ATTACHMENT I

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Revised Technical Specifications

MARKUP OF FINAL DRAFT

OF TECHNICAL SPECIFICATIONS

DEFINITIONS

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ADMINISTRATIVE CONTROLS

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SECTION 1.0

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.

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CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

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

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

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

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 Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, ~~March 1976~~ Rev. 1, October, 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

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FREQUENCY NOTATION

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

GAS DECAY TANK SYSTEM

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

IDENTIFIED LEAKAGE

1.14 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.

MEMBER(S) OF THE PUBLIC

1.15 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, vendors or members of the Armed Forces using property located within the SITE BOUNDARY. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

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

[illegible]

[illegible]

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.20 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.

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

1.22 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.

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QUADRANT POWER TILT RATIO

1.23 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

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2200 MWt.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.26 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

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.30 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.

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THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.32 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

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

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which 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.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 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.

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TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

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

*Excluding decay heat.

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

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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, for 3 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

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REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

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

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See next page

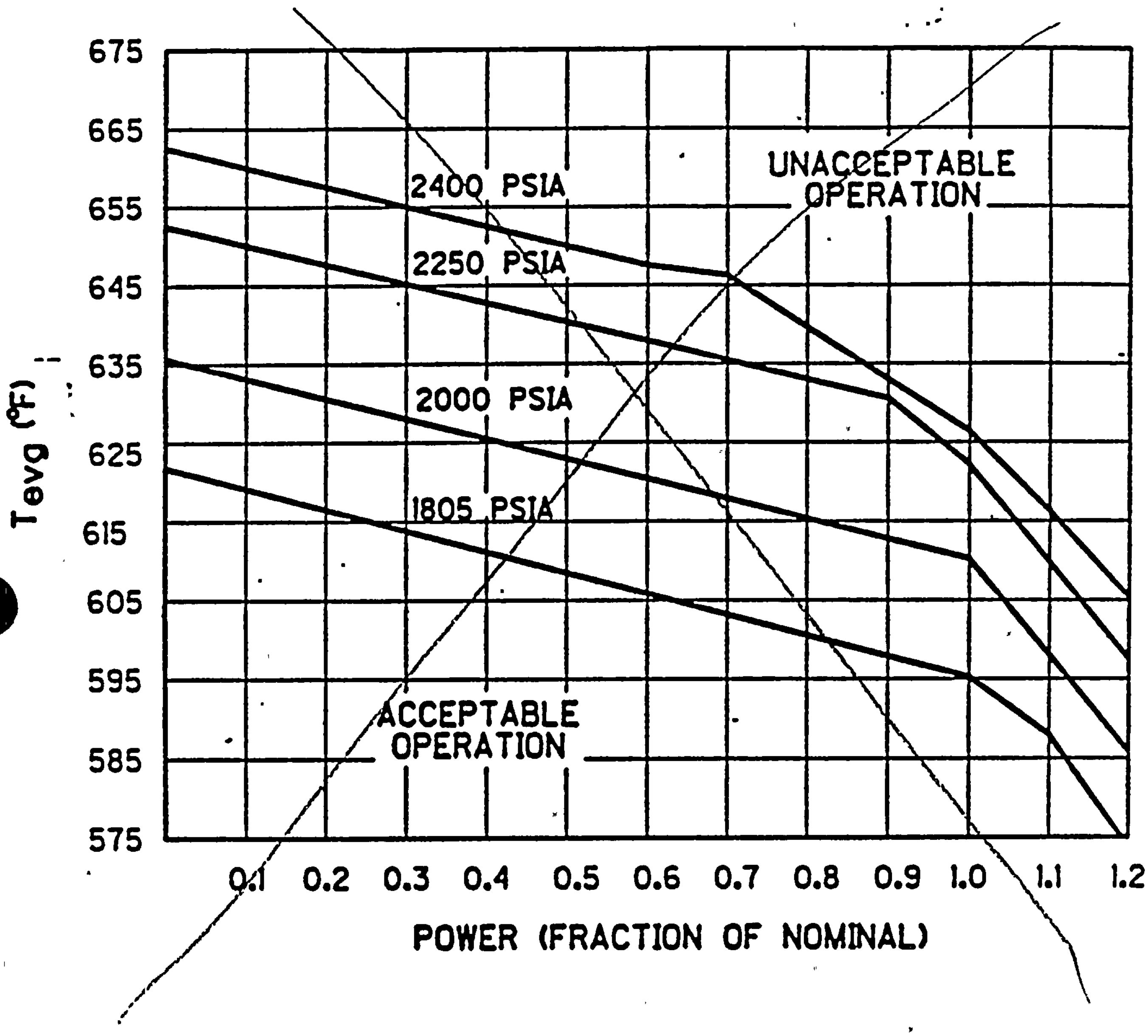

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

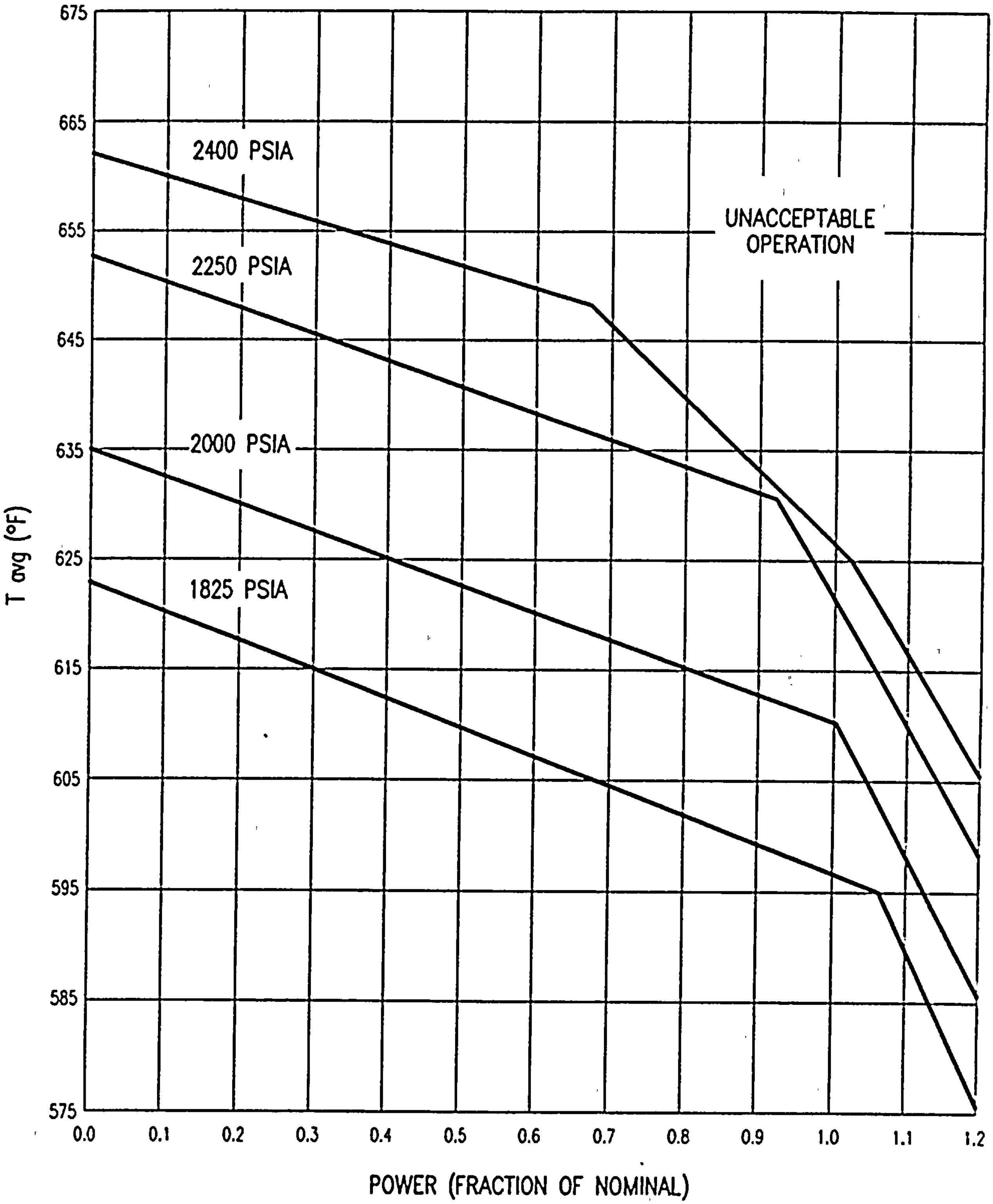

FIGURE 2.1—1 REACTOR CORE SAFETY LIMIT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

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

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE #
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP**	≤[ ]% of RTP**
b. Low Setpoint	≤25% of RTP**	≤[ ]% of RTP**
3. Intermediate Range, Neutron Flux	≤25% of RTP**	≤[ ]% of RTP**
4. Source Range, Neutron Flux	≤10^5 cps	≤[ ] x 10^5 cps
5. Overtemperature ΔT	See Note 1	
6. Overpower ΔT	See Note 3	
7. Pressurizer Pressure-Low	≥1835 psig	≥[ ] psig
8. Pressurizer Pressure-High	≤2385 psig	≤[ ] psig
9. Pressurizer Water Level-High	≤92% of instrument span	≤[ ]% of instrument span
10. Reactor Coolant Flow-Low	≥90% of loop design flow*	≥[ ]% of loop design flow*
11. Steam Generator Water Level Low-Low	≥15% of narrow range instrument span	≥[ ]% of narrow range instrument span

*Loop design flow = 89,500 gpm
**RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE #
12. Steam/Feedwater Flow Mismatch Coincident With	Feed Flow $\leq 0.64 \times 10^6$ lb/hr below steam flow	Feed Flow $\leq$[ ] $\times 10^6$ lb/hr below steam flow
Steam Generator Water Level-Low	$\geq$15% of narrow range instrument span	$\leq$[ ]% of narrow range instrument span
13. Undervoltage - 4.16 kV Busses A and B	$\geq$2496 volts-each bus	$\geq$[ ] volts-each bus
14. Underfrequency - Trip of Reactor Coolant Pump Breaker(s) Open	$\geq$56.1 Hz	$\geq$[ ] Hz
15. Turbine Trip		
a. Auto Stop Oil Pressure	$\geq$45 psig	$\geq$[ ] psig
b. Turbine Stop Valve Closure	Fully Closed ***	Fully Closed ***
16. Safety Injection Input from ESF	N. A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq$[ ] amp

***Limit switch is set when Turbine Stop Valves are fully closed.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE #
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	≤10% of RTP**	≤[ ]% of RTP**
2) Turbine First Stage Pressure	≤10% Turbine Power	≤[ ]% Turbine Power
c. Power Range Neutron Flux, P-8	≤45% of RTP**	≤[ ]% of RTP**
d. Power Range Neutron Flux, P-10	≥10% of RTP**	≥[ ]% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

**RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1}{1+\tau_1 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_2 S)}{(1+\tau_3 S)}\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right] + K_3(P - P') - f(\Delta I)\right\}$$

Where:	ΔT	=	Measured ΔT by RTD Instrumentation;
	$\frac{1}{1+\tau_1 S}$	=	Lag compensator on measured ΔT;
	τ_1	=	Time constants utilized in the lag compensator for ΔT, ~~τ_1 = 2 s;~~ τ_1 + RTD Response time = 2.5 s;
	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
	K_1	=	1.095;
	K_2	=	0.0107/°F;
	$\frac{1+\tau_2 S}{1+\tau_3 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
	τ_2, τ_3	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_2 = 25s, τ_3 = 3 s;
	T	=	Average temperature, °F;
	$\frac{1}{1+\tau_4 S}$	=	Lag compensator on measured T_{avg};
	τ_4	=	Time constant utilized in the measured T_{avg} lag compensator, ~~τ_4 = 2 s;~~ τ_4 + RTD response time = 2.5 s;
	T'	$\leq$	574.2°F (Nominal T_{avg} at RATED THERMAL POWER);
	K_3	=	0.000453/psig;
	P	=	Pressurizer pressure, psig;

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TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

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

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

(1) For $q_t - q_b$ between - 14% and + 10%, f (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 14%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 10%, the ΔT Trip Setpoint shall be automatically reduced by 3.5% of its value at RATED THERMAL POWER.

NOTE 2: (This note number is not used.)

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \le \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right) \left(\frac{1}{1 + \tau_4 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_4 S}\right) - T''\right] - f(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1}{1 + \tau_1 S}$	=	As defined in Note 1,
τ_1	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	=	1.09,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_5 S}{1 + \tau_5 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_5	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_5 = 10 s,
$\frac{1}{1 + \tau_4 S}$	=	As defined in Note 1,
τ_4	=	As defined in Note 1,

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00068/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 574.2°F),
S	=	As defined in Note 1, and
$f(\Delta I)$	=	As defined in Note 1.

NOTE 4: (This note number is not used.)

If no allowable valve is specified as indicated by [ ], the trip set point shall also be the allowable value.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

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2.1 SAFETY LIMITS

BASES

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 relationship 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: there must be at least a 95 percent probability with 95 percent confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

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

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 and 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:

$$F^N_{\Delta H} \leq 1.62\ [1+ 0.3\ (1-P)]$$

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 limit assuming the axial power imbalance is within the limits of the f (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core Safety Limits.

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2.1.1 REACTOR CORE (Continued)

Fuel rod bowing reduces the values of DNB ratio (DNBR). The penalties are calculated pursuant to "Fuel Rod Bow Evaluation," WCAP-8691-P-A Revision 1 (Proprietary) and WCAP-8692 Revision 1 (Non-Proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

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 and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The RCS piping, valves and fittings are designed to ANSI B31.1 which permits a maximum transient pressure of 120% of design pressure of 2485 psig. The Safety Limit of 2735 psig is therefore more conservative than the ANSI B31.1 design criteria and consistent with associated ASME Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) 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 Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

To accommodate the instrument drift that may occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable. If no value is listed in the Allowable column, the setpoint value is the limiting setting.

The methodology to derive the Trip Setpoints includes an allowance for instrument uncertainties. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the 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.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

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 for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

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.

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 is 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 Protection 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 (about 4 seconds), 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.

Table 2.2-1, Note 1, provides the equation defining overtemperature ΔT. Inherent in the equation are time delays on measured ΔT and T_{avg} attributable to RTDs and lag compensators. A time delay of 2.5 seconds is introduced through a combination of RTD response time and adjustments of time constants τ_1 and τ_4.

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Overpower ΔT

The Overpower ΔT trip prevents power density anywhere in the core from exceeding 118% of the design power density. This 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, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

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 first stage 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 safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

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

Reactor Coolant Flow

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

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage pressure at approximately 10%

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Reactor Coolant Flow (Continued)

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 45% 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.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Water Level-Low trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 0.64×10^6 lbs/hour. The Steam Generator Water Level-Low portion of the trip is activated when the water level drops below 15%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage - 4.16 kV Bus A and B Trips

The 4.16 kV Bus A and B Undervoltage trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoint assures a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. The delay is set so that the time required for a signal to reach the Reactor trip breakers following the trip of at least one undervoltage relay in both of the associated Units 4.16 kV busses shall not exceed 1.3 seconds. On decreasing

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Undervoltage and - 4.16 kV Bus A and B Trips (Continued)

power the Undervoltage Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

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

Safety Injection Input from ESF

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

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage pressure at approximately 10% of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7, an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Underfrequency sensors are also installed on the 4.16 kV busses to detect underfrequency and initiate breaker trip on underfrequency. The underfrequency trip setpoints preserve the coast down energy of the reactor coolant pumps, in case of a grid frequency decrease so DNB does not occur.

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Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

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

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

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops, and one or more reactor coolant pump breakers open. On decreasing power, the P-8 interlock automatically blocks the trip on low flow in one coolant loop or one coolant pump breaker open.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. P-10 also provides input to P-7. The trip setpoint on increasing power shall be $\geq$ 10% and the reset point shall be less than or equal to 10%.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

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

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

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, 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 MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

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3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:

a. Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously.

b. Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and

c. Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are different for each unit, this will be identified in parentheses, footnotes or body of the requirement.

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

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

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

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (CONTINUED)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

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

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

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

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

4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses, footnotes or body of the requirement.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to the applicable value shown in Figure 3.1-1.

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

ACTION:

With the SHUTDOWN MARGIN less than the applicable value shown in Figure 3.1-1, immediately initiate and continue boration at greater than or equal to ~~10~~ 4 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the applicable value shown in Figure 3.1-1:

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

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

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

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

*See Special Test Exceptions Specification 3.10.1.

