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NORTH ANNA POWER STATION UNIT 2 TECHNICAL SPECIFICATIONS

APPENDIX "A" TO LICENSE NO. NPF-7

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NORTH ANNA

POWER STATION

UNIT 2

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

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INDEX

DEFINITIONS

SECTION	PAGE

1.0 DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION | PAGE

BASES

SECTION PAGE

3/4.3 INSTRUMENTATION

3/4.4 REACTOR COOLANT SYSTEM

BASES

SECTION PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

BASES

SECTION	PAGE
3/4.7 PLANT SYSTEMS	

BASES

SECTION PAGE

DESIGN FEATURES

INDEX

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

OPERATIONAL MODE - MODE

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

ACTION

1.5 ACTION shall be that part of the Specification which prescribes remedial measures required under designated conditions.

OPERABLE - OPERABILITY

1.6 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, normal and emergency electrical power sources, 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).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

   b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

2. All equipment hatches are closed and sealed,

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

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

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

1.10 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.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

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

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

CORE ALTERATION

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

SHUTDOWN MARGIN

1.13 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.

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 system.

UNIDENTIFIED LEAKAGE

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

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

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

QUADRANT POWER TILT RATIO

1.18 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.

DOSE EQUIVALENT I-131

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

STAGGERED TEST BASIS

1.20 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,

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

FREQUENCY NOTATION

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

AXIAL FLUX DIFFERENCE

1.24 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

PHYSICS TESTS

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

$\bar{E}$ - AVERGE DISINITEGRATION ENERGY

1.26 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

TABLE 1.1

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.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 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.

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.

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.

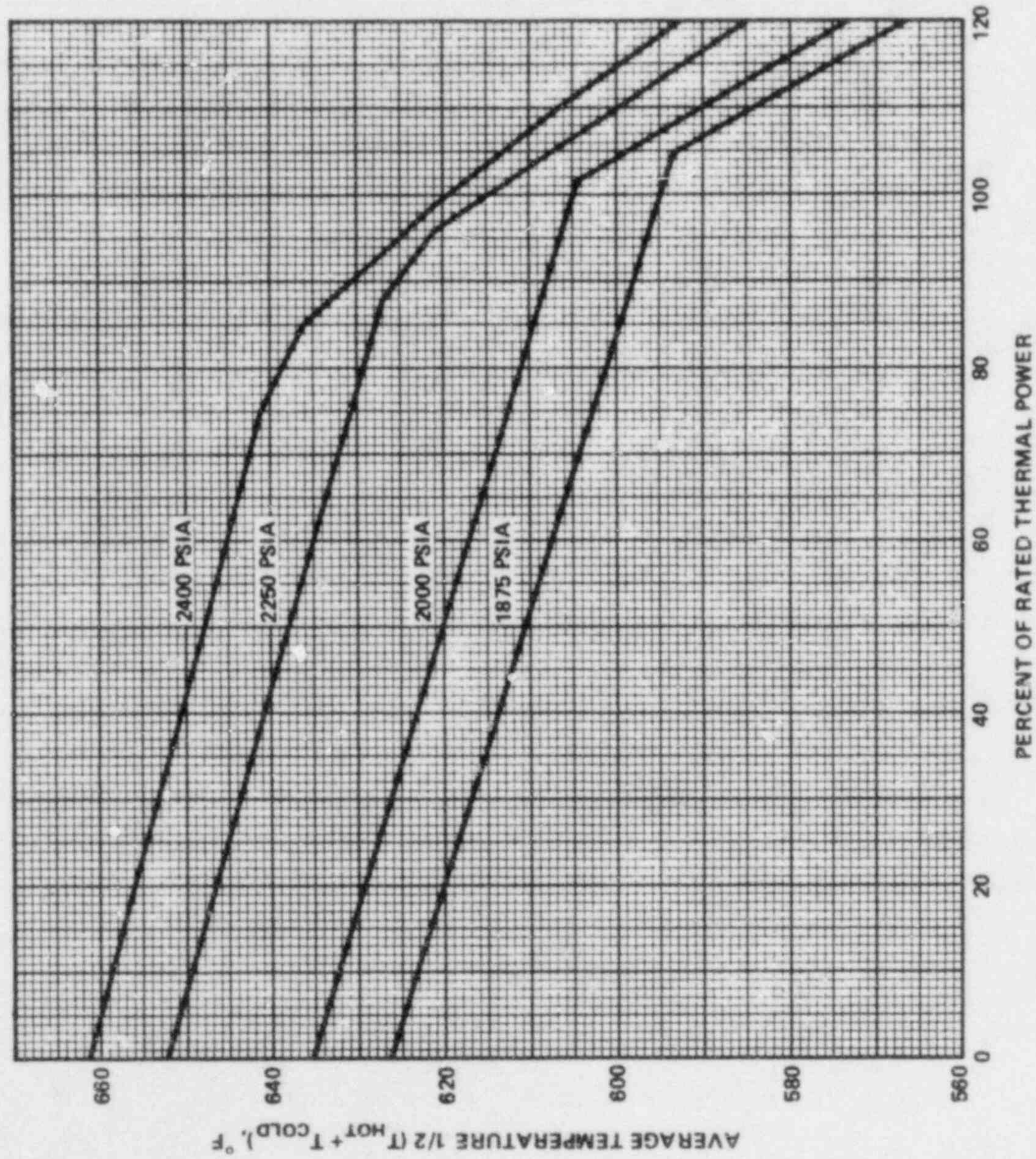

Figure 2.1-1 Reactor Core Safety Limits for Three Loop Operation, 100% Flow

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REACTOR CORE SAFETY LIMIT - TWO LOOP OPERATION

(ONE LOOP ISOLATED)

FIGURE 2.1-2

This page left blank pending NRC approval of ECCS evaluation of two loops in operation with the third loop not isolated.

REACTOR CORE SAFETY LIMIT - TWO LOOP OPERATION

(LOOP STOP VALVES OPEN)

FIGURE 2.1-3

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 92,800 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	$\geq$ 18% of narrow range instrument span-each steam generator	$\geq$ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq$ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span--each steam generator	$\leq$ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pump Busses	$\geq$ 2905 volts-each bus	$\geq$ 2870 volts-each bus
16. Underfrequency-Reactor Coolant Pump Busses	$\geq$ 56.1 Hz - each bus	$\geq$ 56.0 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 40 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 0% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 580.3°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
$K_1 = 1.141$	$K_1 = ()$	$K_1 = ()$
$K_2 = 0.0128$	$K_2 = (\quad)$	$K_2 = (\quad)$
$K_3 = 0.000608$	$K_3 = (\quad)$	$K_3 = (\quad)$

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

(i) for $q_t - q_b$ between - 34 percent and + 10 percent, f_1 (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).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 34 percent, the ΔT trip setpoint shall be automatically reduced by 3 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 10 percent, the ΔT trip setpoint shall be automatically reduced by 1.25 percent of its value at RATED THERMAL POWER.

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*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 \ (T-T") - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 580.3°F

K_4 = 1.086

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00116/°F for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

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 a part of these Technical Specifications.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

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

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

The curves of Figures 2.1-1, 2.1-2 and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

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

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

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% 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 values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

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

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 10 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above 1.30 for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. 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 proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T 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 High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. 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.

Operation with a reactor coolant loop out of service below the 3 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 2 loop operation exclusive of the Overtemperature Delta T set point. Two loop operation above the 3 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature Delta T channels and raising the P-8 set point to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure. The low pressure trip is blocked below P-7.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System. The pressurizer high water level trip is blocked automatically below the P-7 setpoint.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent (P-7) of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above 31% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow.

This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature Delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature Delta T trip set point adjusted to the value specified for 2 loop operation, the P-8 trip at 71% RATED THERMAL POWER with the loop stop valves closed in the nonoperating loop, will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system. The steam generator water level low-low trip is blocked when the loop stop valves are closed. A steam generator water level high-high signal trips the turbine which in turn trips the reactor if above the P-7 setpoint.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level 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 Protection 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 1.616 x 106 lbs/hour of full steam flow at RATED THERMAL POWER. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, 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 and Underfrequency - Reactor Coolant Pump Busses

The reactor trip due to the Undervoltage and Underfrequency on the Reactor Coolant Pump Busses provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.1 seconds." The undervoltage and underfrequency trips are automatically blocked when reactor power is below the P-7 setpoint.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. 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 resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point 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.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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, the unit shall be placed in a MODE in which the specificaton does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within 1 hour;
2. At least HOT SHUTDOWN within the next 6 hours; and
3. At least COLD SHUTDOWN within the following 30 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.

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

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours. This specification is not applicable in MODES 5 or 6.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing 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 Adderda 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).

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} Greater Than 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 10 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 1.77% delta k/k:

a. Within one 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 increased by an amount at least equal to 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 1.0, 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 1.0, 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.

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*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 10 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 1.77% delta k/k:

a. Within one 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 increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

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.

BORON DILUTION

VALVE POSITION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities:

a. 2-CH-140 or

b. 2-CH-160, 2-CH-156, FCV-2114B and FCV-2113B.

APPLICABILITY: MODES 3, 4, 5, and 6.

ACTION:

With the above valves not locked, sealed or otherwise secured in the closed position; 1) suspend all operations involving positive reactivity changes or CORE ALTERATIONS, 2) lock, seal or otherwise secure the valves in the closed position within 15 minutes, and 3) verify that the SHUTDOWN MARGIN is greater than or equal to 1.77% delta k/k within 60 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The above listed valves shall be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after a planned boron dilution or makeup activity.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of core life, hot zero THERMAL POWER condition, and

b. Less negative than -4.0×10^{-4} delta k/k/°F for the all rods withdrawn, end of core life at RATED THERMAL POWER.

APPLICABILITY: Specification 3.1.1.4.a. - MODES 1 and 2* only#.
Specification 3.1.1.4.b. - MODES 1, 2 and 3 only#.

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a. above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

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

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b. above, be in HOT SHUTDOWN within 12 hours.

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

#See Special Test Exception 3.10.3

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.4 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 3.1.1.4.a. above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -3.1 x 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 indicated the MTC is more negative than -3.1 x 10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.4.b., at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 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

4.1.1.5 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 1.0.

*See Special Test Exception 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump through a charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7.a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7.b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

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.

b. At least once per 31 days be 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.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

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

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull to lock position.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% delta k/k at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 340°F or prior to cooldown below 340°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F by verifying that the control switch is in the pull to lock position.

#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system and at least one associated heat tracing system with:

1. A minimum contained borated water volume of 835 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,000 gallons,
2. Between 2000 and 2100 ppm of boron, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume of the tank, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 35°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system and at least one associated heat tracing system with:
   1. A contained borated water volume of between 4450 and 16,280 gallons,
   2. Between 20,000 and 22,500 ppm of boron, and
   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:
   1. A contained borated water volume of between 475,058 and 487,000 gallons,
   2. Between 2000 and 2100 ppm of boron, and
   3. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage 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 1.77% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,
2. Verifying the contained borated water volume of each water source, and
3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

c. With one rod inoperable due to causes other than those addressed by ACTION "a" above or misaligned from its bank step counter demand height by more than ± 12 steps (indicated position), 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 rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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.

d) Either:

1) 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 high neutron flux trip setpoint is reduced to is less than or equal to 85% of RATED THERMAL POWER, or

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

SURVEILLANCE REQUIREMENTS

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

4.1.3.1.2 Each rod not fully inserted 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 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 Rupture (Loss of Coolant Accident)

Major Secondary Pipe Rupture

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

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator per group 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 immediately 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 50% of RATED THERMAL POWER within 8 hours.

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

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

   2. Reduce THERMAL POWER to LESS THAN 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*With the reactor trip system breakers in the closed position.
#See Special Test Exception 3.10.5.

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

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

1. Less than or equal to 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

2. Less than or equal to 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

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 one 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 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 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 two 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 group position using the above figures, or

c. Be in 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.

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*See Special Test Exceptions 3.10.2 and 3.10.3.

