressure protection by limiting mass input capability and requiring adequate pressure relief capacity. Limiting mass input capability requires all Safety Injection (SIH) pumps and all but one centrifugal charging pump to be incapable of injection into the RCS. The pressure relief capacity requires either two redundant relief valves or a depressurized RCS and an RCS vent of sufficient size. One relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum mass input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the Cold Overpressure Protection modes and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve.

If a loss of RCS inventory or reduction in SHUTDOWN MARGIN event occurs, the appropriate response will be to correct the situation by starting RCS makeup pumps. If the loss of inventory or SHUTDOWN MARGIN is significant, this may necessitate the use of additional RCS makeup pumps that are being maintained not capable of injecting into the RCS in accordance with Technical Specification 3.4.9.3. The use of these additional pumps to restore RCS inventory or SHUTDOWN MARGIN will require entry into the associated ACTION statement. The ACTION statement requires immediate action to comply with the specification. The restoration of RCS inventory or SHUTDOWN MARGIN can be considered to be part of the immediate action to restore the additional RCS makeup pumps to a not capable of injecting status. While recovering RCS inventory or SHUTDOWN MARGIN, RCS pressure will be maintained below the P/T limits. After RCS inventory or SHUTDOWN MARGIN has been restored, the additional pumps should be immediately made not capable of injecting and the ACTION statement exited.

OVERPRESSURE PROTECTION SYSTEMS (continued)

PORV Requirements

As designed, the PORV Cold Overpressure Protection (COPPS) is signaled to open if the RCS pressure approaches a limit determined by the COPPS actuation logic. The COPPS actuation logic monitors both RCS temperature and RCS pressure and determines when the nominal setpoint of Figure 3.4-4a or Figure 3.4-4b is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure setpoint for that temperature. The calculated pressure setpoint is then compared with RCS pressure measured by a wide range pressure channel. If the measured pressure meets or exceeds the calculated value, a PORV is signaled to open.

The use of the PORVs is restricted to three and four RCS loops unisolated: for a loop to be considered isolated, both RCS loop stop valves must be closed. If more than one loop is isolated, then the PORVs must have their block valves closed or COPPS must be blocked. For these cases, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized RCS and an RCS vent. This is necessary because the PORV mass and heat injection transients have only been analyzed for a maximum of one loop isolated, the use of the PORVs is restricted to three and four RCS loops unisolated.

The RHR suction relief valves have been qualified for all mass injection transients for any combination of isolated loops. In addition, the heat injection transients not prohibited by the Technical Specifications have also been considered in the qualification of the RHR suction relief valves.

Figure 3.4-4a and Figure 3.4-4b present the PORV setpoints for COPPS. The setpoints are staggered so only one valve opens during a low temperature overpressure transient. Setting both valves to the values of Figure 3.4-4a and Figure 3.4-4b within the tolerance allowed for the calibration accuracy, ensures that the isothermal P/T limits will not be exceeded for the analyzed isothermal events.

When a PORV is opened, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

OVERPRESSURE PROTECTION SYSTEMS

RHR Suction Relief Valve Requirements

The isolation valves between the RCS and the RHR suction relief valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valves with setpoint tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 4) for Class 2 relief valves.

When the RHR system is operated for decay heat removal or low pressure letdown control, the isolation valves between the RCS and the RHR suction relief valves are open, and the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at acceptable pressure levels in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting mass or heat input transient, and maintaining pressure below the P/T limits for the analyzed isothermal events.

For an RCS vent to meet the flow capacity requirement, it requires removing a Pressurizer safety valve, removing a Pressurizer manway, or similarly establishing a vent by opening an RCS vent valve provided that the opening meets the relieving capacity requirements. The vent path must be above the level of reactor coolant, so as not to drain the RCS when open.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABLE SAFETY ANALYSIS

Safety analyses (Ref. 5) demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits for the analyzed isothermal events. In MODES 1, 2, AND 3, and in MODE 4, with RCS cold leg temperature exceeding 226°F, the pressurizer safety valves will provide RCS overpressure protection in the ductile region. At 226°F and below, overpressure prevention is provided by two means: (1) two OPERABLE relief valves, or (2) a depressurized RCS with a sufficiently sized RCS vent, consistent with ASME Section XI, Appendix G for temperatures less than RT_{NDT} + 50°F. Each of these means has a limited overpressure relief capability.

The required RCS temperature for a given pressure increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the Technical Specification curves are revised, the cold overpressure protection must be re-evaluated to ensure its functional requirements continue to be met using the RCS relief valve method or the depressurized and vented RCS condition.

Transients capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch

Heat Input Transients

a. Inadvertent actuation of Pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The Technical Specifications ensure that mass input transients beyond the OPERABILITY of the cold overpressure protection means do not occur by rendering all Safety Injection Pumps and all but one centrifugal charging pump incapable of injecting into the RCS whenever an RCS cold leg is ≤ 226°F.

The Technical Specifications ensure that energy addition transients beyond the OPERABILITY of the cold overpressure protection means do not occur by limiting reactor coolant pump starts. LCO 3.4.1.4.1, "Reactor Coolant Loops and Coolant Circulation - COLD SHUTDOWN - Loops Filled," LCO 3.4.1.4.2, "Reactor Coolant

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OVERPRESSURE PROTECTION SYSTEMS (continued)

Loops and Coolant Circulation - COLD SHUTDOWN - Loops Not Filled," and LCO 3.4.1.3, "Reactor Coolant Loops and Coolant Circulation - HOT SHUTDOWN" limit starting the first reactor coolant pump such that it shall not be started when any RCS loop wide range cold leg temperature is ≤ 226°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature. The restrictions ensure the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressurization event beyond the capability of the COPPS to mitigate. The COPPS utilizes the pressurizer PORVs and the RHR relief valves to mitigate the limiting mass and energy addition events, thereby protecting the isothermal reactor vessel beltline P/T limits. The restrictions will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no restrictions are necessary to start additional RCPs for reactor vessel protection. In addition, this restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

The RCP starting criteria are based on the equipment used to provide cold overpressure protection. A maximum temperature differential of 50°F between the steam generator secondary sides and RCS cold legs will limit the potential energy addition to within the capability of the pressurizer PORVs to mitigate the transient. The RHR relief valve are also adequate to mitigate energy addition transients constrained by this temperature differential limit, provided all RCS cold leg temperature are at or below 150°F. The ability of the RHR relief valves to mitigate energy addition transients when RCS cold leg temperature is above 150°F has not been analyzed. As a result, the temperature of the steam generator secondary sides must be at or below the RCS cold leg temperature if the RHR relief valves are providing cold overpressure protection and the RCS cold leg temperature is above 150°F.

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OVERPRESSURE PROTECTION SYSTEMS (continued)

The cold overpressure transient analyses demonstrate that either one relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when RCS letdown is isolated and only one centrifugal charging pump is operating. Thus, the LCO allows only one centrifugal charging pump capable of injecting when cold overpressure protection is required.

The cold overpressure protection enabling temperature is conservatively established at a value ≤ 226°F based on the criteria provided by ASME Section XI, Appendix G.

PORV Performance

The analyses show that the vessel is protected against non-ductile failure when the PORVs are set to open at the values shown in Figures 3.4-4a and 3.4-4b within the tolerance allowed for the calibration accuracy. The curves are derived by analyses for both three and four RCS loops unisolated that model the performance of the PORV cold overpressure protection system (COPPS), assuming the limiting mass and heat transients of one centrifugal charging pump injecting into the RCS, or the energy addition as a result of starting an RCP with temperature asymmetry between the RCS and the steam generators. These analyses consider pressure overshoot beyond the PORV opening setpoint resulting from signal processing and valve stroke times.

The PORV setpoints in Figures 3.4-4a and 3.4-4b will be updated when the P/T limits conflict with the cold overpressure analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement. Revised limits are determined using neutron fluence projections and the results of testing of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System (Except the Pressurizer)," discuss these evaluations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints as do the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 426.8 psig and 453.2 psig will pass flow greater than that required for the limiting cold overpressure transient while maintaining RCS pressure less than the isothermal P/T limit curve. Assuming maximum relief flow requirements during the limiting cold overpressure event, an RHR suction relief valve will maintain RCS pressure to ≤ 110% of the nominal lift setpoint.

Although each RHR suction relief valve is a passive spring loaded device, which meets single failure criteria, its location within the RHR System precludes meeting single failure criteria when spurious RHR suction isolation valve or RHR suction valve closure is postulated. Thus the loss of an RHR suction relief

OVERPRESSURE PROTECTION SYSTEMS (continued)

valve is the worst case single failure. Also, as the RCS P/T limits are revised to reflect change in toughness in the reactor vessel materials, the RHR suction relief valve's analyses must be re-evaluated to ensure continued accommodation of the design bases cold overpressure transients.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of $\geq$ 2.0 square inches is capable of mitigating the limiting cold overpressure transient. The capacity of this vent size is greater than the flow of the limiting transient, while maintaining RCS pressure less than the maximum pressure on the isothermal P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the isothermal P/T limit curves are revised.

The RCS vent is a passive device and is not subject to active failure.

The RCS vent satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

OVERPRESSURE PROTECTION SYSTEMS (continued)

LCO

This LCO requires that cold overpressure protection be OPERABLE and the maximum mass input be limited to one charging pump. Failure to meet this LCO could lead to the loss of low temperature overpressure mitigation and violation of the reactor vessel isothermal P/T limits as a result of an operational transient.

To limit the mass input capability, the LCO requires a maximum of one centrifugal charging pump capable of injecting into the RCS.

The elements of the LCO that provides low temperature overpressure mitigation through pressure relief are:

1. Two OPERABLE PORVs; or

   A PORV is OPERABLE for cold overpressure protection when its block valve is open, its lift setpoint is set to the nominal setpoints provided for both three and four loops unisolated by Figure 3.4-4a or 3.4-4b and when the surveillance requirements are met.

2. Two OPERABLE RHR suction relief valves; or

   An RHR suction relief valve is OPERABLE for cold overpressure protection when its isolation valves from the RCS are open and when its setpoint is at or between 426.8 psig and 453.2 psig, as verified by required testing.

3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

4. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of $\geq$ 2.0 square inches.

Each of these methods of ovepressure prevention is capable of mitigating the limiting cold overpressure transient.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 226°F, in MODE 5, and in MODE 6 when the head is on the reactor vessel. The Pressurizer safety valves provide RCS overpressure protection in the ductile region (i.e. > 226°F). When the reactor head is off, overpressurization cannot occur.

LCO 3.4.9.1 "Pressure/Temperature Limits" provides the operational P/T limits for all MODES. LCO 3.4.2, "Safety Valves," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and 4 when all RCS cold leg temperatures are > 226°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a rapid increase in RCS pressure when little or no time exists for operator action to mitigate the event.

ACTIONS

a. and b.

With two or more centrifugal charging pumps capable of injecting into the RCS, or with any SIH pump capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted mass input capability to the RCS reflects the urgency of removing the RCS from this condition.

Required ACTION a. is modified by a Note that permits two centrifugal charging pumps capable of RCS injection for ≤ 1 hour to allow for pump swaps. This is a controlled evolution of short duration and the procedure prevents having two charging pumps simultaneously out of pull-to-lock while both charging pumps are capable of injecting into the RCS.

c.

In MODE 4 when any RCS cold leg temperature is ≤ 226°F, with one required relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within an allowed outage time (AOT) of 7 days. Two relief valves in any combination of the PORVs and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

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OVERPRESSURE PROTECTION SYSTEMS (continued)

The AOT in MODE 4 considers the facts that only one of the relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. The RCS must be depressurized and a vent must be established within the following 12 hours if the required relief valve is not restored to OPERABLE within the required AOT of 7 days.

d.

The consequences of operational events that will overpressurize the RCS are more severe at lower temperatures (Ref. 8). Thus, with one of the two required relief valves inoperable in MODE 5 or in MODE 6 with the head on, the AOT to restore two valves to OPERABLE status is 24 hours.

The AOT represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE relief valve to protect against overpressure events. The RCS must be depressurized and a vent must be established within the following 12 hours if the required relief valve is not restored to OPERABLE within the required AOT of 24 hours.

e.

The RCS must be depressurized and a vent must be established within 12 hours when both required Cold Overpressure Protection relief valves are inoperable.

The vent must be sized $\geq$ 2.0 square inches to ensure that the flow capacity is greater than that required for the worst case cold overpressure transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible non-ductile failure of the reactor vessel.

The time required to place the plant in this Condition is based on the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1

Performance of an ANALOG CHANNEL OPERATIONAL TEST is required within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The ANALOG CHANNEL OPERATIONAL TEST will verify the setpoint in accordance with the nominal values given in Figures 3.4-4a and 3.4-4b. PORV actuation could depressurize the RCS; therefore, valve operation is not required.

OVERPRESSURE PROTECTION SYSTEMS (continued)

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required once per 24 months to adjust the channel so that it responds and the valve opens within the required range and accuracy to a known input.

The PORV block valve must be verified open and COPPS must be verified armed every 72 hours to provide a flow path and a cold overpressure protection actuation circuit for each required PORV to perform its function when required. The valve is remotely verified open in the main control room. This Surveillance is performed if credit is being taken for the PORV to satisfy the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the open position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure transient.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify the PORV block valve remains open.

4.4.9.3.2

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying the RHR suction valves, 3RHS*MV8701A and 3RHS*M8701C, are open when suction relief valve 3RHS*RV8708A is being used to meet the LCO and by verifying the RHR suction valves, 3RHS*MV8702B and 3RHS*MV8702C, are open when suction relief valve 3RHS*RV8708B is being used to meet the LCO. Each required RHR suction relief valve shall also be demonstrated OPERABLE by testing it in accordance with 4.0.5. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction valves are verified to be open every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

The ASME Code, Section XI (Ref. 9), test per 4.0.5 verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

OVERPRESSURE PROTECTION SYSTEMS (continued)

4.4.9.3.3

The RCS vent of $\geq$ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent valve that cannot be locked open.

b. Once every 31 days for a valve that is locked, sealed, or secured in position or any other passive vent path. A removed Pressurizer safety valve fits this category.

This passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO.

4.4.9.3.4 and 4.4.9.3.5

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SIH pumps and all but one centrifugal charging pump are verified incapable of injecting into the RCS.

The SIH pumps and charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternate methods of control may be employed using at least two independent means to prevent an injection into the RCS. This may be accomplished through any of the following methods: 1) placing the pump in pull to lock (PTL) and pulling its UC fuses, 2) placing the pump in pull to lock (PTL) and closing the pump discharge valve(s) to the injection line, 3) closing the pump discharge valve(s) to the injection line and either removing power from the valve operator(s) or locking manual valves closed, and 4) closing the valve(s) from the injection source and either removing power from the valve operator(s) or locking manual valves closed.