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SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 When in MODE 1 or 2, the overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

See next Page

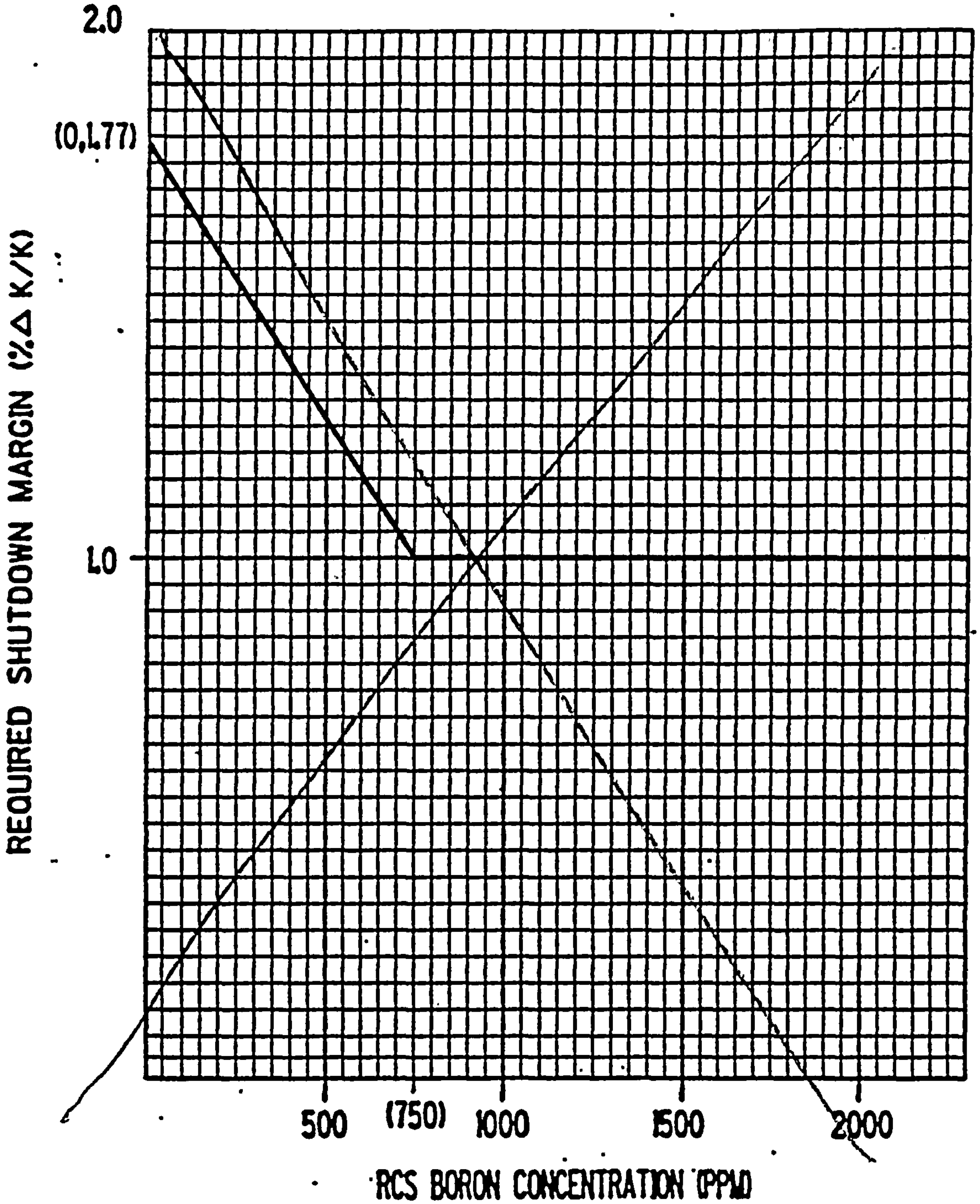

Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant
Boron Concentration

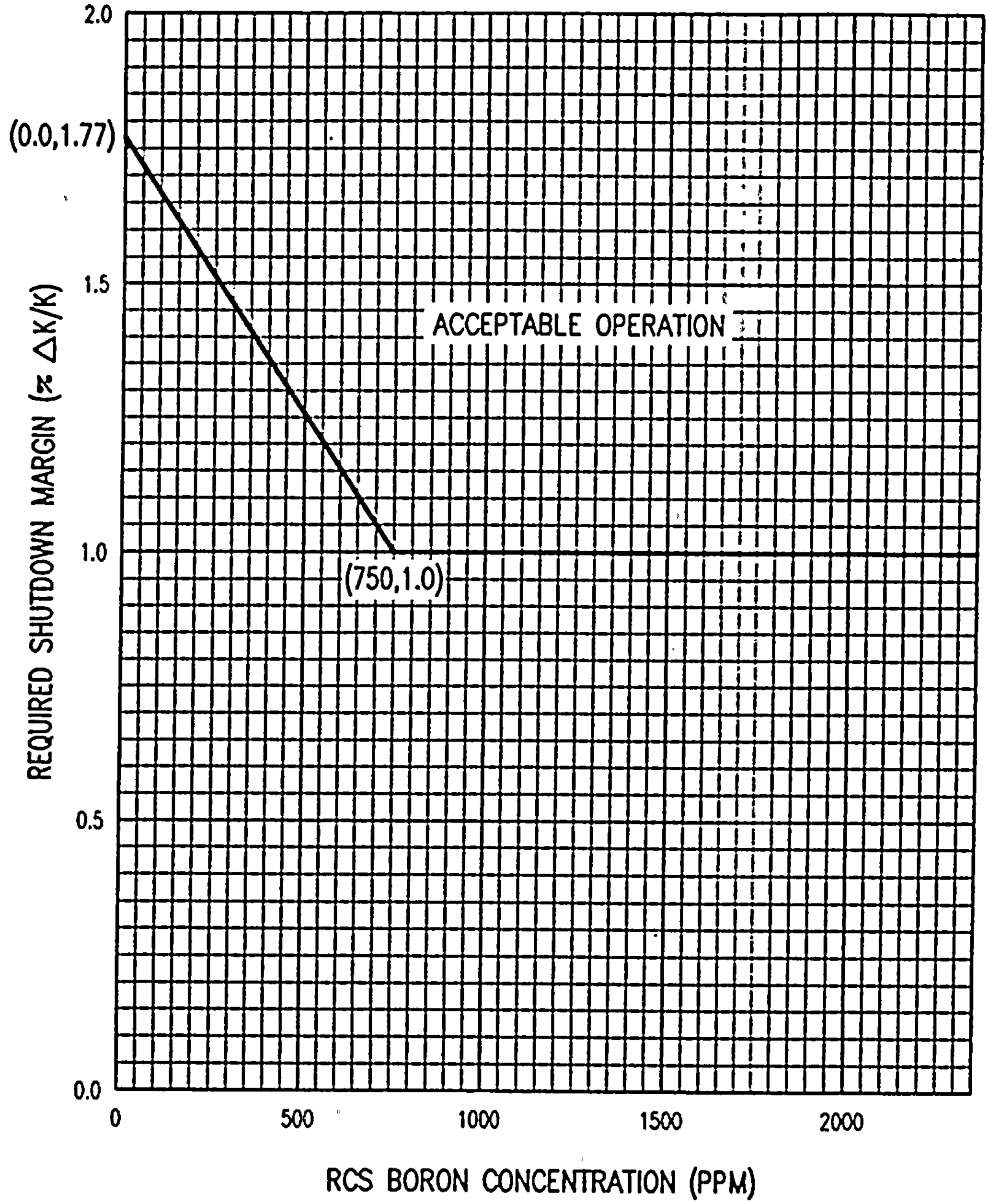

FIGURE 3.1—1 REQUIRED SHUTDOWN MARGIN vs RCS BORON CONCENTRATION

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 4 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$:

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

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^\circ F$ for all rods withdrawn, beginning of the cycle life (BOL), hot zero THERMAL POWER (HZP) conditions; and

b. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^\circ F$ from HZP to 70% RATE THERMAL POWER condition; and

c. Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^\circ F$ from 70% RATED THERMAL POWER decreasing linearly to less positive than or equal to 0 $\Delta k/k/^\circ F$ at 100% RATED THERMAL POWER conditions; and

d. Less negative than -3.5×10^{-4} $\Delta k/k/^\circ F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

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

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a, b or c 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 or equal to limits described in 3.1.1.3a, b and c above 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.

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

**See Special Test Exceptions Specification 3.10.3.

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LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3d., at least once per 14 EFPD during the remainder of the fuel cycle.

c. Perform design calculation to verify conformance to Specifications 3.1.1.3b and c.

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REACTIVITY CONTROL SYSTEMS

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 541°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

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 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 541°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 547°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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3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A flow path from the boric acid storage tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.4a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.4b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145°F when a flow path from the boric acid tanks is used, and

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

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FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

a. The source path from a boric acid storage tank via a boric acid transfer pump to the charging pump suction*, and

b. At least one of the two source paths from the refueling water storage tank to the charging pump suction; and,

c. The flow path from the charging pump discharge to the Reactor Coolant System via the regenerative heat exchanger.

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

ACTION:

a. With no boration source path from a boric acid storage tank OPERABLE,

   1. Demonstrate the OPERABILITY of the second source path from the refueling water storage tank to the charging pump suction by verifying the flow path valve alignment; and

   2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δ k/k at 200°F within the next 6 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δ k/k at 200°F within the next 6 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a SHUTDOWN MARGIN equivalent to 1% Δ k/k at 200°F and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

*The flow required in Specification 3.1.2.2.a above shall be isolated from the other unit.

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SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 145°F when it is a required water source;

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

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 4 gpm to the RCS.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps with independent power supplies shall be OPERABLE.

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

ACTION:

a. With two charging pumps OPERABLE and powered from a common power supply, restore at least two charging pumps from independent power supplies to OPERABLE status within 7 days or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δ k/k at 200°F within the next 6 hours; restore at least two charging pumps from independent power supplies to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With only one charging pump OPERABLE, restore any two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δ k/k at 200°F within 6 hours; restore any two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable to ACTION a, provided the 7 day limit of ACTION a is not exceeded.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The required charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5 The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume of 500 gallons,

2) A boron concentration between 20,000 ppm and 22,500 ppm, and

3) A minimum solution temperature of 145°F.

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

1) A minimum indicated borated water volume of 20,000 gallons,

2) A minimum boron concentration of 1950 ppm, and

3) A minimum solution temperature of 39°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the indicated borated water volume, and

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

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SURVEILLANCE REQUIREMENTS (Continued)

b. By verifying the RWST temperature is above its limit whenever the outside air temperature is less than 39°F at the following frequencies:

1) Within one hour when the outside temperature is below 39°F for 23 consecutive hours, and

2) At least once per 24 hours when the outside temperature is below 39°F.

FD

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume of 3080 gallons,

2) A boron concentration between 20,000 ppm and 22,500 ppm, and

3) A minimum solution temperature of 145°F.

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

1) A minimum indicated borated water volume of 320,000 gallons,

2) A minimum boron concentration of 1950 ppm,

3) A minimum solution temperature of 39°F, and

4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.5 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the indicated borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for 23 consecutive hours, and

2) At least once per 24 hours while the outside temperature exceeds its limits.

FD

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least two independent channels of heat tracing shall be OPERABLE for the boric acid storage tank and for the heat traced portions of the associated flow paths required by Specification 3.1.2.2.

APPLICABILITY: MODES 1, 2, 3 and 4
MODES 5 and 6 (when the boric acid storage tank is the borated water source per Specification 3.1.2.4)

ACTION:

MODES 1, 2, 3 and 4

With only one channel of heat tracing on either the boric acid storage tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6

With only one channel of heat tracing on either the boric acid storage tank or on the heat traced portion of an associated flow path OPERABLE, operations involving CORE ALTERATIONS or positive reactivity additions may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to 145°F at least once per 8 hours; otherwise, suspend all activities involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each heat tracing channel for the boric acid storage tank and associated flow path required by Specification 3.1.2.2 shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 7 days by verifying the tank and flow path temperatures to be greater than or equal to 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

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 (Analog Rod Position Indication) of the group step counter demand position within one hour after rod motion.

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 more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (Analog Rod Position Indication), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than ± 12 steps (Analog Rod Position Indication), POWER OPERATION may continue provided that within one hour either:

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

   2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-2; 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:

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.c and 3.1.3.1.c.3.d below are demonstrated, and

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

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) 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.

SURVIELLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within ± 12 steps (Analog Rod Position Indication) of the group step counter demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time invervals 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 31 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)

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Analog Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the respective actual and demanded shutdown and control rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

   All Shutdown Banks: ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

   Control Bank A and B: ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

   Control Banks C and D: ± 12 steps of the group demand counters for withdrawal range of 0-228 steps.

b. Group demand counters; ± 2 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating 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 75% of RATED THERMAL POWER within 8 hours.

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

   1. Verify that all analog 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 75% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Analog Rod Position Indication System agree within 12 steps (allowing for one hour thermal soak after rod motion) 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 Analog Rod Position Indication System at least once per 4 hours.

4.1.3.2.2 Each of the above required analog rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST performed in accordance with Table 4.1-1.

TABLE 4.1-1

ROD POSITION INDICATOR SURVEILLANCE REQUIREMENTS

Functional Unit	Check	Calibration	Operational Test
Individual Rod Position	S	R	M
Demand Position	S	N/A	R

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

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

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

4.1.3.3.2 A CHANNEL CHECK CALIBRATION AND ANALOG CHANNEL OPERATIONAL TEST shall be performed per Table 4.1-1.

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

#See Special Test Exceptions Specification 3.10.4.

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 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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.

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 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

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

a. Within 15 minutes prior to withdrawal of any rods in control banks 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 1.0

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-2.

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

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

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, 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 1.0

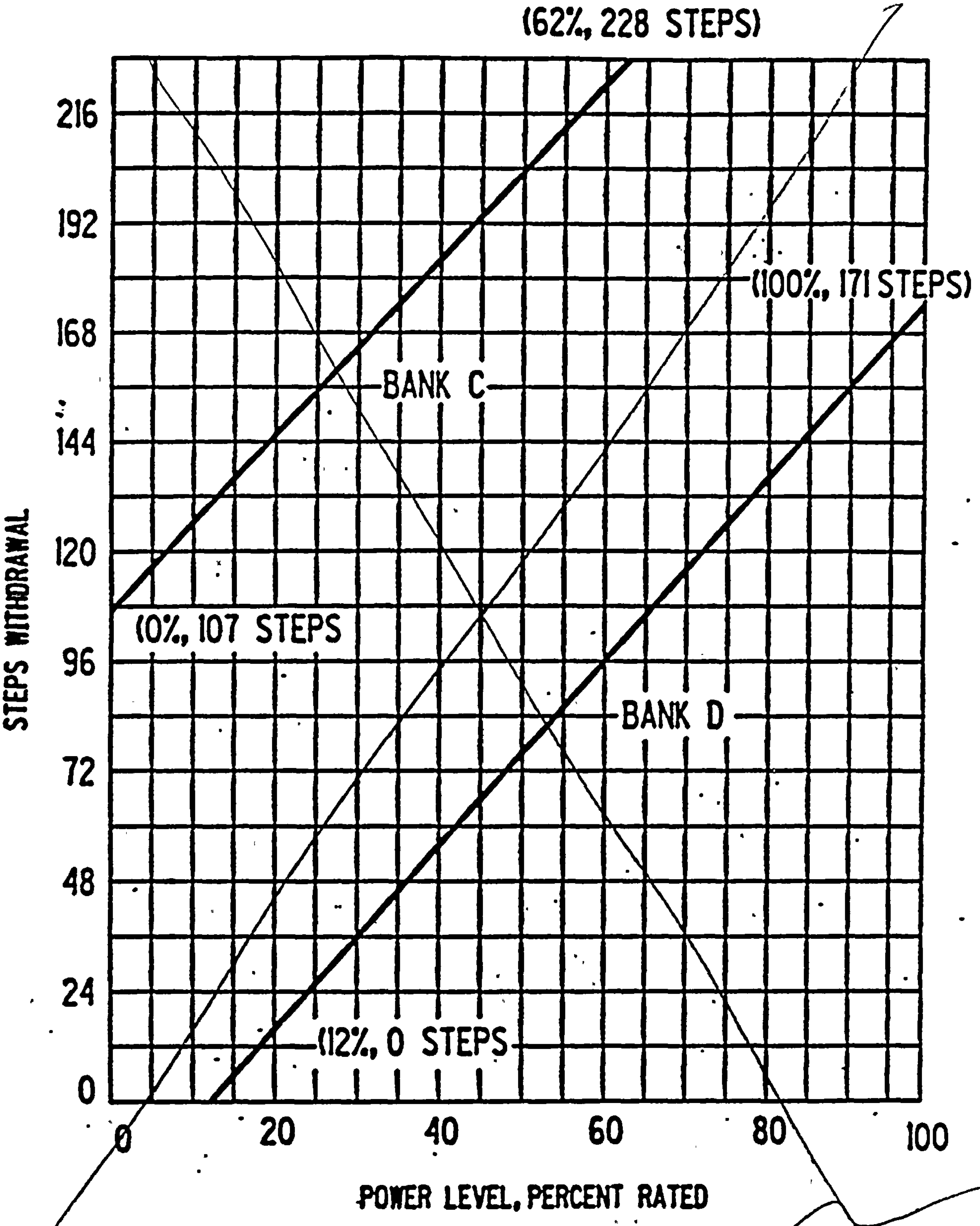

FIGURE 3.1-2

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
THREE LOOP OPERATION

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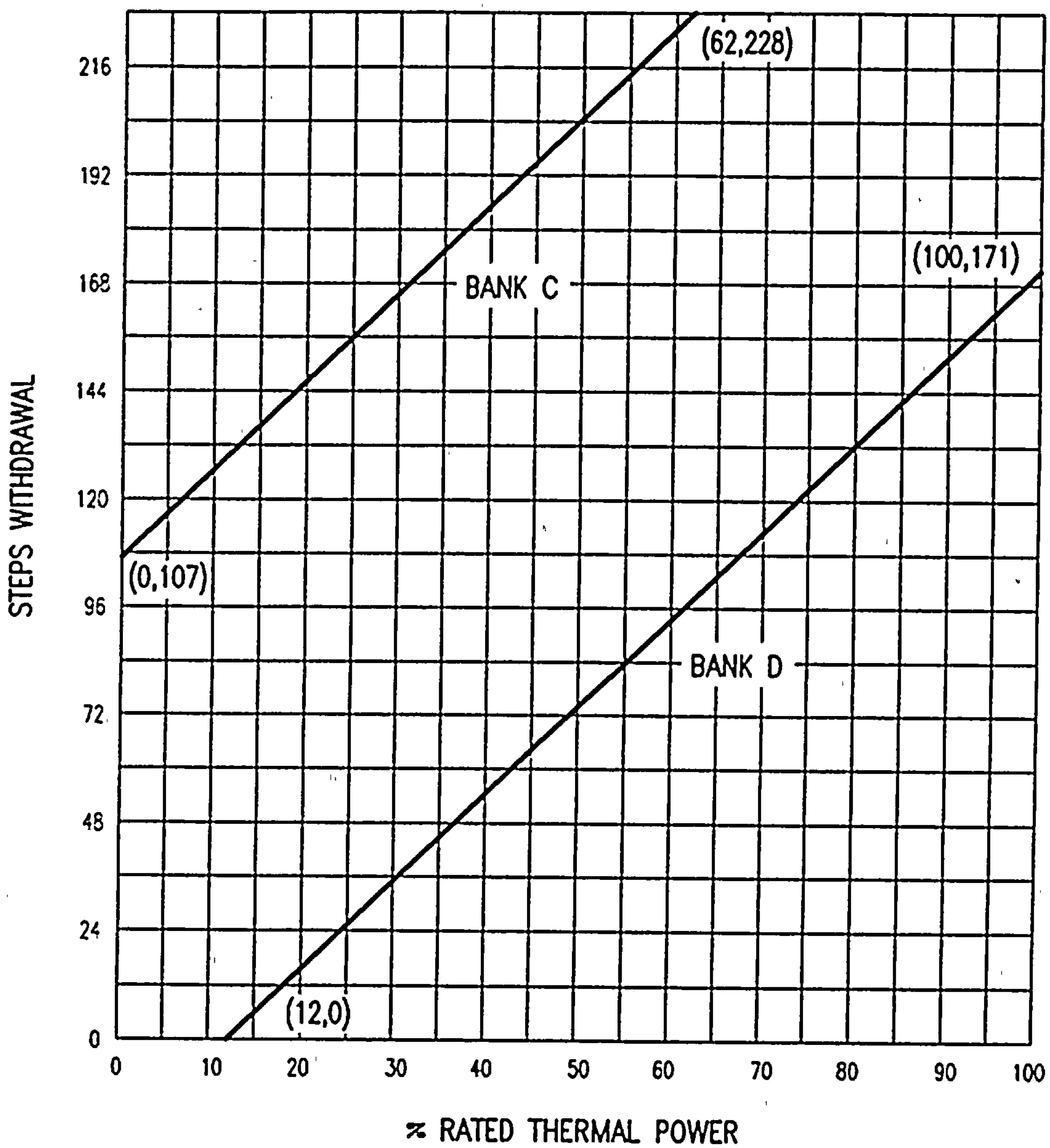

FIGURE 3.1—2 ROD BANK INSERTION LIMITS vs. % THERMAL POWER

TURKEY POINT — UNITS 3 & 4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ± 5% target band (flux difference units) about the target flux difference.

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

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

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

ACTION:

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

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

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

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

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

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

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater

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*See Special Test Exceptions Specification 3.10.2.

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

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

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

   1) At least once per 7 days when the alarm used to monitor the AFD is OPERABLE, and

   2) At least once per hour for the first 6 hours after restoring the alarm used to monitor the AFD to OPERABLE status.*

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

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

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

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

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

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

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*Performance of a functional test to demonstrate OPERABILITY of the alarm used to monitor the AFD may be substituted for this requirement.