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

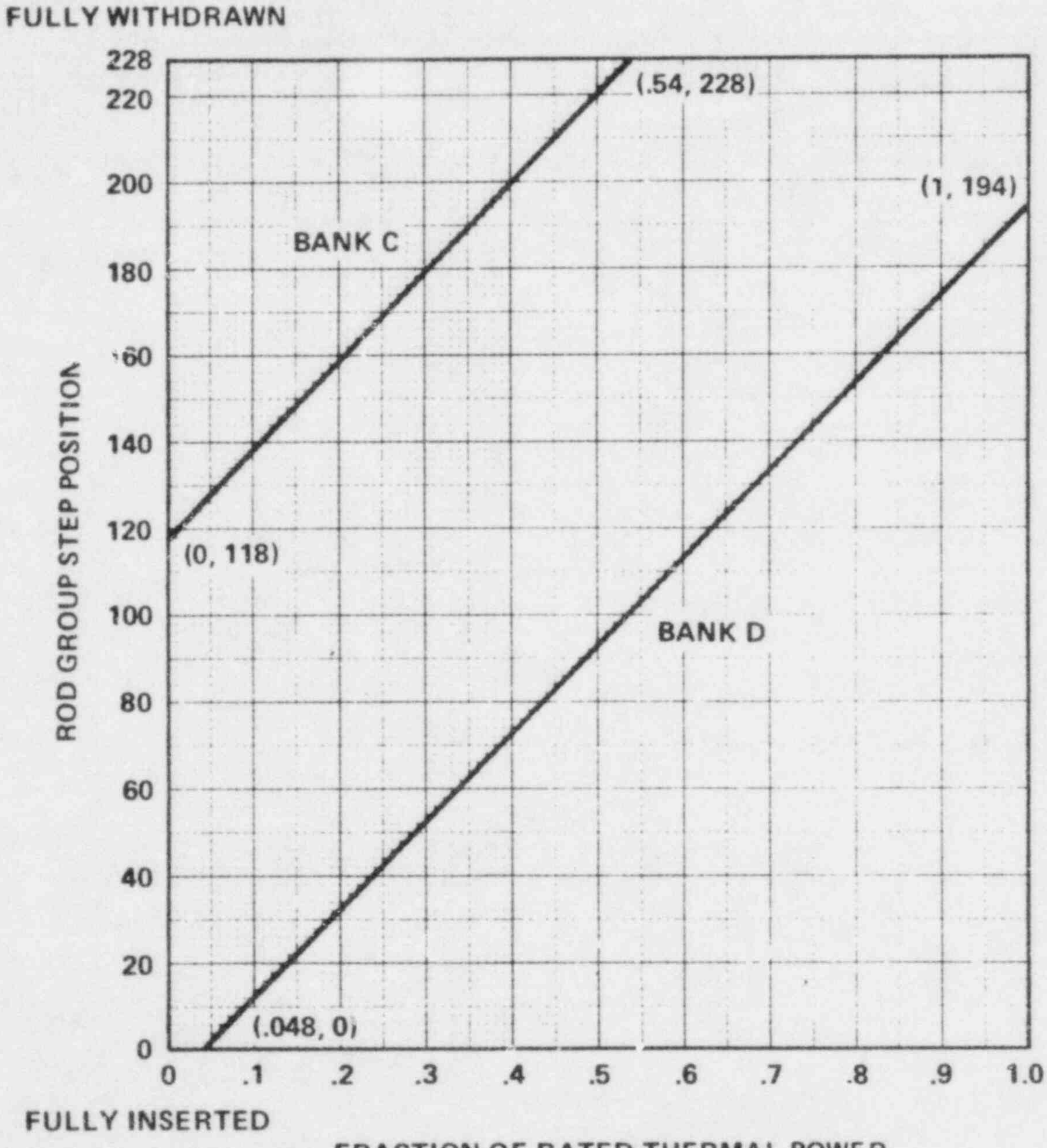

Figure 3.1-1 Rod Group Insertion Limits Versus Thermal Power

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FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER TWO LOOP OPERATION

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

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.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 5% target band about the target flux difference and with THERMAL POWER:

1. Above 81% of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 81% of RATED THERMAL POWER.

2. Between 50% and 81% of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 81% of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION 2.a.1, above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

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

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE 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 AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when at least 2 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one 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 one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

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 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

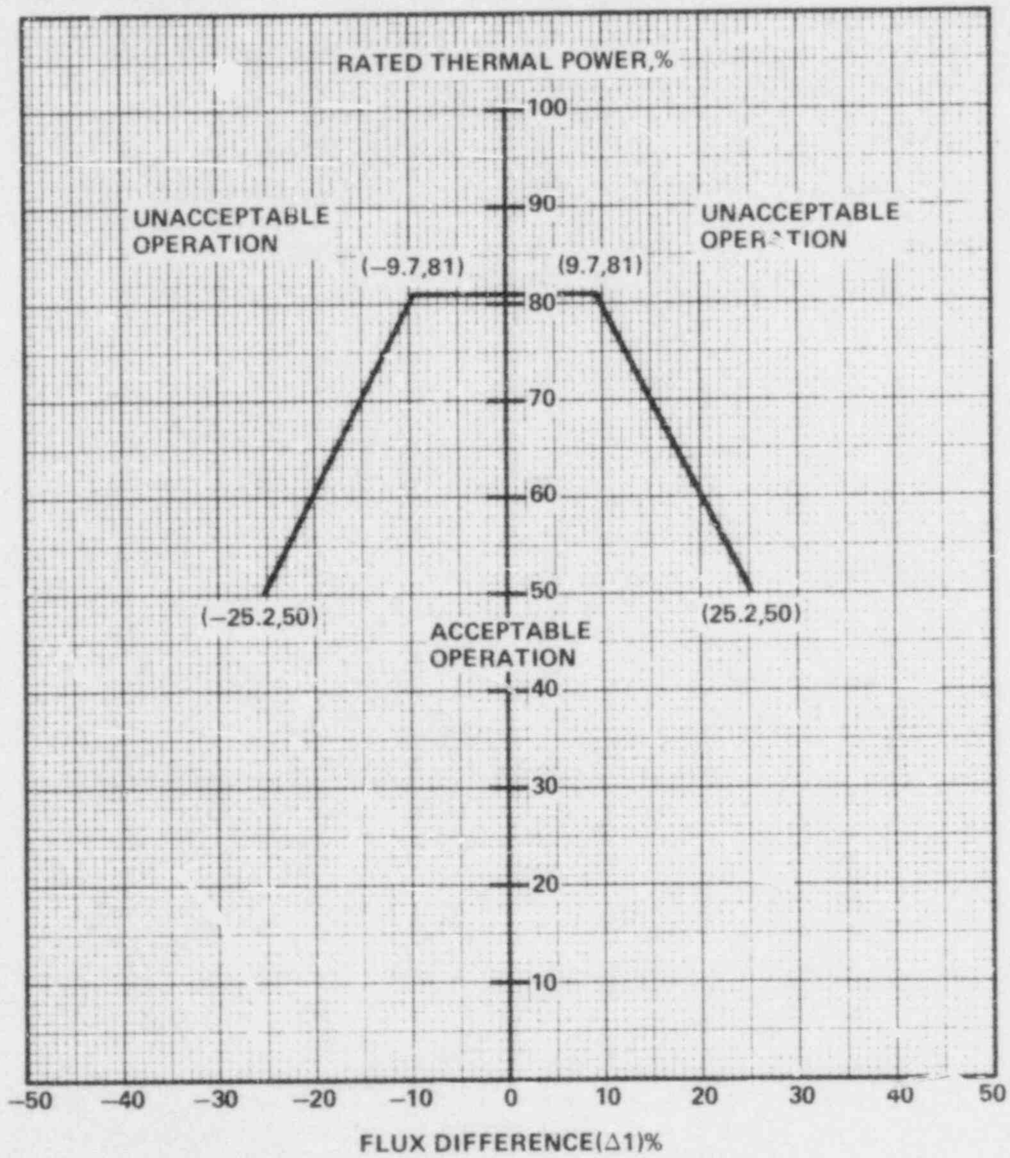

Figure 3.2–1 Axial Flux Difference Limits as a Function of Rated Thermal Power

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \left[\frac{2.10}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.20] [K(Z)] \text{ for } P \leq 0.5$$

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

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

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

   1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

   2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\overline{R}$.

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

2. The relationship:

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

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

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

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

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

b) At least once per 31 EFPD, whichever occurs first.

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

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. F_{xy}^{RTP} less than or equal to 1.71 for all core planes containing bank "D" control rods, and
2. F_{xy}^{RTP} less than or equal to 1.55 for all unrodded core planes.

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

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive (17 x 17 fuel elements).
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}:

1. The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit, and
2. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, 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.

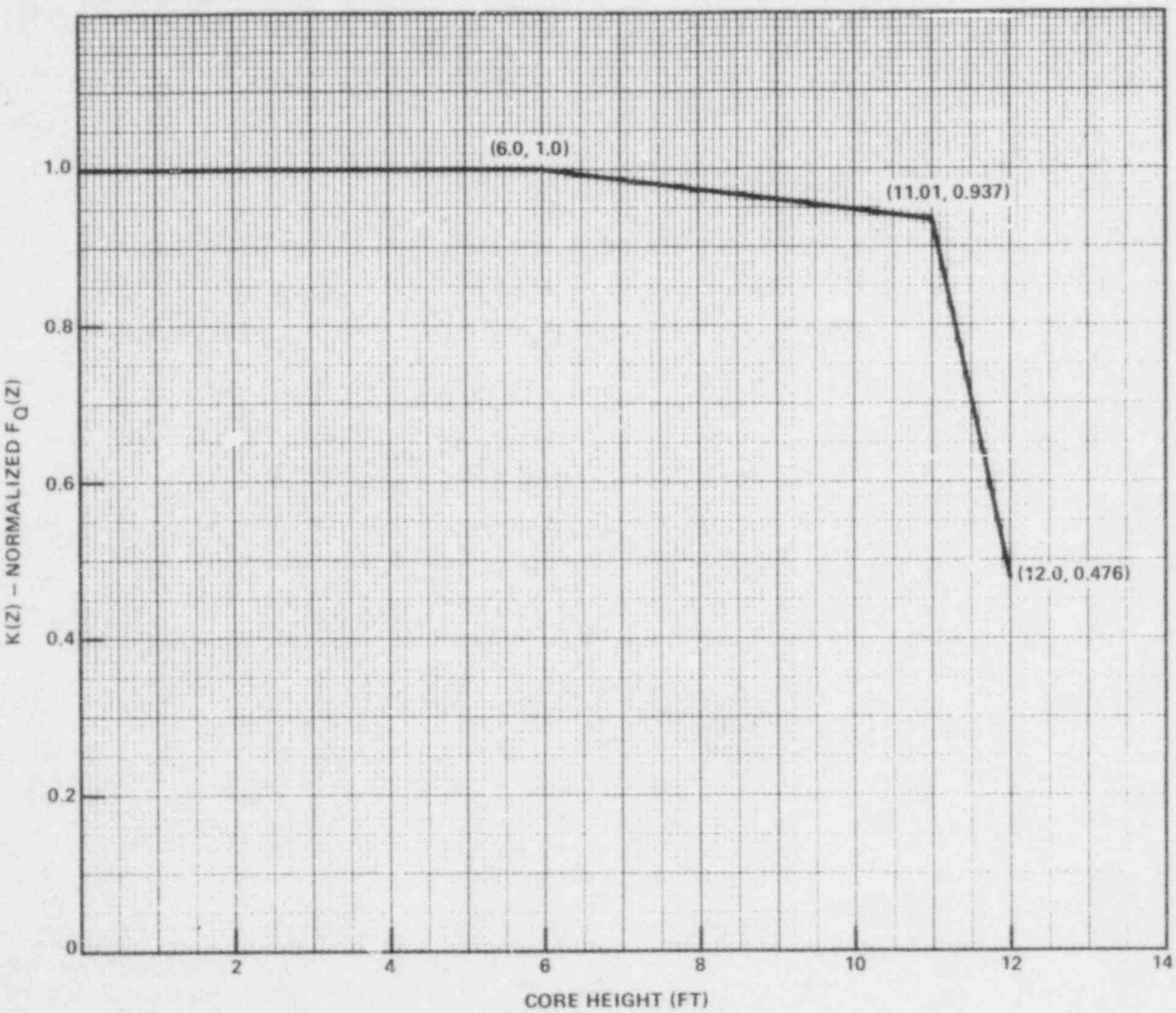

Figure 3.2–2 K(Z) – Normalized $F_Q(Z)$ as a Function of Core Height

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

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

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

where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ *, and

RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1.