An SIH pump may be energized for testing or for filling the Accumulators provided it is incapable of injecting into the RCS.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.
2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.
3. Generic Letter 88-11
4. ASME, Boiler and Pressure Vessel Code, Section III
5. FSAR, Chapter 15
6. 10CFR50, Section 50.46
7. 10CFR50, Appendix K
8. Generic Letter 90-06
9. ASME, Boiler and Pressure Vessel Code, Section XI

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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 required to meet the guidance of "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. The "operating bypass" designed for the isolation valves is applicable to MODES 1, 2, and 3 with Pressurizer pressure above P-11 setpoint. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

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 and with some valves out of normal injection lineup, on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System is required to be available to support charging pump operation. The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System consists of two redundant trains, each capable of providing 100% of the required flow. Each train has a two position, "Off" and "Auto," remote control switch. With the remote control switches for each train in the "Auto" position, the system is capable of automatically transferring operation to the redundant train in the event of a low flow condition in the operating train. The associated fans do not receive any safety related automatic start signals (e.g., Safety Injection Signal).

Placing the remote control switch for a Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train in the "Off" position to start the redundant train or to perform post maintenance testing to verify availability of the redundant train will not affect the availability of that train, provided appropriate administrative controls have been established to ensure the remote control switch is immediately returned to the "Auto" position after the completion of the specified activities or in response to plant conditions. These administrative controls include the use of an approved procedure and a designated individual at the control switch for the respective Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train who can rapidly respond to instructions from procedures, or control room personnel, based on plant conditions.

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.

Surveillance Requirement 4.5.2.b.1 requires verifying that the ECCS piping is full of water. The ECCS pumps are normally in a standby, nonoperating mode, with the exception of the operating centrifugal charging pump(s). As such, the ECCS flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly when required to inject into the RCS. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gases (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

This Surveillance Requirement is met by:

- VENTING ECCS pump casings and the accessible discharge piping high points including the ECCS pump suction crossover piping (i.e., downstream of valves 3RSS*MV8837A/B and 3RSS*MV8838A/B to safety injection and charging pump suction).

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ECCS SUBSYSTEMS (Continued)

- VENTING of the nonoperating centrifugal charging pumps at the suction line test connection. The nonoperating centrifugal charging pumps do not have casing vent connections and VENTING the suction pipe will assure that the pump casing does not contain voids and pockets of entrained gases.

- using an external water level detection method for the water filled portions of the RSS piping upstream of valves 3RSS*MV8837A/B and 3RSS*MV8838A/B. When deemed necessary by an external water level detection method, filling and venting to reestablish the acceptable water levels may be performed after entering LCO ACTION statement 3.6.2.2 since VENTING without isolation of the affected train would result in a breach of the containment pressure boundary.

The following ECCS subsections are exempt from this Surveillance:

- the operating centrifugal charging pump(s) and associated piping - as an operating pump is self VENTING and cannot develop voids and pockets of entrained gases.

- the RSS pumps, since this equipment is partially dewatered during plant operation.

- the RSS heat exchangers, since this equipment is laid-up dry during plant operation.

- the RSS piping that is not maintained filled with water during plant operation.

Surveillance Requirement 4.5.2.C.2 requires that the visual inspection of the containment be performed at least once daily if the containment has been entered that day and when the final containment entry is made. This will reduce the number of unnecessary inspections and also reduce personnel exposure.

The Emergency Core Cooling System (ECCS) has several piping cross connection points for use during the post-LOCA recirculation phase of operation. These cross-connection points allow the Recirculation Spray System (RSS) to supply water from the containment sump to the safety injection and charging pumps. The RSS has the capability to supply both Train A and B safety injection pumps and both Train A and B charging pumps. Operator action is required to position valves to establish flow from the containment sump through the RSS subsystems to the safety injection and charging pumps since the valves are not automatically repositioned. The quarterly stroke testing (Technical Specification 4.0.5) of the ECC/RSS recirculation flowpath valves discussed below will not result in subsystem inoperability (except due to other equipment manipulations to support valve testing) since these valves are manually aligned in accordance with the Emergency Operating Procedures (EOPs) to establish the recirculation flowpaths. It is expected the valves will be returned to the normal pre-test position following termination of the surveillance testing in response to the accident. Failure to restore any valve to the normal pre-test position will be indicated to the Control Room

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ECCS SUBSYSTEMS (Continued)

Operators when the ESF status panels are checked, as directed by the EOPs. The EOPs direct the Control Room Operators to check the ESF status panels early in the event to ensure proper equipment alignment. Sufficient time before the recirculation flowpath is required is expected to be available for operator action to position any valves that have not been restored to the pretest position, including local manual valve operation. Even if the valves are not restored to the pre-test position, sufficient capability will remain to meet ECCS post-LOCA recirculation requirements. As a result, stroke testing of the ECCS recirculation valves discussed below will not result in a loss of system independence or redundancy, and both ECCS subsystems will remain OPERABLE.

When performing the quarterly stroke test of 3SIH*MV8923A, the control switch for safety injection pump 3SIH*PIA is placed in the pull-to-lock position to prevent an automatic pump start with the suction valve closed. With the control switch for 3SIH*PIA in pull-to-lock, the Train A ECCS subsystem is inoperable and Technical Specification 3.5.2, ACTION a., applies. This ACTION statement is sufficient to administratively control the plant configuration with the automatic start of 3SIH*PIA defeated to allow stroke testing of 3SIH*MV8923A. In addition, the EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, and an additional single failure does not occur (an acceptable assumption since the Technical Specification ACTION statement limits the plant configuration time such that no additional equipment failure need be postulated). During the injection phase the redundant subsystem (Train B) is fully functional, as is a significant portion of the Train A subsystem. During the recirculation phase, the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps, and the Train B RSS subsystem can supply water from the containment sump to the B safety injection pump.

When performing the quarterly stroke test of 3SIH*MV8923B, the control switch for safety injection pump 3SIH*PIB is placed in the pull-to-lock position to prevent an automatic pump start with the suction valve closed. With the control switch for 3SIH*PIB in pull-to-lock, the Train B ECCS subsystem is inoperable and Technical Specification 3.5.2, ACTION a., applies. This ACTION statement is sufficient to administratively control the plant configuration with the automatic start of 3SIH*PIB defeated to allow stroke testing of 3SIH*MV8923B. In addition, the EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, and an additional single failure does not occur (an acceptable assumption since the Technical Specification ACTION statement limits the plant configuration time such that no additional equipment failure need be postulated). During the injection

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ECCS SUBSYSTEMS (Continued)

phase the redundant subsystem (Train A) is fully functional, as is a significant portion of the Train B subsystem. During the recirculation phase, the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps and the Train A safety injection pump. The Train B RSS subsystem cannot supply water from the containment sump to any of the remaining pumps.

When performing the quarterly stroke test of 3SIH*MV8807A or 3SIH*MV8807B, 3SIH*MV8924 is closed first to prevent the potential injection of RWST water into the RCS through the operating charging pump. When 3SIH*MV8924 is closed, it is not necessary to declare either ECCS subsystem inoperable. Although expected to be open for post-LOCA recirculation, sufficient time is expected to be available post-LOCA to identify and open 3SIH*MV8924 either from the Control Room or locally at valve. The EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, even if a single failure is postulated. The failure to open 3SIH*MV8924 due to mechanical binding or the loss of power to ECCS Train A could be the single failure. If a different single failure is postulated, restoration of 3SIH*MV8924 can be accomplished. The closure of 3SIH*MV8924 has no affect on the injection phase. During the recirculation phase, assuming 3SIH*MV8924 remains closed (i.e., the single failure), the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps, and the Train B RSS subsystem can supply water from the containment sump to the Train A and B safety injection pumps. If power is lost to ECCS Train A and 3SIH*MV8924 is not opened locally (i.e., the single failure), cold leg recirculation can be accomplished by using RSS Train B to supply containment sump water via 3SIH*P1B to the RCS cold legs and 3SIL*MV8809B can be opened to supply containment sump water via RSS Train B to the RCS cold legs. Hot leg recirculation can be accomplished by using RSS Train B to supply containment sump water via 3SIH*P1B to the RCS hot legs and maintaining 3SIL*MV8809B open to supply containment sump water via RSS Train B to the RCS cold legs.

ECCS Subsystems: Auxiliary Building RPCCW Ventilation Area Temperature Maintenance:

In MODES 1, 2, 3 and 4, two trains of 4 heaters each, powered from class 1E power supplies, are required to support charging pump OPERABILITY during cold weather conditions. These heaters are required whenever outside temperature is less than or equal to 17°F.

When outside air temperature is below 17°F, if both trains of heaters in the RPCCW Ventilation Area are available to maintain at least 65°F in the Charging Pump and Reactor Component Cooling Water Pump areas of the Auxiliary Building, both charging pumps are OPERABLE for MODES 1, 2 and 3.

When outside air temperature is below 17°F, if one train of heaters in the RPCCW Ventilation Area is available to maintain at least 32°F in the Charging Pump and Reactor Component Cooling Water Pump areas of the Auxiliary Building, the operating charging pump is OPERABLE, for MODE 4.

With less than 4 OPERABLE heaters in either train, the corresponding train of charging is inoperable. This condition will require entry into the applicable ACTION statement for LCOs 3.5.2 and 3.5.3.

LCO 3.5.2 ACTION statement "b", and LCO 3.5.3 ACTION statement "c" address special reporting requirements in response to ECCS actuation with water injection to the RCS. The special report completion is not a requirement for logging out of the ACTION statements that require the reports.

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, and (2) the reactor will remain subcritical in the cold condition following a large break (LB) LOCA, assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

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 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum/minimum solution temperatures for the RWST in MODES 1, 2, 3 and 4 are based on analysis assumptions.

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BACKGROUND

Trisodium phosphate (TSP) dodecahydrate is stored in porous wire mesh baskets on the floor or in the sump of the containment building to ensure that iodine, which may be dissolved in the recirculated reactor cooling water following a loss of coolant accident (LOCA), remains in solution. TSP also helps inhibit stress corrosion cracking (SCC) of austenitic stainless steel components in containment during the recirculation phase following an accident.

Fuel that is damaged during a LOCA will release iodine in several chemical forms to the reactor coolant and to the containment atmosphere. A portion of the iodine in the containment atmosphere is washed to the sump by containment sprays (i.e., Quench Spray and/or Containment Recirculation Spray). The emergency core cooling water is borated for reactivity control. This borated water causes the sump solution to be acidic. In a low pH (acidic) solution, dissolved iodine will be converted to a volatile form. The volatile iodine will evolve out of solution into the containment atmosphere, significantly increasing the levels of airborne iodine. The increased levels of airborne iodine in containment contribute to the radiological releases and increase the consequences from the accident due to containment atmosphere leakage.

After a LOCA, the components of the core cooling and containment spray systems will be exposed to high temperature borated water. Prolonged exposure to the core cooling water combined with stresses imposed on the components can cause SCC. The SCC is a function of stress, oxygen and chloride concentrations, pH, temperature, and alloy composition of the components. High temperatures and low pH, which would be present after a LOCA, tend to promote SCC. This can lead to the failure of necessary safety systems or components.

Adjusting the pH of the recirculation solution to levels above 7.0 prevents a significant fraction of the dissolved iodine from converting to a volatile form. The higher pH thus decreases the level of airborne iodine in containment and reduces the radiological consequences from containment atmosphere leakage following a LOCA. Maintaining the solution pH $\geq$ 7.0 also reduces the occurrence of SCC of austenitic stainless steel components in containment. Reducing SCC reduces the probability of failure of components.

Granular TSP dodecahydrate is employed as a passive form of pH control for post LOCA containment spray and core cooling water. Baskets of TSP are placed on the floor or in the sump of the containment building to dissolve

BACKGROUND (continued)

from released reactor coolant water and containment sprays after a LOCA. Recirculation of the water for core cooling and containment sprays then provides mixing to achieve a uniform solution pH. The dodecahydrate form of TSP is used because of the high humidity in the containment building during normal operation. Since the TSP is hydrated, it is less likely to absorb large amounts of water from the humid atmosphere and will undergo less physical and chemical change than the anhydrous form of TSP.

APPLICABLE SAFETY ANALYSES

The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution based on the recirculation water pH being ≥ 7.0. The radionuclide releases from the containment atmosphere and the consequences of a LOCA would be increased if the pH of the recirculation water were not adjusted to 7.0 or above.

LIMITING CONDITION FOR OPERATION

The TSP is required to adjust the pH of the recirculation water to ≥ 7.0 after a LOCA. A pH ≥ 7.0 after a LOCA is necessary to prevent significant amounts of iodine released from fuel failures and dissolved in the recirculation water from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the release of radionuclides and the consequences of the accident. A pH ≥ 7.0 is also necessary to prevent SCC of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is based upon the extreme cases of water volume and pH possible in the containment sump after a large break LOCA. The minimum required volume is the volume of TSP that will achieve a sump solution pH of ≥ 7.0 when taking into consideration the maximum possible sump water volume and the minimum possible pH. The amount of TSP needed in the containment building is based on the mass of TSP required to achieve the desired pH. However, a required volume is specified, rather than mass, since it is not feasible to weigh the entire amount of TSP in containment. The minimum required volume is based on the manufactured density of TSP dodecahydrate. Since TSP can have a tendency to agglomerate from high humidity in the containment building, the density may increase and the volume decrease during normal plant operation. Due to possible agglomeration and increase in density, estimating the minimum volume of TSP in containment is conservative with respect to achieving a minimum required pH.

APPLICABILITY

In MODES 1, 2, 3, and 4, a design basis accident (DBA) could lead to a fission product release to containment that leaks to the secondary containment boundary. The large break LOCA, on which this system's design is based, is a full-power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and reactor coolant system pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequence of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the SLCRS is not required to be OPERABLE.

ACTIONS

If it is discovered that the TSP in the containment building sump is not within limits, action must be taken to restore the TSP to within limits. During plant operation, the containment sump is not accessible and corrections may not be possible.

The 7-day Completion Time is based on the low probability of a DBA occurring during this period. The Completion Time is adequate to restore the volume of TSP to within the technical specification limits.

If the TSP cannot be restored within limits within the 7-day Completion Time, the plant must be brought to a MODE in which the LCO does not apply. The specified Completion Times for reaching MODES 3 and 4 are those used throughout the technical specifications; they were chosen to allow reaching the specified conditions from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Surveillance Requirement 4.5.5

Periodic determination of the volume of TSP in containment must be performed due to the possibility of leaking valves and components in the containment building that could cause dissolution of the TSP during normal operation. A Frequency of once per 24 months is required to determine visually that a minimum of 974 cubic feet is contained in the TSP Storage Baskets. This requirement ensures that there is an adequate volume of TSP to adjust the pH of the post LOCA sump solution to a value $\geq$ 7.0.