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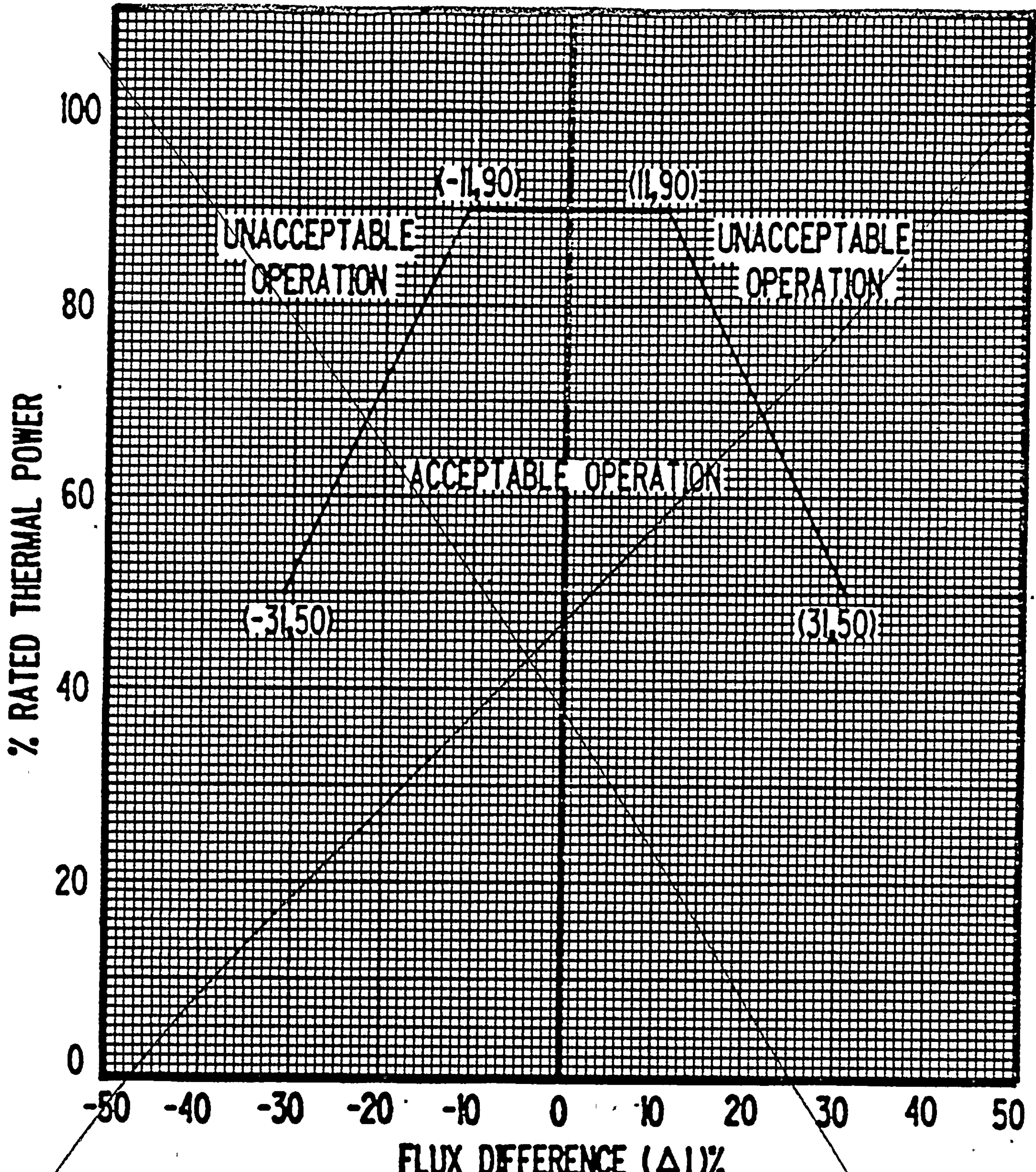

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FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

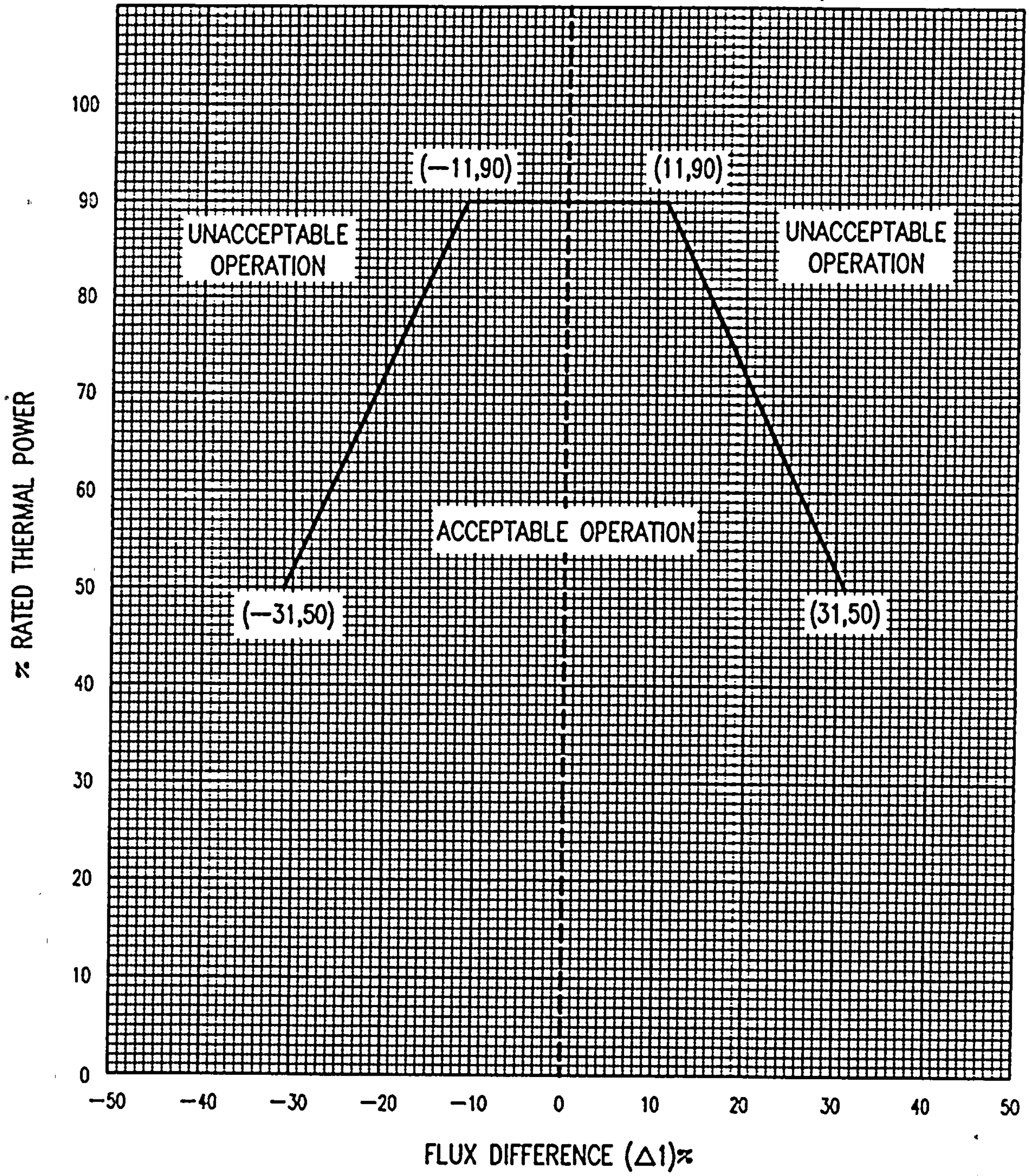

FIGURE 3.2—1 AXIAL FLUX DIFFERENCE LIMITS vs. % THERMAL POWER

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{P} X\ [K(Z)] \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{0.5} X\ [K(Z)] \text{ for } P \leq 0.5$$

where: $[F_Q]^L$ = 2.32 limit

$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$,

$[F_Q]^M$ = The Measured Value, and

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

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F_Q^M(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^M(Z)$ exceeds $F_Q^L(Z)$ 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 Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1% $F_Q^M(Z)$ exceeds the $F_Q^L(Z)$; and

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

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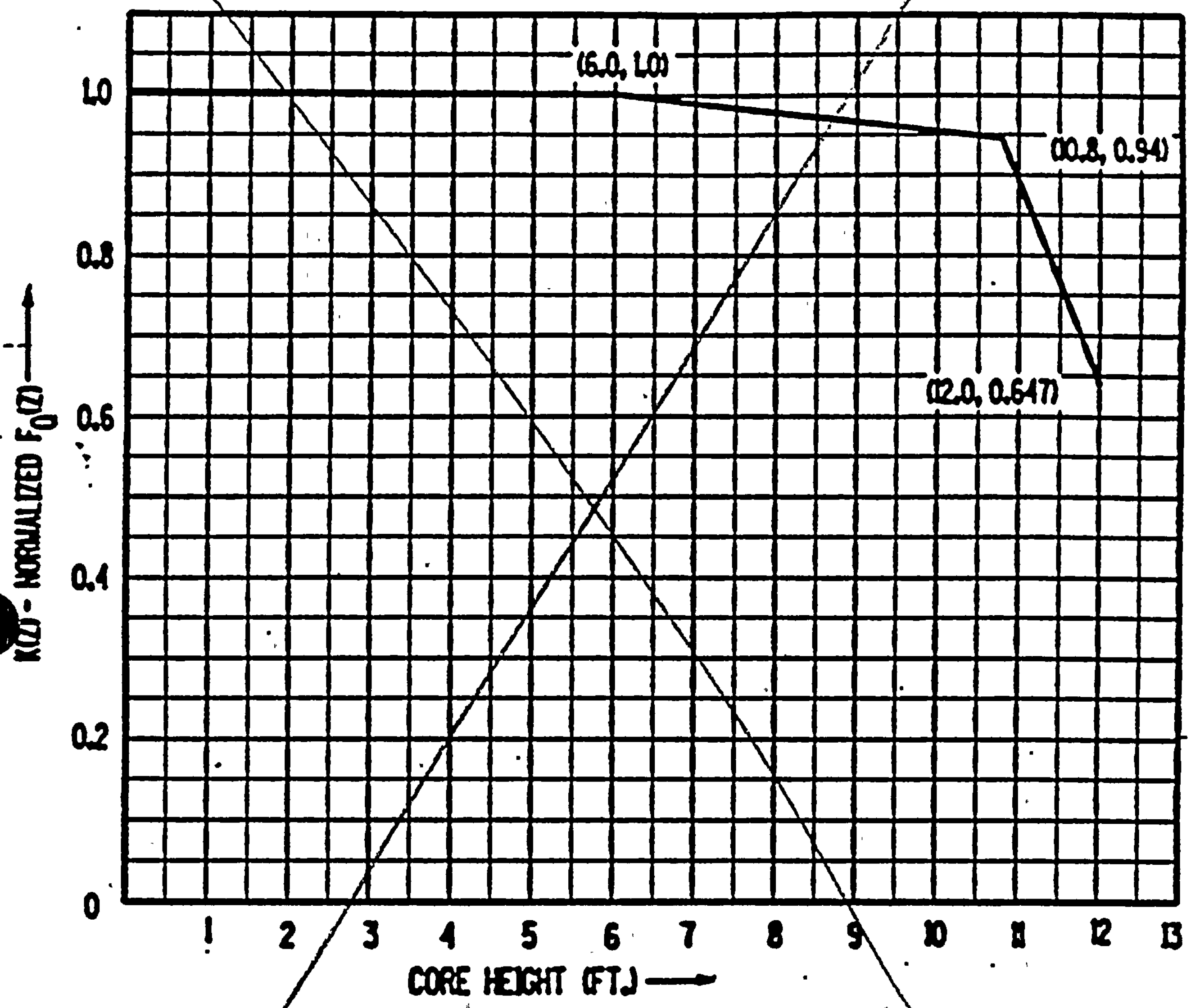

FIGURE 3.2-2
K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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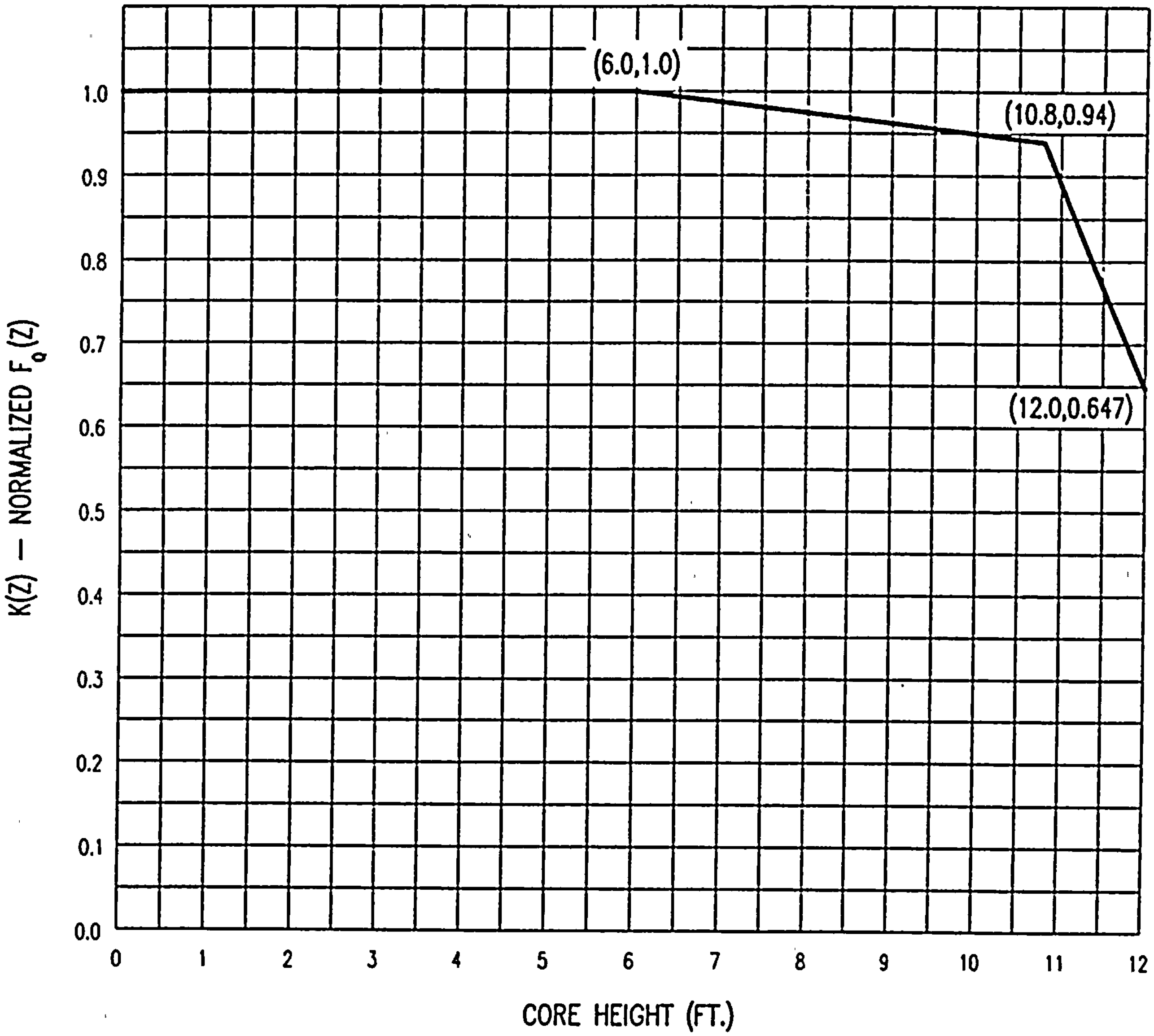

FIGURE 3.2—2 K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

.2.2.1 If $[F_Q]^P$ as predicted by approved physics calculations is greater than $[F_Q]^L$ and P is greater than P_T* as defined in 4.2.2.2, $F_Q(Z)$ shall be evaluated by MIDS (Specification 4.2.2.2), BASE LOAD (Specification 4.2.2.3) or RADIAL BURNDOWN (Specification 4.2.2.4) to determine if F_Q is within its limit $[F_Q]^P$ = Predicted F_Q).

If $[F_Q]^P$, is less than $[F_Q]^L$ or P is less than P_T, $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit as follows:

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

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verifying that the requirements of Specification 3.2.2 are satisfied.

c. $F_Q^M(Z) \leq F_Q^L(Z)$

Where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowance for manufacturing tolerances and measurement uncertainty and $F_Q^L(Z)$ is the F_Q limit defined in 3.2.2.

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

1. Prior to exceeding 75% of RATED THERMAL POWER,** after refueling,

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

e. With the relationship specified in Specification 4.2.2.1.c above not being satisfied:

1) Calculate the percent $F_Q^M(Z)$ exceeds its limit by the following expression:

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

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

---

*P_T = Reactor power level at which predicted F_Q would exceed its limit.

*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 obtained.

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2) The following action shall be taken:

   a) Comply with the requirements of Specification 3.2.2 for $F_Q^M(Z)$ exceeding its limit by the percent calculated above.

4.2.2.2 MIDS

Operation is permitted at power above P_T where P_T equals the ratio of $[F_Q]^L$ divided by $[F_Q]^P$ if the following Augmented Surveillance (Movable Incore Detection System, MIDS) requirements are satisfied:

a. The axial power distribution shall be measured by MIDS when required such that the limit of $[F_Q]^L/P$ times Figure 3.2.2 is not exceeded. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation (Z).

   1. If $F_j(Z)$ exceeds $[F_j(Z)]_s$* as defined in the bases by $\leq$ 4%, immediately reduce thermal power one percent for every percent by which $[F_j(Z)]_s$ is exceeded.

   2. If $F_j(Z)$ exceeds $[F_j(Z)]_s$ by > 4% immediately reduce thermal power below P_T. Corrective action to reduce $F_j(Z)$ below the limit will permit return to thermal power not to exceed current P_L** as defined in the bases.

b. $F_j(Z)$ shall be determined to be within limits by using MIDS to monitor the thimbles required per Specification 4.2.2.2.c at the following frequencies.

   1. At least once every 24 hours, and

   2. Immediately following and as a minimum at 2, 4 and 8 hours following the events listed below and every 24 hours thereafter.

      1) Raising the thermal power above P_T, or

      2) Movement of control-bank D more than an accumulated total of 15 steps in any one direction.

c. MIDS shall be operable when the thermal power exceeds P_T with:

   1. At least two thimbles available for which $\bar{R}_j$ and $_j$ as defined in the bases have been determined.

---

*$[F_j(Z)]_s$ is the alarm setpoint for MIDS.

**P_L is reactor thermal power expressed as a fraction of the Rated Thermal Power that is used to calculate $[F_j(Z)]_s$.

SURVEILLANCE REQUIREMENTS (Continued)

2. At least two movable detectors available for mapping $F_j(Z)$.

3. The continued accuracy and representativeness of the selected thimbles shall be verified by using the most recent flux map to update the $\bar{R}$ for each selected thimble. The flux map must be updated at least once per 31 effective full power days.

where:

$\bar{R}$ = Total peaking factor from a full flux map ratioed to the axial peaking factor in a selected thimble.

j - The thimble location selected for monitoring.

4.2.2.3 Base Load

Base Load operation is permitted at powers above P_T if the following requirements are satisfied:

a. Either of the following preconditions for Base Load operation must be satisfied.

1. For entering Base Load operation with power less than P_T,

   a) Maintain THERMAL POWER between $P_T/1.05$ and P_T for at least 24 hours,

   b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24-hour period.

   c) After 24 hours have elapsed, take a full core flux map to determine $F_Q^M(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

   d) Calculate P_{BL} per 4.2.2.3b.