ACTION:

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

a. 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,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to

ACTION Continued

exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

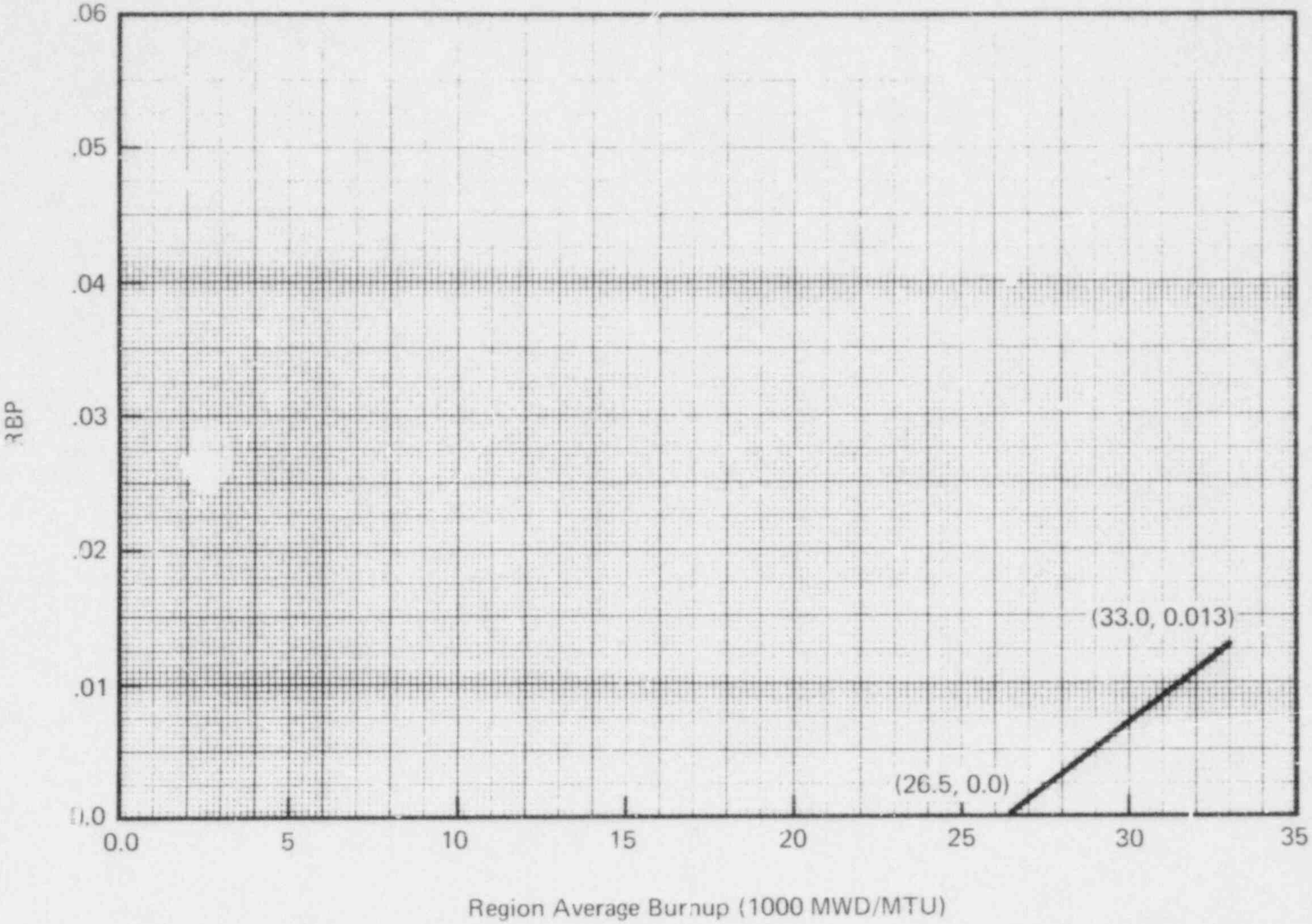

Figure 3.2-3 Rod Bow Penalty Fraction Versus Region Average Burnup

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 QUARANT POWER TILT RATIO at least once per hour until:

(a) Either 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:

a) Either 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 1.0 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

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

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.

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.

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:

(a) Either 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 1.0, 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.

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.

LIMITING CONDITION FOR OPERATION (Continued)

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:
   (a) Either 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 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.

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

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

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 of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops In Operation	LIMITS 2 Loops In Operation** & Loop Stop Valves Open	LIMITS 2 Loops In Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$585°F		
Pressurizer Pressure	$\geq$2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$278,400 gpm		

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

**Values dependent on NRC approval of ECCS evaluation for these conditions.

POWER DISTRIBUTION LIMITS

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.10]\,[K(Z)]}{(\bar{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

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

d. $\bar{R}_j$, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns above 90% of RATED THERMAL POWER in accordance with:

$$\bar{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}

e. σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\bar{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\bar{R}_j - R_{ij})^2\right]^{1/2}}{\bar{R}j}$$

f. The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system, respectively.

g. The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 above 90% OF RATED THERMAL POWER#.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by less than or equal to 4 percent, reduce THERMAL POWER one percent for every percent by

---

The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

ACTION (Continued)

which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to 90% or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by greater than 4 percent, reduce THERMAL POWER to 90% or less of RATED THERMAL POWER within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.8 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

   a) Increasing the THERMAL POWER above 90% of RATED THERMAL POWER, or

   b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

SURVEILLANCE REQUIREMENTS (Continued)

a) Increasing the THERMAL POWER above 90% of RATED THERMAL POWER, or

b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

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

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3.1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Three Loop Operation	3	2	2	1, 2	$7^{\#}$
Two Loop Operation	3	1**	2	1, 2	9
9. Pressurizer Pressure-Low	3	2	2	1, 2	$7^{\#}$
10. Pressurizer Pressure--High	3	2	2	1, 2	$7^{\#}$
11. Pressurizer Water Level--High	3	2	2	1, 2	$7^{\#}$
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	$7^{\#}$
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	$7^{\#}$
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	$7^{\#}$
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	$7^{\#}$

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. Undervoltage-Reactor Coolant Pump Busses	3-1/bus	2	2	1	7#
17. Underfrequency-Reactor Coolant Pump Busses	3-1/bus	2	2	1	7#
18. Turbine Trip					
A. Low Autc Stop Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	4	1	7#
19. Safe.y Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10
B. Above P-7	1/breaker	2	1/breaker per operating loop	1	11
21. Reactor Trip Breakers	2	1	2	1, 2, and *	1
22. Automatic Trip Logic	2	1	2	1, 2, and *	1

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above the P-6, (Block of Source Range Reactor Trip), setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and 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 the redundant channel(s) per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to $\leq$75% of RATED THERMAL and the Power Range, Neutron Flux trip setpoint is reduced to $\leq$85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors, is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

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

a. Below the P-6, (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6, (Block of Source Range Reactor Trip) setpoint, but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6, (Block of Source Range Reactor Trip) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6, (Block of Source Range Reactor Trip) setpoint, operation may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, 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 - Not applicable.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - Not applicable

TABLE 3.3-1 (Continued)

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below the P-8, (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint, within the next 2 hours. Operation below the P-8, (Block of Low Reactor Coolant Pump Flow and Reactor Coolant Pump Breaker Position) setpoint, may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-6	1 of 2 Intermediate range above setpoint (increasing power level)	1×10^{-10}	$< 3 \times 10^{-10}$	Allows manual block of source range reactor trip
	2 of 2 Intermediate range below setpoint (decreasing power level)	5×10^{-11}	$> 3 \times 10^{-11}$	Defeats the block of source range reactor trip
P-10	2 of 4 Power range above setpoint (increasing power level)	10%	<11%	Allows manual block of power range (low setpoint) and intermediate range reactor trips and intermediate range rod stop. Blocks source range reactor trip.
	3 of 4 Power range below setpoint (decreasing power level)	8%	>7%	Defeats the block of power range (low setpoint) and intermediate range reactor trips and intermediate range range rod stop. Input to P-7
P-7	2 of 4 Power range above setpoint or 1 of 2 Turbine Impulse chamber pressure above setpoint (Power level increasing)	10% Pressure equivalent to 10% RATED THERMAL POWER	<11% <11%	Allows reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurize low pressure, and Pressurizer high level.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip on: Low flow or reactor coolant pump breakers open in more than one loop, Undervoltage (RCP busses), Underfrequency (RCP busses), Turbine Trip, Pressurizer low pressure, and Pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Permit reactor trip on low flow or reactor coolant pump breaker open in a single loop.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Blocks reactor trip on low flow or reactor coolant pump breaker open in a single loop.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 4.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

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

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pump Busses	$\leq$ 1.2 seconds
17. Underfrequency-Reactor Coolant Pump Busses	$\leq$ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21. Reactor Trip Breakers	NOT APPLICABLE
22. Automatic Trip Logic	NOT APPLICABLE

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M	1, 2
3.	Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4.	Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5.	Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2, and *
6.	Source Range, Neutron Flux	S(7)	R(6)	M, S/U(1)	2, 3, 4, 5, and *
7.	Overtemperature ΔT	S	R(6)	M	1, 2
8.	Overpower ΔT	S	R(6)	M	1, 2
9.	Pressurizer Pressure--Low	S	R	M	1, 2
10.	Pressurizer Pressure--High	S	R	M	1, 2
11.	Pressurizer Water Level--High	S	R	M	1, 2
12.	Loss of Flow - Single Loop	S	R	M	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow - Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pump Busses	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pump Busses	N.A.	R	M	1
18. Turbine Trip				
A. Low Auto Stop Oil Pressure	N.A.	N.A.	S/U(1)	N.A.
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	N.A.
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	1
21. Reactor Trip Breaker	N.A.	N.A.	M(5) and S/U(1)	1, 2, and *
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, and *

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference > 2 percent.

(3) - Compare incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference ≥ 3 percent.

(4) - Manual ESF functional input check every 18 months.

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

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

(7) - Below the P-6, (Block of Source Range Reactor Trip), setpoint.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14*
d. Pressurizer Pressure -- Low-Low	3	2	2	1, 2, 3#	14*
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		14*
Two Loops Operating	3/operating steam line	2###/steam line twice in either operating steam line	2/operating steam line		15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Flow in Two Steam Lines-High				1, 2, 3##	
Three Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*
Two Loops Operating	2/operating steam	1###/any operating steam line	1/operating steam line		15
COINCIDENT WITH EITHER					
T_{avg}--Low-Low				1, 2, 3##	
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		14*
Two Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any operating loop		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Three Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops		14*
Two Loops Operating	1 pressure/ loop	1### pressure in any operating loop	1 pressure any operating loop		15
2. CONTAINMENT SPRAY					
a. Manual	2 sets 2 switches/set	1 set	2 sets	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) From Safety Injection Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Phase "B" Isolation					
1) Manual	2 sets 2 switches/set	1 set	2	1, 2, 3, 4	18
2) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
3) Containment Pressure--High-High	4	2	3	1, 2, 3	16
4. STEAM LINE ISOLATION					
a. Manual	2/steam line	1/steam line	2/operating steam line	1, 2, 3	21
b. Automatic Actuation Logic	2	1	2	1, 2, 3	20
c. Containment Pressure--Intermediate High-High	3	2	2	1, 2, 3	14
d. Steam Flow in Two Steam Lines--High				1, 2, 3##	
Three Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH EITHER					
T_{avg}--Low-Low				1, 2, 3##	
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		14*
Two Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any operating loop		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Three Loops Operating	1 pressure/loop	1 pressure any 2 loops	1 pressure any 2 loops		14*
Two Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure any operating loop		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT			TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6.	AUXILIARY FEEDWATER PUMP START						
	a.	Manual Initiation	2	1	2	1, 2, 3	21
	b.	Automatic Actuation Logic	2	1	2	1, 2, 3	20
	c.	Steam Generator Water Level Low-Low	3/stm. gen.	2/stm. gen in any operating stm. gen.	2/stm. gen.	1, 2, 3	14
	d.	SI	See #1 above (All SI initiating functions and requirements)				
	e.	Station Blackout	2	2	2	1, 2, 3	18
	f.	Main feed pump trip	2/pump	1/pump	1/pump	1, 2	18
7.	LOSS OF POWER						
	a.	4.16 kv Emergency Bus Under Voltage (Loss of Voltage)	3/Bus	2 Bus	2/Bus	1, 2, 3, 4	19*
	b.	4.16 Kv Emergency Bus Under Voltage (Grid Degraded Voltage)	3/Bus	2 Bus	2/Bus	1, 2, 3, 4	19*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11, (Pressurizer Pressure Block of Safety Injection), setpoint.