The periodic verification is required every refueling outage, since access to the TSP baskets is only feasible during outages. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

3/4.6 CONTAINMENT SYSTEMS

BASES

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 SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

Primary CONTAINMENT INTEGRITY is required in MODES 1 through 4. This requires an OPERABLE containment automatic isolation valve system. In MODES 1, 2 and 3 this is satisfied by the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure. In MODE 4 the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure are not required to be OPERABLE. Automatic actuation of the containment isolation system in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. Since the manual actuation pushbuttons portion of the containment isolation system is required to be OPERABLE in MODE 4, the plant operators can use the manual pushbuttons to rapidly postion all automatic containment isolation valves to the required accident position. Therefore, the containment isolation actuation pushbuttons satisfy the requirement for an OPERABLE containment automatic isolation valve system in MODE 4.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates, as specified in the Containment Leakage Rate Testing Program, ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than 0.75 La during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The Limiting Condition for Operation defines the limitations on containment leakage. The leakage rates are verified by surveillance testing as specified in the Containment Leakage Rate Testing Program, in accordance with the requirements of Appendix J. Although the LCO specifies the leakage rates at accident pressure, Pa, it is not feasible to perform a test at such an exact value for pressure. Consequently, the surveillance testing is performed at a pressure greater than or equal to Pa to account for test instrument uncertainties and stabilization changes. This conservative test pressure ensures that the measured leakage rates

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3/4.6.1.2 CONTAINMENT LEAKAGE (continued)

are representative of those which would occur at accident pressure while meeting the intent of the LCO. This test methodology is in accordance with the Containment Leakage Rate Testing Program.

The surveillance testing for measuring leakage rates are in accordance with the Containment Leakage Rate Testing Program.

The enclosure building bypass leakage paths are listed in the "Technical Requirements Manual." The addition or deletion of the enclosure building bypass leakage paths shall be made in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operations Review Committee.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The ACTION requirements are modified by a Note that allows entry and exit to perform repairs on the affected air lock components. This means there may be a short time during which the containment boundary is not intact (e.g., during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

ACTION a. is only applicable when one air lock door is inoperable. With only one air lock door inoperable, the remaining OPERABLE air lock door must be verified closed within 1 hour. This ensures a leak tight containment barrier is maintained by use of the remaining OPERABLE air lock door. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, the remaining OPERABLE air lock door must be locked closed within 24 hours and then verified periodically to ensure an acceptable containment leakage boundary is maintained. Otherwise, a plant shutdown is required.

ACTION b. is only applicable when the air lock door interlock mechanism is inoperable. With only the air lock interlock mechanism inoperable, an OPERABLE air lock door must be verified closed within 1 hour. This ensures a leak tight containment barrier is maintained by use of an OPERABLE air lock door. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, an OPERABLE air lock door must be locked closed within 24 hours and then verified periodically to ensure an acceptable containment leakage boundary is maintained. Otherwise, a plant shutdown is required. In addition, entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock) is permitted.

ACTION c. is applicable when both air lock doors are inoperable, or the air lock is inoperable for any other reason excluding the door interlock mechanism. With both air lock doors inoperable or the air lock otherwise inoperable, an evaluation of the overall containment leakage rate per Specification 3.6.1.2

3/4.6.1.3 CONTAINMENT AIR LOCKS (continued)

shall be initiated immediately, and an air lock door must be verified closed within 1 hour. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in the air lock have failed a seal test or if overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per Specification 3.6.1.1) would be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. The 1 hour requirement is consistent with the requirements of Technical Specification 3.6.1.1 to restore CONTAINMENT INTEGRITY. In addition, the air lock and/or at least one air lock door must be restored to OPERABLE status within 24 hours or a plant shutdown is required.

Surveillance Requirement 4.6.1.3.a verifies leakage through the containment air lock is within the requirements specified in the Containment Leakage Rate Testing Program. The containment air lock leakage results are accounted for in the combined Type B and C containment leakage rate. Failure of an air lock door does not invalidate the previous satisfactory overall air lock leakage test because either air lock door is capable of providing a fission product barrier in the event of a design basis accident.

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 is performed in accordance with the Containment Leakage Rate Testing Program, which ensures that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests. While the leakage rate limitation is specified at accident pressure, P_a, the actual surveillance testing is performed by applying a pressure greater than or equal to P_a. This higher pressure accounts for test instrument uncertainties and test volume stabilization changes which occurs under actual test conditions.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does not exceed the design pressure of 60 psia during LOCA conditions. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature. The limits on the pressure and average air temperature are consistent with the assumptions of the safety analysis. The minimum total containment pressure of 10.6 psia is determined by summing the minimum permissible air partial pressure of 8.9 psia and the maximum expected vapor pressure of 1.7 psia (occurring at the maximum permissible containment initial temperature of 120°F).

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 60 psia in the event of a LOCA. A visual inspection, in accordance with the Containment Leakage Rate Testing Program, is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be locked closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The Type C testing frequency required by 4.6.1.2 is acceptable, provided that the resilient seats of these valves are replaced every other refueling outage.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH SPRAY SYSTEM and RECIRCULATION SPRAY SYSTEM

The OPERABILITY of the Containment Spray Systems ensures that containment depressurization and iodine removal will occur in the event of a LOCA. The pressure reduction, iodine removal capabilities and resultant containment leakage are consistent with the assumptions used in the safety analyses.

LCO 3.6.2.2

One Recirculation Spray System consists of:

- Two OPERABLE containment recirculation heat exchangers
- Two OPERABLE containment recirculation pumps

The Containment Recirculation Spray System (RSS) consists of two parallel redundant subsystems which feed two parallel 360 degree spray headers. Each subsystem consists of two pumps and two heat exchangers. Train A consists of 3RSS*P1A and 3RSS*P1C. Tain B consists of 3RSS*P1B and 3RSS*P1D.

Bases Change

The design of the Containment RSS is sufficiently independent so that an active failure in the recirculation spray mode, cold leg recirculation mode, or hot leg recirculation mode of the ECCS has no effect on its ability to perform its engineered safety function. In other words, the failure in one subsystem does not affect the capability of the other subsystem to perform its designated safety function of assuring adequate core cooling in the event of a design basis LOCA. As long as one subsystem is OPERABLE, with one pump capable of assuring core cooling and the other pump capable of removing heat from containment, the RSS system meets its design requirements.

The LCO 3.6.2.2. ACTION applies when any of the RSS pumps, heat exchangers, or associated components are declared inoperable. All four RSS pumps are required to be OPERABLE to meet the requirements of this LCO 3.6.2.2. During the injection phase of a Loss Of Coolant Accident all four RSS pumps would inject into containment to perform their containment heat removal function. The minimum requirement for the RSS to adequately perform this function is to have at least one subsystem available. Meeting the requirements of LCO 3.6.2.2. ensures the minimum RSS requirements are satisfied.

Bases Change

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 Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these 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. FSAR Table 6.2-65 lists all containment isolation valves. The addition or deletion of any containment isolation valve shall be made in accordance with Section 50.59 of 10CFR50 and approved by the committee(s) as described in the QAP Topical Report.

For the purposes of meeting this LCO, the safety function of the containment isolation valves is to shut within the time limits assumed in the accident analyses. As long as the valves can shut within the time limits assumed in the accident analyses, the valves are OPERABLE. Where the valve position indication does not affect the operation of the valve, the indication is not required for valve OPERABILITY under this LCO. Position indication for containment isolation valves is covered by Technical Specification 6.8.4.e., Accident Monitoring Instrumentation. Failed position indication on these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3. Maintaining the valves OPERABLE, when position indication fails, facilitates troubleshooting and correction of the failure, allowing the indication to be restored "as soon as practicable."

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration.

If the containment isolation valve on a closed system becomes inoperable, the remaining barrier is a closed system since a closed system is an acceptable alternative to an automatic valve. However, actions must still be taken to meet Technical Specification ACTION 3.6.3.d and the valve, not normally considered as a containment isolation valve, and closest to the containment wall should be put into the closed position. No leak testing of the alternate valve is necessary to satisfy the ACTION statement. Placing the manual valve in the closed position sufficiently deactivates the penetration for Technical Specification compliance.

Closed system isolation valves applicable to Technical Specification ACTION 3.6.3.d are included in FSAR Table 6.2-65, and are the isolation valves for those penetrations credited as General Design Criteria 57. The specified time (i.e., 72 hours) of Technical Specification ACTION 3.6.3.d is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3 and 4. In the event the affected penetration is isolated in accordance with 3.6.3.d, the affected penetration flow path must be verified to be isolated on a periodic basis, (Surveillance Requirement 4.6.1.1.a). This is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The frequency of once per 31 days in this surveillance for verifying that each affected penetration flow path is isolated is appropriate considering the valves are operated under administrative controls and the probability of their misalignment is low.

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For the purposes of meeting this LCO, neither the containment isolation valve, nor any alternate valve on a closed system have a leakage limit associated with valve OPERABILITY.

The opening of containment isolation valves on an intermittent basis under administrative controls includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The appropriate administrative controls, based on the above considerations, to allow containment isolation valves to be opened are contained in the procedures that will be used to operate the valves. Entries should be placed in the Shift Manager Log when these valves are opened or closed. However, it is not necessary to log into any Technical Specification ACTION Statement for these valves, provided the appropriate administrative controls have been established.

Opening a closed containment isolation valve bypasses a plant design feature that prevents the release of radioactivity outside the containment. Therefore, this should not be done frequently, and the time the valve is opened should be minimized. The determination of the appropriate administrative controls for containment isolation valves requires an evaluation of the expected environmental conditions. This evaluation must conclude environmental conditions will not preclude access to close the valve, and this action will prevent the release of radioactivity outside of containment through the respective penetration.

When the Residual Heat Removal (RHR) System is placed in service in the plant cooldown mode of operation, the RHR suction isolation remotely operated valves 3RHS*MV8701A and 3RHS*MV8701B, and/or 3RHS*MV8702A and 3RHS*MV8702B are opened. These valves are normally operated from the control room. They do not receive an automatic containment isolation closure signal, but are interlocked to prevent their opening if Reactor Coolant System (RCS) pressure is greater than approximately 412.5 psia. When any of these valves are opened, either one of the two required licensed (Reactor Operator) control room operators can be credited as the operator required for administrative control. It is not necessary to use a separate dedicated operator.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. Containment hydrogen concentration is also important in verifying the adequacy of mitigating actions. The requirement to perform a hydrogen sensor calibration at least every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two point calibration, utilizing gas containing approximately one percent hydrogen gas for one of the calibration points, and gas containing approximately four percent hydrogen gas for the other calibration point.

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3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

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 or the Mechanical Vacuum Pumps are capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Post-LOCA performance of the hydrogen recombiner blowers is based on a series of equations supplied by the blower manufacturer. These equations are also the basis of the acceptance criteria used in the surveillance procedure. The required performance was based on starting containment conditions before the LOCA of 10.59 psia (total pressure), 120°F and 100% relative humidiy.

The surveillance procedure shall use the following methods to verify acceptable blower flow rate:

1. Definitions and constants

   CFM = cubic feet per minute

   RPM = revolutions per minute

   Blower RPM = 3550

   Blower ft^3/revolution = .028 ft^3

   Standard CFM = gas volume converted to conditions of 68°F and 14.7 psia.

2. Measure and record the following information:

   Pcontainment--Average of 3LMS*P934, 935, 936, and 937 (psia)

   Pout--From 3HCS*PI1A or B (psia)

   Tc--Containment temperature (°F)

   Pin--Measure with a new inlet gauge or calculate from Equation 3a below (psia)

   scfm measured--See Procedure/Form 3613A.3-1

   ΔP_f--From Table 2 (psi)

   A--As found Slip Constant

   Accuracy--Instrument accuracy range from Table 1.

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

3. Calculate as found slip constant (A)

   a. Pin = Pcontainment - ΔP_f

   b.

$$A = \frac{3550 - \left(\left[\frac{scfm_{measured} - Accuracy}{0.028 \times 0.95}\right] \times \left[\frac{14.7}{Pin} \times \frac{Tc + 460}{528}\right]\right)}{\left(\left[\frac{Pout}{Pin} \times 14.7\right] - 14.7\right)^{1/2} \times \left(\frac{14.7}{Pin} \times \frac{Tc + 460}{528}\right)^{1/2}}$$

4. Calculate expected postaccident flow rate using A calculated in Step 3.

   a. Slip RPM

      $= A \times (4.937)^{1/2} \times 1.218$

   b. Actual Inlet CFM

      ACFM = .028 (3550 - Slip RPM)

   c. Standard CFM

      scfm = ACFM 0.725

   d. Postaccident scfm Minimum = scfm × 0.95

   e. Acceptance Flow Rate

Postaccident scfm minimum ≥ 41.52 scfm.

Table 1 Accuracy Range (Ref. 2)

scfm (measured)	Accuracy Range
30 to < 40	9.13 scfm
40 to < 50	6.98 scfm
50 to < 60	5.81 scfm
60 to < 90	5.17 scfm

Table 2 Inlet Piping Loss (Ref. 1)

scfm Measured (Unadjusted)	ΔP_f (psi)
30	.21
40	.31
50	.52
60	.73
70	.98
80	1.28

Bases Change of 8-25-2005

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

References:
1. Calculation 90-RPS-722GM, "Flow Acceptance Criteria for 3HCS*RBNR 1A/B Blowers 3HCS*C1A/B."
2. Calculation PA 90-LOE-0132GE, "Hydrogen Recombiner Flow Error Analysis."

The acceptance flow rate is the required flow rate at the worst case containment conditions 24 hours after the LOCA. The analysis assumes the recombiners are started no later than 24 hours after the accident. The 18-month surveillance shall verify the gas temperature and blower flow rate concurrently.

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

3/4.6.5.1 STEAM JET AIR EJECTOR

The closure of the isolation valves in the suction of the steam jet air ejector ensures that: (1) the containment internal pressure may be maintained within its operation limits by the mechanical vacuum pumps, and (2) the containment atmosphere is isolated from the outside environment in the event of a LOCA. These valves are required to be closed for containment isolation.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

Background

The OPERABILITY of the Supplementary Leak Collection and Release System (SLCRS) ensures that radioactive materials that leak from the primary containment into the Secondary Containment following a Design Basis Accident (DBA) are filtered out and adsorbed prior to any release to the environment.

SLCRS Ductwork Integrity:

The Supplementary Leak Collection and Release System (SLCRS) remains OPERABLE with the following bolting configuration:

a. For 3HVR*DMPF44:

- Eight bolts properly installed on the ductwork access panels.
- At least one bolt must be installed in each corner area.
- The remaining bolts should be installed in the center area of each side.

b. For 3HVR*DMPF29:

- 12 bolts properly installed on the ductwork access panel.
- At least one bolt must be installed in each corner area.
- The remaining bolts should be approximately equally spaced along each side with two bolts per side.

With the above bolting specified for 3HVR*DMPF44 and 3HVR*DMPF29, reference (1) concluded the following:

- Any leakage around the plates is minimal and causes negligible effect on the performance of the SLCRS system.
- Assures the gasket will not be extruded from between the plate and duct flange when the SLCRS fans are started.
- The remaining bolts may be installed with the fans running.
- Provides adequate structural integrity in the seismic event based on engineering analysis.

Applicable Safety Analyses

The SLCRS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis assumes that only one train of the SLCRS and one train of the auxiliary building filter system is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction of the airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from the containment is determined for a LOCA.