2. For entering Base Load operation with power greater than P_T,

   a) Maintain THERMAL POWER between P_T and the power limit determined in 4.2.2.2 for at least 24 hours, and maintain Augmented Surveillance requirements of 4.2.2.2 during this period.

   b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24-hour period,

c) After 24 hours have elapsed, take a full core flux map to determine $F_Q^M(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

d) Calculate P_{BL} per 4.2.2.3b.

b. Base Load operation is permitted provided:

1. THERMAL POWER is maintained between P_T and P_{BL} or between P_T and 100% (whichever is most limiting).

2. AFD (Delta-I) is maintained within a ± 2% or ± 3% target band.

3. Full core flux maps are taken at least once per 31 effective Full Power Days.

P_{BL} and P_T are defined as:

$$P_{BL} = \frac{[F_Q]^L \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL} \times 1.09}$$

$$P_T = [F_Q]^L/[F_Q]^P$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ with no allowance for manufacturing tolerances or measurement uncertainty. For the purpose of this Specification $[F_Q^M(Z)]$ shall be obtained between elevations bounded by 10% and 90% of the active core height. $[F_Q]^L$ is the F_Q limit. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation.

The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6. The 9% uncertainty factor accounts for manufacturing tolerance, measurement error, rod bow and any burnup and power dependent peaking factor increases.

c. During Base Load operation, if the THERMAL POWER is decreased below P_T, then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

d. If any of the conditions of 4.2.2.3b are not maintained, reduce THERMAL POWER to less than or equal to P_T, or, within 15 minutes initiate the Augmented Surveillance (MIDS) requirements of 4.2.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

.2.2.4 RADIAL BURNDOWN

Operation is permitted at powers above P_T if the following Radial Burndown conditions are satisfied:

a. Radial Burndown operation is restricted to use at powers between P_T and P_{RB} or P_T and 1.00 (whichever is most limiting). The maximum relative power permitted under Radial Burndown operation, P_{RB}, is equal to the minimum value of the ratio of $[F_Q^L(Z)]/[F_Q(Z)]_{RB}$ Meas.

where: $[F_Q(Z)]_{RB\ Meas.} = [F_{xy}(Z)]_{Map}$ Meas. x $F_z(Z)$ x 1.09 and $[F_Q^L(Z)]$ is equal to $[F_Q^L]$ x K(Z).

-- b. A full core flux map to determine $[F_{xy}(Z)]_{Map}$ Meas. shall be taken within the time period specified in Section 4.2.2.1d.2. For the purpose of the specification, $[F_{xy}(Z)]_{Map}$ Meas. shall be obtained between the elevations bounded by 10% and 90% of the active core height.

c. The function $F_z(Z)$, provided in the Peaking Factor Limit Report (6.9.1.6), is determined analytically and accounts for the most perturbed axial power shapes which can occur under axial power distribution control. The uncertainty factor of 9% accounts for manufacturing tolerances, measurement error, rod bow, and any burnup dependent peaking factor increases.

d. Radial Burndown operation may be utilized at powers between P_T and P_{RB}, or, P_T and 1.00 (whichever is most limiting) provided that the AFD (Delta-I) is within ± 5% of the target axial offset.

e. If the requirements of Section 4.2.2.4d are not maintained, then the power shall be reduced to less than or equal to P_T, or within 15 minutes Augmented Surveillance of hot channel factors shall be initiated if the power is above P_T.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specifications 4.2.2.1, 4.2.2.2, 4.2.2.3 or 4.2.2.4 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.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.62\ [1.0 + 0.3\ (1-P)],$$

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

1. Restore $F^N_{\Delta H}$ to within the above limit, 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.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, 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 $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit 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.

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SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 When a measurement of $F^N_{\Delta H}$ is taken, the measured $F^N_{\Delta H}$ shall be increased by 4% to account for measurement error.

4.2.3.3 This corrected $F^N_{\Delta H}$ shall be determined to be within its limit through incore flux mapping:

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.

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.

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*See Special Test Exceptions Specification 3.10.2.

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'IMITING CONDITION FOR OPERATION (Continued)

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.

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.

d. The provisions of Specification 3.0.4 are not applicable.

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 Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either 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 either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report in accordance with 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

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3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg} $\leq$ 576.6°F

b. Pressurizer Pressure $\geq$ 2209 psig*, and

c. Reactor Coolant System Flow $\geq$277,900 gpm

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

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

4.2.5.2 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS flow rate shall be demonstrated by measurement once per 18 months.

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

MITING 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.

JRVEILLANCE 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 Requirement specified in Table 4.3-1.

TABLE

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	6
6. Overpower ΔT	3	2	2	1, 2	6
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6
10. 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	2/loop	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
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6
12. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coin-cident with 1 stm./feed-water flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	12
14. Underfrequency-Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2/bus	1	11
15. Turbine Trip (Above P-7)					
a. Autostop Oil Pressure	3	2	2	1	12
b. Turbine Stop Valve Closure	2	2	2	1	12

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine First Stage Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
B. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

**When the Reactor Trip System breakers are in the open position, one or both of the backup NIS instrumentation channels may be used to satisfy this requirement. For backup NIS testing requirements, see Specification 3/4.3.3.3, ACCIDENT MONITORING.

***Reactor Coolant Pump breaker A is tripped by underfrequency sensor UF-3A1 (UF-4A1) or UF-3B1(UF-4B1). Reactor Coolant Pump breakers B and C are tripped by underfrequency sensor UF-3A2(UF-4A2) or UF-3B2(UF-4B2).

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

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

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 1 hour,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per 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 per Specification 4.2.4.2.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, 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 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER 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 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

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

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement; be in at least HOT STANDBY within 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

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

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
12. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.	1, 2
13. Undervoltage - 4.16 kV Busses A and B	N.A.	R	N.A.	N.A.	N.A.	1
14. Underfrequency - Trip of Reactor Coolant Pump Breakers(s) Open	N.A.	R	N.A.	N.A.	N.A.	1
15. Turbine Trip						
a. Autostop 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. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7 (includes P-10 input and Turbine First Stage Pressure)	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1

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
17. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7,14)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(13),R(15)	N.A.	1, 2, 3*, 4*, 5*

[illegible]

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.

(1) If not performed in previous 7 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) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. 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) With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Monthly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of 1/2 decade above the existing count rate.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the undervoltage and shunt trip attachment of the Reactor Trip Breakers.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(13) Remote manual undervoltage trip when breaker placed in service.

(14) Interlock Logic Test shall consist of verifying that the interlock is in its required state by observing the permissive annunciator window.

(15) Automatic undervoltage trip.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water).					
a. Manual Initiation	2	1	2	1, 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3*	15

E 3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam line in any two steam lines	1/steam generator in any two steam lines	1, 2, 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	15
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

E 3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation.)				
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions requirements.				
4) Containment Radioactivity-High	2##	1	1	1, 2, 3, 4	16
4. Steam Line Isolation					
a. Manual Initiation (individual)	1/operating steam line	1/operating steam line	1/operating steam line	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15

3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line	1/steam line	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	15
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level--Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	15
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of All Main Feedwater Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D (2 instantaneous relays per load center) Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C 4D (2 inverse time relays per load center) Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4,6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radio-activity--High	2	1	1	1, 2, 3, 4,6**	16
d. Containment Isolation Manual Phase A or Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

TABLE 3.3-2 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the Pressurizer Pressure Interlock Setpoint of 2000 psig.

##Channels are for particulate radioactivity and for gaseous radioactivity.

###Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

*Trip function may be blocked in this MODE below the T_{avg}--Low Interlock Setpoint.

**Only during CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - 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 or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With less than the Minimum Channels OPERABLE requirement, comply with the ACTION statement requirements of Specification 3.3.3.1 Item 1a of Table 3.3-4.

ACTION 17 - 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.

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 19 - 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 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

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

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the control room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE#
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)		
a. Manual Initiation	N.A.	N.A.
-- b. Automatic Actuation Logic	N.A.	N.A.
c. Containment Pressure--High	$\leq$6 psig	$\leq$[ ] psig
d. Pressurizer Pressure--Low	$\geq$1715 psig	$\geq$[ ] psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	$\leq$150 psi	$\leq$[ ] psi
f. Steam Line Flow--High	$\leq$A function defined as follows: A Δp corresponding to 0.64×10^6 lbs/hr at 0% load increaing linearly to a Δp corresponding to 3.84×10^6 lbs/hr at full load.	[ ]
Coincident with: Steam Generator Pressure--Low	$\geq$600 psig	$\geq$[ ] psig
or T_{avg}--Low	$\geq$531°F	$\geq$[ ]°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High Coincident with:	$\leq$30.0 psig	$\leq$[ ] psig
Containment Pressure--High	$\leq$6.0 psig	$\leq$[ ] psig

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE#
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
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-High	$\leq$30.0 psig	$\leq$[ ] psig
Coincident with: Containment Pressure--High	$\leq$6.0 psig	$\leq$[ ] psig
c. Containment Ventilation Isolation		
1) Containment Isolation Manual Phase A or Phase B	N.A.	N.A.
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.	
4) Containment Radio-activity--High (1)	Particulate (R-11) <6.1 x 10^5 CPM Gaseous (R-12) See (2)	[ ]

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE#
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-High Coincident with:	≤30.0 psig	≤[ ] psig
Containment Pressure--High	≤6.0 psig	≤[ ] psig
f. Steam Line Flow--High	≤A function defined as follows: A Δp corresponding to 0.64×10^6 lbs/hr at 0% load increaing linearly to a Δp corresponding to 3.84×10^6 lbs/hr at full load.	[ ]
Coincident with: Steam Line Pressure--Low	≥600 psig	≥[ ] psig
or T_{avg}--Low	≥531°F	≥[ ]°F
5. Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥15% of narrow range instrument span.	≥[ ]% of narrow range instrument span.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE#
6. Auxiliary Feedwater (Continued)			
d. Bus Stripping		See Item 7. below for all Bus Stripping Setpoints and Allowable Values.	
e. Trip of All Main Feedwater Pump Breakers.		N.A.	N.A.
7. Loss of Power			
a. 4.16 kV Busses A and B (Loss of Voltage)		N.A.	N.A.
b. 480V Load Centers (Instantaneous Relays) Degraded Voltage			
Load Center			
	3A	436V±5V (10 sec ± 1 sec delay)	[ ]
	3B	416V±5V (10 sec ± 1 sec delay)	[ ]
	3C	417V±5V (10 sec ± 1 sec delay)	[ ]
	3D	428V±5V (10 sec ± 1 sec delay)	[ ]
	4A	415V±5V (10 sec ± 1 sec delay)	[ ]
	4B	414V±5V (10 sec ± 1 sec delay)	[ ]
	4C	401V±5V (10 sec ± 1 sec delay)	[ ]
	4D	403V±5V (10 sec ± 1 sec delay)	[ ]
Coincident with: Safety Injection		See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE#
7. Loss of Power (Continued)		
c. 480V Load Centers (Inverse Time Relays) Degraded Voltage		
Load Center		
3A	419V±5V(60 sec ±30 sec delay)	[ ]
3B	426V±5V(60 sec ±30 sec delay)	[ ]
3C	427V±5V(60 sec ±30 sec delay)	[ ]
3D	436V±5V(60 sec ±30 sec delay)	[ ]
4A	427V±5V(60 sec ±30 sec delay)	[ ]
4B	424V±5V(60 sec ±30 sec delay)	[ ]
4C	413V±5V(60 sec ±30 sec delay)	[ ]
4D	412V±5V(60 sec ±30 sec delay)	[ ]
8. Engineering Safety Features Actuation System Interlocks		
a. Pressurizer Pressure	$\leq$2000 psig	$\leq$[ ] psig
b. T_{avg}--Low	$\geq$531°F	[ ]
9. Control Room Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE#
9. Control Room Isolation (Continued)		
c. Containment Radioactivity--High (1)	Particulate (R-11) <6.1 x 10^5 CPM Gaseous (R-12) See (2)	[ ]
d. Containment Isolation Manual Phase A or Phase B	N.A.	N.A.
e. Air Intake Radiation Level	$\leq$ 2 mR/hr	~~[ ]~~ 2.83 mR/hr

TABLE NOTATIONS

(1) Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in Specification 3.11.2.1.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

#If no allowable value is specified so indicated by [ ], the trip setpoint shall also be the allowable value.

4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL 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. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)						
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(4)
c. Containment Pressure--High	N.A.	R	N.A.	~~Q M~~ N.A.	M(1)	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M(6)	N.A.	N.A.	1, 2, 3(4)
e. High Differential Pressure Between the Steam Line Header and any Steam Line	S	R	M(6)	N.A.	N.A.	1, 2, 3(4)
f. Steam Line Flow--High Coincident with:	S	R	M(6)	N.A.	N.A.	1, 2, 3(4)
Steam Generator Pressure--Low	S	R	M(6)	N.A.	N.A.	1, 2, 3(4)
or						
T_{avg}--Low	S	R	M(6)	N.A.	N.A.	1, 2, 3(4)

TABLE 4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2. Containment Spray						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
3. Containment Isolation						
a. Phase "A" Isolation						
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	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.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3, 4

TABLE 4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)						
3) Containment Pressure--High-High Coincident	N.A.	R	N.A.	R	M(1)	1, 2, 3
with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
c. Containment Ventilation Isolation						
1) Containment Isolation Manual Phase A or Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A. ~~M(1)~~	~~1, 2, 3, 4~~
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
4) Containment Radioactivity--High	S	R	M	N.A.	N.A.	1, 2, 3, 4
4. Steam Line Isolation						
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(4)

TABLE 4[illegible] (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation (Continued)						
c. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	R	M(1)	1, 2, 3
d. Steam Line Flow--High	S(4)	R	M(6)	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	S(4)	R	M(6)	N.A.	N.A.	1, 2, 3
or T_{avg}--Low	S(4)	R	M(6)	N.A.	N.A.	1, 2, 3
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
6. Auxiliary Feedwater (2)						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2, 3

TABLE 4.[illegible] (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)						
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
d. Bus Stripping	N.A.	R	N.A.	R	N.A.	1, 2, 3
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A.	N.A.	R	N.A.	1, 2
7. Loss of Power						
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	1, 2, 3, 4
b. 480V Load Centers 3A,3B,3C,3D and 4A,4B,4C,4D (Instantaneous Relays) Degraded Voltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. 480V Load Centers 3A,3B,3C,3D and 4A,4B,4C,4D (Inverse Time Relays) Degraded Voltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST#	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Engineering Safety Features Actuation System Interlocks						
a. Pressurizer Pressure	N.A.	R	M(6)	N.A.	N.A.	1, 2, 3(4)
b. T_{avg}--Low	N.A.	R	M(6)	N.A.	N.A.	1, 2, 3(4)
9. Control Room Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	(3)
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Containment Radioactivity--High	S	R	M	N.A.	N.A.	(5)
d. Containment Isolation Manual Phase A or Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	S	R	M	N.A.	N.A.	All

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.
(3) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment or in the spent fuel pool.
(4) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.
(5) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.
(6) Test of alarm function not required when alarm locked in.

#At least once per 18 months each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

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3/4.3.3 MONITORING INSTRUMENTATION

ADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, 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 3.3-4.

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 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $<6.1\times10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 MODES 5 AND 6
b. RCS Leakage Detection Particulate Radioactivity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity - High Gaseous	1	1	**	$<5.5\times10^{-2}$ $\frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity-High Gaseous#	1	1	**	$<2.8\times10^{-2}$ $\frac{\mu Ci}{cc}$ (SPING) or $<1.0\times10^6$ CPM (PRMS)	28
3. Control Room Air Intake Radiation Level	1	2	All	$\leq$ ~~2~~ 2.83 mR/hr	29

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(F)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

-- Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in Specification 3.11.2.1.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment sump level monitoring system is OPERABLE,

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,

3) A Reactor Coolant System water inventory balance is performed at least once per 8 hours during steady state operation except when operating in shutdown cooling mode, and

4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (except during CORE ALTERATION or movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During CORE ALTERATION or movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pool area involving spent fuel manipulations.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

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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. Containment Atmosphere Radiaoctivity--High	S	R	M	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	S	R	M	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M	1, 2, 3, 4
2. Spent Fuel Pool Areas				
a. Unit 3 Radioactivity--High Gaseous	S	R	M	*
b. Unit 4 (Plant Vent) Radioactivity--High Gaseous# (SPING and PRMS)	S	R	M	*
~~3. Control Room Air Intake Radiation Level~~	~~S~~	~~R~~	~~M~~	~~All~~

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.
# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUADRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

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

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

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

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

b. Monitoring the QUADRANT POWER TILT RATIO*, or

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

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

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

b. Monitoring the QUADRANT POWER TILT RATIO*, or

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

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*Exception to the 16 detector thimble requirement of monitoring the QUADRANT POWER TILT RATIO is acceptable when performing Specification 4.2.4.2 using two sets of four symmetric thimbles.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

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SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
1.	Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2.	Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3.	Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4.	Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5.	Reactor Coolant Pressure - Wide Range	2	1	1, 2, 3	31, 32
6.	Pressurizer Water Level	2	1	1, 2, 3	31, 32
7.	Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8.	Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9.	PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10.	PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12.	Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13.	Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
14.	In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core	1, 2, 3	31, 32
15.	Containment High Range Area Radiation	2	1	1, 2, 3	34
16.	Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17.	Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18.	Containment Hydrogen Monitors	2	1	1, 2	35
19.	High Range-Noble Gas Effluent Monitors				
	a. Plant Vent Exhaust	1	1	ALL	34
	b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
	c. Condenser Air Ejectors	1	1	1, 2, 3	34
	d. Main Steam Lines	1	1	1, 2, 3	34
20.	RWST Water Level	2	1	1, 2, 3	31, 32
21.	Steam Generator Water Level (Narrow Range)	2/stm. gen.	1/stm. gen.	1, 2, 3	31, 32
~~22.~~	~~Steam Generator Water Level (Wide Range)~~	~~1/stm. gen.~~	~~1/stm. gen.~~	~~1, 2, 3~~	~~32~~
~~23.~~ 22.	Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.
2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 — With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either 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.

ACTION 32 — With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either 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.

ACTION 33 — Close the associated block valve ~~an~~ and open its circuit breaker.

ACTION 34 — With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

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

ACTION 35 — With one or both hydrogen monitor(s) inoperable, comply with Action Requirements of Specification 3.6.5.

ACTION 36 — With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 — With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 7 days. If repairs are not feasible without shutting down, 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 system to OPERABLE status.

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TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 38 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 48 hours. If repairs are not feasible without shutting down:

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

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 system to OPERABLE status; and

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

ACTION 39 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement~~s~~, comply with the provisions of Specification 3.~~8~~ 6.4. for an inoperable containment isolation valve.

verify position by an alternate means (e.g., administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 72 hours, or

TABLE 4.3-4

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Wide Range)	M	R
2. Containment Pressure (Narrow Range)	M	R
3. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
5. Reactor Coolant Pressure - Wide Range	M	R
6. Pressurizer Water Level	M	R
7. Auxiliary Feedwater Flow Rate	M	R
8. Reactor Coolant System Subcooling Margin Monitor	M	R
9. PORV Position Indicator (Primary Detector)	M	R
10. PORV Block Valve Position Indicator	M	R
11. Safety Valve Position Indicator (Primary Detector)	M	R
12. Containment Water Level (Narrow Range)	M	R
13. Containment Water Level (Wide Range)	M	R
14. In Core Thermocouples (Core Exit Thermocouples)	M	R
15. Containment - High Range Area Radiation Monitor	M	R*
16. Reactor Vessel Level Monitoring System	M	R
17. Neutron Flux, Backup NIS (Wide Range)	M	R
18. Containment Hydrogen Monitor	M	R*
19. High Range - Noble Gas Effluent Monitors		
a. Plant Vent Exhaust	M	R
b. Unit 3 - Spent Fuel Pit Exhaust	M	R
c. Condenser Air Ejectors	M	R
d. Main Steam Lines	M	R
20. RWST Water Level	M	R
21. Steam Generator Water Level (Narrow Range)	M	R
~~22. Steam Generator Water Level (Wide Range)~~	~~M~~	~~R~~
2322. Containment Isolation Valve Position Indication	M	R

*Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-6 shall be OPERABLE.