##Trip function may be blocked in this MODE below the P-12, (T_{avg} Block of Safety Injection) setpoint.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided the other Channel is OPERABLE.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST, provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the blocked condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be blocked for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-3 (Continued)

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour and the Minimum Channels OPERABLE requirement is met, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the 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 requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 20 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1 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 be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels above setpoint	2000 psig	$\leq$ 2010 psig	P-11 prevents manual block of safety injection actuation on low-low pressurizer pressure.
	With 2 of 3 pressurizer pressure channels below setpoint	1980 psig	$\leq$ 1990 psig	P-11 allows manual block of safety injection actuation on low-low pressurizer pressure.
P-12	With 2 of 3 T_{avg} channels above setpoint	543°F (Nominal)	$\leq$ 545°F	P-12 prevents manual block of safety injection actuation on high steam line flow.
	With 2 of 3 T_{avg} channels below setpoint	543°F (Nominal)	$\geq$ 541°F	P-12 allows manual block of safety injection actuation on high steam line flow.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 17 psia	$\leq$ 18.5 psia
d. Pressurizer Pressure--Low-Low	$\geq$ 1765 psig	$\geq$ 1755 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	$\leq$ A function defined as follows: a Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load	$\leq$ A function defined as follows: a Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load
	$T_{avg} \geq 543°F$	$T_{avg} \geq 542°F$
	$\geq$ 600 psig steam line pressure	$\geq$ 585 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 27.75 psia	$\leq$ 29.25 psia
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High	$\leq$ 27.75 psia	$\leq$ 29.25 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--Intermediate High-High	$\leq$ 17.8 psia	$\leq$ 19.3 psia
d. Steam Flow in Two Steam lines--High Coincident with T_{avg}--Low-Low Or Steam Line Pressure--Low	$\leq$ A function defined as follows: a Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load. $T_{avg} \geq 543°F$ $\geq$ 600 psig steam line pressure	$\leq$ A function defined as follows: a Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load. $T_{avg} \geq 542°F$ $\geq$ 585 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUES
6.	AUXILIARY FEEDWATER PUMP START		
a.	Manual	Not applicable	Not applicable
b.	Automatic Actuation Logic	Not applicable	Not applicable
c.	Steam Generator Water Level Low-Low	≥ 18% of narrow range instrument span each steam generator	≥ 17% of narrow range instrument span each steam generator
d.	S.I.	See 1 above (All S.I. Setpoints)	
e.	Station Blackout	≥ 57.5% Transfer Bus Voltage	≥ 52.5% Transfer Bus Voltage
f.	Trip of Main Feed Pump	N.A.	N.A.
7.	LOSS OF POWER		
a.	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2999 ± 60 volts with a 2.2 ± 0.03 second time delay	2912 ± 60 volts with a 3 ± 0.03 second time delay
b.	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3744 ± 1.4 volts with a 60 ± 3 second time delay	3619 ± 1.4 volts with a 75 ± 3 second time delay

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}/28.0^{(3)}$
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable
3. Pressurizer Pressure--Low-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/13.0^{(2)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}$
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 13.0^{(2)}/23.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}/28.0^{(3)}$
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq 15.0^{(2)}/25.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 5.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq 20.0^{(2)}/30.0^{(3)}$
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 10.0
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 13.0^{(2)}/23.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}/28.0^{(3)}$
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 8.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure--High-High	
a. Containment Quench Spray	$\leq$ 60.0
b. Containment Isolation - Phase "B"	$\leq$ 60.0
8. Containment Pressure--Intermediate High-High	
a. Steam Line Isolation	$\leq$ 7.0
9. Steam Generator Water Level--Low-Low	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0
10. Station Blackout	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0
11. Main Feedwater Pump Trip	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0
12. Steam Generator Water Level--High-High	
a. Turbine Trip - Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0
13. Loss of Power	
a. 4.16 kv Emergency Bus Undervoltage (Loss of voltage)	$\leq$ 13.3(4)
b. 4.16 kv Emergency Bus Undervoltage (Degraded voltage)	$\leq$ 11.5(4) with SI Signal $\leq$ 74.0(4) with no SI Signal

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, and Low Head Safety Injection pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) The response times shown are based on the time from when the signal reaches the trip setting until the diesel generator is supplying the emergency bus.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	1, 2, 3
d. Pressurizer Pressure--Low-Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low-Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. AUXILIARY FEEDWATER PUMPS				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Steam Generator Water	S	R	M	1, 2, 3
d. S.I.	See 1 above (all S.I. Surveillance Requirements)			
e. Station Blackout	N.A.	R	N.A.	1, 2, 3
f. Main Feedwater Pump Trip	N.A.	N.A.	R	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	M	1, 2, 3, 4
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage	N.A.	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once every other 31 days.

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

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area Criticality Monitor #	1	*	$\leq$ 15 mR/hr	$10^{-4} - 10^{+1}$ R/hr	22
b. Containment-Purge & Exhaust Isolation	1	6	$\leq$ 50 mR/hr	$10^{-4} - 10^{+1}$ R/hr	25
2. PROCESS MONITORS					
a. Ventilation Vent #					
i. Gaseous Gross Activity	1	**	$\leq 1 \times 10^{-5}$ μCi/ml	$10 - 10^{6}$ cpm	24
ii Particulate Gross Activity	1	**	$\leq 2 \times 10^{-9}$ μCi/ml	$10 - 10^{6}$ cpm	24
b. Containment					
i. Gaseous Activity					
a)Purge & Exhaust Isolation	1	6	$\leq 3.6 \times 10^{3}$ cpm	$10 - 10^{6}$ cpm	25
b)RCS Leakage Detection	1	1, 2, 3 & 4	N/A	$10 - 10^{6}$ cpm	23
ii. Particulate Activity					
a)Purge & Exhaust Isolation	1	6	$\leq 1 \times 10^{5}$ cpm	$10 - 10^{6}$ cpm	25
b)RCS Leakage Detection	1	1, 2, 3 & 4	N/A	$10 - 10^{6}$ cpm	23

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
# Common to Unit 1 and Unit 2

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Criticality Monitor #	S	R	M	*
b. Containment-Purge & Exhaust Isolation	S	R	M	6
2. PROCESS MONITORS				
a. Ventilation Vent #				
i. Gaseous Gross Activity	S	R	M	**
ii. Particulate Gross Activity	S	R	M	**
b. Containment				
i. Gaseous Activity				
a)Purge & Exhaust Isolation	S	R	M	6
b)RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity				
a)Purge & Exhaust Isolation	S	R	M	6
b)RCS Leakage Detection	S	R	M	1, 2, 3, & 4

*With fuel in the storage pool or building
**With irradiated fuel in the storage pool
#Common to Unit 1 and Unit 2

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall shall be OPERABLE with:

a. At least 75% of the detector thembles,

b. A minimum of 2 detector thimbles per core quadrant, and

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

APPLICABILITY:

When the movable incore detection system is used for:

a. Recalibration of the ecore neutron flux detection system,

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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE, at least once per 24 hours, by normalizing each detector output to be used during its use 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)

AUXILIARY SHUTDOWN PANEL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

AUXILIARY SHUTDOWN PANEL MONITORING INSTRUMENTATION*

INSTRUMENT	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant Temperature - Average	530-630°F	1
2. Pressurizer Pressure	1700-2500 psig	1
3. Pressurizer Level	0-100%	1
4. Auxiliary Feed Pump Discharge Header Pressure	500-1500 psig	1
5. Emergency Condensate Storage Tank Level	0-100%	1
6. Charging Flow	0-150 GPM	1
7. Main Steam Line Pressure	0-1400 psig	1
8. Steam Generator Level	0-100%	1
9. Relay Room Positive Ventilation	0-.50 inches of H_2O	1

*Located at Elevation 254' in the Emergency Switchgear and Relay Room.

TABLE 4.3-6

AUXILIARY SHUTDOWN PANEL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant Temperature - Average	M	R
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Auxiliary Feed Pump Discharge Header Pressure	M	R
5. Emergency Condensate Storage Tank Level	M	R
6. Charging Flow	M	R
7. Main Steam Line Pressure	M	R
8. Steam Generator Level	M	R
9. Relay Room Positive Ventilation	M	R

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the total number of channels shown in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Containment Pressure	2	1
2.	Reactor Coolant Outlet Temperature-T_{hot} (wide range)	2	1
3.	Reactor Coolant Inlet Temperature-T_{cold} (wide range)	2	1
4.	Reactor Coolant Pressure-Wide Range	1	1
5.	Pressurizer Water Level	1	1
6.	Steam Line Pressure	2/steam generator	1/steam generator
7.	Steam Generator Water Level-Narrow Range	2/steam generator	1/steam generator
8.	Refueling Water Storage Tank Water Level	1	1
9.	Boric Acid Tank Solution Level	1	1
10.	Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
11.	Reactor Coolant System Subcooling Margin Monitor	2	1
12.	PORV Position Indicator	2/valve	1/valve
13.	PORV Block Valve Position Indicator	1/valve	1/valve
14.	Safety Valve Position Indicator	1/valve	1/valve

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature-T_{hot} (wide range)	M	R
3. Reactor Coolant Inlet Temperature-T_{cold} (wide range)	M	R
4. Reactor Coolant Pressure-Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level-Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Boric Acid Tank Solution Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position Indicator	M	R

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11 inoperable:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

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

4.3.3.7.3 The non-supervised circuits between the local panels in Specification 4.3.3.7.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	MINIMUM DETECTORS REQUIRED HEAT	SMOKE
1. Reactor Coolant Pumps	1/pump*	
2. Control Room		
a. Under floor-loop 1	2	2
b. Under floor-loop 2	2	
c. Normal Air Supply#		1
d. Emergency Air Supply		1
3. Primary Cable Vault and Tunnel	2	3
4. Service Building Cable Vault and Tunnel	5	4
5. Emergency Switchgear Room Emergency Air Supply		1
6. Station Battery Rooms		1/room
7. Diesel Generators	2/room	
8. Fuel Oil Pump House#		
a. Room 1	1	1
b. Room 2	1	1
9. Motor Control Center		2
10. Auxiliary Building Charcoal Filters (Common with Unit 1)		
a. Intake Side	3/room	
b. Outlet Side	3/room	

*A RCP bearing or motor temperature may be substituted for an inoperable RCP heat detector provided the bearing or motor temperature(s) is monitored at least once per hour when the RCP is in operation.

#Common to Units 1 and 2

AXIAL POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The axial power distribution monitoring system (APDMS) shall be OPERABLE with:

a. At least two detector thimbles available for which $\bar{R}$ has been determined from full incore flux maps. These two thimbles shall be those having the lowest uncertainty, σ, covering the full configuration of permissible rod patterns permitted at RATED THERMAL POWER.

b. At least two movable detectors, with associated devices and readout equipment, available for mapping $F_j(Z)$ in the above required thimbles.

APPLICABILITY: When the APDMS is used for monitoring the axial power distribution*#.

ACTION: With the APDMS inoperable, do not use the system for determining the Axial Power Distribution. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 The full incore flux maps used to determine $\bar{R}$ and for monitoring $F_j(Z)$ shall be updated at least once per 31 days. The continued accuracy and representativeness of the selected thimbles shall be verified by using their latest flux maps to update the $\bar{R}$ for each representative thimble. The original uncertainty, σ, shall not be updated, except as follows:

*Except as provided in Specification 4.2.6.1.b.

#The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.8.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$

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

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 with power removed from the loop stop valve operators.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENT

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

4.4.1.2 At least once per 31 days, with the reactor coolant loops in operation by verifying that the power is removed from the loop stop valve operators.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
2. Reactor coolant Loop B and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

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

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

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

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

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

4. Residual Heat Removal Subsystem A,**

5. Residual Heat Removal Subsystem B.**

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

APPLICABILITY: MODES 4 and 5.

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; be in COLD SHUTDOWN within 20 hours.

b. With no 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 coolant loop to operation.

---

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 340°F unless 1) the pressurizer water volume is less than 457 cubic feet or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

**The offsite or emergency power source may be inoperable in MODE 5.

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

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

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

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

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

ISOLATED LOOP

LIMITING CONDITION FOR OPERATION

3.4.1.2 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops, unless the loop has been drained for maintenance.

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

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's stop valves; either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.2 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops at least once per 24 hours and within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop.