The SLCRS is not normally in operation. The SLCRS starts on a SIS signal. The modeled SLCRS actuation in the safety analysis (the Millstone 3

Bases Change

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

FSAR Chapter 15, Section 15.6) is based upon a worst-case response time following an SI initiated at the limiting setpoint. One train of the SLCRS in conjunction with the Auxiliary Building Filter (ABF) system is capable of drawing a negative pressure (0.4 inches water gauge at the auxiliary building 24'6" elevation) within 120 seconds after a LOCA. This time includes diesel generator startup and sequencing time, system startup time, and time for the system to attain the required negative pressure after starting.

LCO

In the event of a DBA, one SLCRS is required to provide the minimum postulated iodine removal assumed in the safety analysis. Two trains of the SLCRS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single-active failure. The SLCRS works in conjunction with the ABF system. Inoperability of one train of the ABF system also results in inoperability of the corresponding train of the SLCRS. Therefore, whenever LCO 3.7.9 is entered due to the ABF train A (B) being inoperable, LCO 3.6.6.1 must be entered due to the SLCRS train A (B) being inoperable.

When a SLCRS LCO is not met, it is not necessary to declare the secondary containment inoperable. However, in this event, it is necessary to determine that a loss of safety function does not exist. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed.

Applicability

In MODES 1, 2, 3, and 4, a DBA could lead to a fission product release to containment that leaks to the secondary containment. The large break LOCA, on which this system's design is based, is a full-power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and reactor coolant system pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the SLCRS is not required to be OPERABLE.

ACTIONS

With one SLCRS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The OPERABLE train is capable of providing 100 percent of the iodine removal needs for a DBA. The 7-day Completion Time is based on consideration of such factors as the reliability of the OPERABLE redundant SLCRS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs. If the SLCRS cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and MODE 5 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full-power conditions in an orderly manner and without challenging plant systems.

Bases Change of 8-25-2005

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

Surveillance Requirements

a

Cumulative operation of the SLCRS with 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 31-day frequency was developed in consideration of the known reliability of fan motors and controls. This test is performed on a STAGGERED TEST BASIS once per 31-days.

b, c, e, and f

These surveillances verify that the required SLCRS filter testing is performed in accordance with Regulatory Guide 1.52, Revision 2. ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The surveillances include testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system."

This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits, as well as providing trend data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and the sample canisters are typically removed due to the 18 month criteria.

d

The automatic startup ensures that each SLCRS train responds properly. The once per 24 months frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance was performed with the reactor at power. The surveillance verifies that the SLCRS starts on a SIS test signal. It also includes the automatic functions to isolate the other ventilation systems that are not part of the safety-related postaccident operating configuration and to start up and to align the ventilation systems

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

that flow through the secondary containment to the accident condition.

- The main steam valve building ventilation system isolates.
- Auxiliary building ventilation (normal) system isolates.
- Charging pump/reactor plant component cooling water pump area cooling subsystem aligns and discharges to the auxiliary building filters and a filter fan starts.
- Hydrogen recombiner ventilation system aligns to the postaccident configuration.
- The engineered safety features building ventilation system aligns to the postaccident configuration.

References:

1. Engineering analysis, Memo MP3-DE-94-539, "Bolting Requirements for Access Panels on Dampers 3HVR*DMPF29 & 44, dated June 16, 1994.

Bases Change

3/4.6.6.2 SECONDARY CONTAINMENT

The Secondary Containment is comprised of the containment enclosure building and all contiguous buildings (main steam valve building [partially], engineering safety features building [partially], hydrogen recombiner building [partially], and auxiliary building). The Secondary Containment shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

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

Secondary Containment ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Supplementary Leak Collection and Release System, and Auxiliary Building Filter System will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

The SLCRS and the ABF fans and filtration units are located in the auxiliary building. The SLCRS is described in the Millstone Unit No. 3 FSAR, Section 6.2.3.

In order to ensure a negative pressure in all areas within the Secondary Containment under most meteorological conditions, the negative pressure acceptance criterion at the measured location (i.e., 24'6" elevation in the auxiliary building) is 0.4 inches water gauge.

LCO

The Secondary Containment OPERABILITY must be maintained to ensure proper operation of the SLCRS and the auxiliary building filter system and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

Applicability

Maintaining Secondary Containment OPERABILITY prevents leakage of radioactive material from the Secondary Containment. Radioactive material may enter the Secondary Containment from the containment following a LOCA. Therefore, Secondary Containment is required in MODES 1, 2, 3, and 4 when a design basis accident such as a LOCA could release radioactive material to the containment atmosphere.

3/4.6.6.2 SECONDARY CONTAINMENT (continued)

In MODES 5 and 6, the probability and consequences of a DBA are low due to the RCS temperature and pressure limitation in these MODES. Therefore, Secondary Containment is not required in MODES 5 and 6.

ACTIONS

In the event Secondary Containment OPERABILITY is not maintained, Secondary Containment OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a DBA occurring during this time period. Therefore, it is considered that there exists no loss of safety function while in the ACTION Statement.

Inoperability of the Secondary Containment does not make the SLCRS fans and filters inoperable. Therefore, while in this ACTION Statement solely due to inoperability of the Secondary Containment, the conditions and required ACTIONS associated with Specification 3.6.6.1 (i.e., Supplementary Leak Collection and Release System) are not required to be entered. If the Secondary Containment OPERABILITY cannot be restored to OPERABLE status within the required completion time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full-power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

4.6.6.2.1

Maintaining Secondary Containment OPERABILITY requires maintaining each door in each access opening in a closed position except when the access opening is being used for normal entry and exit. The normal time allowed for passage of equipment and personnel through each access opening at a time is defined as no more than 5 minutes. The access opening shall not be blocked open. During this time, it is not considered necessary to enter the ACTION statement. A 5-minute time is considered acceptable since the access opening can be quickly closed without special provisions and the probability of occurrence of a DBA concurrent with equipment and/or personnel transit time of 5 minutes is low.

The 31-day frequency for this surveillance is based on engineering judgment and is considered adequate in view of the other indications of access opening status that are available to the operator.

3/4.6.6.2 SECONDARY CONTAINMENT (continued)

4.6.6.2.2

The ability of a SLCRS to produce the required negative pressure during the test operation within the required time provides assurance that the Secondary Containment is adequately sealed.

With the SLCRS in postaccident configuration, the required negative pressure in the Secondary Containment is achieved in 110 seconds from the time of simulated emergency diesel generator breaker closure. Time delays of dampers and logic delays must be accounted for in this surveillance. The time to achieve the required negative pressure is 120 seconds, with a loss-of-offsite power coincident with a SIS. The surveillance verifies that one train of SLCRS in conjunction with the ABF system will produce a negative pressure of 0.4 inches water gauge at the auxiliary building 24'6" elevation relative to the outside atmosphere in the Secondary Containment. For the purpose of this surveillance, pressure measurements will be made at the 24'6" elevation in the auxiliary building. This single location is considered to be adequate and representative of the entire Secondary Containment due to the large cross-section of the air passages which interconnect the various buildings within the Secondary Containment. In order to ensure a negative pressure in all areas inside the Secondary Containment under most meteorological conditions, the negative pressure acceptance criterion at the measured location is 0.4 inch water gauge. It is recognized that there will be an occasional meteorological condition under which slightly positive pressure may exist at some localized portions of the boundary (e.g., the upper elevations on the down-wind side of a building). For example, a very low outside temperature combined with a moderate wind speed could cause a slightly positive pressure at the upper elevations of the containment enclosure building on the leeward face. The probability of occurrence of meteorological conditions which could result in such a positive differential pressure condition in the upper levels of the enclosure building has been estimated to be less than 2% of the time.

The probability of wind speed within the necessary moderate band, combined with the probability of extreme low temperature, combined with the small portion of the boundary affected, combined with the low probability of airborne radioactive material migrating to the upper levels ensures that the overall effect on the design basis dose calculations is insignificant.

The SLCRS system and fan sizing was based on an estimated infiltration rate. The fan flow rates are verified within a minimum and maximum on a monthly basis. Initial testing verified that the drawdown criterion was met at the lowest acceptable flow rate. The new standard Technical Specification (NUREG-1431) 3.6.6.2 surveillance requirement requires that the drawdown

3/4.6.6.2 SECONDARY CONTAINMENT (continued)

criterion be met while not exceeding a maximum flow rate. It is assumed that the purpose of this flow limit is to ensure that adequate attention is given to maintain the SLCRS boundary integrity and not using excess system capacity to cover for boundary degradation.

The SLCRS system was designed with minimal margin and, therefore, does not have excess capacity that can be substituted for boundary integrity. Additionally, since SLCRS fan flow rates are verified to be acceptable on a more frequent basis than the drawdown test surveillance, and by means of previous testing the minimum flow rate is acceptable, verifying a flow rate during the drawdown test would not provide an added benefit. Historical SLCRS flow measurements show a lack of repeatability associated with the inaccuracies of air flow measurement. As a result, the more reliable verification of system performance is the actual negative pressure generated by the drawdown test and a measured flow rate would add little.

3/4.6.6.3 SECONDARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the Secondary Containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide a secondary boundary surrounding the primary containment that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

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, 1971 Edition. The design minimum total relieving capacity for all valves on all of the steam lines is 1.579×10^7 lbs/h which is 105% of the total secondary steam flow of 1.504×10^7 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

The OPERABILITY of the main steam Code safety valves is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The lift settings for the main steam Code safety valves are listed in Table 3.7-3. This table allows a ± 3% setpoint tolerance (allowable value) on the lift setting for OPERABILITY to account for drift over an operating cycle.

Each main steam Code safety valve is demonstrated OPERABLE with lift settings as shown in Table 3.7-3, in accordance with Technical Specification 4.0.5. During this testing, the main steam Code safety valves are OPERABLE provided the actual lift settings are within ± 3% of the required lift setting. A footnote to Table 3.7-3 requires that the lift setting be restored to within ± 1% of the required lift setting following testing to allow drift during the next operating cycle. However, if the testing is done at the end of the operating cycle when the plant is being shut down for refueling, restoration to ± 1% of the specified lift setting is not required for valves that will not be used (e.g., replaced) for the next operating cycle. While the lift settings are being restored to within ± 1% of the required lift setting, the main steam Code safety valves remain OPERABLE provided the actual lift setting is within ± 3% of the required lift setting.

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:

$$Hi\ \phi = (100/Q)\,\frac{(w_s h_{fg} N)}{K}$$

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES (Continued)

where:

$Hi\phi$ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$

h_{fg} = heat of vaporization for steam at the highest MSSV opening pressure including tolerance (± 3%) and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

SAFETY VALVES (Continued)

w_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the OPERABLE MSSVs at the highest OPERABLE MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the OPERABLE MSSVs at the highest OPERABLE MSSV operating pressure, excluding the three highest capacity MSSVs. The following plant specific safety valve flow rates were used:

SG Safety Valve Number (Bank No.)	Main Steam System	
	Set Pressure (psia)	Flow (lbm/hr per loop)
1	1200	893,160
2	1210	900,607
3	1220	908,055
4	1230	915,502
5	1240	922,950

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater (AFW) System ensures a makeup water supply to the steam generators (SGs) to support decay heat removal from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply, assuming the worst case single failure. The AFW System consists of two motor driven AFW pumps and one steam turbine driven AFW pump. Each motor driven AFW pump provides at least 50% of the AFW flow capacity assumed in the accident analysis. After reactor shutdown, decay heat eventually decreases so that one motor driven AFW pump can provide sufficient SG makeup flow. The steam driven AFW pump has a rated capacity approximately double that of a motor driven AFW pump and is thus defined as a 100% capacity pump.

Given the worst case single failure, the AFW System is designed to mitigate the consequences of numerous design basis accidents, including Feedwater Line Break, Loss of Normal Feedwater, Steam Generator Tube Rupture, Main Steam Line Break, and Small Break Loss of Coolant Accident.

Basis Change of 8-25-2005

AUXILIARY FEEDWATER SYSTEM (Continued)

In addition, given the worst case failure, the AFW is designed to supply sufficient makeup water to replace SG inventory loss as the RCS is cooled to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

Surveillance Requirement 4.7.1.2.1 verifies that each AFW pump's total head at a recirculation flow test point is greater than or equal to the required total head. This surveillance ensures that the AFW pump performance has not degraded during the operating cycle. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed with recirculation flow. This test confirms one point on the pump curve and is indicative of overall performance. This test confirms component OPERABILITY is used to trend performance and to detect incipient failures by indicating abnormal performance. The total head specified in Surveillance Requirement 4.7.1.2.1 does not include a margin for test measurement uncertainty. This consideration shall be addressed at the implementing procedure level.

Motor driven auxiliary feedwater pumps and associated flow paths are OPERABLE in the following alignment during normal operation below 10% RATED THERMAL POWER.

- Motor operated isolation valves (3FWA*MOV35A/B/C/D) are open in MODE 1, 2 and 3,
- Control valves (3FWA*HV31A/B/C/D) may be throttled or closed during alignment, operation and restoration of the associated motor driven AFW pump for steam generator inventory control.

The motor operated isolation valves must remain fully open due to single failure criteria (the valves and associated pump are powered from the opposite electrical trains).

The Turbine Driven Auxiliary Feedwater (TDAFW) pump and associated flow paths are OPERABLE with all control and isolation valves fully open in MODE 1, 2 and 3. Due to High Energy Line Break analysis, the TDAFW pump cannot be used for steam generator inventory control during normal operation below 10% RATED THERMAL POWER.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK

The OPERABILITY of the demineralized water storage tank (DWST) with a 334,000 gallon minimum measured water volume ensures that sufficient water is available to maintain the reactor coolant system at HOT STANDBY conditions for 10 hours with steam discharge to the atmosphere, concurrent with a total loss-of-offsite power, and with an additional 6-hour cooldown period to reduce reactor coolant temperature to 350°F. The 334,000 gallon required water volume contains an allowance for tank inventory not usable because of tank discharge line location, other tank physical characteristics, and surveillance measurement uncertainty considerations. The inventory requirement is conservatively based on 120°F water temperature which maximizes inventory required to remove RCS decay heat. In the event of a feedline break, this inventory requirement includes an allowance for 30 minutes of spillage before operator action is credited to isolate flow to the line break.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK (Continued)

If the combined condensate storage tank (CST) and DWST inventory is being credited, there are 50,000 gallons of unusable CST inventory due to tank discharge line location, other physical characteristics, level measurement uncertainty and potential measurement bias error due to the CST nitrogen blanket. To obtain the Surveillance Requirement 4.7.1.3.2's DWST and CST combined volume, this 50,000 gallons of unusable CST inventory has been added to the 334,000 gallon DWST water volume specified in LCO 3.7.1.3 resulting in a 384,000 gallons requirement (334,000 + 50,000 = 384,000 gallons).

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 Part 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

BACKGROUND

The main steam line isolation valves (MSIVs) isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by low steam generator pressure, high containment pressure, or steam line pressure negative rate (high). The MSIVs fail closed on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.3.