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

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-6 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).

b. With more than one-half of the Function A fire detection instrument in any fire zone shown in Table 3.3-6 inoperable, or with any Function B fire detection instruments shown in Table 3.3-6 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).

d ~~c~~. The provisions of Specifications 3.0.3 are not applicable.

c. With the fire watch patrol not established at the 18 foot level of the turbine area, restore the fire watch patrol within one hour, or prepare and submit a Special Report to the Commission.

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 3.3-6

FIRE DETECTION INSTRUMENTS
FOR ESSENTIAL EQUIPMENT

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS		
FIRE ZONE AREA	HEAT (x/y)*	FLAME (x/y)	SMOKE (x/y)
4 - Aux. Bldg. Corridor E. 10'			(2/0)
5 - Chem. Drain/Laundry/Shower Tank Room			(2/0)
9 - Laundry/Chemical Drain Tank Room			(1/0)
10 - Pipeway			(11/0)
11 - Unit 3 RHR Heat Exchanger Room			(5/0)***
12 - RHR Pump 3A Room			(2/0)***
13 - RHR Pump 3B Room			(2/0)***
14 - Unit 4 RHR Heat Exchanger Room			(5/0)***
15 - RHR Pump 4A Room			(2/0)***
16 - RHR Pump 4B Room			(2/0)***
19 - Unit 3 W Elect Penet Room			(5/0)***
20 - Unit 3 S Elect Penet Room			(11/0)
21 - Instrument Shop			(2/0)
22 - Radioactive Laboratory			(2/0)
26 - Unit 4 N Elect Penet Room			(8/0)
27 - Unit 4 W Elect Penet Room			(6/0)
30 - Unit 4 Piping and Valve Room			(4/0)***
40 - Unit 3 Piping and Valve Room			(4/0)***
45 - Unit 4 Charging Pump Room	(0/4)		(3/0)
47 - Unit 4 Component Cooling Water Area	(0/4)	(5/2)***	
54 - Unit 3 Component Cooling Water Area	(0/4)	(4/2)***	
55 - Unit 3 Charging Pump Room	(0/4)		(3/0)
58 - Aux Bldg Corridor, El. 18'			(18/0)
59 - Unit 4 Containment Electrical Penet. Area**			(10/0)
60 - Unit 3 Containment Electrical Penet. Area**			(16/0)
61 - Reactor Control Rod Eqpmt Room - Unit 4			(4/0)
62 - Computer Room			(11/0)
63 - Reactor Control Rod Eqpmt Room - Unit 3			(4/0)
67 - 4160V Switchgear 4B			(10/0)
68 - 4160V Switchgear 4A			(6/0)
70 - 4160V Switchgear 3B			(10/0)
71 - 4160V Switchgear 3A			(6/0)
72 - Emergency Diesel B	(0/3	(1/0)	(1/0)
73 - Emergency Diesel A	(0/3	(1/0)	(1/0)
74 - Emergency Day Tank Room B	(1/1)		
75 - Emergency Day Tank Room A	(1/1)		
76 - Unit 4 Turbine Lube Oil Reservoir	(1/0)		
79A- North-South Breezeway	(0/6)		(4/0)
81 - Unit 4 Main Transformer	(1/0)		
82 - Unit 4 Aux Transformer Area	(1/0)		
84 - Unit 3 and 4 Aux Feedwater Pump Area (DC Enclosure Bldg)			(3/0)

TABLE 3.3-6 (Continued)

FIRE DETECTION INSTRUMENTS FOR ESSENTIAL EQUIPMENT

INSTRUMENT LOCATION FIRE ZONE AREA	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)*	 FLAME (x/y)	 SMOKE (x/y)
87 - Unit 3 Aux Transformer Area	(1/0)		
93 - 480V Load Center 4A and 4B			(1/0)
94 - 480V Load Center 4C and 4D			(2/0)
95 - 480V Load Center 3A and 3B			(1/0)
96 - 480V Load Center 3C and 3D			(2/0)
7 - Mechanical Equipment Room			(1/0)
98 - Cable Spreading Room			(16/15)
101- RPI Inverter and MG Sets			(1/0)
102- Battery Rack 4B	(1/0)		
103- Battery Rack 3A	(1/0)		
104- RPI Inverter and MG Sets			(2/0)
106- Control Room	(1/0)		(17/0)
108A- Train A Inverters			(3/4)
108B- Train B Inverters			(4/4)
109- Battery Rack 4A	(1/0)		
110- Battery Rack 3B	(1/0)		
113- Unit 4 Feedwater Platform		(2/0)***	
116- Unit 3 Feedwater Platform		(2/0)***	
119- Unit 4 Intake Cooling Water Pump Area		(4/0)***	
120- Unit 3 Intake Cooling Water Pump Area		(4/0)***	
132- Control Room Electrical Chase			(1/0)
N/A - 18' level of the Turbine Area	(N/A)#	(N/A)#	(N/A)#

TABLE NOTATIONS

*(x/y): x is number of Function A (early warning fire dectection and notification only) instruments.

y is number of Function B (actuation of Fire Suppression Systems and early warning fire detection and notification) instruments.

**: The fire detection instruments located within the containment are not required to be operable during the performance of Type A Containment Leakage Rate Test.

***: Installed to meet the requirements of 10 CFR Part 50, Appendix R, Section III.G.

#: A fire watch patrol shall be established to inspect the 18 foot level of the Turbine Area once each hour.

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INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

PPLICABILITY: At all times, except as indicated in Table 3.3-7.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-7. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1*	35
b. Steam Generator Blowdown Effluent Line	1**	36
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1*	37
b. Steam Generator Blowdown Effluent Line	1**/Steam generator	37

*Applicable during liquid effluent releases.

**Applicable during blowdown operations.

TABLE 3.3-7 (Continued)

TABLE NOTATION

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross (beta or gamma) radioactivity at a lower limit of detection of no more than 1×10^{-7} microcuries/ml or analyzed isotopically (Gamma) at a lower limit of detection of at least 5×10^{-7} microcurie/ml:

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

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

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

TA 3.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluents Line	D	P	R(2)*	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. Flow Rate Measurment Devices				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R*	Q
b. Steam Generator Blowdown Effluent Lines	D(3)	N.A.	R	Q

*Channel calibration frequency shall be at least once per 18 months.

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measures levels above the Alarm/Trip Setpoint.

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

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

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INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-8

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-8. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GAS DECAY TANK SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	45
b. Effluent System Flow Rate Measuring Device	1	*	46
2. WASTE GAS DISPOSAL SYSTEM (Explosive Gas Monitoring System)			
a. Hydrogen and Oxygen Monitors	1	**	49
3. Condenser Air Ejector Vent System			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	#	47
b. Iodine Sampler	1	##	48
c. Particulate Sampler	1	##	48
d. Effluent System Flow Rate Measuring Device	1	##	46
e. Sampler Flow Rate Measuring Device	1	##	46

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TABLE 3.3-[illegible] (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Effluent System Flow Rate Measuring Device	1	*	46
e. Sampler Flow Rate Measuring Device	1	*	46
5. Unit 3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Sampler Flow Rate Measuring Device	1	*	46

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TABLE 3.3-8 (Continued)

TABLE NOTATION

* At all times.
** During GAS DECAY TANK SYSTEM operation.
\# Applies during MODE 1, 2, 3 and 4.
\## Applies during MODE 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 and analyzed at least weekly.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least a) once per 8 hours during degassing operations, and b) once per day during other operations.

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM					
a. Noble Gas Activity Monitor - Provising Alarm and Automatic Termination of Release (Plant Vent Monitor)	P	P	R(3)	Q(1)	*
b. Effluent System Flow Rate Measuring Device	P	N.A.	R	N.A.	*
2. GAS DECAY TANK SYSTEM (Explosive Gas Monitoring System)					
a. Hydrogen and Oxygen Monitors	D	N.A.	Q(4,5)	M	**
3. Condenser Air Ejector Vent System					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R(3)	Q(2)	#
b. Iodine Sampler	W	N.A.	N.A.	N.A.	##
c. Particulate Sampler	W	N.A.	N.A.	N.A.	##
d. Efflent System Flow Rate Measuring Device	D	N.A.	R	N.A.	##

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TABLE 4.[illegible] (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Condenser Air Ejector Vent System (Continued)					
e. Sample Flow Rate Measuring Device	D	N.A.	R	N.A.	##
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	(3,6)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Efflent System Flow Rate Measuring Device	D	N.A.	(6)	N.A.	*
e. Sampler Flow Rate Measureng Device	D	N.A.	(6)	N.A.	*

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TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Unit 3 Spent Fuel Pit Building Vent					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Measuring Device	D	N.A.	R	N.A.	*

TABLE NOTATION

* At all times.
** During GAS DECAY TANK SYSTEM operation.
\# Applies during MODE 1, 2, 3 and 4.
\#\# Applies during MODE 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measures levels above the Alarm/Trip Setpoint.
d

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that if the instrument indicates measured levels above the Alarm Setpoint, alarm annunciation occurs in the control room (for PRMS only) and in the computer room (for SPING only).

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

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TABLE 4.3-6 (Continued)

TABLE NOTATIONS (Continued)

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(6) CHANNEL CALIBRATION frequency shall be at least once per 18 months.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.

\CTION:

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

SUREVEILLANCE REQUIREMENTS

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

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

c. Reactor Coolant Loop C 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 three reactor coolant loop in operation and the Reactor Trip breakers in the closed position, within 1 hour open the Reactor Trip 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 10% 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 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

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

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

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

-- d. RHR Loop A, and

e. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

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

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

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

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

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REACTOR COOLANT SYSTEM

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 ~~at~~ 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 10%.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

**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.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

.HUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

--

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*While in MODE 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

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

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3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of indicated level, and at least two groups of pressurizer heaters each having a capacity of at least 125 kW and capable of being supplied by emergency power.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by demonstrating the capability to power the heaters from the emergency power.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each power-operated relief valve (PORV) block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 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 Specification 3.4.4 or is closed to provide an isolation function.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

2) Tubes in those areas where experience has indicated potential problems, 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 in the inservice inspection may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with 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. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months following replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, 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 resulting in rapid depressurization of the primary system, or

   4) A main steam line or feedwater line break resulting in rapid depressurization of the affected steam generator.

SURVEILLANCE REQUIREMENTS (Continued)

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 because it may become unserviceable prior to the next inspection 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 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 from the point of entry (cold leg side) completely around the U-bend and to the bottom of the hot leg; and

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.

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

4.4.5.5 Reports

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

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

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR Part 50.72 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.

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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	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

Table Notation

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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

STEAM GENERATOR TUBE INSPECTION

SAMPLE SIZE	1st SAMPLE INSPECTION		2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
	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 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 Section 4.4.5.5c.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s 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 Section 4.4.5.5C.	N/A	N/A

$S = \frac{9}{n}\%$ Where 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. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment Sump Level Monitoring System is OPERABLE;

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;

3) A Reactor Coolant System water inventory balance is performed at least once per 8 hours during steady state operation except when operating in shutdown cooling mode; and

4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.

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

b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

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

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REACTOR COOLANT SYSTEM

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 primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

e. Leakage as specified in Table 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:

   1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

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*Test pressures less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

LIMITING CONDITION FOR OPERATION (Continued)

2. The leakage* from the remaining isolation valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily unless they are manual valves located inside containment.

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

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm 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

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

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

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

c. Performance of a Reactor Coolant System water inventory balance within 12 hours after achieving steady-state operation** and at least once per 24 hours thereafter during steady-state operation, except that not more than 48 hours shall elapse between any two successive inventory balances; and

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

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

a. At least once per 18 months.

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

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

*To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

**RCS average coolant temperature being changed by less than 5°F/hour.

LIMITING CONDITION FOR OPERATION (Continued)

d. Following valve actuation due to automatic or manual action or flow through the valve:

1. Within 24 hours by verifying valve closure, and

2. Prior to entering Mode 2 by verifying leakage rate.

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
Unit 3	Unit 4	High-Head Safety Injection Check Valves
3-874A	4-874A	Loop A, hot leg
3-875A	4-875A	cold leg
3-873A	4-873A	cold leg
3-874B	4-874B	Loop B, hot leg
3-875B	4-875B	cold leg
3-873B	4-873B	cold leg
3-875C	4-875C	Loop C, cold leg
3-873C	4-873C	cold leg
		Residual Heat Removal Line Check Valves
3-876-A	4-876A	Loop A, cold leg
	4-876E	
3-876B	4-876B	Loop B, cold leg
3-876D	4-876D	
3-876C	4-876C	Loop C, cold leg
3-876E		
MOV3-750	MOV4-750	Loop A, hot leg to RHR
MOV3-751	MOV4-751	Loop A, hot leg to RHR

ACCEPTABLE LEAKAGE LIMITS

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable provided that the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride**	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride**	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with average reactor coolant temperature less than or equal to 250°F.

**Not required when reactor is defueled and RCS forced circulation is unavailable.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least 5 times per week not to exceed 72 hours between samples
Chloride**	At least 5 times per week not to exceed 72 hours between samples
Fluoride**	At least 5 times per week not to exceed 72 hours between samples

*Not required with average reactor coolant temperature less than or equal to 250°F.

**Not required when reactor is defueled and RCS forced circulation is unavailable.

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REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

b. Less than or equal to 100/Ē 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 average reactor coolant temperature less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than 100/Ē microcurie per gram, be in at least HOT STANDBY with average reactor coolant temperature 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/Ē microcuries per gram, perform the sampling and analysis requirements of Item 6.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

*With the average reactor coolant temperature greater than or equal to 500°F.

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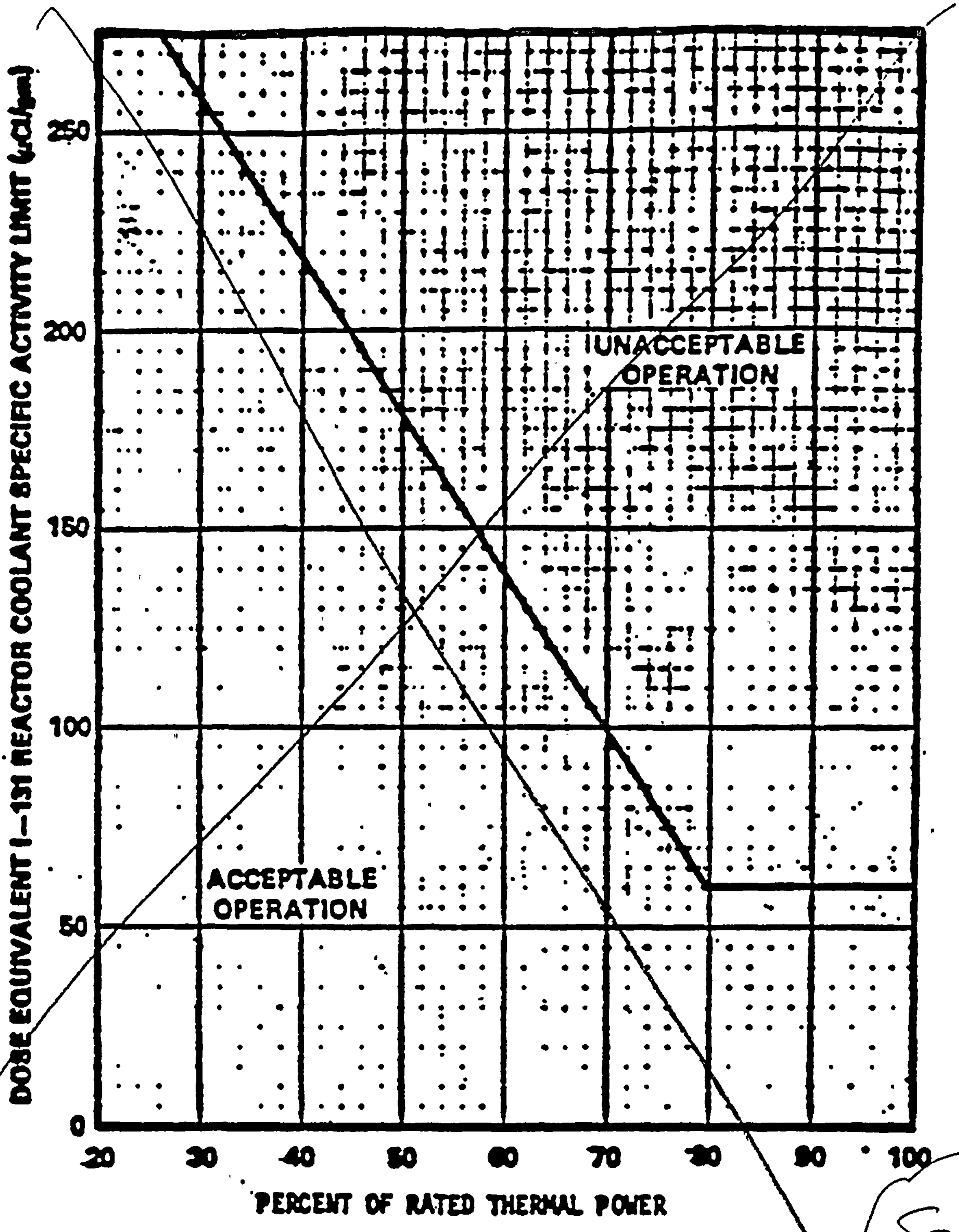

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FIGURE 3.4-1

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

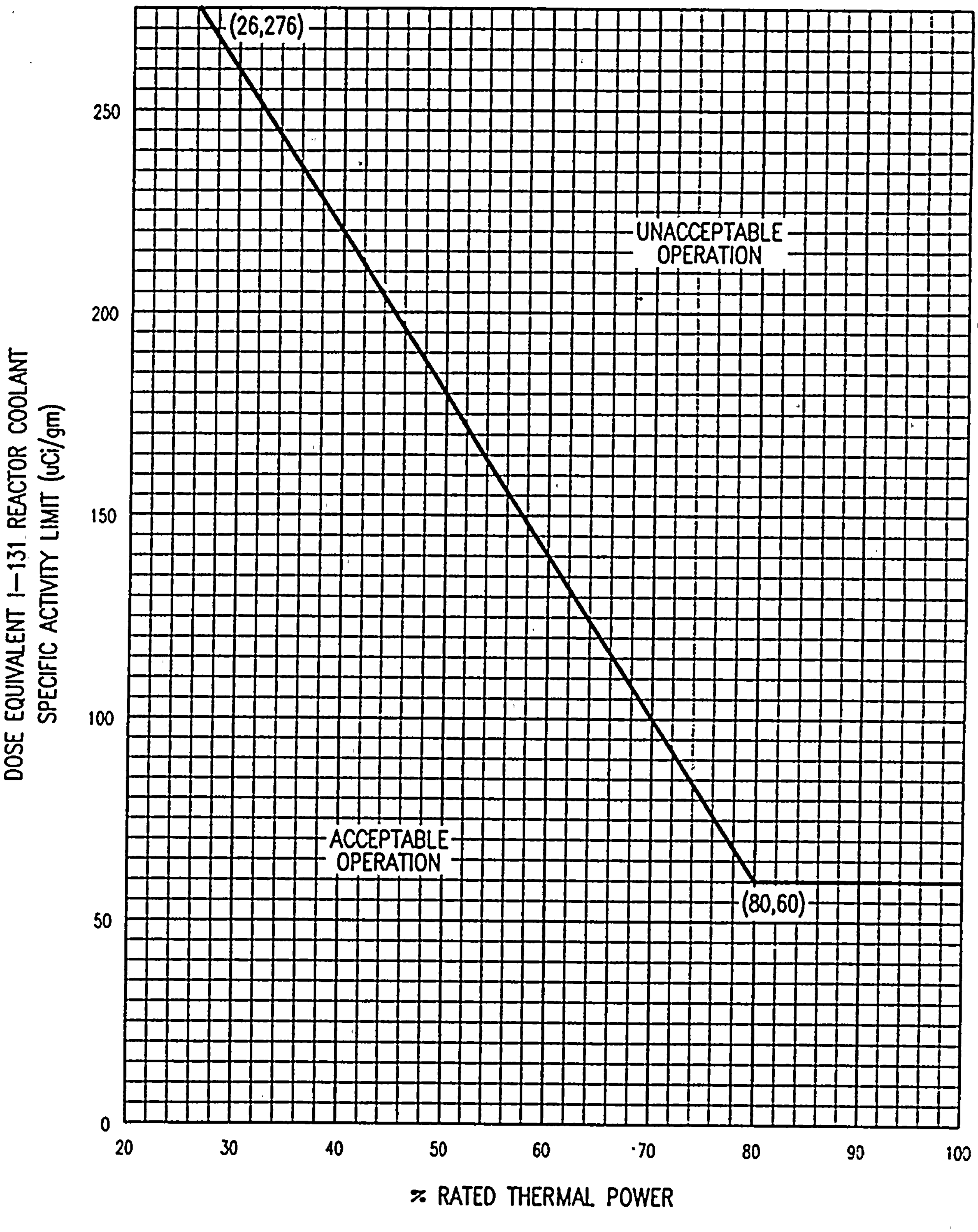

FIGURE 3.4—1
DOSE EQUIVALENT I—131 Rx COOLANT SPECIFIC ACTIVITY LIMIT vs % THERMAL POWER
(WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1.0uCi/gram DOSE EQUIVALENT I—131.)