I[illegible]ED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.3 A reactor coolant loop cold leg stop valve shall remain closed until:

a. The isolated loop has been operating on a recirculation flow of greater than or equal to 125 gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops.

b. The reactor is subcritical by at least 1.77 percent $\Delta k/k$.

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The reactor shall be determined to be subcritical by at least 1.77 percent $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 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.

SURVEILLANCE REQUIREMENTS

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

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

SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 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 HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

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

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

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 Two power relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With one 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 in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

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

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve trough one complete cycle of full travel.

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 125 kw of pressurizer heaters and a water volume of less than or equal to 1240 cubic feet.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply for the pressurizer heaters either restore the inoperable emergency power supply 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 breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator in a non-isolated reactor coolant loop shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators in non-isolated reactor coolant loops 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 required Specification 4.0.5.

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

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

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

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

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%, and

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

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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

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

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

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

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

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

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 of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in 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 into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

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

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

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

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

   4. A major steam line or feedwater line break.

REACTOR COOLANT SYSTEM

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 20% of the nominal wall thickness caused by degradation.

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

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

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

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. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1.8 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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

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

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C–2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S. G.</td><td>C–1</td><td>None</td></tr>
<tr><td>C–2</td><td>Plug defective tubes</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–3</td><td rowspan="3">Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.

Prompt notification to NRC pursuant to specification 6.9.1</td><td>All other S. G.s are C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S. G.s C–2 but no additional S. G. are C–3</td><td>Perform action for C–2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S. G. is C–3</td><td>Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1</td><td>N/A</td><td>N/A</td></tr>
</table>

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The containment atmosphere particulate radioactivity monitoring system, and

b. The containment atmosphere gaseous radioactivity monitoring system, or

c. The containment sump discharge flow measurement system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

b. Containment sump discharge flow measurement system-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

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

e. 30 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 GPM leakage from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

*The leakage limit for any RHR system isolation valve shown in Table 3.4-1 shall be 5 GPM.

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 particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

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

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

e. Monitoring the reactor head flange leakoff temperature 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 pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 4 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
2-SI-85 2-SI-93 2-SI-107 2-SI-119	High head safety injection to cold legs and hot legs
MOV-2836 MOV-2869A, B	High head safety injection off charging header
MOV-2867C, D	Boron injection tank outlet valves
2-SI-91 2-SI-99 2-SI-105	Low head safety injection to cold legs
2-SI-126 2-SI-128	Low head safety injection to hot legs
2-SI-151 2-SI-170 2-SI-153 2-SI-185 2-SI-168 2-SI-187	Accumulator discharge check valves
MOV-2700 MOV-2701 MOV-2720A, B	RHR system isolation valves
MOV-2890 A, B, C & D	Low head safety injection to cold legs and hot legs

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

b. With any one or more chemistry parameters in excess of the 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.

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 T_{avg} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	MINIMUM ANALYSIS FREQUENCIES
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

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

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 μCi/gram DOSE EQUIVALENT I-131, and

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

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

ACTION:

MODES 1, 2 and 3*

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

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant greater than

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

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

2. Fuel burnup by core region,

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

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

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

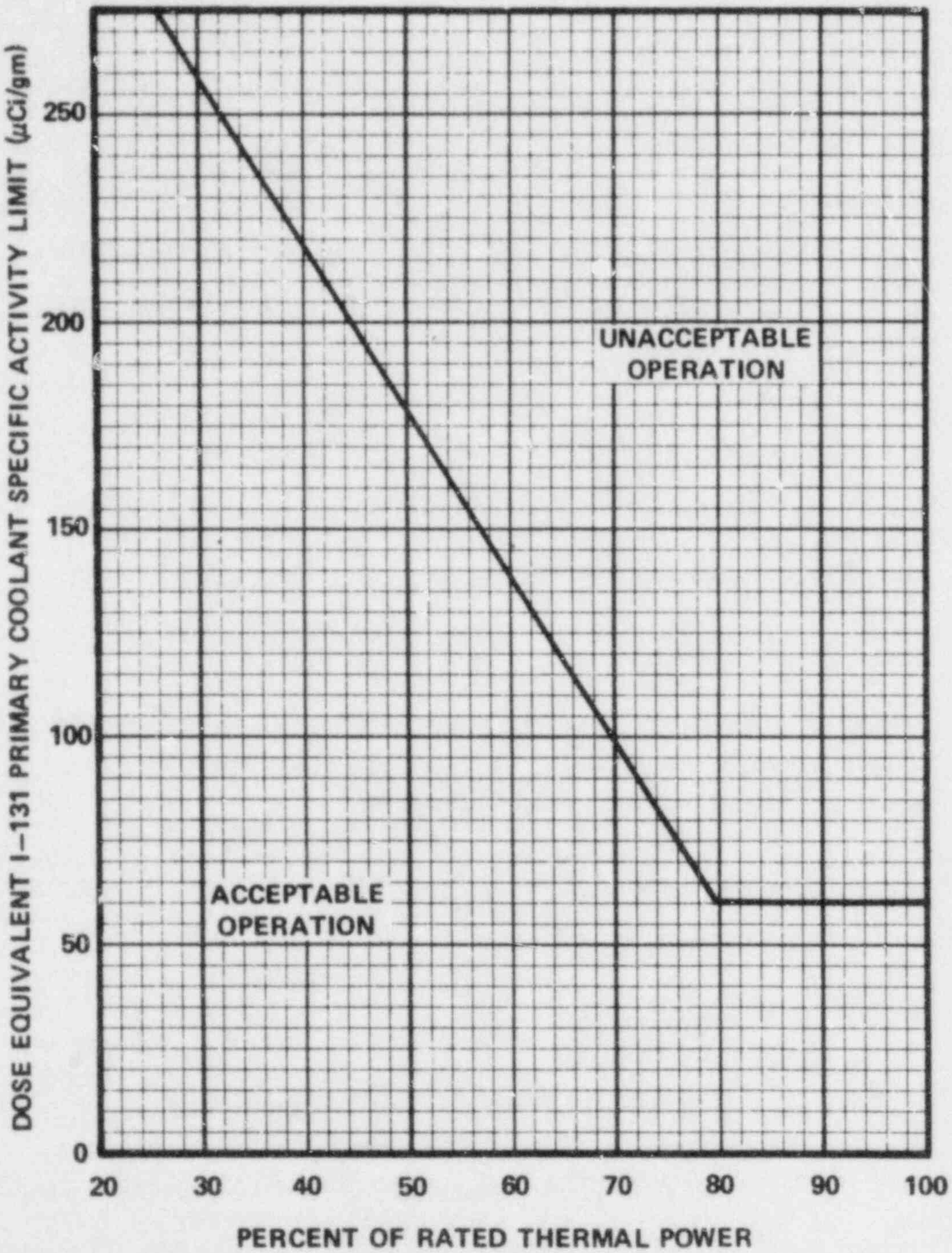

FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

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

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 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, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