APPLICABLE SAFETY ANALYSIS

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6.2. It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.1.5. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting temperature case for the containment analysis is the SLB inside containment, at 75% power with mass and energy releases based on offsite power available following turbine trip, and failure of the MSIV on the affected steam generator to close.

At hot zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The reactor is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

(Bases Change)

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES (continued)

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIVs is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB upstream of the MSIV at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during POWER OPERATION. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

e. The MSIVs are also utilized during other events, such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

Bases Change of 8-25-2005

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES (continued)

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10CFR100 limits or the NRC Staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODES 2, 3, and 4 except when closed and deactivated when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODES 1, 2, and 3 the MSIVs are required to close within 10 seconds to ensure the accident analysis assumptions are met. In MODE 4 the MSIVs are required to close within 120 seconds to ensure the accident analysis assumptions are met. An engineering evaluation has determined that a Reactor Coolant System (RCS) temperature greater than or equal to 320°F is required to provide sufficient steam energy to provide the motive force to operate the MSIVs. Therefore, below an RCS temperature of 320°F the MSIVs are not OPERABLE and are required to be closed.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

MODE 1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides a passive barrier for containment isolation.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES (continued)

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging plant systems.

MODES 2, 3, and 4

Since the MSIVs are required to be OPERABLE in MODES 2, 3, and 4, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis. The MSIVs may be opened to perform Surveillance Requirement 4.7.1.5.2.

The 8 hour Completion Time is consistent with that allowed in MODE 1.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day verification time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems. The Action Statement is modified by a note indicating that separate condition entry is allowed for each MSIV.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 DELETED

SURVEILLANCE REQUIREMENTS (continued)

4.7.1.5.2 This surveillance demonstrates that MSIV closure time is less than 10 seconds (120 seconds for MODE 4 only) on an actual or simulated actuation signal, when tested pursuant to Specification 4.0.5. A simulated signal is defined as any of the following engineering safety features actuation system instrumentation functional units per Technical Specifications Table 4.3-2: 4.a.1) manual initiation, individual, 4.a.2) manual initiation system, 4.c. containment pressure high-2, 4.d. steam line pressure low, or 4.e. steam line pressure - negative rate high. The MSIV closure time is assumed in the accident analyses. This surveillance is normally performed upon returning the plant to operation following a refueling outage. The test is normally conducted in MODES 3 or 4 with the plant at suitable (appropriate) conditions (e.g., pressure and temperature). The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of valve closure when the unit is generating power.

This surveillance requirement is modified by an exception that will allow entry into and operation in MODES 3 and 4 prior to performing the test to establish conditions consistent with those under which the acceptance criterion was generated. Successful performance of this test within the required frequency is necessary to operate in MODES 3 and 4 with the MSIVs open, to enter MODE 2 from MODE 3, and for plant operation in MODE 1. If this surveillance has not been successfully performed within the required frequency, the MSIVs are inoperable and are required to be closed.

In MODE 4 only, the MSIVs can be considered OPERABLE if the closure time is less than 120 seconds. An engineering evaluation has determined that a RCS temperature greater than or equal to 320°F is required to provide sufficient steam energy to provide the motive force to operate the MSIVs. Therefore, below an RCS temperature of 320°F the MSIVs are not OPERABLE and are required to be closed.

3/4.7.1.6 STEAM GENERATOR ATMOSPHERIC RELIEF BYPASS LINES

The OPERABILITY of the steam generator atmospheric relief bypass valve (SGARBV) lines provides a method to recover from a steam generator tube rupture (SGTR) event during which the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to limit the primary to secondary break flow into the ruptured steam generator. The time required to limit the primary to secondary break flow for an SGTR event is more critical than the time required to cooldown to RHR entry conditions. Because of these time constraints, these valves and associated flow paths must be OPERABLE from the control room. The number of SGARBVs required to be OPERABLE from the control room to satisfy the SGTR accident analysis requires consideration of single failure criteria. Four SGARBV are required to be OPERABLE to ensure the credited steam release pathways available to conduct a unit cooldown following a SGTR.

For other design events, the SGARBVs provide a safety grade method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the steam bypass system or the steam generator atmospheric relief valves be unavailable. Prior to operator action to cooldown, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below design limits.

Each SGARBV line consists of one SGARBV and an associated block valve (main steam atmospheric relief isolation valve, 3MSS*MOV18A/B/C/D). These block valves are used in the event a steam generator atmospheric relief valve (SGARV) or SGARBV fails to close. Because of the electrical power relationship between the SGARBV and the block valves, if a block valve is maintained closed, the SGARBV flow path is inoperable because of single failure consideration.

The bases for the required ACTIONS can be found in NUREG 1431, Rev. 1.

The LCO APPLICABILITY and ACTION statements uses the terms "MODE 4 when steam generator is relied upon for heat removal" and "in MODE 4 without reliance upon steam generator for heat removal." This means that those steam generators which are credited for decay heat removal to comply with LCO 3.4.1.3 (Reactor Coolant System, HOT SHUTDOWN) shall have an OPERABLE SGARBV line. See Bases Section 3/4.4.1 for more detail.

3/4.7.2 DELETED

3/4.7.3 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Reactor Plant 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.

The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System is required to be available to support reactor plant component cooling water pump operation. The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System consists of two redundant trains, each capable of providing 100% of the required flow. Each train has a two position, "Off" and "Auto," remote control switch. With the remote control switches for each train in the "Auto" position, the system is capable of automatically transferring operation to the redundant train in the event of a low flow condition in the operating train. The associated fans do not receive any safety related automatic start signals (e.g., Safety Injection Signal).

Placing the remote control switch for a Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train in the "Off" position to start the redundant train or to perform post maintenance testing to verify availability of the redundant train will not affect the availability of that train, provided appropriate administrative controls have been established to ensure the remote control switch is immediately returned to the "Auto" position after the completion of the specified activities or in response to plant conditions. These administrative controls include the use of an approved procedure and a designated individual at the control switch for the respective Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train who can rapidly respond to instructions from procedures, or control room personnel, based on plant conditions.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the 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.

An OPERABLE service water loop requires one OPERABLE service water pump and associated strainer. Two OPERABLE service water loops, with one OPERABLE service water pump and associated strainer per loop, will provide sufficient core (and containment) decay heat removal during a design basis accident coincident with a loss of offsite power and a single failure.

Bases Change of 8-25-2005

3/4.7.5 ULTIMATE HEAT SINK

BACKGROUND

The ultimate heat sink (UHS) for Millstone Unit No. 3 is Long Island Sound. It serves as a heat sink for both safety and nonsafety-related cooling systems. Sensible heat is discharged to the UHS via the service water and circulating water systems.

LIMITING CONDITION FOR OPERATION

The UHS is required to be OPERABLE and is considered OPERABLE if the average water temperature is less than or equal to 75°F. The limitation on the UHS temperature ensures that cooling water at or less than the design temperature (75°F) is available to either (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits. It 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," March 1974.

The Circulating Water System has six condenser inlet waterboxes, each contains a temperature measurement device. The average UHS temperature is normally obtained from the plant process computer by averaging the six Circulating Water System condenser inlet waterbox temperature measurements. Given potential condenser waterbox temperature instrumentation failure(s), or that a waterbox is not operating or a process computer failure, other methods may be used to determine the average UHS temperature. For example, if one condenser waterbox instrument has failed, the average UHS temperature may be based on five condenser inlet waterbox temperature measurements. For the purposes of determining average UHS temperature, if condenser waterbox inlet temperature is used, the average should be based on no less than 3 measurements. If the process computer condenser waterbox inlet temperature average is based on less than three measurements, the average is automatically flagged to users as potentially in error. Using local Service Water System temperature instruments (two or more) is an acceptable alternative for determining average UHS temperature.

It has been concluded that using the average of multiple condenser waterbox inlet temperature measurements is sufficiently representative of the UHS temperature to assure OPERABILITY of the UHS. The only exception to this conclusion is when a condenser thermal backwash evolution is being conducted. During this evolution, there is a potential for significant intake structure temperature stratification. Therefore, during condenser thermal backwashing evolutions, the average UHS temperature shall be monitored by temperature instruments in the service water system to assure OPERABILITY of the UHS.

APPLICABILITY

In MODES 1, 2, 3, AND 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

ACTION STATEMENT

When the UHS temperature is above 75°F, the ACTION Statement for the LCO requires that the UHS temperature be monitored for 12 hours, and the plant be placed in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours in the event the UHS temperature does not drop below 75°F during the 12-hour monitoring period.

The 12-hour interval is based on operating experience related to trending of the parameter variations during the applicable MODES. During this period, the UHS temperature will be monitored on an increased frequency. If the trend shows improvement, and if the trend of the UHS temperature gives reasonable expectations that the temperature will decrease below 75°F during the 12 hour monitoring period, the UHS temperature will be continued to be monitored during the remaining portion of the 12-hour period. However, if it becomes apparent that the UHS temperature will remain above 75°F throughout the 12-hour monitoring period, conservative action regarding compliance with the ACTION Statement should be taken.

An evaluation was conducted to qualify the risk significance of various Chapter 15 initiating events and earthquakes during periods of elevated UHS temperature. It concluded that a seismic event was not credible for the time periods with elevated UHS temperature.

With respect to the service water loads, the limiting Condition II and III Chapter 15 event initiators are those that add additional heat loads to the service water system. A loss of offsite power event is limiting because of the added loads due to the diesel generator and the residual heat removal heat exchanger. A steam generator tube rupture event is limiting because of the addition of the safety injection and diesel generator loads without isolation of the turbine plant component cooling water loads (no loss of offsite power or containment depressurization actuation signal). Although the risk significance of a Condition IV accident occurring during the period of elevated UHS temperature is considered to be negligibly small compared to that of Condition II and III events, a Loss of Coolant Accident with or without a LOP was also evaluated. These scenarios have been evaluated with the additional consideration of a single failure. The evaluation investigated whether or not these events could be resolved with an elevated UHS temperature. It was determined that Millstone Unit No. 3 could recover from these events, even with an elevated temperature of 77°F.

This evaluation provides the basis for the ACTION statement requirement to place the plant in HOT STANDBY within six hours and in COLD SHUTDOWN within the next 30 hours, if the UHS temperature goes above 77°F during the 12-hour monitoring period.

SURVEILLANCE REQUIREMENTS

For the surveillance requirements, the UHS temperature is measured at the locations described in the LCO write-up provided in this section.

Surveillance Requirement 4.7.5.a verifies that the UHS is capable of providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature. The 24-hour frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This surveillance requirement verifies that the average water temperature of the UHS is less than or equal to 75°F.

Surveillance Requirement 4.7.5.b requires that the UHS temperature be monitored on an increased frequency whenever the UHS temperature is greater than 70°F during the applicable MODES. The intent of this Surveillance Requirement is to increase the awareness of plant personnel regarding UHS temperature trends above 70°F. The frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

3/4.7.6 DELETED

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

BACKGROUND

The control room emergency ventilation system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. Additionally, the system provides temperature control for the control room during normal and post-accident operations.

The control room emergency ventilation system is comprised of the control room emergency air filtration system and a temperature control system.

The control room emergency air filtration system consists of two redundant systems that recirculate and filter the control room air. Each control room emergency air filtration system consists of a moisture separator, electric heater, prefilter, upstream high efficiency particulate air (HEPA) filter, charcoal adsorber, downstream HEPA filter, and fan. Additionally, ductwork, valves or dampers, and instrumentation form part of the system.

Normal Operation

A portion of the control room emergency ventilation system is required to operate during normal operations to ensure the temperature of the control room is maintained at or below 95°F.

Bases Change of 8-25-2005

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

BACKGROUND (Continued)

Post Accident Operation

The control room emergency ventilation system is required to operate during post-accident operations to ensure the temperature of the control room is maintained and to ensure the control room will remain habitable during and following accident conditions.

The following sequence of events occurs upon receipt of a control building isolation (CBI) signal or a signal indicating high radiation in the air supply duct to the control room envelope.

1. The control room boundary is isolated to prevent outside air from entering the control room to prevent the operators from being exposed to the radiological conditions that may exist outside the control room. The analysis for a loss of coolant accident assumes that the highest releases occur in the first hour after a loss of coolant accident.

2. After 60 seconds, the control room envelope pressurizes to 1/8 inch water gauge by the control room emergency pressurization system. This action provides a continuous PURGE of the control room envelope and prevents inleakage from the outside environment. Technical Specification 3/4.7.8 provides the requirements for the control room envelope pressurization system.

3. Control room pressurization continues for the first hour.

4. After one hour, the control room emergency ventilation system will be placed in service in the filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). To run the control room emergency air filtration system in the filtered pressurization mode, the air supply line must be manually opened.

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room. For all postulated design basis accidents except a Fuel Handling Accident, the radiation exposure to personnel occupying the control room shall be 5 rem or less whole body, or its equivalent for the duration of the accident, consistent with the requirements of General Design Criterion 19 of Appendix "A," 10 CFR 50. For a Fuel Handling Accident, the radiation exposure to personnel occupying the control room shall be 5 rem TEDE or less, consistent with the requirements of 10 CFR 50.67. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

Bases Change of 8-25-2005

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

LIMITING CONDITION FOR OPERATION

Two independent control room emergency air filtration systems are required to be OPERABLE to ensure that at least one is available in the event the other system is disabled.

A control room emergency air filtration system is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. moisture separator, heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The integrity of the control room habitability boundary (i.e., walls, floors, ceilings, ductwork, and access doors) must be maintained such that the control building habitability zone can be maintained at its design positive pressure if required to be aligned in the filtration pressurization mode. However, the LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

ACTIONS a., b., and c. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. ACTIONS d. and e. are applicable in MODES 5 and 6, and whenever fuel is being moved within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment or the storage pool.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS

MODES 1, 2, 3, and 4

a. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. In this condition, the remaining control room emergency air filtration system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room emergency air filtration system function. The 7-day completion time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

If the inoperable train cannot be restored to an OPERABLE status within 7 days, the unit must be placed in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems.

b. With both control room emergency air filtration systems inoperable, except due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room emergency air filtration system must be restored to OPERABLE status within 1 hour, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

c. With both control room emergency air filtration systems inoperable due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. The control room boundary must be restored to OPERABLE status within 24 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the control room emergency air filtration systems cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be

Basis Change of 8-25-2005

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS (Continued)

available to address these concerns for intentional and unintentional entry in to this condition. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour allowed outage time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the control room boundary.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool

d. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE control room emergency air filtration system in the recirculation mode or suspend the movement of fuel. Initiating and maintaining operation of the OPERABLE train in the recirculation mode ensures: (i) OPERABILITY of the train will not be compromised by a failure of the automatic actuation logic; and (ii) active failures will be readily detected.

e. With both control room emergency air filtration systems inoperable, or with the train required by ACTION 'd' not capable of being powered by an OPERABLE emergency power source, actions must be taken to suspend all operations involving the movement of fuel. This action places the unit in a condition that minimizes risk. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.7.a

The control room environment should be checked periodically to ensure that the control room temperature control system is functioning properly. Verifying that the control room air temperature is less than or equal to 95°F at least once per 12 hours is sufficient. It is not necessary to cycle the control room ventilation chillers. The control room is manned during operations covered by the technical specifications. Typically, temperature aberrations will be readily apparent.