T[illegible] 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT ANY 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. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
6. 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

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

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

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

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

c. A maximum temperature change of less than or equal to 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

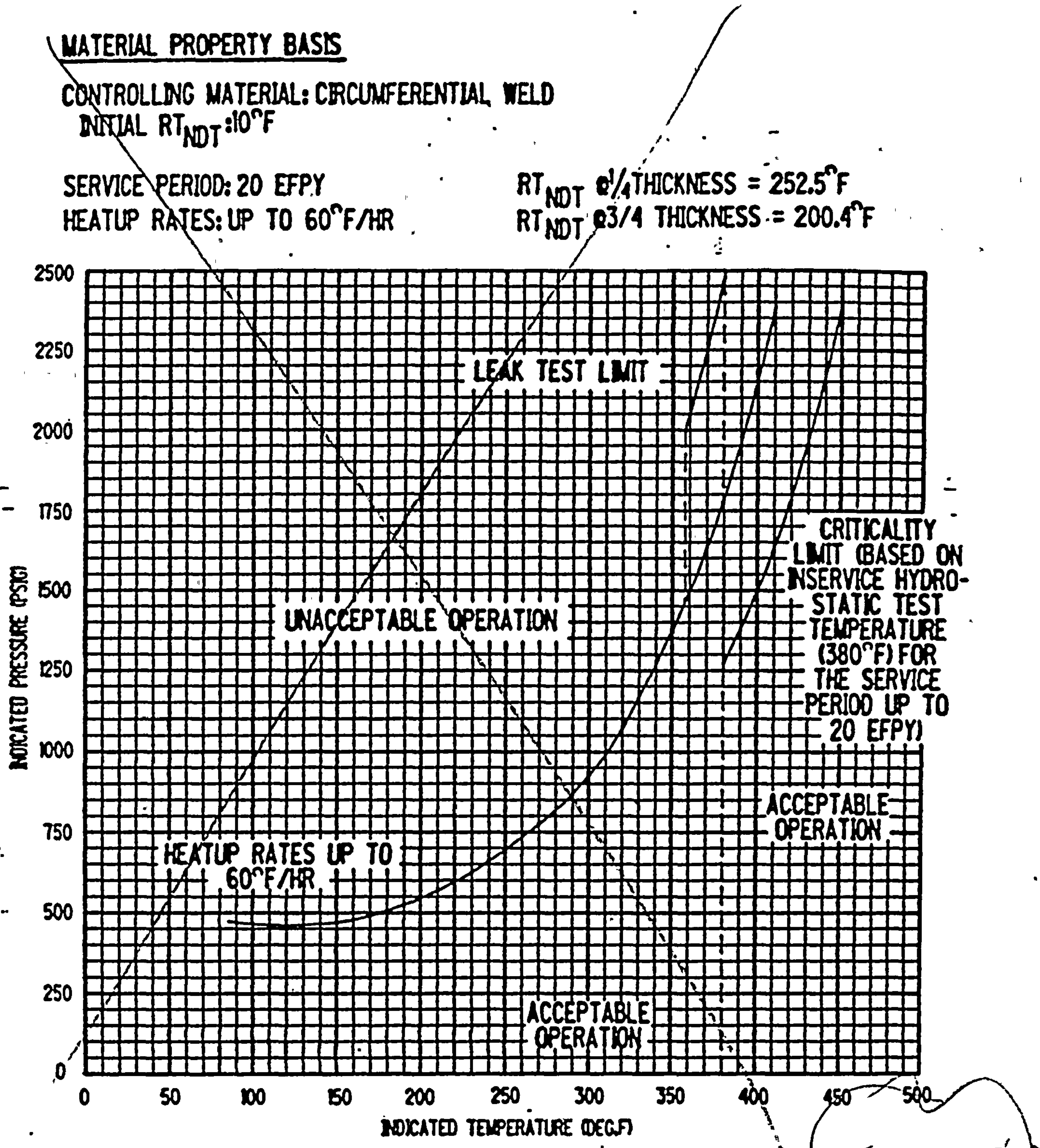

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FIGURE 3.4-2

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (60°F/hr) - APPLICABLE UP TO 20 EFPY

[illegible] [illegible]

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 20 EFPY — RT_{NDT} @ 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 60°F/HR — RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

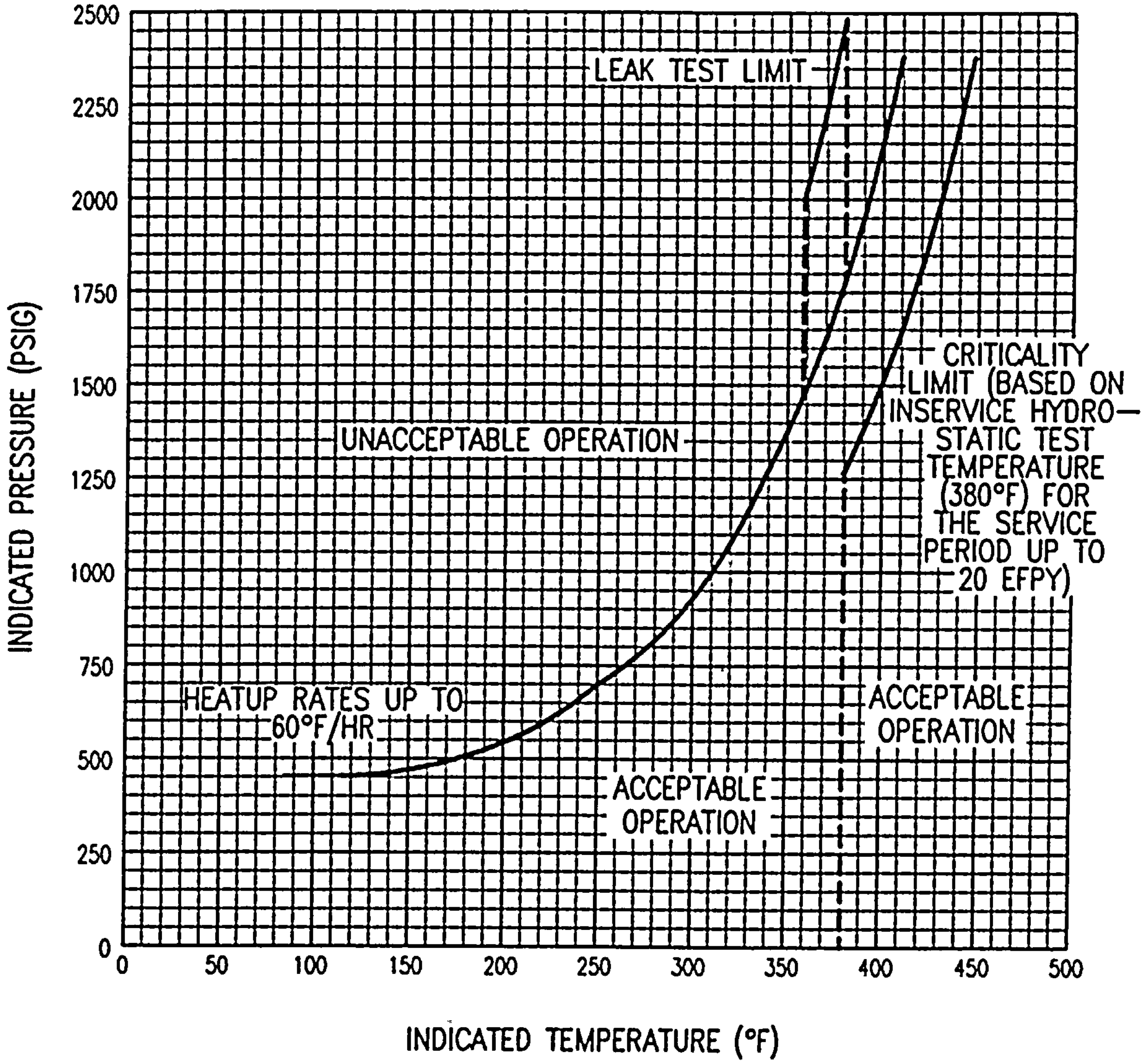

FIGURE 3.4—2 REACTOR COOLANT HEATUP LIMITATIONS (<60°F/HR)

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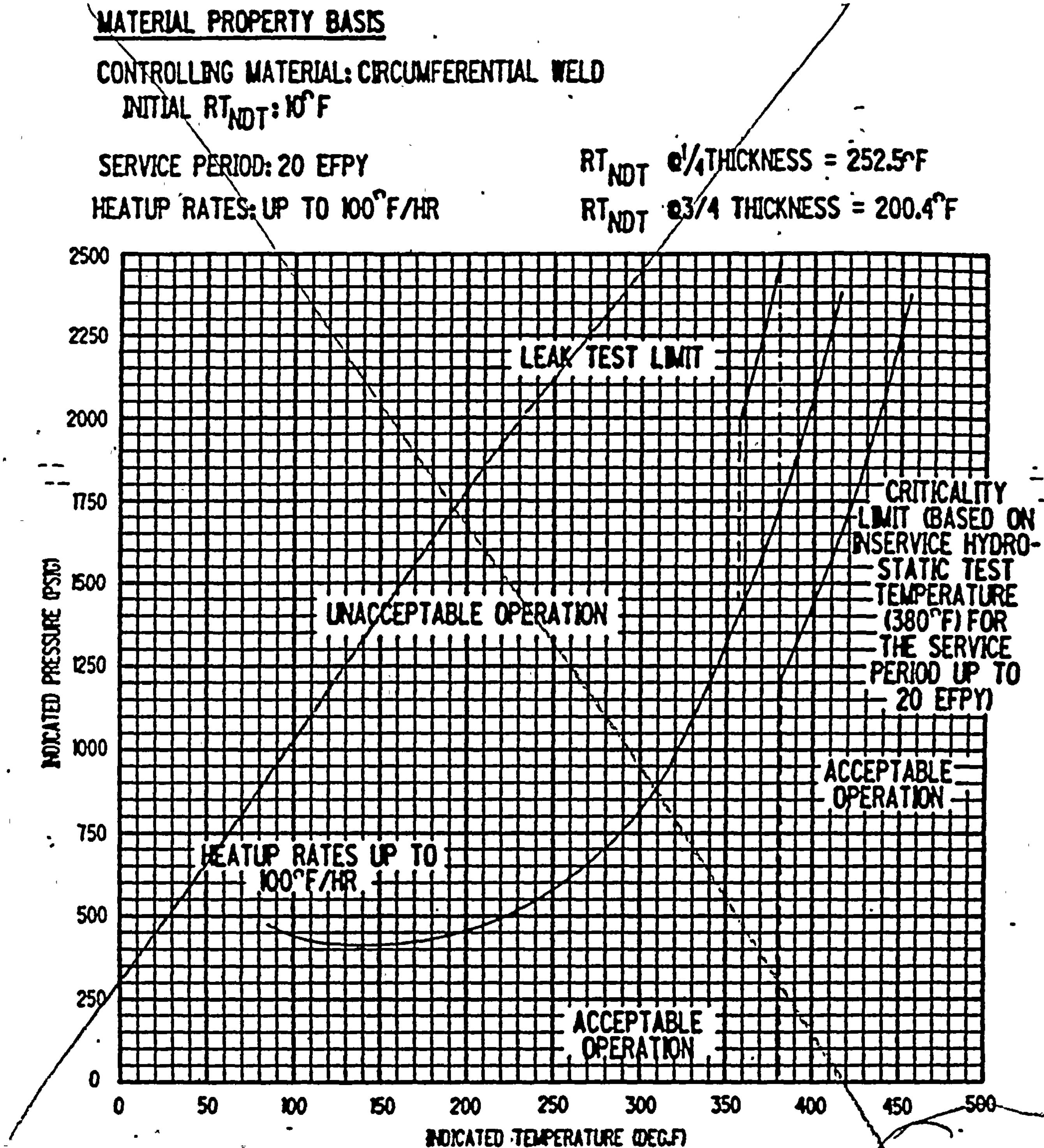

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FIGURE 3.4-3

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (100°F/hr) - APPLICABLE UP TO 20 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 20 EFPY — RT_{NDT} @ 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 100°F/HR — RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

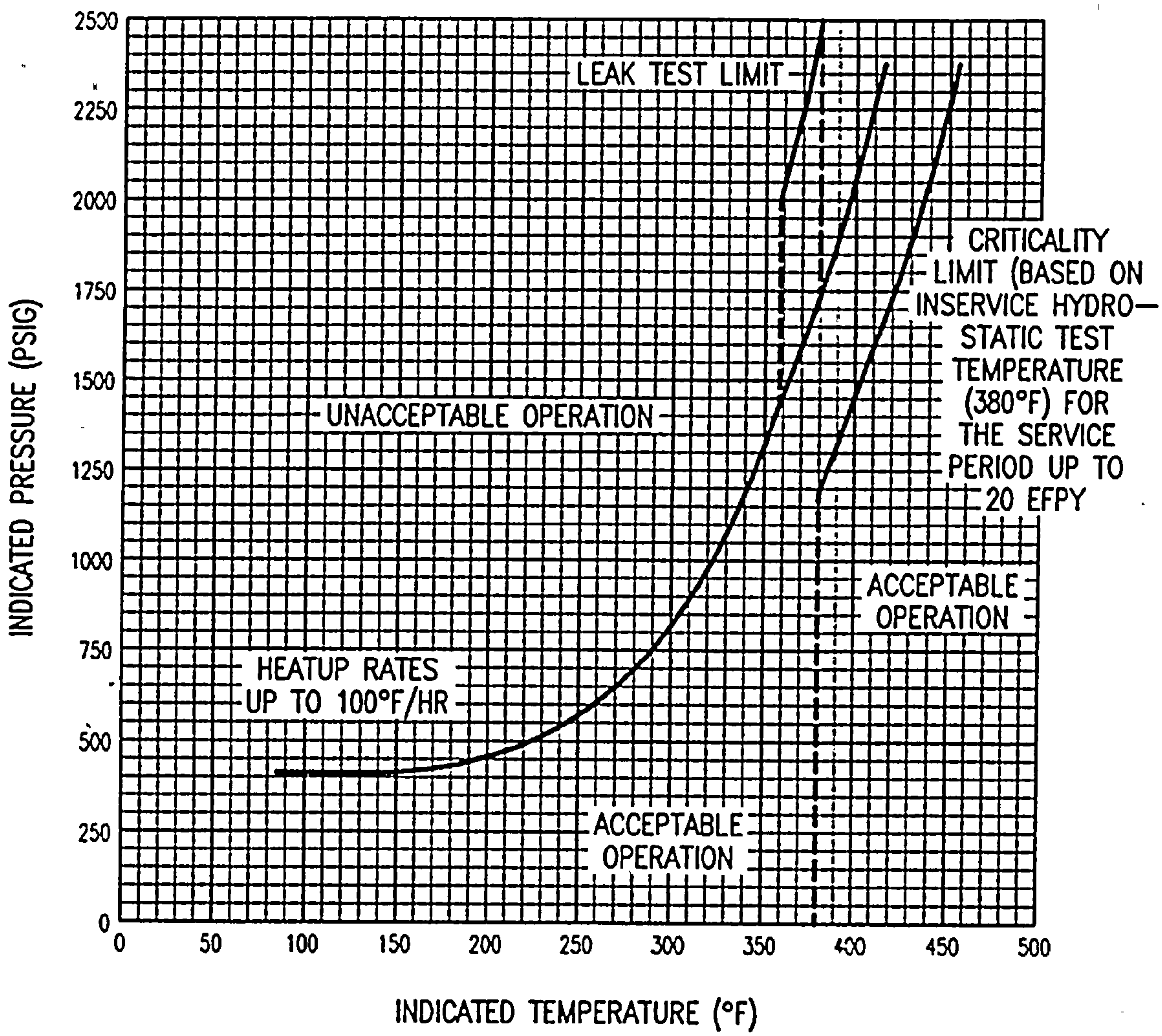

FIGURE 3.4—3 REACTOR COOLANT HEATUP LIMITATIONS (<100°F/HR)