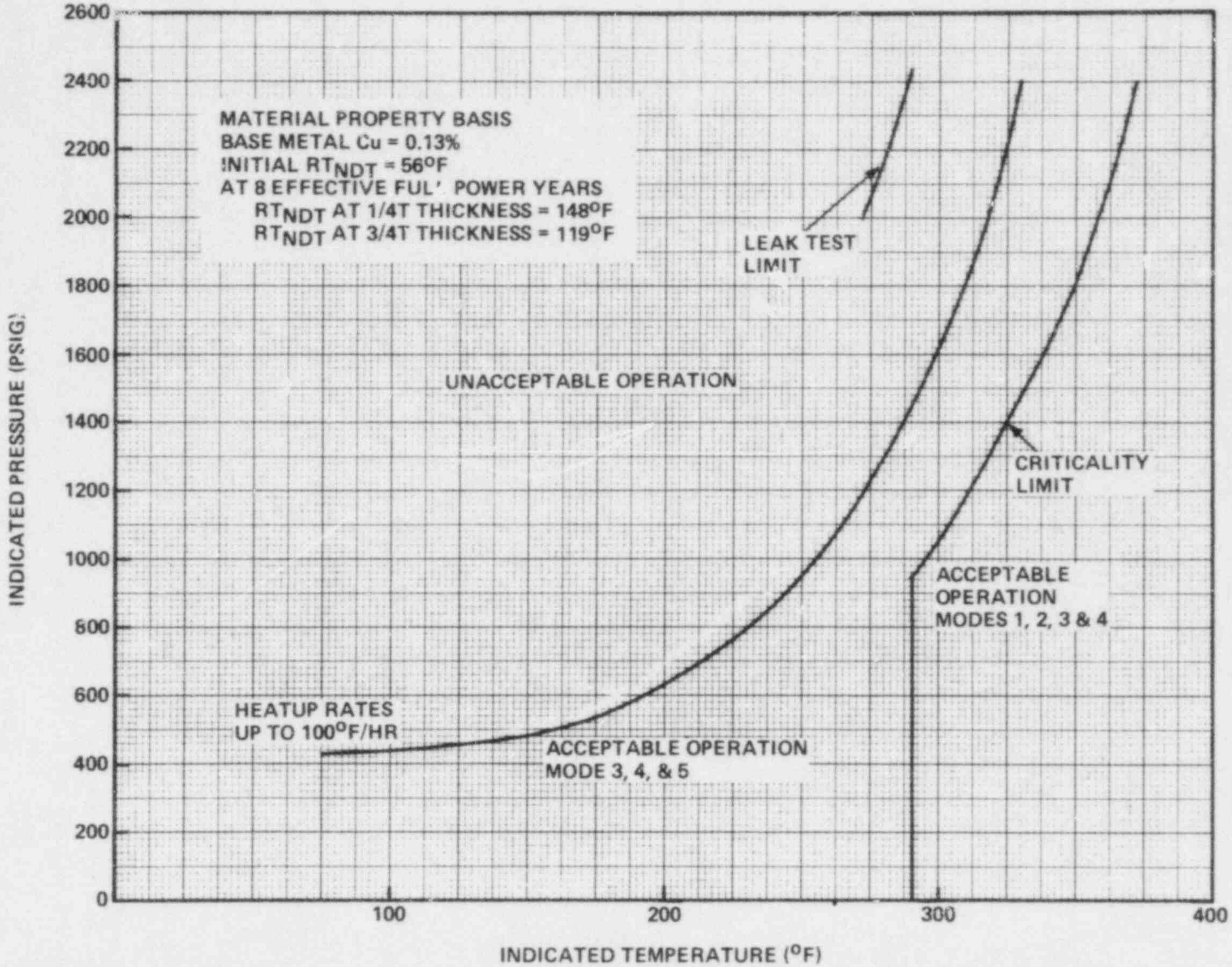

Figure 3.4-2 Reactor Coolant System Temperature-Pressure Heatup Limitations

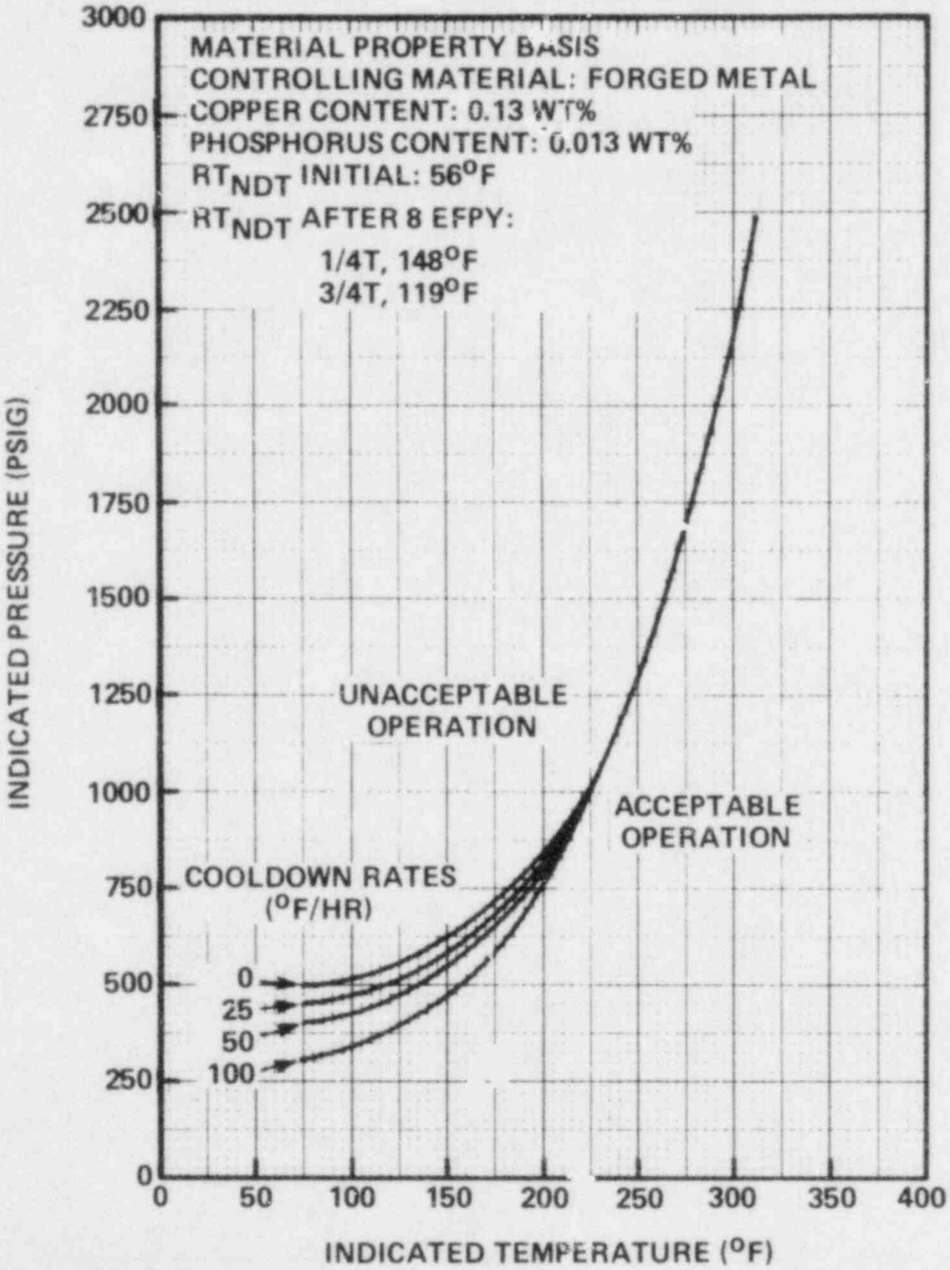

Figure 3.4-3 Reactor Coolant System Temperature-Pressure Cooldown Limitations

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F or cooldown of 200°F, in any one hour period, and

b. A maximum spray water temperature and pressurizer 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.

LIMITING CONDITION FOR OPERATION

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

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 475 psig whenever any RCS cold leg temperature is less than or equal to 340°F, and 2) less than or equal to 405 psig whenever any RCS cold leg temperature is less than 140°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches, or

c. A maximum pressurizer water volume of 457 cubic feet with all RCS cold leg temperatures greater than or equal to 320°F.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 340°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL 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.

c. Verifying the PORV key switch is in the AUTO position and the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

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

4.4.9.3.3 The pressurizer water volume shall be verified to be less than or equal to 457 cubic feet at least once per 12 hours when the pressurizer is being used for overpressure protection

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

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES.

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

STEAM GENERATOR SUPPORTS

LIMITING CONDITION FOR OPERATION

3.4.10.2 The temperature of the steam generator supports shall be maintained:

a. Greater than 225°F for A572 material monitored at a middle level corner during operation and at a top level corner during heatup of the supports.

b. Less than 355°F at the monitored top level corner.

c. Greater than 85°F for A36 material monitored at a bottom level corner during heatup.

APPLICABILITY: With pressurizer pressure greater than 1000 psig.

ACTION:

With the temperature of any steam generator support outside the above limits, restore the temperature to within the limit within 4 hours or be below 1000 psig within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.2.1 The steam generator support temperatures for A572 material shall be verified to be within the specified limits at least once per 12 hours.

4.4.10.2.2 The steam generator support temperatures for A36 material shall be verified to be within the specified limit prior to exceeding a pressurizer pressure of greater than 1000 psig.

4.4.10.2.3 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7580 and 7756 gallons

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 599 and 667 psig.

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 one hour or be in HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig. Power lock out of valves is not permitted in MODE 3 when below 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 5% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that the breaker supplying power to the isolation valve operator is locked in the off position.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

1. When a simulated RCS pressure signal exceeds 2010 psig,

2. Upon receipt of a safety injection test signal.

ECCS SUBSYSTEMS - T_{avg} GREATER THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

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

c. The provisions of Specification 3.0.4 are not applicable to Specifications 3.5.2.a and 3.5.2.b for one hour following heatup above 340°F or prior to cooldown below 340°F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

SURVEILLANCE REQUIREMENTS (Continued)

Valve Number	Valve Function	Valve Position
a. MOV-2890A	a. LHSI to hot leg	a. closed
b. MOV-2890B	b. LHSI to hot leg	b. closed
c. MOV-2836	c. Ch pump to cold leg	c. closed
d. MOV-2869A	d. Ch pump to hot leg	d. closed
e. MOV-2869B	e. Ch pump to hot leg	e. closed

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

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

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

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

d. At least once per 18 months by:

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

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

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

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump, and

      b) Low head safety injection pump.

SURVEILLANCE REQUIREMENTS (Continued)

f. By verifying that each of the following pumps develop the indicated discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump greater than or equal to 2410 psig.

   2. Low head safety injection pump greater than or equal to 156 psig

g. By verifying that the following manual valves requiring adjustment to prevent pump "runout" and subsequent component damage are locked and tagged in the proper position for injection:

   1. Within 4 hours following completion of any repositioning or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

1.	2-SI-89	Loop A Cold Leg
2.	2-SI-97	Loop B Cold Leg
3.	2-SI-103	Loop C Cold Leg
4.	2-SI-116	Loop A Hot Leg
5.	2-SI-111	Loop B Hot Leg
6.	2-SI-123	Loop C Hot Leg

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

   1. For high head safety injection lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is $\geq$ 384 gpm, and

      b) The total pump flow rate is $\leq$ 650 gpm.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank upon being manually realigned or from the containment sump when the suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

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

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

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

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F.

EMERGENCY CORE COOLING SYSTEMS

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

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 340°F by verifying that the control switch is in the pull to lock position.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of at least 900 gallons,

b. Between 20,000 and 22,500 ppm of boron, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection 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 HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

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

b. At least once per 24 hours 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.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A contained borated water volume of between 475,058 and 487,000 gallons.

b. Between 2000 and 2100 ppm of boron, and

c. A solution temperature between 40°F and 50°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1., and

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals with gas at Pa, greater than or equal to 40.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

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*Except valves, blind flanges and deactiviated automatic valves which are located inside the containment and are locked sealed or otherwise sealed 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.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.1 percent by weight of the containment air per 24 hours at P_a, greater than or equal to 40.6 psig, or

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, greater than or equal to 40.6 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at P_a, greater than or equal to 40.6 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 .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 .75 L_a a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .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 Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a

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

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a, greater than or equal to 40.6 psig.

d. Type B and C tests shall be conducted with gas at P_a, greater than or equal to 40.6 psig, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Penetrations using continuous leakage monitoring systems

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

f. Type B test for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, greater than or equal to 40.6 psig, at intervals no greater than once per 3 years.

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

h. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, greater than or equal to 40.6 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

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

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

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

 4. The provisions of Specification 3.0.4 are not applicable.

b. With a 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. *After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage through the bubble flow detector when the volume between the door seals is pressurized to greater than or equal to P_a, 40.6 psig, for at least 2 minutes,

*Exemption to Appendix "J" of 10 CFR 50.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 6 months by conducting an overall air lock leakage test at greater than or equal to P_a, 40.6 psig, and by verifying that the overall air lock leakage rate is within its limit#, and

c. At least once per 18 months during shutdown 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.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal air partial pressure shall be maintained greater than or equal to 8.9 psia and within the acceptable operation range of the applicable RWST water temperature limit lines and bulk air temperature limit lines shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment internal air partial pressure less than 8.9 psia or above the applicable RWST water temperature limit line or bulk air temperature line shown on Figure 3.6-1, restore the internal air partial pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

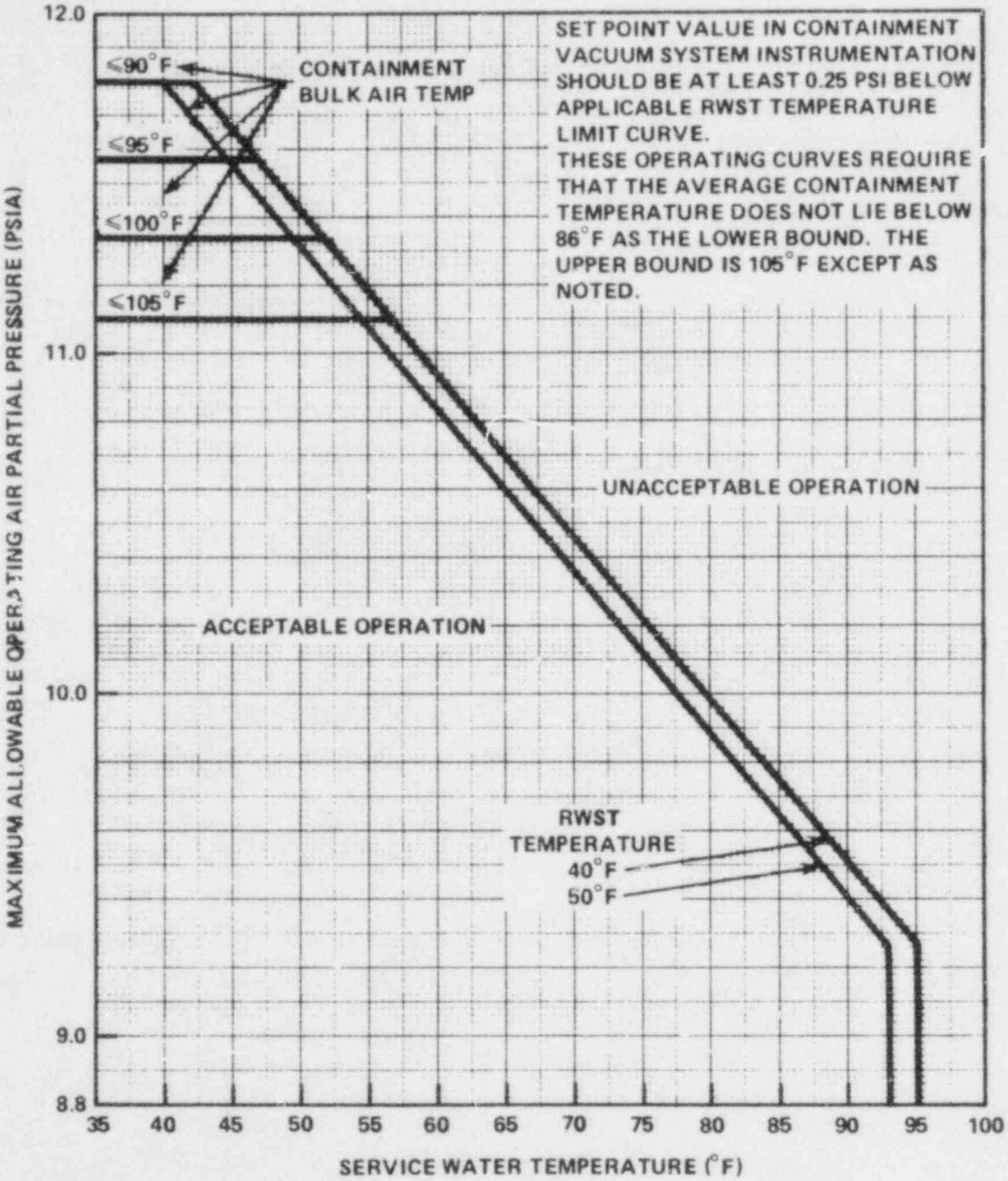

Figure 3.6-1 Maximum Allowable Primary Containment Air Partial Pressure Versus Service Water Temperature and RWST Water Temperature

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall be maintained greater than or equal to 86°F and less than or equal to 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature greater than 105°F or less than 86°F, restore the average air temperature to within the limit within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The primary containment average air temperature shall be the weighted average of at least the minimum number of temperatures at the following locations and shall be determined at least once per 24 hours:

Location			Weight Factor(WF)	Min. No. of Temperature Detectors
a.	Containment dome	Elev. ~ 390	0.04789	1
b.	Inside crane wall	Elev. ~ 329	0.09373	2
c.	Annulus	Elev. ~ 329	0.02283 (0.02935)*	2
d.	Annulus	Elev. ~ 238	0.08309	1
e.	Cubicles	Elev. ~ 268	**	1

4.6.1.5.2 The average containment air temperature shall be determined by the following relationship:

$$T_{containment} = \frac{1.0}{\sum_{i=1}^{n} \frac{WF_i}{T_i}} \quad \text{where}$$

WF_i is the weight factor for the temperature T_i, of the i^{th} temperature measurement.