4.7.7.b

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing the trains once every 31 days on a STAGGERED TEST BASIS provides an adequate check of this system. This surveillance requirement verifies a system flow rate of 1,120 cfm ± 20%. Additionally, the system is required to operate for at least 10 continuous hours with the heaters energized. These operations are sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters due to the humidity in the ambient air.

Bases Change of 8-25-2005

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.c

The performance of the control room emergency filtration systems should be checked periodically by verifying the HEPA filter efficiency, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The frequency is at least once per 24 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.

ANSI N510-1980 will be used as a procedural guide for surveillance testing.

4.7.7.c.1

This surveillance verifies that the system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 1,120 cfm ± 20%. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in the regulatory guide.

4.7.7.c.2

This surveillance requires that a representative carbon sample be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978 and that a laboratory analysis verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters. The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

4.7.7.c.3

This surveillance verifies that a system flow rate of 1,120 cfm ± 20%, during system operation when testing in accordance with ANSI N510-1980.

4.7.7.d

After 720 hours of charcoal adsorber operation, a representative carbon sample must be obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and a laboratory analysis must verify that the representative carbon sample meets the laboratory testing criteria of ASTM D3803-89 and Millstone Unit 3 specific parameters.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

The maximum surveillance interval is 900 hours, per Surveillance Requirement 4.0.2. The 720 hours of operation requirement originates from Nuclear Regulatory Guide 1.52, Table 2, Note C. This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data.

4.7.7.e.1

This surveillance verifies that the pressure drop across the combined HEPA filters and charcoal adsorbers banks at less than 6.75 inches water gauge when the system is operated at a flow rate of 1,120 cfm ± 20%. The frequency is at least once per 24 months.

4.7.7.e.2

This surveillance verifies that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch water gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas and outside atmosphere during positive pressure system operation. The frequency is at least once per 24 months.

The intent of this surveillance is to verify the ability of the control room emergency air filtration system to maintain a positive pressure while running in the filtered pressurization mode.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

During the first hour, the control room pressurization system creates and maintains the positive pressure in the control room. This capability is verified by Surveillance Requirement 4.7.8.C, independent of Surveillance Requirement 4.7.7.e.2. A CBI signal will automatically align an operating filtration system into the recirculation mode of operation due to the isolation of the air supply line to the filter.

After the first hour of an event with the potential for a radiological release, the control room emergency ventilation system will be aligned in the filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). Alignment to the filtered pressurization mode requires manual operator action to open the air supply line.

4.7.7.e.3

This surveillance verifies that the heaters can dissipate 9.4 ± 1 kW at 480V when tested in accordance with ANSI N510-1980. The frequency is at least once per 24 months. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

4.7.7.f

Following the complete or partial replacement of a HEPA filter bank, the OPERABILITY of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1,120 cfm ± 20%.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.g

Following the complete or partial replacement of a charcoal adsorber bank, the OPERABILITY of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfied the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow of 1,120 cfm ± 20%.

References:

(1) Nuclear Regulatory Guide 1.52, Revision 2

(2) MP3 UFSAR, Table 1.8-1, NRC Regulatory Guide 1.52

(3) NRC Generic Letter 91-04

(4) Condition Report (CR) #M3-99-0271

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

BACKGROUND

The control room envelope pressurization system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The control room envelope pressurization system consists of two banks of air bottles with its associated piping, instrumentation, and controls. Each bank is capable of providing the control room area with one-hour of air following any event with the potential for radioactive releases.

Control Room Envelope OPERABILITY is satisfied while:

- Door 352 (C-49-1) is closed (East door)
- Door 351 (C-47-1) is closed, but C-47-1A, ATD/Missile Shield, is not closed (West doors)

Normal Operation

During normal operations, the control room envelope pressurization system is required to be on standby.

Post Accident Operation

The control room envelope pressurization system is required to operate during post-accident operations to ensure the control room will remain habitable during and following accident conditions.

The sequence of events which occurs upon receipt of a control building isolation (CBI) signal or a signal indicating high radiation in the air supply duct to the control room envelope is described in Bases Section 3/4.7.7.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the control room envelope pressurization system ensures that: (1) breathable air is supplied to the control room, instrumentation rack room, and computer room, and (2) a positive pressure is created and maintained within the control room envelope during control building isolation for the first hour following any event with the potential for radioactive releases. Each system is capable of providing an adequate air supply to the control room for one hour following an initiation of a control building isolation signal. After one hour, operation of the control room emergency ventilation system would be initiated.

LIMITING CONDITION FOR OPERATION

Two independent control room envelope pressurization systems are required to be OPERABLE to ensure that at least one is available in the event the other system is disabled.

A control room envelope pressurization system is OPERABLE when the associated:

a. air storage bottles are OPERABLE; and

b. piping and valves are OPERABLE.

The integrity of the control room habitability boundary (i.e., walls, floors, ceilings, ductwork, and access doors) must be maintained. However, the LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

ACTIONS a., b., c., and d. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. ACTIONS e. and f. are applicable in MODES 5 and 6, and whenever fuel is being moved within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel that is impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment or the storage pool.

ACTIONS

MODES 1, 2, 3, and 4

a. With one control room envelope pressurization system inoperable, action must be taken either to restore the inoperable system to an OPERABLE status within 7 days, or place the unit in HOT STANDBY within six hours and COLD SHUTDOWN within the next 30 hours.

   The remaining control room envelope pressurization system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room envelope pressurization system. The 7-day completion time is based on the low probability of a design basis accident occurring during this time period and the ability of the remaining train to provide the required capability.

   The completion times for the unit to be placed in HOT STANDBY and COLD SHUTDOWN are reasonable. They are based on operating experience, and they permit the unit to be placed in the required conditions from full power conditions in an orderly manner and without challenging unit systems.

b. With both control room envelope pressurization systems inoperable, except due to an inoperable control room boundary or during performance of Surveillance Requirement 4.7.8.c, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room envelope pressurization system must be restored to OPERABLE status within 1 hour, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Basis Change of 8-25-2005

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

ACTIONS (Continued)

c. With both control room envelope pressurization systems inoperable due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. The control room boundary must be restored to OPERABLE status within 24 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the control room envelope pressurization systems cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry in to this condition. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour allowed outage time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the control room boundary.

d. With both control room envelope pressurization systems inoperable during the performance of Surveillance Requirement 4.7.8.c and the system not being tested under administrative control, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room envelope pressurization system must be restored to OPERABLE status within 4 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The administrative controls for the system not being tested consist of a dedicated operator, in constant communication with the control room, who can rapidly restore this system to OPERABLE status. Allowing both control room envelope pressurization systems to be inoperable for 4 hours under administrative control is acceptable since the system not being tested is inoperable only because it is isolated. Therefore, the system can be rapidly restored if needed. The other completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

ACTIONS (Continued)

MODES 5 and 6, and fuel movement within containment or the spent fuel pool

e. With one control room envelope pressurization system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, immediately suspend the movement of fuel. This action places the unit in a condition that minimizes potential radiological exposure to Control Room personnel. This action does not preclude the movement of fuel to a safe position.

   The remaining control room envelope pressurization system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room envelope pressurization system. The 7-day completion time is based on the low probability of a design basis accident occurring during this time period and the ability of the remaining train to provide the required capability.

   Stud tensioning may continue in MODE 6 and a MODE change to MODE 5 is permitted with a control room envelope pressurization system inoperable (Reference 1).

f. With both control room envelope pressurization systems inoperable, immediately suspend the movement of fuel. This action places the unit in a condition that minimizes potential radiological exposure to Control Room personnel. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.8.a

This surveillance requires verification that the air bottles are properly pressurized. Verifying that the air bottles are pressurized to greater than or equal to 2200 psig will ensure that a control room envelope pressurization system will be capable of supplying the required flow rate. The frequency of the surveillance is at least once per 7 days. It is based on engineering judgment and has been shown to be appropriate through operating experience.

4.7.8.b

This surveillance requires verification of the correct position of each valve (manual, power operated, or automatic) in the control room envelope pressurization system flow path. It helps ensure that the control room envelope pressurization system is capable of performing its intended safety function by verifying that an appropriate flow path will exist. The surveillance applies to those valves that could be mispositioned. This surveillance does not apply to valves that have been locked, sealed, or secured in position, because these positions are verified prior to locking, sealing, or securing.

The frequency of the surveillance is at least once per 31 days on a STAGGERED TEST BASIS. It is based on engineering judgment and has been shown to be appropriate through operating experience.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.c

The performance of the control room envelope pressurization system should be checked periodically. The frequency is at least once per 24 months and following any major alteration of the control room envelope pressure boundary.

A major alteration is a change to the control room envelope pressure boundary that: (1) results in a breach greater than analyzed for acceptable pressurization and requires nonroutine work evolutions to restore the boundary. A nonroutine work evolution is one which makes it difficult to determine As-Found and As-Left conditions. Examples of routine work evolution include: (1) opening and closing a door, and (2) repairing cable and pipe penetrations because the repairs are conducted in accordance with procedures and are verified via inspections. For these two examples, there is a high level of assurance that the boundary is restored to the As-Found condition.

This surveillance requires at least once per 24 months or following a major alteration of the control room envelope pressure boundary by:

- Verifying the control room envelope is isolated in response to a Control Building Isolation Test signal,
- Verifying, after a 60 second time delay following a Control Building Isolation Test signal, the control room envelope pressurizes to greater than or equal to 0.125 inch water gauge relative to adjacent areas and outside atmosphere; and
- Verifying the positive pressure of Technical Specification 4.7.8.c.2 is maintained for greater than or equal to 60 minutes.

Changes in conditions outside the control room envelope cause pressure spikes which are reflected on the differential pressure indicator, 3HVC-PDI 113.

Pressure spikes or fluctuations which result in the differential pressure momentarily dropped below the 0.125 inch water gauge acceptance criteria are acceptable providing the following conditions are met:

1. Differential pressure remains positive at all times.
2. Differential pressure is only transitorily below the acceptance criteria.
3. Differential pressure returns to a value above the acceptance criteria.

3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

The control room envelope pressurization system design basis criteria is set at ≥ 0.125 inch water gauge criteria to account for wind effects, thermal column effects, and barometric pressure changes. Pressurizing the control room envelope of 0.125 inch water gauge above the initial atmospheric pressure ensures it will remain at a positive pressure during subsequent changes in outside conditions over the next 60 minutes. Since the surveillance requirement is verified by actual reference to outside pressure, allowances are provided for differential pressure fluctuations caused by external forces. The 0.125 inch water gauge acceptance criteria provides the margin for these fluctuations. This meets the requirements of Regulatory Guide 1.78 and NUREG-800, Section 6.4 and is consistent with the assumptions in the Control Room Operator DBA dose calculation.

4.7.8.c.1

This surveillance verifies that the control room envelope is isolated following a control building isolation (CBI) test signal.

4.7.8.c.2

This surveillance verifies that the control room envelope pressurizes to greater than or equal to 1/8 inch water gauge, relative to the outside atmosphere, after 60 seconds following receipt of a CBI test signal.

4.7.8.c.3

This surveillance verifies that the positive pressure developed in accordance with Surveillance Requirement 4.7.8.c.2 is maintained for greater than or equal to 60 minutes. This capability is independent from the requirements regarding the control room emergency filtration system contained in Technical Specification 3/4.7.7. Also, following the first hour, the control room emergency ventilation system is responsible for ensuring that the control room envelope remains habitable.

References:

(1) NRC Routine Inspection Report 50-423/87-33, dated February 10, 1988.

(2) NRC Generic Letter 91-04.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. The charging pump/reactor plant component cooling water pump ventilation system must be operational to ensure operability of the auxiliary building filter system and the supplementary leak collection and release system. 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 operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

LCO 3.7.9 Action Statement:

With one Auxiliary Building Filter System inoperable, restoration to OPERABLE status within 7 days is required.

The 7 days restoration time requirement is based on the following: The risk contribution is less for an inoperable Auxiliary Building Filter System, than for the charging pump or reactor plant component cooling water (RPCCW) systems, which have a 72 hour restoration time requirement. The Auxiliary Building Filter System is not a direct support system for the charging pumps or RPCCW pumps. Because the pump area is a common area, and as long as the other train of the Auxiliary Building Filter System remains OPERABLE, the 7 day restoration time limit is acceptable based on the low probability of a DBA occurring during the time period and the ability of the remaining train to provide the required capability. A concurrent failure of both trains would require entry into LCO 3.0.3 due to the loss of functional capability. The Auxiliary Building Filter System does support the Supplementary Leak Collection and Release System (SLCRS) and the LCO Action statement time of 7 days is consistent with that specified for SLCRS (See LCO 3.6.6.1).

Surveillance Requirement 4.7.9.c

Surveillance requirement 4.7.9.c requires that after 720 hours of operation a charcoal sample must be taken and the sample must be analyzed within 31 days after removal.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system." This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing

Bases Change

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System, and associated filters and fans, ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. 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 operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System is required to be available to support the Auxiliary Building Filter System and the Supplementary Leak Collection and Release System (SLCRS). The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System consists of two redundant trains, each capable of providing 100% of the required flow. Each train has a two position, "Off" and "Auto," remote control switch. With the remote control switches for each train in the "Auto" position, the system is capable of automatically transferring operation to the redundant train in the event of a low flow condition in the operating train. The associated fans do not receive any safety related automatic start signals (e.g. Safety Injection Signal).

Placing the remote control switch for a Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train in the "Off" position to start the redundant train or to perform post maintenance testing to verify availability of the redundant train will not affect the availability of that train, provided appropriate administrative controls have been established to ensure the remote control switch is immediately returned to the "Auto" position after the completion of the specified activities or in response to plant conditions. These administrative controls include the use of an approved procedure and a designated individual at the control switch for the respective Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train who can rapidly respond to instructions from procedures, or control room personnel, based on plant conditions.

3/4.7.10 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. For the purpose of declaring the affected system OPERABLE with the inoperable snubber(s), an engineering evaluation may be performed, in accordance with Section 50.59 of 10 CFR Part 50.

LCO 3.7.9 ACTION statement:

With one Auxiliary Building Filter System inoperable, restoration to OPERABLE status within 7 days is required.

The 7 days restoration time requirement is based on the following: The risk contribution is less for an inoperable Auxiliary Building Filter System, than for the charging pump or reactor plant component cooling water (RPCCW) systems, which have a 72 hour restoration time requirement. The Auxiliary Building Filter System is not a direct support system for the charging pumps or RPCCW pumps. Because the pump area is a common area, and as long as the other train of the Auxiliary Building Filter System remains OPERABLE, the 7 day restoration time limit is acceptable based on the low probability of a DBA occurring during the time period and the ability of the remaining train to provide the required capability. A concurrent failure of both trains would require entry into LCO 3.0.3 due to the loss of functional capability. The Auxiliary Building Filter System does support the Supplementary Leak Collection and Release System (SLCRS) and the LCO ACTION statement time of 7 days is consistent with that specified for SLCRS (See LCO 3.6.6.1).