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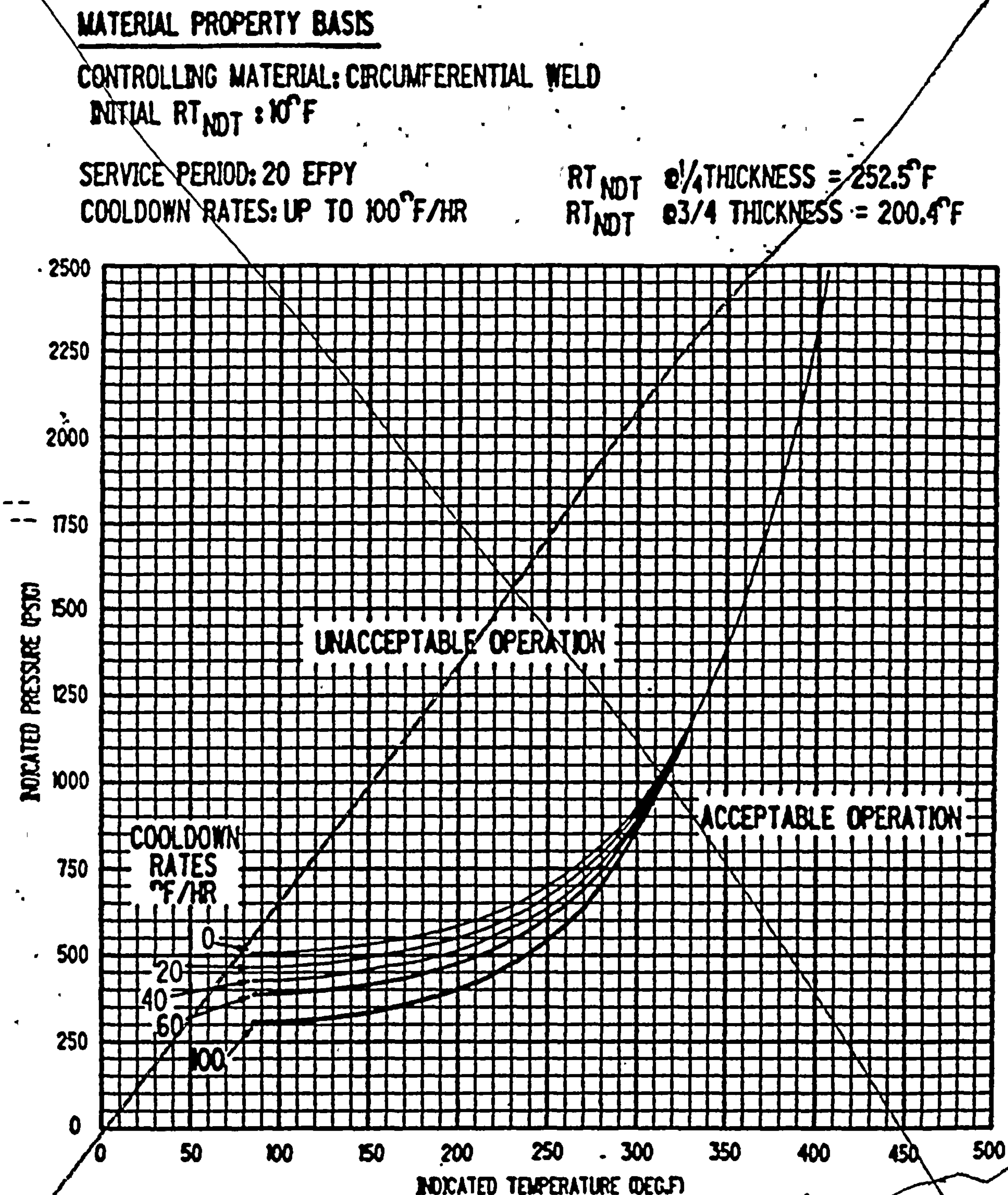

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FIGURE 3.4-4

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (100°F/hr) - APPLICABLE UP TO 20 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 20 EFPY
COOLDOWN RATES: UP TO 100°F/HR

RT_{NDT} @ 1/4 THICKNESS = 252.5°F

RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

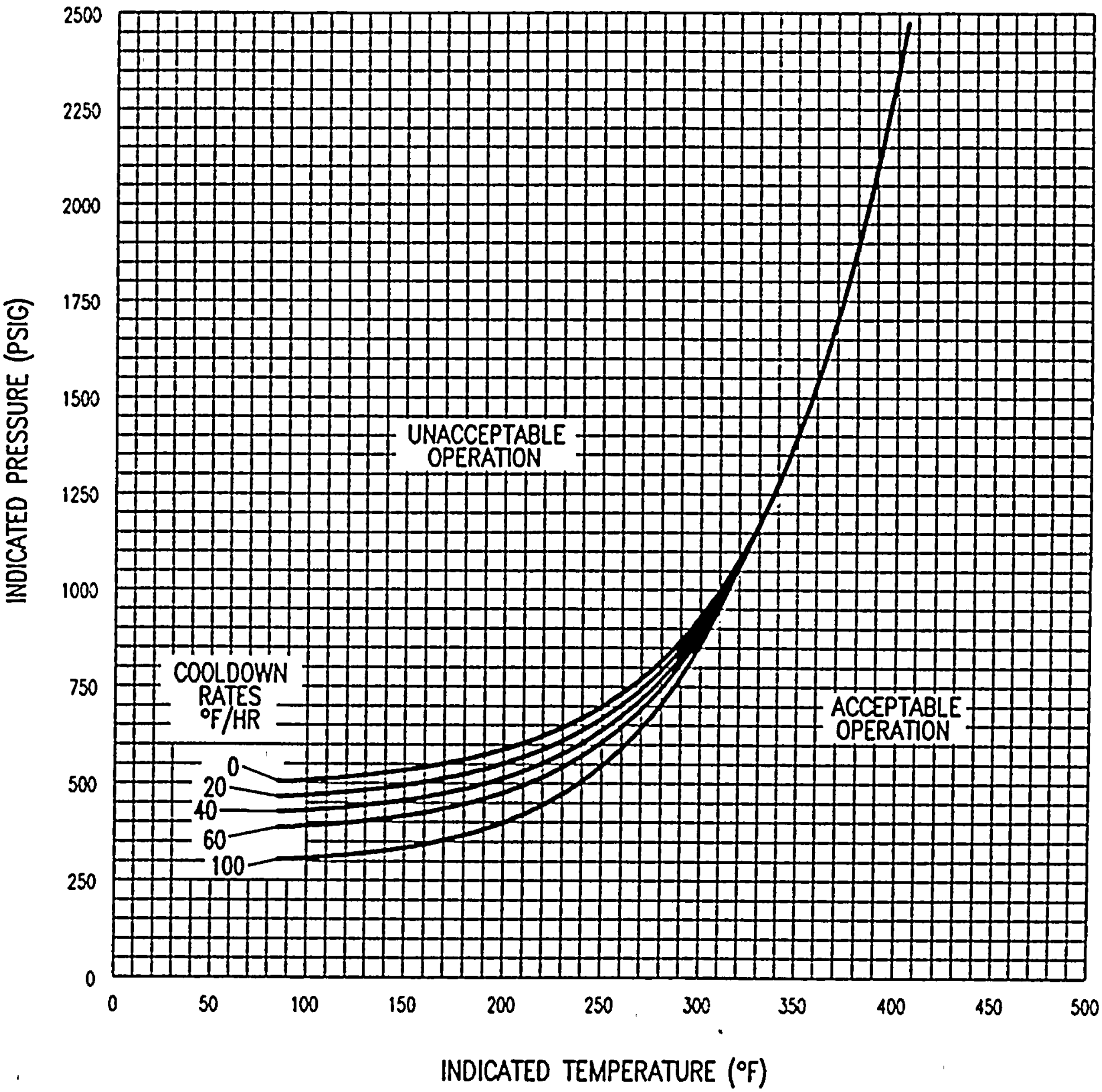

FIGURE 3.4—4 REACTOR COOLANT COOLDOWN LIMITATIONS (0—100°F/HR)

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TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 3

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
U	30°	0.49	Standby
V	--	---	Specimen withdrawn at 12 years
W	40°	0.34	Standby
X	50°	0.34	33 years
Y	150°	0.49	Standby
Z	230°	0.34	Standby

UNIT 4

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
U	30°	0.49	Standby
V	290°	0.79	24 years
W	40°	0.34	Standby
X	50°	0.34	Standby
Y	150°	0.49	Standby
Z	230°	0.34	Standby

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LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and below an RCS average coolant temperature of 275°F at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a) Two power-operated relief valves (PORVs) with a lift setting of 415 ± 15 psig, or

b) The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

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

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours, and

b. In MODE 4 with RCS average coolant temperature less than or equal to 275°F, and in MODE 5 or in Mode 6 with the reactor vessel head on:

1. With one PORV inoperable, perform at least one of the following within the next 7 days:

a) Restore the inoperable PORV to OPERABLE status, or

b) Depressurize and vent the RCS through at least a 2.20 square inch vent, or

c) Depressurize and maintain a RCS vent through at least one open PORV and open associated block valve.

2. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.20 square inch vent within 24 hours.

3. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

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SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

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

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

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

d. While the PORVs are required to be OPERABLE, the backup air supply shall be verified OPERABLE at least once per 24 hours.

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify the high pressure injection flow path to the RCS is isolated at least once per 24 hours by closed valves with power removed or by locked closed manual valves.

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

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES

`CTION:

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

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

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

SURVEILLANCE REQUIREMENTS

\.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

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

a. Reactor vessel head, and

b. Pressurizer steam space

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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 its circuit breaker open,

b. An indicated borated water volume of between 6545 and 6665 gallons,

c. A boron concentration of between 1950 and 2350 ppm,

d. A nitrogen cover-pressure of between 600 and 675 psig, and

e. A water level and pressure channel OPERABLE.

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 4 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

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2) Verifying that each accumulator isolation valve is open by control room indication (power may be restored to the valve operator to perform this surveillance if redundant indicator is inoperable).

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 31 days:

   1) When the RCS pressure is above 1000 psig, by verifying that the power to the isolation valve operator is disconnected by a locked open breaker, and

   2) Each accumulator water level and pressure channel shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

d. At least once per 18 months:

   1) Each accumulator water level and pressure channel shall be demonstrated OPERABLE by the performance of a CHANNEL CALIBRATION, and

   2) Each accumulator check valve shall be checked for operability.

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

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection (SI) pumps with discharge aligned to the RCS cold legs,

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps with discharge aligned to the RCS cold legs,

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

-- e. Two OPERABLE flow paths capable of taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s), restore the inoperable component or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water in 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 since ~~(date to be provided later)~~. January 1, 1990.

c. With one Safety Injection pump inoperable, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

d. With two Safety Injection Pumps inoperable, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to T_{avg} exceeding 380°F. Safety Injection flow paths may be isolated when T_{avg} is less than 380°F.

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SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS component and flow path shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
864A and B	Supply from RWST to ECCS	Open
862A and B	RWST Supply to RHR pumps	Open
863A and B	RHR Recirculation	Closed
866A and B	H.H.S.I. to Hot Legs	Closed
HCV-758*	RHR HX Outlet	Open

To permit temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

b. At least once per 31 days by:

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

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, and

3) Verifying that each RHR Pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.

c. At least once per 92 days by:

1) Verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions ~~in accordance with Figure 3.5-1~~ when tested pursuant to Specification 4.0.5.

SI pump $\geq$ 1126 psid at a metered flowrate $\geq$ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

$\geq$ 1156 psid at a metered flowrate $\geq$ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

*Air Supply to HCV-758 shall be verified shut off and sealed closed once per 31 days.

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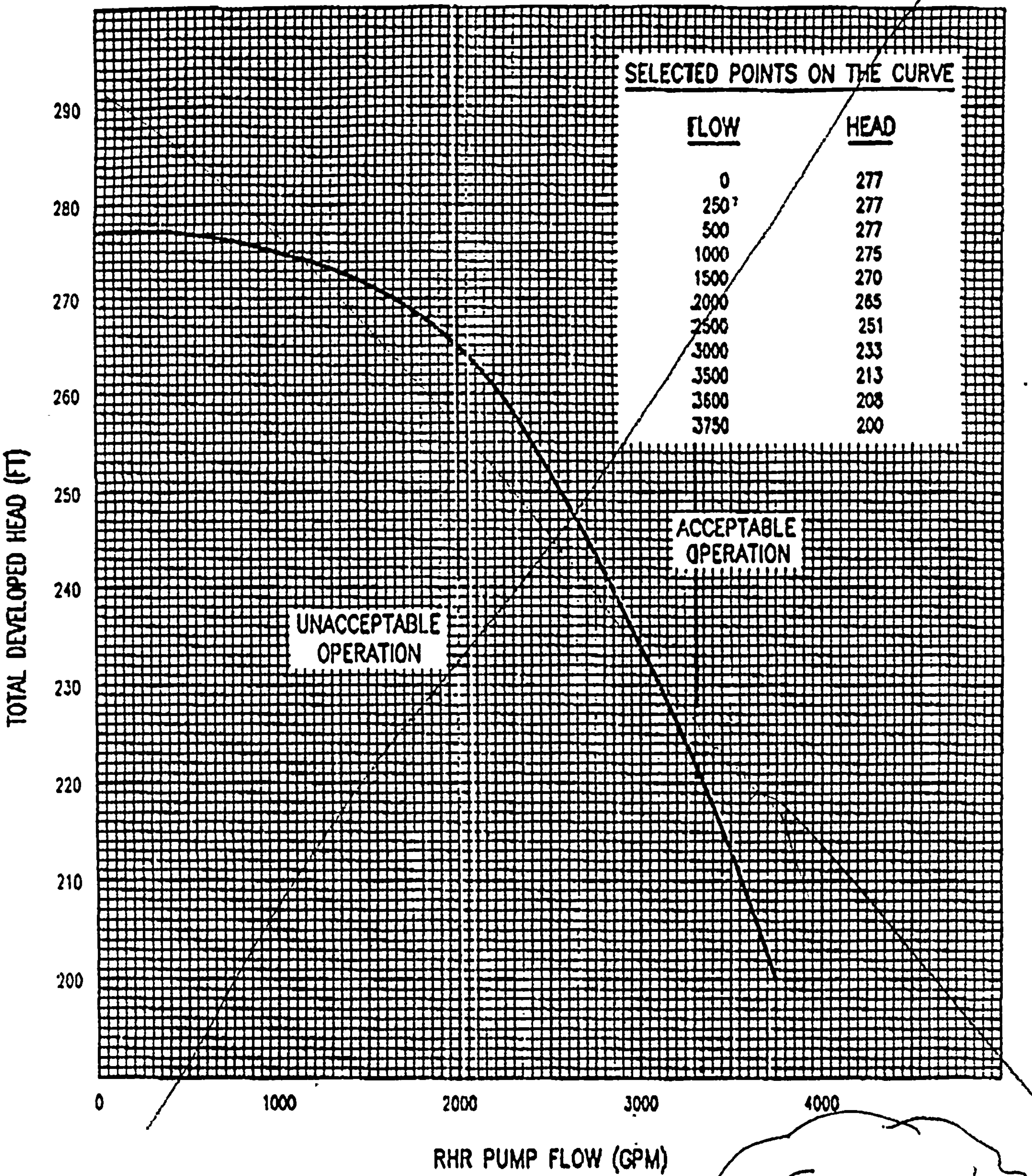

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Figure 3.5-1
RHR Pump Curve

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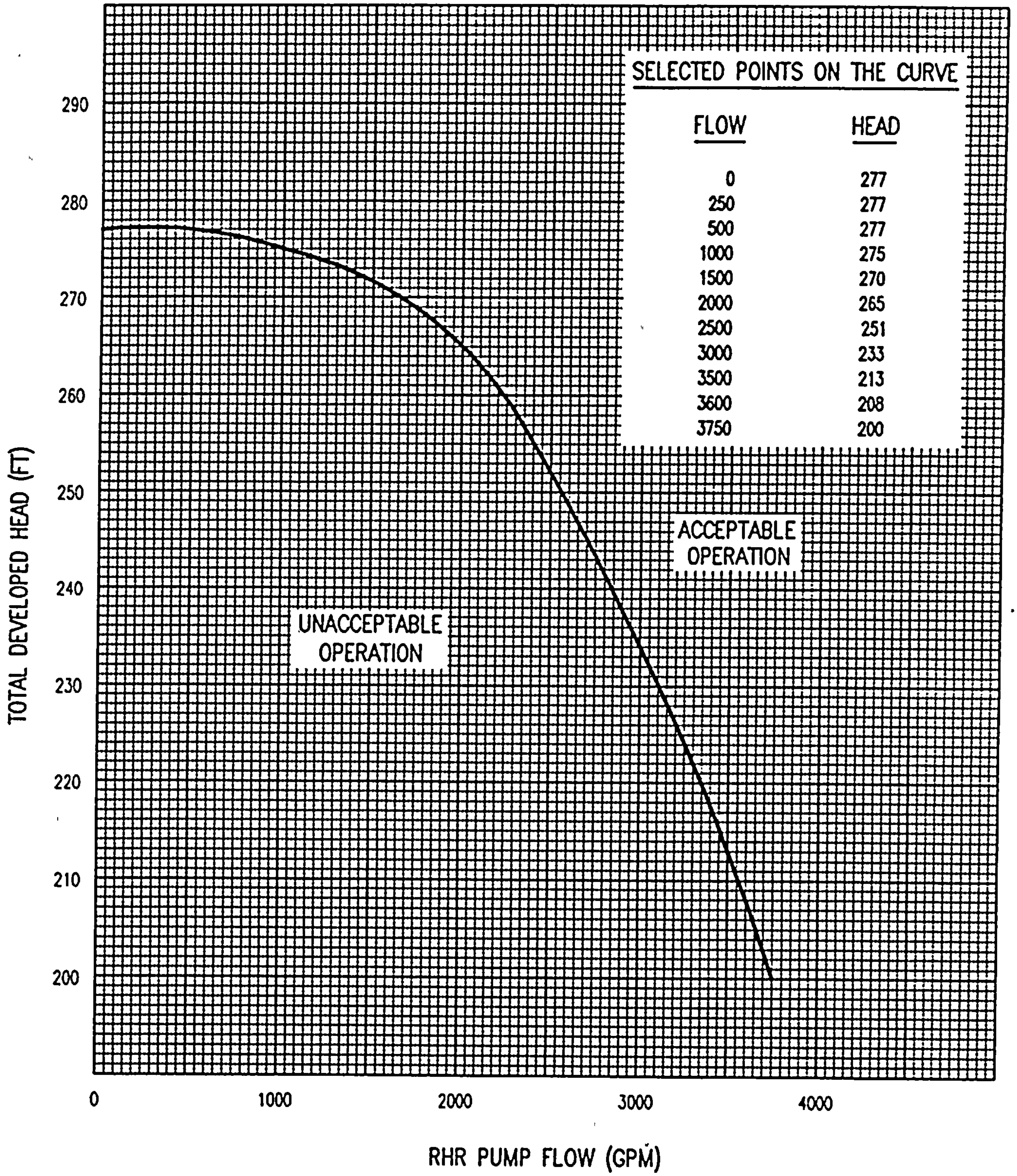

FLOW	HEAD
0	277
250	277
500	277
1000	275
1500	270
2000	265
2500	251
3000	233
3500	213
3600	208
3750	200

FIGURE 3.5—1
RESIDUAL HEAT REMOVAL PUMP MINIMUM ACCEPTABLE PERFORMANCE CURVE

SURVEILLANCE REQUIREMENTS

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

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

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

e. At least once per 18 months by:

   1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 525 psig the interlocks cause the valves to automatically close and prevent the valves from being opened, and

   2) Verifying correct interlock action to ensure that the RWST is isolated from the RHR System during RHR System operation and to ensure that the RHR System cannot be pressurized from the Reactor Coolant System unless the above RWST Isolation Valves are closed.

   3) A visual inspection of the containment sump and verifying that the 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.

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

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

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

      a) Safety Injection pump, and

      b) RHR pump.

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SURVEILLANCE REQUIREMENTS

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 components are required to be OPERABLE, and

2) At least once per 18 months.