* Weight factor to be used for pressurizer cubicle at Elev. 268.

**Weight factor to be used for cubicles A=0.03932, B=0.03597., C=0.03619

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete and liner, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
2. Verifying the temperature of the borated water in the refueling water storage tank is within the limits shown on Figure 3.6-1.

b. Verifying that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 123 psig when tested pursuant to Specification 4.0.5.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--high-high signal.
2. Verifying that each spray pump starts automatically on a Containment Pressure--high-high signal.

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

CONTAINMENT RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The containment recirculation spray system shall be OPERABLE with:

a. Four separate and independent containment recirculation spray subsystems, each composed of a spray pump, associated heat exchanger and flow path.

b. Two separate and independent outside recirculation spray pump casing cooling subsystems, each composed of a casing cooling pump, and flow path capable of transferring fluid from the casing cooling tank to the suction of the outside recirculation spray pumps.

c. One casing cooling tank shall be OPERABLE with:

1. Contained borated water volume of at least 116,500 gallons.
2. Between 2000 and 2100 ppm boron concentration.
3. A solution temperature greater than or equal to 35° and less than or equal to 50°F.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

a. With one containment recirculation spray subsystem or casing cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours; restore the inoperable subsystem to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

b. With the casing cooling tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2.1 Each containment recirculation spray subsystem and casing cooling subsystem 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying, that on recirculation flow, each outside recirculation spray pump develops a discharge pressure of greater than or equal to 115 psig and each casing cooling pump develops a discharge pressure of greater than or equal to 46 psig when tested pursuant to Specification 4.0.5.

c. At least once per 18 months by:

1. Verifying that on a Containment Pressure--High-High signal, each casing cooling pump starts automatically without time delay, and each recirculation spray pump starts automatically with the following time delays: inside 195 ± 9.75 seconds, outside 210 ± 21 seconds.

2. Verifying that each automatic valve in in the flow path actuates to its correct position on a Containment Pressure--high-high test signal.

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

4.6.2.2.2 The casing cooling tank shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the casing cooling tank temperature.

CHEMICAL ADDITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The chemical addition system shall be OPERABLE with:

a. A chemical addition tank containing a volume of between 4800 and 5500 gallons of between 12 and 13 percent by weight NaOH solution, and

b. A chemical addition flow path capable of adding NaOH solution from the chemical addition tank to both containment quench spray system pumps via the RWST.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.3 The chemical addition system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and
2. Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure -- high-high test signal.

d. At least once per 5 years by verifying individual flow from the RWST and the chemical addition tank thru the drain lines in the cross connection between the respective tanks.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each weight or spring loaded check valve testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is less than 1.2 psid and opens when the differential pressure in the direction of flow is greater than or equal to 1.2 psid but less than 5.0 psid.

b. 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 the applicable cycle test, above, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.

d. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is less than 1.2 psid and opens when the differential pressure in the direction of flow is greater than or equal to 1.2 psid but less than or equal to 5.0 psid.

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

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
A.	PHASE "A" ISOLATION		
1.	MOV-2380	Reactor Coolant Pump Seal Water Return	10
2.	MOV-2381	Reactor Coolant Pump Seal Water Return	10
3.	HCV-2200A	Reactor Coolant Letdown Line	10
4.	HCV-2200B	Reactor Coolant Letdown Line	10
5.	HCV-2200C	Reactor Coolant Letdown Line	10
6.	TV-2204	Reactor Coolant Letdown Line	10
7.	TV-SI200	Nitrogen to Pressurizer Relief Tank and SI Accumulators	60
8.	TV-DG200A	Primary Drains Transfer Tank Pump Discharge	60
9.	TV-DG200B	Primary Drains Transfer Tank Pump Discharge	60
10.	TV-DA200A	Containment Sump Pump Discharge to Waste Drain Tanks	60
11.	TV-DA200B	Containment Sump Pump Discharge to Waste Drain Tanks	60
12.	TV-BD200A	Steam Generator Blowdown	60
13.	TV-BD200B	Steam Generator Blowdown	60
14.	TV-BD200C	Steam Generator Blowdown	60

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
15.	TV-BD200D	Steam Generator Blowdown	60
16.	TV-BD200E	Steam Generator Blowdown	60
17.	TV-BD200F	Steam Generator Blowdown	60
18.	TV-RM200A	Air Radiation Monitor Return	60
19.	TV-RM200D#	Air Radiation Monitor Return	60
20.	TV-RM200B	Air Radiation Monitor Supply	60
21.	TV-RM200C	Air Radiation Monitor Supply	60
22.	TV-2519A	Primary Grade Water	10
23.	TV-VG200A	Primary Vent Header	60
24.	TV-VG200B	Primary Vent Header	60
25.	TV-SI201	Safety Injection Accumulators to Waste Gas Charcoal Filters	60
26.	HCV-2936	Safety Injection Accumulators to Waste Gas Charcoal Filters	10
27.	TV-SS204A	Pressurizer Relief Tank Sample	60
28.	TV-SS204B	Pressurizer Relief Tank Sample	60
29.	TV-SS200A	Pressurizer Liquid Space Sample	60
30.	TV-SS200B	Pressurizer Liquid Space Sample	60
31.	TV-SS206A	Primary Coolant Hot Leg Sample	60
32.	TV-SS206B	Primary Coolant Hot Leg Sample	60

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
33.	TV-SS202A	Primary Coolant Cold Leg Sample	60
34.	TV-SS202B	Primary Coolant Cold Leg Sample	60
35.	TV-LM200A	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
36.	TV-LM200B	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
37.	TV-LM200C	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
38.	TV-LM200D	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
39.	TV-LM200E	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
40.	TV-LM200F	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
41.	TV-LM200G	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
42.	TV-LM200H	Reactor Containment Leakage Monitoring Lines to Open Pressure Taps	60
43.	TV-SS201A	Pressurizer Vapor Space Sample	60
44.	TV-SS201B	Pressurizer Vapor Space Sample	60
45.	TV-SV202-1#	Condenser Air Ejector Vent	60
46.	TV-SV203	Condenser Air Ejector Vent	60

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
47.	TV-CV250A	Containment Vacuum Pump Suction	60
48.	TV-CV250B	Containment Vacuum Pump Suction	60
49.	TV-CV250C	Containment Vacuum Pump Suction	60
50.	TV-CV250D	Containment Vacuum Pump Suction	60
51.	TV-SS203	Residual Heat Removal System Sample Lines	60
52.	TV-LM201A	Reactor Containment Leakage Monitoring Lines to Reference System	60
53.	TV-LM201B	Reactor Containment Leakage Monitoring Lines to Reference System	60
54.	TV-LM201C	Reactor Containment Leakage Monitoring Lines to Reference System	60
55.	TV-LM201D	Reactor Containment Leakage Monitoring Lines to Reference System	60
56.	TV-2859	Safety Injection Test Line	10
57.	TV-2842	Safety Injection Test Line	10
58.	TV-SS212A	Steam Generator Surface Sample	60
59.	TV-SS212B	Steam Generator Surface Sample	60
60.	TV-MS209#	Main Steam Drains to Condenser	60
61.	TV-MS210#	Main Steam to Blowdown	60
62.	TV-SV202-2#	Condenser Air Ejector Vent	60
63.	FCV-AS200A#	Condenser Air Ejector Steam Supply	60
64.	FCV-AS200B#	Condenser Air Ejector Steam Supply	60

TABLE 3.6-1 (Cont.)

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
65. TV-IA201A	Containment Instrument Air Supply	60
66. TV-IA201B	Containment Instrument Air Supply	60
67. TV-IA202A	Containment Instrument Air Return	60
68. TV-IA202B#	Containment Instrument Air Return	60
B. PHASE "B" ISOLATION		
1. TV-CC203A	Component Cooling Water From RHR System and Excess Letdown Heat Exchanger	60
2. TV-CC203B	Component Cooling Water From RHR System and Excess Letdown Heat Exchanger	60
3. TV-CC201A	Reactor Coolant Pump Thermal Barrier Cooling Water Return	60
4. TV-CC201B	Reactor Coolant Pump Thermal Barrier Cooling Water Return	60

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
5.	TV-CC200A	Chilled Water From Containment Air Recirculation Coils	60
6.	TV-CC200B	Chilled Water From Containment Air Recirculation Coils	60
7.	TV-CC200C	Chilled Water From Containment Air Recirculation Coils	60
8.	TV-CC205A	Chilled Water From Containment Air Recirculation Coils	60
9.	TV-CC205B	Chilled Water From Containment Air Recirculation Coils	60
10.	TV-CC205C	Chilled Water From Containment Air Recirculation Coils	60
11.	TV-CC204A	Reactor Coolant Pumps, Cooling Water In	60
12.	TV-CC204B	Reactor Coolant Pumps, Cooling Water In	60
13.	TV-CC204C	Reactor Coolant Pumps, Cooling Water In	60
14.	TV-CC202A	Reactor Coolant Pumps and Shroud Cooling Cooling Water Out	60
15.	TV-CC202B	Reactor Coolant Pumps and Shroud Cooling, Cooling Water Out	60
16.	TV-CC202C	Reactor Coolant Pumps and Shroud Cooling, Cooling Water Out	60

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
17.	TV-CC202D	Reactor Coolant Pumps and Shroud Cooling, Cooling Water Out	60
18.	TV-CC202E	Reactor Coolant Pumps and Shroud Cooling, Cooling Water Out	60
19.	TV-CC202F	Reactor Coolant Pumps and Shroud Cooling, Cooling Water Out	60
20.	TV-BD200A	Steam Generator Blowdown	50
21.	TV-BD200B	Steam Generator Blowdown	60
22.	TV-BD200C	Steam Generator Blowdown	60
23.	TV-BD200D	Steam Generator Blowdown	60
24.	TV-BD200E	Steam Generator Blowdown	60
25.	TV-BD200F	Steam Generator Blowdown	60

C. CONTAINMENT PURGE AND EXHAUST (VENTILATION DUCTS)

1.	MOV-HV200A*	Purge Supply	NA
2.	MOV-HV200B*	Purge Supply	NA
3.	MOV-HV202*	Alternate Supply	NA
4.	MOV-HV200C*	Purge Exhaust	NA
5.	MOV-HV200D*	Purge Exhaust	NA
6.	MOV-HV201*	Bypass	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
D. MANUAL		
1. 2-SI-47*	Safety Injection Accumulator Make Up	NA
2. 2-RH-38*	Residual Heat Removal System to Refueling Water Storage Tank	NA
3. 2-RH-37*	Residual Heat Removal System to Refueling Water Storage Tank	NA
4. 2-HC-13*	Discharge From Atmosphere Clean-up System (Hydrogen Recombiner)	NA
5. 2-HC-29*	Discharge From Atmosphere Clean-up System (Hydrogen Analyzer)	NA
6. 2-HC-18*	Discharge From Atomsphere Clean-up System (Hydrogen Recombiner)	NA
7. 2-HC-33*	Discharge From Atmosphere Clean-up System (Hydrogen Analyzer)	NA
8. 2-DA-7*	Primary Vent Pot Vent	NA
9. 2-DA-9*	Primary Vent Pot Vent	NA
10. 2-CH-233#*	Reactor Coolant Pump Seal Water Supply	NA
11. 2-CH-237#*	Reactor Coolant Pump Seal Water Supply	NA