Surveillance Requirement 4.7.9.c

Surveillance requirement 4.7.9.c requires that after 720 hours of operation a charcoal sample must be taken and the sample must be analyzed within 31 days after removal.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system." This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and sample canisters are typically removed due to the 18 month criteria.

3/4.7.10 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. For the purpose of declaring the affected system OPERABLE with the inoperable snubber(s), an engineering evaluation may be performed, in accordance with Section 50.59 of 10 CFR Part 50.

Snubbers are classified and grouped by design and manufacturer but not by size. Snubbers of the same manufacturer but having different internal mechanisms are classified as different types. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity

3/4.7.10 SNUBBERS (Continued)

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 Section 50.59 of 10 CFR Part 50.

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 asafety-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 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 5% 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

3/4.7.10 SNUBBERS (Continued)

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 indicatin 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.

3/4.7.11 DELETED

3/4.7.14 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±2.2°F.

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 Appendix A to 10 CFR Part 50.

LCO 3.8.1.1.a

LCO 3.8.1.1.a requires two independent offsite power sources. With both the RSST and the NSST available, either power source may supply power to the vital busses to meet the intent of Technical Specification 3.8.1.1. The FSAR, and Regulatory Guide 1.32, 1.6, and 1.93 provide the basis for requirements concerning off-site power sources. The basic requirement is to have two independent offsite power sources. The requirement to have a fast transfer is not specifically stated. An automatic fast transfer is required for plants without a generator output trip breaker, where power from the NSST is lost on a turbine trip. The surveillance requirement for transfer from the normal circuit to the alternate circuit is required for a transfer from the NSST to the RSST in the event of an electrical failure. There is no specific requirement to have an automatic transfer from the RSST to the NSST.

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 in part on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. Technical Specification 3.8.1.1 ACTION Statements b.2 and c.2 provide an allowance to avoid unnecessary testing of the other OPERABLE diesel generator. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generator, Surveillance Requirement 4.8.1.1.2.a.5 does not have to be performed. If the cause of inoperability exists on the other OPERABLE diesel generator, the other OPERABLE diesel generator would be declared inoperable upon discovery, ACTION Statement e. would be entered, and appropriate actions will be taken. Once the failure is corrected, the common cause failure no longer exists, and the required ACTION Statements (b., c., and e.) will be satisfied.

If it can not be determined that the cause of the inoperable diesel generator does not exist on the remaining diesel generator, performance of Surveillance Requirement 4.8.1.1.2.a.5, within the allowed time period, suffices to provide assurance of continued OPERABILITY of the diesel generator. If the inoperable diesel generator is restored to OPERABLE status prior to the determination of the impact on the other diesel generator, evaluation will continue of the possible common cause failure. This continued evaluation is no

longer under the time constraint imposed while in ACTION Statements b.2 or c.2.

The determination of the existence of a common cause failure that would affect the remaining diesel generator will require an evaluation of the current failure and the applicability to the remaining diesel generator. Examples that would not be a common cause failure include, but are not limited to:

1. Preplanned preventative maintenance or testing; or

2. An inoperable support system with no potential common mode failure for the remaining diesel generator; or

3. An independently testable component with no potential common mode failure for the remaining diesel generator.

When one diesel generator is inoperable, there is an additional ACTION requirement (b.3 and c.3) 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.

If one Millstone Unit No. 3 diesel generator is inoperable in MODES 1 through 4, a 72 hour allowed outage time is provided by ACTION Statement b.5 to allow restoration of the diesel generator, provided the requirements of ACTION Statements b.1, b.2, and b.3 are met. This allowed outage time can be extended to 14 days if the additional requirements contained in ACTION Statement b.4 are also met. ACTION Statement b.4 requires verification that the Millstone Unit No. 2 diesel generators are OPERABLE as required by the applicable Millstone Unit No. 2 Technical Specification (2 diesel generators in MODES 1 through 4, and 1 diesel generator in MODES 5 and 6) and the Millstone Unit No. 3 SBO diesel generator is available. The term verify, as used in this context, means to administratively check by examining logs or other information to determine if the required Millstone Unit No. 2 diesel generators and the Millstone Unit No. 3 SBO diesel generator are out of service for maintenance or other reasons. It does not mean to perform Surveillance Requirements needed to demonstrate the OPERABILITY of the required Millstone Unit No. 2 diesel generators or availability of the Millstone Unit No. 3 SBO diesel generator.

When using the 14 day allowed outage time provision and the Millstone Unit No. 2 diesel generator requirements and/or Millstone Unit No. 3 SBO diesel generator requirements are not met, 72 hours is allowed for restoration of the required Millstone Unit No. 2 diesel generators and the Millstone Unit No. 3 SBO diesel generator. If any of the required Millstone Unit No. 2 diesel generators and/or Millstone Unit No. 3 SBO diesel generator are not restored within 72 hours, and one Millstone Unit No. 3 diesel generator is still inoperable, Millstone Unit No. 3 is required to shut down.

Bases Change of 8-25-2005

The 14 day allowed outage time for one inoperable Millstone Unit No. 3 diesel generator will allow performance of extended diesel generator maintenance and repair activities (e.g., diesel inspections) while the plant is operating. To minimize plant risk when using this extended allowed outage time the following additional Millstone Unit No. 3 requirements must be met:

1) The charging pump and charging pump cooling pump in operation shall be powered from the bus not associated with the out of service diesel generator. In addition, the spare charging pump will be available to replace an inservice charging pump if necessary.

2) The extended diesel generator outage shall not be scheduled when adverse or inclement weather conditions and/or unstable grid conditions are predicted or present.

3) The availability of the Millstone Unit No. 3 SBO DG shall be verified by test performance within 30 days prior to allowing a Millstone Unit No. 3 EDG to be inoperable for greater than 72 hours.

4) All activity in the switchyard shall be closely monitored and controlled. No elective maintenance within the switchyard that could challenge offsite power availability shall be scheduled.

5) A contingency plan shall be available (OP 3314J, Auxiliary Building Emergency Ventilation and Exhaust) to provide alternate room cooling to the charging and CCP pump area (24'6" Auxiliary Building) in the event of a failure of the ventilation system prior to commencing an extended diesel generator outage.

In addition, the plant configuration shall be controlled during the diesel generator maintenance and repair activities to minimize plant risk consistent with the Configuration Risk Management Program, as required by 10 CFR 50.65(a)(4).

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.

LCO 3.8.1.1 ACTION statement b.3 and c.3

Required ACTION Statement b.3 and c.3 requires that all systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel as a source of emergency power be verified OPERABLE.

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

Technical Specification 3.8.1.1.b.1 requires each of the diesel generator day tanks contain a minimum volume of 278 gallons. Technical Specification 3.8.1.2.b.1 requires a minimum volume of 278 gallons be contained in the required diesel generator day tank. This capacity ensures that a minimum usable volume of 189 gallons is available. This volume permits operation of the diesel generators for approximately 27 minutes with the diesel generators loaded to the 2,000 hour rating of 5335 kw. Each diesel generator has two independent fuel oil transfer pumps. The shutoff level of each fuel oil transfer pump provides for approximately 60 minutes of diesel generator operation at the 2000 hour rating. The pumps start at day tank levels to ensure the minimum level is maintained. The loss of the two redundant pumps would cause day tank level to drop below the minimum value.

Technical Specification 3.8.1.1.b.2 requires a minimum volume of 32,760 gallons be contained in each of the diesel generator's fuel storage systems. Technical Specification 3.8.1.2.b.2 requires a minimum volume of 32,760 gallons be contained in the required diesel generator's fuel storage system. This capacity ensures that a minimum usable volume (29,180 gallons) is available to permit operation of each of the diesel generators for approximately three days with the diesel generators loaded to the 2,000 hour rating of 5335 kW. The ability to cross-tie the diesel generator fuel oil supply tanks ensures that one diesel generator may operate up to approximately six days. Additional fuel oil can be supplied to the site within twenty-four hours after contacting a fuel oil supplier.

Surveillance Requirements 4.8.1.1.2.a.6 (monthly) and 4.8.1.1.2.b.2 (once per 184 days) and 4.8.1.1.2.j (18 months test)

The Surveillances 4.8.1.1.2.a.6 and 4.8.1.1.2.b.2 verify that the diesel generators are capable of synchronizing with the offsite electrical system and loaded to greater than or equal to continuous rating of the machine. A minimum time of 60 minutes is required to stabilize engine temperatures, while

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minimizing the time that the diesel generator is connected to the offsite source. Surveillance Requirement 4.8.1.1.2.j requires demonstration once per 18 months that the diesel generator can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which are at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the diesel generator. The load band is provided to avoid routine overloading of the diesel generator. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain diesel generator OPERABILITY. The load band specified accounts for instrumentation inaccuracies using plant computer and for the operational control capabilities and human factor characteristics. The note (*) acknowledges that momentary transient outside the load range shall not invalidate the test.

Surveillance Requirements 4.8.1.1.2.a.5 (Monthly), 4.8.1.1.2.b.1 (Once per 184 Days), 4.8.1.1.2.g.4.b (18 Month Test), 4.8.1.1.2.g.5 (18 Month Test) and 4.8.1.1.2.g.6.b (18 Month Test)

Several diesel generator surveillance requirements specify that the emergency diesel generators are started from a standby condition. Standby conditions for a diesel generator means the diesel engine coolant and lubricating oil are being circulated and temperatures are maintained within design ranges. Design ranges for standby temperatures are greater than or equal to the low temperature alarm setpoints and less than or equal to the standby "keep-warm" heater shutoff temperatures for each respective sub-system.

Surveillance Requirement 4.8.1.1.2.j (18 Month Test)

The existing "standby condition" stipulation contained in specification 4.8.1.1.2.a.5 is superseded when performing the hot restart demonstration required by 4.8.1.1.2.j.

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

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1975 & 1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." Sections 5 and 6 of IEEE Std 450-1980 replaced Sections 4 and 5 of IEEE Std 450-1975, otherwise the balance of IEEE Std 450-1975 applies.

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 4.8-2a 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 4.8-2a 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 DELETED

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. 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 specified in the CORE OPERATING LIMITS REPORT includes a conservative uncertainty allowance of 50 ppm boron. The boron concentration, specified in the CORE OPERATING LIMITS REPORT, provides for boron concentration measurement uncertainty between the spent fuel pool and the RWST. 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.

MODE ZERO shall be the Operational MODE where all fuel assemblies have been removed from containment to the Spent Fuel Pool. Technical Specification Table 1.2 defines MODE 6 as "Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed." With no fuel in the vessel the definition for MODE 6 no longer applies. The transition from MODE 6 to MODE ZERO occurs when the last fuel assembly of a full core off load has been transferred to the Spent Fuel Pool and has cleared the transfer canal while in transit to a storage location. This will:

- Ensure Technical Specifications regarding sampling the transfer canal boron concentration are observed (4.9.1.1.2);
- Ensure that MODE 6 Technical Specification requirements are not relaxed prematurely during fuel movement in containment.

3/4.9.1.2 BORON CONCENTRATION IN SPENT FUEL POOL

During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment. This is accomplished in Region 1, 2, and 3 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in some fuel storage regions, the limits on fuel burnup, fuel enrichment and minimum fuel decay time, and the use of blocking devices in certain fuel storage locations.

The boron requirement in the spent fuel pool specified in 3.9.1.2 ensures that in the event of a fuel assembly handling accident involving either a single dropped or misplaced fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to 0.95.

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 penetration closure and OPERABILITY ensure that a release of radioactive material within containment to the environment will be minimized. The OPERABILITY, closure restrictions, and administrative controls are sufficient to minimize the release of radioactive material from a fuel element rupture based upon the lack of containment pressurization potential during the movement of fuel within containment. The containment purge valves are containment penetrations and must satisfy all requirements specified for a containment penetration.

This specification is applicable during the movement of new and spent fuel assemblies within the containment building. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel impacted upon is damaged. Therefore, the movement of either new or irradiated fuel can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment.

Containment penetrations, including the personnel access hatch doors and equipment access hatch, can be open during the movement of fuel provided that sufficient administrative controls are in place such that any of these containment penetrations can be closed within 30 minutes. Following a Fuel Handling Accident, each penetration, including the equipment access hatch, is closed such that a containment atmosphere boundary can be established. However, if it is determined that closure of all containment penetrations would represent a significant radiological hazard to the personnel involved, the decision may be made to forgo the closure of the affected penetration(s). The containment atmosphere boundary is established when any penetration which provides direct access to the outside atmosphere is closed such that at least one barrier between the containment atmosphere and the outside atmosphere is established. Additional actions beyond establishing the containment atmosphere boundary, such as installing flange bolts for the equipment access hatch or a containment penetration, are not necessary.

Administrative controls for opening a containment penetration require that one or more designated persons, as needed, be available for isolation of containment from the outside atmosphere. Procedural controls are also in place to ensure cables or hoses which pass through a containment opening can be quickly removed. The location of each cable and hose isolation device for those cables and hoses which pass through a containment opening is recorded to ensure timely closure of the containment boundary. Additionally, a closure plan is developed for each containment opening which includes an estimated time to close the containment opening. A log of personnel designated for containment closure is maintained, including identification of which containment openings each person has responsibility for closing. As necessary, equipment will be pre-staged to support timely closure of a containment penetration.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

The ability to close the equipment access hatch penetration within 30 minutes is verified each refueling outage prior to the first fuel movement in containment with the equipment access hatch open. Prior to opening a containment penetration, a review of containment penetrations currently open is performed to verify that sufficient personnel are designated such that all containment penetrations can be closed within 30 minutes. Designated personnel may have other duties, however, they must be available such that their assigned containment openings can be closed within 30 minutes. Additionally, each new work activity inside containment is reviewed to consider its effect on the closure of the equipment access hatch, at least one personnel access hatch door, and/or other open containment penetrations. The required number of designated personnel are continuously available to perform closure of their assigned containment openings whenever fuel is being moved within the containment.

Controls for monitoring radioactivity within containment and in effluent paths from containment are maintained consistent with General Design Criterion 64. Local area radiation monitors, effluent discharge radiation monitors, and containment gaseous and particulate radiation monitors provide a defense-in-depth monitoring of the containment atmosphere and effluent releases to the environment. These monitors are adequate to identify the need for establishing the containment atmosphere boundary. When containment penetrations are open during a refueling outage under administrative control for extended periods of time, routine grab samples of the containment atmosphere, equipment access hatch, and personnel access hatch will be required.

The containment atmosphere is monitored during normal and transient operations of the reactor plant by the containment structure particulate and gas monitor located in the upper level of the Auxiliary Building or by grab sampling. Normal effluent discharge paths are monitored during plant operation by the ventilation particulate samples and gas monitors in the Auxiliary Building.