RHR System
Valve Number

HCV-*-758
MOV-*-872
FCV-*-605

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

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, the following ECCS components and flow path shall be OPERABLE:

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

c. An OPERABLE flow path capable of (1) taking suction from the refueling water storage tank upon being manually realigned and (2) transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no OPERABLE ECCS flow path from the refueling water storage tank, restore at least one ECCS flow path to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the components 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 since ~~(date to be provided later)~~ January 1, 1990.

SURVEILLANCE REQUIREMENTS

4.5.3 The above ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storge tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A minimum boron concentration of 1950 ppm of boron,

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) 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 required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 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 CONTAINMENT INTEGRITY shall be demonstrated:

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

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than 50 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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*Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

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

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CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.25% by weight of the containment air per 24 hours at P_a, 49.9 psig.

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 49.9 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 L_a the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

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

1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the following equation:

$$| L_c - (L_{am} + L_o) | \leq 0.25 L_a$$

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

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

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

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 49.9 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Purge supply and exhaust isolation valves, and

3) Equipment access opening which shall be tested at least once every 12 months and after each use.

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

f. Purge supply and exhaust isolation valves seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2, as applicable;

g. The provisions of Specification 4.0.2 are not applicable.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 49.9 psig.

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

ACTION:

a. With one containment air lock door inoperable:

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

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the ~~following applicable action is performed at least~~ 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.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests at not less than 50 psig, and verifying the overall air lock leakage rate is within its limit at least once per 6 months.*

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

*The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +3 psig.

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

ACTION:

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

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SURVEILLANCE REQUIREMENTS

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 125°F and shall not exceed 120°F by more than 336 equivalent hours* during a calendar year.

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

ACTION:

With the containment average air temperature greater than 125°F or greater than 120°F for more than 336 equivalent hours* during a calendar year, reduce the average air temperature to within the applicable 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:

Approximate Location

a.	0° Azimuth	58 feet elevation
b.	120° Azimuth	58 feet elevation
c.	240° Azimuth	58 feet elevation

*Equivalent hours are determined from actual hours using the time-temperature relationships that support the environmental qualification requirements of 10 CFR 50.49.

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With more than one tendon (not including exempted* tendons) with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one exempted* tendon with an observed lift-off force at the accessible end below 86% of the predicted lower limit, restore the tendon to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specifications 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 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 Tendons. The containment tendons' structural integrity shall be demonstrated every fifth year from the date of the initial structural integrity test. The tendons' structural integrity shall be demonstrated by:

*Exempted in accordance with IWL-2521.1(a). Lift-off forces observed at the accessible end below 90% of the predicted lower limit shall be reported to the Commission for information only.

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a. Determining that a random but representative sample** of at least 12 tendons (3 dome, 4 vertical, and 5 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon (not including exempted* tendons) in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable.

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (dome, vertical, and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

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

   2) There are no changes in the presence or physical appearance of the sheathing filler-grease, and

   3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6, -0%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

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*Exempted in accordance with IWL-2521.1(a). Lift-off forces observed at the accessible end below 90% of the predicted lower limit shall be reported to the Commission for information only.

**After the process of randomly selecting tendons is performed, inaccessible tendons may be exempted in accordance with IWL-2521.1(a). Substitute tendons shall be selected that are located as close as possible to the exempted tendons. The accessible end of exempted tendons shall have the lift-off force measured.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. Assuring that the observed lift-off force for each tendons exceeds the minimum required lift-off force. Required lift-off forces shall be calculated individually for each surveillance tendon prior to the beginning of each surveillance, and should consider such factors as:

   1) Prestressing history;
   2) Friction losses; and
   3) Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) Minimum grease coverage exists for the different parts of the anchorage system, and

   2) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through visual inspection that no unacceptable levels of corrosion exist on the end anchorages and no unacceptable cracking exists in the concrete adjacent to the end anchorages. Determination of acceptance levels shall be by engineering evaluation of the areas in question. If unacceptable conditions are found, the tendons inspected during the previous surveillance shall be examined to determine whether the corrosion levels or concrete cracking have increased since the previous surveillance. Inspection of adjacent concrete surfaces shall be performed concurrently with the containment tendon surveillance (Technical Specification 4.6.1.6.1).

4.6.1.6.3 Containment Surfaces. In accordance with 10 CFR 50, Appendix J. Section V. A, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed during the shutdown for (but prior to) each Type A containment leakage rate test (Technical Specification 4.6.1.2.1). The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with 10 CFR 50, Appendix J, Section V. A. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. The containment purge supply and exhaust isolation valves shall be sealed closed to the maximum extent practicable but may be open for purge system operation for pressure control, for environmental conditions control, for ALARA and respirable air quality considerations for personnel entry and for surveillance tests that require the valve to be open.

b. The purge supply and exhaust isolation valves shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

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

ACTION:

a. With a containment purge supply and/or exhaust isolation valve(s) open for reasons other than given in 3.6.1.7.a above, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Specification 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status or isolate the penetrations such that the measured leakage rate does not exceed the limits of Specification 4.6.1.7.2 within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7.a at least once per 31 days.

4.6.1.7.2 At least once per 6 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.3 At least once per 18 months, the mechanical stop on each containment purge supply and exhaust isolation valve shall be verified to be in place and that the valves will open no more than 33 or 30 degrees, respectively.

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3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST.

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

ACTION:

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

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

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

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested pursuant to Specification 4.0.5:

Containment Spray Pump $\geq$241.6 psid while aligned in recirculation mode.

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

2) Verifying that each spray pump starts automatically on a containment spray actuation test signal. The manual isolation valves in the spray lines at the containment shall be locked closed for the performance of these tests.

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

CONTAINMENT SYSTEMS

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

ACTION:

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes, and

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

b. At least once per 18 months by verifying that each unit starts automatically on a safety injection (SI) test signal.

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CONTAINMENT SYSTEMS

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

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

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter 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

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release or (3) after every 720 hours of system operation by:

   1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbons at the system flow rate of 37,500 cfm ±10%;

   2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ANSI N-510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position C.6a of Regulatory Guide 1.52, Rev. 2; and

   3) Verifying a system flow rate of 37,500 cfm ±10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation when tested in accordance with ANSI N510-1975;

[illegible]

URVEILLANCE REQUIREMENTS (Continued)

c. After maintenance affecting flow distribution, by performance of a visual inspection and an air distribution test at a system flow rate of 37,500 cfm ±10%;

d. At least once per 18 months by:

1) Verifying that the system starts on a Safety Injection test signal and;

2) Verifying that the filter cooling solenoid valves can be opened by operator action and are opened automatically on a loss of flow signal.

e. After each complete or partial replacement of a HEPA filter bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP test aerosol while operating the system at a flow rate of 37,500 cfm ±10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of halogenated hydrocarbon while operating the system at a flow rate of 37,500 cfm ±10%.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

*With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

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

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

c. Verifying that on a Containment Ventilation Isolation test signal, each purge, exhaust and instrument air bleed valve actuates to its isolation position.

4.6.4.3 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.

3/4.6.5 COMBUSTIBLE GAS CONTROL

YDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.5 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

`CTION:

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

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

4.6.5.2 The flow path to each hydrogen monitor shall be demonstrated OPERABLE at least once per 31 days by a system walkdown to verify that each accessible manual, power operated, or automatic valve is in its correct position and that power is available to those components related to the operability of the flowpath.

3/4.6.6 POST ACCIDENT CONTAINMENT VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 A Post Accident Containment Vent System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

'ith the Post Accident Containment Vent System inoperable, restore the Post Accident Containment Vent System to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 The Post Accident Containment Vent System shall be demonstrated OPERABLE:

a. At least once per 31 days by demonstrating system flow path operability via a system walkdown to verify that each accessible manual valve is in its correct position.

b. At least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after 720 hours of system operation or (4) after replacement of a filter by:

   1) A visual inspection of the system for foreign materials and gasket deterioration and verifying that the filter system satisfies the penetration and bypass leakage testing acceptance criteria of less than 1% for DOP and halogenated hydrocarbon tests conducted at a design flow rate of 55 cfm ±10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample performed in accordance with ANSI N510-1975, meets the methyl iodide removal criteria of greater than or equal to 90% and that any charcoal failing to meet the criteria be replaced with charcoal that meets or exceeds the criteria of Position C.6.a of Regulatory Guide 1.52, Revision 2.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber is less than 6 inches Water Gauge at a flow rate of 55 cfm ± 10%,

2) Visual inspection of the system and operation of all valves.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	82
2	54
3	27

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (±1%)*	ORIFICE SIZE SQUARE INCHES
	Loop A	Loop B	Loop C		
1.	RV1400	RV1405	RV1410	1085 psig	16
2.	RV1401	RV1406	RV1411	1100 psig	16
3.	RV1402	RV1407	RV1412	1115 psig	16
4.	RV1403	RV1408	RV1413	1130 psig	16

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

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the OPERABILITY of both standby feed-water pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train.

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN Within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

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

1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS (Continued)

equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;

2) Verifying by control panel indication and visual observation of equipment that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that power is available to those components which require power for flow path operability.

b. At least once per 18 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

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 1 by verifying normal flow to each steam generator.

TABLE 3.7-3

AUXILIARY FEEDWATER SYSTEM OPERABILITY

UNIT	TRAIN	STEAM SUPPLY FLOWPATH(3)	PUMP	DISCHARGE WATER FLOWPATH(3)
3	1	SG 3C via MOV-3-1405 or SG 3B via MOV-3-1404(1)	A or C(2)	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403 or SG 3B via MOV-3-1404(1)	B or C(2)	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405 or SG 4B via MOV-4-1404(1)	A or C(2)	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403 or SG 4B via MOV-4-1404(1)	B or C(2)	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES:

(1) Steam admission valves MOV-3-1404 and MOV-4-1404 can be aligned to either train (but not both) to restore OPERABILITY in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

(2) During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being powered from, and supplying water to either train, except as noted in ACTION 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater pump (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable.

(3) If any local manual realignment of valves is required when operating the auxiliary feedwater pumps, a dedicated individual, who is in communication with the control room, shall be stationed at the auxiliary feedwater pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5 or 6

An indicated water volume of 185,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

An indicated water volume of 370,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5 or 6

With the CST system inoperable, within 4 hours restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3

1) With the CST system inoperable due to containing less than 370,000 gallons, but greater than or equal to 185,000 gallons within 4 hours restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With the CST system inoperable with less than 185,000 gallons within 1 hour restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.3 The condensate storage tank (CST) system shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limit when the tank is 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 0.10 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 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

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

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

'ODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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PLANT SYSTEMS

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two standby feedwater pumps shall be OPERABLE* and at least 60,000 gallons of water (available volume), shall be in the Demineralized Water Storage Tank**.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one standby feedwater pump inoperable, restore the inoperable pumps to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both standby feedwater pumps inoperable:

   1. Within 24 hours, notify the NRC and provide cause for inoperability and plans to restore pump(s) to OPERABLE status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 60,000 gallons of water in the Demineralized Water Storage Tank restore the available volume to at least 60,000 gallons within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.6.2 At least monthly verify the standby feedwater pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.6.3 At least once per 18 months, verify operability of the respective standby feedwater pump by powering from the non-safety grade diesel generators and providing feedwater to the steam generators.

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*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

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3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps, and

b. Two CCW heat exchangers.

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

ACTION:

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With less than two CCW heat exchangers OPERABLE, restore two heat exchangers to OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE:

a. At least once per 12 hours, by verifying that two heat exchangers and one pump are capable of removing design basis heat loads.

b. At least once per 31 days by: (1) 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 (2) verifying by a performance test the heat exchanger surveillance curves.

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

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Component Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for CCW operability are OPERABLE.

3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with:

a. Three ICW pumps, and
b. Two ICW headers.

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

ACTION:

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With only one ICW header OPERABLE, restore two headers to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

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

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

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 The ultimate heat sink shall be OPERABLE with an average supply water temperature to the Intake Cooling Water System less than or equal to 100°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours. This action shall be applicable to both units simultaneously.

--

SURVEILLANCE REQUIREMENTS

4.7.4 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average supply water temperature* to the Intake Cooling Water System to be within its limit.

*Portable monitors may be used to measure the temperature.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With the Control Room Emergency Ventilation System inoperable, suspend all movement of fuel in the spent fuel pool and restore the inoperable system to OPERABLE status within 84 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If this ACTION applies to both units simultaneously, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

With the Control Room Emergency Ventilation System inoperable, suspend all operations involving CORE ALTERATIONS, movement of fuel in the spent fuel pool, or positive reactivity changes. This ACTION shall apply to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The Control Room Emergency Ventilation 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 120°F;

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

c. At least once per 18 months or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following operational exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (4) after complete or partial replacement of a filter bank by:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal at a system flow rate of 1000 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, and analyzed per ANSI N510-1975, meets the criteria for methyl iodine removal efficiency of greater than or equal to 90% or the charcoal be replaced with charcoal that meets or exceeds the criteria of position C.6.a. of Regulatory Guide 1.52 (Revision 2), and

3). Verifying by a visual inspection the absence of foreign materials and gasket deterioration.

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%;

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation.

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3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 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 4.7.6f. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

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

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type (on any system) are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection (of that system) shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

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

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

**The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.6e. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Functional Tests

For each unit during refueling shutdown, a representative sample of snubbers shall be tested using the following sample plan:

1) At least 10% of the total number of safety related snubbers for the respective unit identified by site records 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 4.7.6e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested;

2) The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following categories;

   A. Snubbers within 5 feet of heavy equipment (ex. valves, pumps, turbines, motors, etc.)

   B. Snubbers within 10 feet of the discharge from a safety relief valve.

3) Snubbers identified by site records as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber OPERABILITY for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved with the specified range of velocity or acceleration in both tension and compression;

2) Snubber release rate, where required, is within the specified range in tension and compression,

3) The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

f. 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.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated under the provisions of 10 CFR Part 21.

Should the results of the evaluation indicate that the failure was caused by either manufacturer or design deficiency, further action shall be taken, if needed, based on manufacturer or engineering recommendations.

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

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SURVEILLANCE REQUIREMENTS (Continued)

g. Snubber Service Life Monitoring Program

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.3m.

Concurrent with the first inservice visual inspection and during refueling shutdown thereafter, the installation and maintenance records for each safety related snubber as identified by site records shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

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

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

[illegible]

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

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

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

4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

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PLANT SYSTEMS

3/4.7.8 FIRE SUPPRESSION SYSTEMS

FIRE WATER SUPPLY AND DISTRIBUTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 The Fire Water Supply and Distribution System shall be OPERABLE with:

a. At least two fire suppression pumps, one electric driven and one diesel driven with their discharge aligned to the fire suppression header,

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

c. An OPERABLE flow path capable of taking suction from the Raw Water Tank I and Raw Water Tank II and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.8.2, 3.7.8.3, and 3.7.8.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specification 3.0.3 are not applicable. This action applies to both units simultaneously.

b. With the Fire Water Supply and Distribution System otherwise inoperable, establish a backup fire water capability within 24 hours. This action applies to both units simultaneously.

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SURVEILLANCE REQUIREMENTS

4.7.8.1.1 The Fire Water Supply and Distribution System shall be demonstrated OPERABLE:

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

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

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

d. At least once per 12 months by performance of a system flush,

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

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

1) Verifying that each automatic valve in the flow path actuates to its correct position,

2) Verifying that the electric-driven pump develops at least 1880 gpm at a system pressure of 130.2 psig, and the diesel-driven pump develops at least 2350 gpm at a system pressure of 130.2 psig, by verifying 3 points on the pump performance curve.

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

4) Verifying that each fire pump starts sequentially to maintain the Fire Water Supply and Distribution System pressure greater than or equal to 125 psig.

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

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least 375 gallons of fuel, and

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

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

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

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

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

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

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

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

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

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

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

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Fire Zones 47 and 54 - Component Cooling Water Areas

b. Fire Zones 45 and 55 - Charging Pump Rooms

c. Fire Zones 79A - North - South Breezeway

d. Fire Zones 72, 73, 74 and 75 - Emergency Diesel Generator and Day Tank Rooms

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment. This ACTION applies to both units simultaneously.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

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

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

c. At least once per 18 months:

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

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

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations given in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-4 inoperable, provide an equivalent capacity fire hose from the nearest equivalent OPERABLE water source. The fire hose shall be of a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station, and shall be stored in a roll at the outlet of the OPERABLE water supply. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. This ACTION applies to both units simultaneously.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations given in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 12 months, by:

    1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

    2) Removing the hose for inspection and re-racking, and

    3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

    1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

    2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

[illegible]

TABLE 3.7-4

FIRE HOSE STATIONS

IDENTIFICATION	LOCATION	FIRE ZONE
HS-03-01	EL. 18' - East of 4160V SWGR Room on Column	88
HS-03-02	EL. 18' - West of 3A Condensate Pump on Pedestal	87
HS-03-03	EL. 18' - Passageway South of SG Feed Pump Room	83
HS-03-04	EL. 30' - East of 480V Load Center on Column	105
HS-03-05	EL. 30' - South End of Mezzanine Deck	105
HS-03-06	EL. 42' - NW End of Turbine Deck	117
HS-03-07	EL. 42' - North of 6A HPFW Heater	117
HS-03-08	EL. 42' - NW Corner of Entrance to Elevator	79
HS-04-01	EL. 18' - South of 4160V SWGR Room on Column	82
HS-04-02	EL. 18' - Passageway South of SG Feed Pump Room	78
HS-04-03	EL. 30' - East of 480V Load Center at Stairway	105
HS-04-04	EL. 30' - South End of Mezzanine Deck	105
HS-04-05	EL. 42' - West End of Turbine Deck	117
HS-04-06	EL. 42' - East Side of Turbine Deck and North of 6A FW Heater	117
HS-04-07	EL. 42' - East Side of Turbine Deck and North of 6B FW Heater	117
HS-04-08	EL. 42' - Southwest Corner of Turbine Deck	117
HS-AB-01	EL. 18' - East-West Passageway at West End	58
HS-AB-02	EL. 18' - East-West Passageway at East End	58
HS-AB-03	EL. 18' - North-South Passageway Outside Unit 3 Charging Pump Room	58
HS-AB-04	EL. 50' - Roof of Unit 3 New Fuel Storage Area	118
HS-AB-05	EL. 50' - Roof of Unit 4 New Fuel Storage Area	118

FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.8.4 The fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specification 3.0.3 and are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.4 Each of the fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

b. At least once per 6 months by visually inspecting each fire hydrant and verifying that the hydrant is not damaged, and

c. At least once per 12 months by:

1) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

2) Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

3) Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

FIRE HYDRANTS

IDENTIFICATION	FIRE ZONE	LOCATION	NUMBER OF HYDRANTS
FH-01	124	NE Corner of Unit 3 near Vehicle Gate into RCA	1
FH-06	NA	W of Nuclear Maintenance Building	1
FH-07	86	Unit 3 Transformer Area	1
FH-08	81	Unit 4 Transformer Area	1
FH-09	76	Unit 4 Turbine-Generator Area	1
FH-10 FH-11	77	Unit 4 Condensate Storage Tank Area	2
FH-12	NA	Unit 4 New Fuel Storage Area	1
FH-13	123	Refueling Water Storage Area	1
FH-17	NA	Nuclear Dry Storage Area	1
FH-16	NA	Steam Generator Storage Area	1
		TOTAL	11

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.9.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, and a functional test is performed at least once per 18 months, and

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.