Administrative controls are also in place to ensure that the containment atmosphere boundary is established if adverse weather conditions which could present a potential missile hazard threaten the plant. Weather conditions are monitored during fuel movement whenever a containment penetration, including the equipment access hatch and personnel access hatch, is open and a storm center is within the plant monitoring radius of 150 miles.

The administrative controls ensure that the containment atmosphere boundary can be quickly established (i.e. within 30 minutes) upon determination that adverse weather conditions exist which pose a significant threat to the Millstone Site. A significant threat exists when a hurricane warning or tornado warning is issued which applies to the Millstone Site, or if an average wind speed of 60 miles an hour or greater is recorded by plant meteorological equipment at the meteorological tower. If the meteorological equipment is inoperable, information from the National Weather Service can be used as a backup in determining plant wind speeds. Closure of containment penetrations, including the equipment access hatch penetration and at least one personnel access hatch door, begin immediately upon determination that a significant threat exists.

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 ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) 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 over fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be less than the activity release assumed in the design basis fuel handling accident, and (2) the resulting geometry will not result in a critical array.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

3/4.9.8.1 HIGH WATER LEVEL

BACKGROUND

The purpose of the Residual Heat Removal (RHR) System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Reactor Plant Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR system for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR system.

3/4.9.8.1 HIGH WATER LEVEL (continued)

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is fission product barrier. One train of the RHR system is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange to prevent this challenge. The LCO does permit deenergizing the RHR pump for short durations, under the conditions that the boron concentration is not diluted. This conditional deenergizing of the RHR pump does not result in a challenge to the fission product barrier.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.10, "Water Level — Reactor Vessel." Requirements for the RHR system in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.8.2, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

LIMITING CONDITION FOR OPERATION

The requirement that at least one RHR loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat an 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 stratification.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path. An operating RHR flow path should be capable of determining the low-end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8-hour period. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzle and RCS to RHR isolation valve testing. During this 1-hour period, decay heat is removed by natural connection to the large mass of water in the refueling cavity.

3/4.9.8.1 HIGH WATER LEVEL (continued)

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operations, except as permitted in the Note to the LCO.

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.

Reduced boron concentrations can occur by the addition of water with lower boron concentration that contained in the RCS. Therefore, actions that result in an unplanned boron dilution shall be suspended immediately.

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

Surveillance Requirement

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR system.

3/4.9.8.2 LOW WATER LEVEL

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR system.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

LIMITING CONDITION FOR OPERATION

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor cooling temperature.

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 procedure to cool the core.

3/4.9.8.2 LOW WATER LEVEL (continued)

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. An operating RHR flow path should be capable of determining the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.8.1, "Residual Removal (RHR) AND Coolant Circulation—High Water Level."

ACTIONS

a. If less than the required number of RHR loops are OPERABLE, actions shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation, or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.8.1, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective action.

b. If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a low boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in ACTIONS 'a' and 'b' concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

Surveillance Requirement

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

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

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

3/4.9.13 SPENT FUEL POOL - REACTIVITY (continued)

use of fixed neutron absorbers in the racks, a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-3.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 are complied with.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region 1 3-OUT-OF-4 storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 Boundary to other Storage Regions and minimize the number of boundaries where a fuel misplacement incident can occur.

3/4.9.13 SPENT FUEL POOL - REACTIVITY (continued)

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 are complied with.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region 1 3-OUT-OF-4 storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 Boundary to other Storage Regions and minimize the number of boundaries where a fuel misplacement incident can occur.

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 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

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

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SECTION 5.0

DESIGN FEATURES

5.1 SITE LOCATION

The Unit 3 Containment Building is located on the site at Millstone Point in Waterford, Connecticut. The nearest site boundary on land is 1719 feet northeast of the containment building wall (1627 feet northeast of the elevated stack), which is the minimum distance to the boundary of the exclusion area as described in 10 CFR 100.3. No part of the site that is closer than these distances shall be sold or leased except to Dominion Nuclear Connecticut, Inc. or its corporate affiliates for use in conjunction with normal utility operations.

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 264 zircaloy-4 or ZIRLO clad fuel rods with an initial composition of natural uranium dioxide or a maximum nominal enrichment of 5.0 weight percent U-235 as fuel material. Limited substitutions of zircaloy-4, ZIRLO or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design bases. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 61 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 95.3% hafnium and 4.5% natural zirconium or 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel.

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5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The storage rack Regions are:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:

   (1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.

   (2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4, fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 total available fuel storage locations. An additional Region 2 rack with 81 storage locations may be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations, for a total of 1860 total available fuel storage locations.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The designated officer shall be responsible for overall operation of the Millstone Station Site and shall delegate in writing the succession to this responsibility. The designated manager shall be responsible for overall Unit safe operation and shall delegate in writing the succession to this responsibility.

6.1.2 The Shift Manager shall be responsible for the control room command function.

6.1.3 Unless otherwise defined, the technical specification titles for members of the staff are generic titles. Unit specific titles for the functions and responsibilities associated with these generic titles are identified in appropriate administrative documents.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Program Topical Report.

b. The designated manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The designated officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 FACILITY STAFF

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

FACILITY STAFF (Continued)

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. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. A radiation protection technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

f. Deleted

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions. These procedures should follow the general guidance of the NRC Policy Statement on working hours (Generic Letter No. 82-12).

*The radiation protection technician 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.

<u>TABLE 6.2-1</u>

<u>MINIMUM SHIFT CREW COMPOSITION</u>

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SM	1	1
SRO	1	None
RO	2	1
PEO	2	1
STA	1*	None

SM - Shift Manager with a Senior Operator license on Unit 3
SRO - Individual with a Senior Operator license on Unit 3
RO - Individual with an Operator license on Unit 3
PEO - Plant Equipment Operator (Non-licensed)
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 crewmember being late or absent.

During any absence of the Shift Manager 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 Manager 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 may be filled by an on-shift Senior Reactor Operator only if that Senior Reactor Operator meets the Shift Technical Advisor qualifications of the Commission Policy Statement on Engineering Expertise on Shift.

6.2.3 Deleted.

6.2.4 SHIFT TECHNICAL ADVISOR

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

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971* for comparable positions. Exceptions to this requirement are specified in the Quality Assurance Program.

6.3.2 If the operations manager does not hold a senior reactor operator license for Millstone Unit No. 3, then the operations manager shall have held a senior reactor operator license at a pressurized water reactor, and the assistant operations manager shall hold a senior reactor operator license for Millstone Unit No. 3.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff that meets or exceeds the requirements as specified in the Quality Assurance Program and 10 CFR Part 55.59 shall be maintained.

6.4.2 Deleted.

6.5 Deleted.

* As of November 1, 2001, applicants for reactor operator and senior reactor operator qualification shall meet or exceed the education and experience guidelines of Regulatory Guide 1.8, Revision 3, May 2000.

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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 applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto;

c. Refueling operations;

d. Surveillance activities of safety related equipment;

e. Not used.

f. Not used.

g. Fire Protection Program implementation;

h. Quality controls for effluent monitoring, using the guidance in Regulatory Guide 1.21, Rev. 1, June 1974; and

i. Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) implementation except for Section I.E, Radiological Environmental Monitoring.

6.8.2 a. The designated manager or designated officer or designated senior officer may designate specific procedures and programs, or classes of procedures and programs to be reviewed in accordance with the Quality Assurance Program Topical Report.

b. Procedures and programs listed in Specification 6.8.1, and changes thereto, shall be approved by the designated manager or designated officer or by cognizant managers or directors who are designated as the Approval Authority by designated manager or designated officer as specified in administrative procedures. The Approval Authority for each procedure and program or class of procedure and program shall be specified in administrative procedures.

c. Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with the Quality Assurance Program Topical Report, prior to implementation. Each procedure of Specification 6.8.1 shall be 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

c. The change is documented, reviewed and approved in accordance with the Quality Assurance Program Topical Report 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 recirculation spray, Safety Injection, charging portion of chemical and volume control, and hydrogen recombiners. 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.

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. 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

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. Deleted

e. Accident Monitoring Instrumentation

A program which will ensure the capability to monitor plant variables and systems operating status during and following an accident. This program shall include those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials (Category 2 and 3 instrumentation as defined in Regulatory Guide 1.97, Revision 2) and provide the following:

1) Preventive maintenance and periodic surveillance of instrumentation,

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 38.57 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.042\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $<0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

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* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

PROCEDURES AND PROGRAMS (Continued)

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the organization responsible for the REMP and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the organization responsible for the REMP within 14 days of implementation.

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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector, 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 Startup Report shall address each of the tests identified in 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.

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 in accordance with 10 CFR 50.4.

6.9.1.2a. Deleted

6.9.1.2b. The results of specific activity analyses in which the reactor 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 the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the 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 radioiodine isotope concentration (μCi/gm) as a function of time for the

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

ANNUAL REPORTS (Continued)

duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit. The report covering the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.3 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

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A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station.

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The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM), and in 10 CFR Part 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses 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 REMODCM, as well as summarized and tabulated results of these analyses and measurements. 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 in the next annual report.

6.9.1.4 RADIOACTIVE EFFLUENT RELEASE REPORT

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A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the REMODCM and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

6.9.1.5 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.6 a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Overtemperature ΔT and Overpower ΔT setpoint parameters for Specification 2.2.1,

2. Shutdown Margin for Specifications 3/4.1.1.1.1, 3/4.1.1.1.2, and 3/4.1.1.2,

3. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

CORE OPERATING LIMITS REPORT (Cont.)

4. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,
5. Control Rod Insertion Limits for Specification 3/4.1.3.6,
6. Axial Flux Difference Limits, target band, and APL^{ND} for Specifications 3/4.2.1.1 and 3/4.2.1.2,
7. Heat Flux Hot Channel Factor, K(z), W(z), APL^{ND}, and $W(z)_{BL}$ for Specifications 3/4.2.2.1 and 3/4.2.2.2.
8. Nuclear Enthalpy Rise Hot Channel Factor, Power Factor Multiplier for Specification 3/4.2.3.
9. DNB Parameters for Specification 3/4.2.5.
10. Shutdown Margin Monitor minimum count rate for Specification 3/4.3.5.
11. Boron Concentration for Specification 3/4.9.1.1.

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," (W Proprietary). (Methodology for Specifications 3.1.1.3--Moderator Temperature Coefficient, 3.1.3.5--Shutdown Bank Insertion Limit, 3.1.3.6--Control Bank Insertion Limits, 3.2.1--Axial Flux Difference, 3.2.2--Heat Flux Hot Channel Factor, 3.2.3--Nuclear Enthalpy Rise Hot Channel Factor, 3.1.1.1.1, 3.1.1.1.2, 3.1.1.2 -- Shutdown Margin, 3.9.1.1 -- Boron Concentration.)
2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC), January 31, 1980--Attachment: Operation and Safety-Analysis Aspects of an Improved Load Follow Package.
3. NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981 Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Revision 2, July 1981.
4. WCAP-10216-P-A-R1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," (W Proprietary). (Methodology for Specifications 3.2.1--Axial Flux Difference [Relaxed Axial Offset Control] and 3.2.2--Heat Flux Hot Channel Factor [W(z) surveillance requirements for F_Q Methodology].)
5. WCAP-9561-P-A, ADD. 3, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS--SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)
6. WCAP-10266-P-A, Addendum 1, "THE 1981 VERSION OF THE WESTINGHOUSE ECCS EVALUATION MODEL USING THE BASH CODE," (W Proprietary). (Methodology for Specification 3.2.2--Heat Flux Hot Channel Factor.)

CORE OPERATING LIMITS REPORT (Cont.)

7. WCAP-11946, "Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Millstone Nuclear Power Station Unit 3," (W Proprietary).

8. WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL.17 USING THE NOTRUMP CODE," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

9. WCAP-10079-P-A, "NOTRUMP - A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

10. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

11. Letter from V. L. Rooney (USNRC) to J. F. Opeka, "Safety Evaluation for Topical Report, NUSCO-152, Addendum 4, 'Physics Methodology for PWR Reload Design,' TAC No. M91815," July 18, 1995.

12. Letter from E. J. Mroczka to the USNRC, "Proposed Changes to Technical Specifications, Cycle 4 Reload Submittal - Boron Dilution Analysis," B13678, December 4, 1990.

13. Letter from D. H. Jaffe (USNRC) to E. J. Mroczka, "Issuance of Amendment (TAC No. 77924)," March 11, 1991.

14. Letter from M. H. Brothers to the USNRC, "Proposed Revision to Technical Specification, Shutdown Margin Requirements and Shutdown Margin Monitor Operability for Modes 3, 4, and 5 (PTSCR 3-16-97), B16447, May 9, 1997.

15. Letter from J. W. Anderson (USNRC) to M. L. Bowling (NNECO), "Issuance of Amendment - Millstone Nuclear Power Station, Unit No. 3 (TAC No. M98699)," October 21, 1998.

16. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis."

17. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model."

18. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Westinghouse Proprietary Class 2). (Methodology for Specification 2.2.1.)

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, within the time period specified for each report.

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area, or

6.12 HIGH RADIATION AREA (cont.)

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

6.12.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designees, and

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of

6.12 HIGH RADIATION AREA (cont.)

radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

6.13 RADIOLOGICAL EFFLUENT MONITORING AND OFFSITE DOSE CALCULATION MANUAL (REMODCM)

a. The REMODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental program; and

b. The REMODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release, reports required by Specification 6.9.1.3 and Specification 6.9.1.4.

Licensee initiated changes to the REMODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1) sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2) a determination that the change(s) will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I of 10 CFR 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by SORC and the approval of the designated officer; and

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire REMODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the REMODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicated the date (i.e., month and year) the change was implemented.

6.14 RADIOACTIVE WASTE TREATMENT

Procedures for liquid and gaseous radioactive effluent discharges from the Unit shall be prepared, approved, maintained and adhered to for all operations involving offsite releases of radioactive effluents. These procedures shall specify the use of appropriate waste treatment systems utilizing the guidance provided in the REMODCM.

The Solid Radioactive Waste Treatment System shall be operated in accordance with the Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

6.15 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the REMODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the REMODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the REMODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the REMODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the REMODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

    1. For noble gases: a dose rate $\leq$ 500 mrem/yr to the whole body and a dose rate $\leq$ 3000 mrem/yr to the skin, and

    2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate $\leq$ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specification 4.0.2 and Specification 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.16 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provided (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the REMODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

c. Participation in a Interlaboratory Comparison Program 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.

6.17 REACTOR COOLANT PUMP FLYWHEEL INSPECTION PROGRAM

This program shall provide for the inspection of each reactor coolant pump flywheel by either qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.

6.18 TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications:

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license or

   2. A change to updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 6.18.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.19 COMPONENT CYCLIC OR TRANSIENT LIMIT

This program provided controls to track the FSAR, Section 3.9N, cyclic and transient occurrences to ensure that components are maintained within the design limits.