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
12.	2-CH-241#*	Reactor Coolant Pump Seal Water Supply	NA
13.	2-SA-65*	Service Air	NA
14.	2-SA-123*	Service Air	NA
15.	(Deleted)		
16.	NA*#	Fuel Transfer Tube (Penetration #65)	NA
17.	2-CV-4*	Air Ejector Suction	NA
18.	2-RC-143*	Dead Weight Pressure Calibrator	NA
19.	2-RC-145*	Dead Weight Pressure Calibrator	NA
20.	1-RP-84*	Refueling Purification Outlet	NA
21.	2-RP-7*	Refueling Purification Outlet	NA
22.	2-RP-6*	Refueling Purification Inlet	NA
23.	1-RP-50*	Refueling Purification Inlet	NA
24.	(Deleted)		
25.	(Deleted)		
26.	(Deleted)		
27.	2-WT-437*	Steam Generator Wet Layup	NA
28.	2-WT-438*	Steam Generator Wet Layup	NA
29.	2-WT-439*	Steam Generator Wet Layup	NA
30.	2-WT-446*	Steam Generator Wet Layup	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
31.	2-WT-447*	Steam Generator Wet Layup	NA
32.	2-WT-448*	Steam Generator Wet Layup	NA
33.	2-SI-83*	High Head Safety Injection, (Boron Injection Tank Bypass)	NA
34.	NA*	Fire Protection Supply (Penetration 34)	NA

E. REMOTE MANUAL

1.	MOV-QS201A*	Quench Spray Pump Discharge	NA
2.	MOV-QS201B*	Quench Spray Pump Discharge	NA
3.	MOV-RS255A#*	Recirc. Spray Pump Suction	NA
4.	MOV-RS255B#*	Recirc. Spray Pump Suction	NA
5.	MOV-2860A#*	LHSI Pump Suction From Containment Sump	NA
6.	MOV-2860B#*	LHSI Pump Suction From Containment Sump	NA
7.	MOV-RS256A*	Recirculation Spray Pump Discharge	NA
8.	MOV-RS256B*	Recirculation Spray Pump Discharge	NA
9.	MOV-SW203A*	Service Water to Recirculation Spray Coolers	NA
10.	MOV-SW203B*	Service Water to Recirculation Spary Coolers	NA
11.	MOV-SW203C*	Service Water to Recirculation Spray Coolers	NA
12.	MOV-SW203D*	Service Water to Recirculation Spray Coolers	NA
13.	MOV-SW204A*	Service Water from Recirculation Spray Coolers	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
14.	MOV-SW204B*	Service Water from Recirculation Spray Coolers	NA
15.	MOV-SW204C*	Service Water from Recirculation Spray Coolers	NA
16.	MOV-SW204D*	Service Water from Recirculation Spray Coolers	NA
17.	TV-CV200*	Containment Air Ejector Suction	NA
18.	MOV-2869A*	High Head Safety Injection to RCS Except Boron Injection Line	NA
19.	MOV-2836*	High Head Safety Injection to RCS Except Boron Injection Line	NA
20.	MOV-2869B*	High Head Safety Injection to RCS Except Boron Injection Line	NA
21.	HCV-2142*	Reactor Coolant Letdown Line From RHR System	NA
22.	TV-SS207A*#	Residual Heat Removal System Sample Lines	NA
23.	TV-SS207B*#	Residual Heat Removal System Sample Lines	NA
24.	MOV-2890A*	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
25.	MOV-2890B*	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
26.	MOV-2890C*	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
27.	MOV-2890D*	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
28.	FCV-2160*	Loop Fill Header	NA
29.	MOV-2289A*	Charging Line	NA
30.	MOV-2867C*	High Head Safety Injection, Boron Injection Tank	NA
31.	MOV-2867D*	High Head Safety Injection, Boron Injection Tank	NA
32.	MOV-RS-200A*	Casing Cooling to Outside Recirculation Spray Pump	NA
33.	MOV-RS-200B*	Casing Cooling to Outside Recirculation Spray Pump	NA
34.	MOV-RS-201A*	Casing Cooling to Outside Recirculation Spray Pump	NA
35.	MOV-RS-201B*	Casing Cooling to Outside Recirculation Spray Pump	NA
F.	CHECK		
1.	2-CC-194	Component Cooling Water to RHR System and Excess Letdown Heat Exchanger	NA
2.	2-CC-199	Component Cooling Water to RHR System and Excess Letdown Heat Exchanger	NA
3.	2-SI-93	High Head Safety Injection, Boron Injection to RCS	NA
4.	2-CC-302	Component Cooling Water to Containment Air Recirculation Coils	NA
5.	2-CC-289	Component Cooling Water to Containment Air Recirculation Coils	NA

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
6.	2-CC-276	Component Cooling Water to Containment Air Recirculation Coils	NA
7.	2-CH-335	Charging Line	NA
8.	2-CC-152	Component Cooling Water to Reactor Coolant Pumps	NA
9.	2-CC-115	Component Cooling Water to Reactor Coolant Pumps	NA
10.	2-CC-78	Component Cooling Water to Reactor Coolant Pumps	NA
11.	2-CH-331	Reactor Coolant Pumps, Seal Water Return	NA
12.	2-SI-136	Safety Injection Accumulator Make Up	NA
13.	2-SI-85	High Head Safety Injection to RCS except Boron Injection Line	NA
14.	2-HC-20	Discharge From Containment Atmosphere Clean-up System	NA
15.	2-HC-15	Discharge From Containmnt Atmosphere Clean-up System	NA
16.	2-CH-308#	Reactor Coolant Pump Seal Water Supply	NA
17.	2-CH-260#	Reactor Coolant Pump Seal Water Supply	NA
18.	2-CH-284#	Reactor Coolant Pump Seal Water Supply	NA
19.	2-IA-428	Air Radiation Monitor Return	NA
20.	2-RC-162	Primary Grade Water	NA
21.	2-CH-332	Loop Fill Header	NA
22.	2-IA-250	Containment Instrument Air Return	NA

TABLE 3.6-1 (Cont.)

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
23.	2-SI-132	Nitrogen to Pressurizer Relief Tank and SI Accumulators	NA
24.	2-SI-126	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
25.	2-SI-128	LHSI Pump Discharge to Reactor Coolant System Hot Legs	NA
26.	2-SI-91	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
27.	2-SI-99	LHSI Pump Discharge to Reactor Coolant System Cold Legs	NA
28.	2-SI-10'	LHSI Pump Discharge do Reactor Coolant System Cold Legs	NA
29.	2-QS-22**	Quench Spray Pump Discharge	NA
30.	2-QS-11**	Quench Spray Pump Discharge	NA
31.	2-RS-30**	Recirculation Spray Pump Discharge	NA
32.	2-RS-20**	Reciruclation Spray Pump Discharge	NA
33.	2-VP-24	Air Ejector Vent	NA
34.	2-SI-119	High Head Safety Injection to RCS Except Boron Injection Line	NA
35.	2-SI-107	High Head Safety Injection to RCS Except Boron Injection Line	NA
36.	2-FW-62#	Feedwater to Steam Generators	NA

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
37.	2-FW-94#	Feedwater to Steam Generators	NA
38.	2-FW-126#	Feedwater to Steam Generators	NA
39.	2-WT-41#	Chemical Feed Lines	NA
40.	2-WT-53#	Chemical Feed Lines	NA
41.	2-WT-69#	Chemical Feed Lines	NA
42.	2-FW-70#	Auxiliary Feedwater to Steam Generator	NA
43.	2-FW-102#	Auxiliary Feedwater to Steam Generator	NA
44.	2-FW-134#	Auxiliary Feedwater to Steam Generator	NA
45.	2-RS-103#	Casing Cooling to Outside Recirculation Spray Pump	NA
46.	2-RS-118#	Casing Cooling to Outside Recirculation Spray Pump	NA
47.	NA	Fire Protection Supply (Penetration 34)	NA

TABLE 3.6-1 (Cont.)

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC.)
G.	STEAM LINE ISOLATION		
1.	TV-MS201A#	Main Steam Line Trip Valve	5
2.	TV-MS201B#	Main Steam Line Trip Valve	5
3.	TV-MS201C#	Main Steam Line Trip Valve	5
H.	RELIEF		
1.	RV-2203	Letdown Line Relief Valve	NA

Valve not subject to Type "C" leakage test.

* Valve position maintained by administrative control

NA - Not applicable

** Weight loaded check valve

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers (shared with Unit 1) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent (± .25%) hydrogen, balance nitrogen, and

b. Four volume percent (± .25%) hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two separate and independent containment hydrogen recombiner systems (shared with Unit 1) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes and is maintained for at least two hours and that each purge blower operates for at least 15 minutes.

b. At least once per 18 months by:

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

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

3. Verifying, during a recombiner system functional test using containment atmospheric air at a flow rate of greater than or equal to 50 scfm, that the heater temperature increases to greater than or equal to 1100°F within 5 hours and is maintained for at least 4 hours.

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

WASTE GAS CHARCOAL FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A waste gas charcoal filter system (shared with Unit 1) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the waste gas charcoal filter system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The waste gas charcoal filter system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Initiating flow through the HEPA filter and charcoal adsorber train using the process vent blowers and verifying that the purge 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 painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 300 cfm ± 10% (except as shown in Specifications 4.6.4.3.e. and f.).

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

   3. Verifying a system flow rate of 300 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

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

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is less than 8.5 inches Water Gauge while operating the filter train at a flow rate of 300 cfm ± 10%.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm ± 10%.

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

STEAM JET AIR EJECTOR

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the inside or outside isolation valve in the steam jet air ejector suction line not closed, restore the valve to the closed position within 1 hour or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 The steam jet air ejector suction line outside isolation valve shall be determined to be in the closed position by a visual inspection prior to increasing the Reactor Coolant System temperature above 350°F and at least once per 31 days thereafter if the valve is not locked, sealed or otherwise secured in the closed position.

4.6.5.1.2 The steam jet air ejector suction line inside isolation valve shall be determined to be in the closed position prior to increasing the Reactor Coolant System temperature above 350°F.

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 of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. 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 Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 2 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop 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 Setpoint trip 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.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	44

TABLE 3.7-1 (Continued)

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER) Loop Stop Valves Closed**	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER) Loop Stop Valves Open**
1		
2		
3		

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ($\pm$ 1%)*	ORIFICE SIZE
a. SV-MS 201 A, B, C	1085 psig	16 in^2
b. SV-MS 202 A, B, C	1095 psig	16 in^2
c. SV-MS 203 A, B, C	1110 psig	16 in^2
d. SV-MS 204 A, B, C	1120 psig	16 in^2
e. SV-MS 205 A, B, C	1135 psig	16 in^2

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5, each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a discharge pressure of greater than or equal to 1260 psig at a flow of greater than or equal to 53 gpm.

2. Verifying that the steam turbine driven pump develops a discharge pressure of greater than or equal to 1380 psig at a flow of greater than or equal to 35 gpm on recirculation flow. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on an auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts automatically upon receipt of an auxiliary feedwater actuation test signal.

c. The auxiliary feedwater system shall be demonstrated OPERABLE prior to entry into MODE 3 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the emergency condensate storage tank through each auxiliary feedwater pump to its associated steam generator.

EMERGENCY CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The emergency condensate storage tank (ECST) shall be OPERABLE with a contained water volume of at least 110,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the ECST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of a 300,000 gallon condensate storage tank as a backup supply to the auxiliary feedwater pumps and restore the emergency condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The emergency condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying that the water level in the condensate storage tank is sufficient to replenish the ECST to 110,000 gallon whenever the condensate storage tank is the supply source for the auxiliary feedwater pumps.

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 μCi/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 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

MAIN STEAM TRIP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam trip valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam trip valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam trip valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed and the provisions of specification 3.0.4 are not applicable provided the main steam trip valve is maintained closed; otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam trip valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

STEAM TURBINE ASSEMBLY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The structural integrity of the steam turbine assembly shall be maintained.

APPLICABILITY: MODES 1 and 2

ACTION: With the structural integrity of the steam turbine assembly not conforming to the above requirement restore the structural integrity of the steam turbine prior to placing it in service.

SURVEILLANCE REQUIREMENTS

4.7.1.6 The structural integrity of the steam turbine assembly shall be demonstrated;

a. At least once per 40 months, during shutdown, by a visual and surface inspection of the steam turbine assembly at all accessible locations, and

b. At least once per 10 years, during shutdown, by disassembly of the turbine and performing a visual, surface and volumetric inspection of all normally inaccessible parts.

PLANT SYSTEMS

TURBINE OVERSPEED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1, 2 and 3

ACTION: With the above required turbine overspeed protection system inoperable, within 6 hours either restore the system to OPERABLE status or isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENT

4.7.1..71 The provisions of Specification 4.0.4 are not applicable.

4.7.1.7.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE;

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

   1. 4 Turbine Throttle valves

   2. 4 Turbine Governor valves

   3. 4 Turbine Reheat Stop valves

   4. 4 Turbine Reheat Intercept valves

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

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

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

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

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

SURVEILLANCE REQUIREMENTS

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

3/4.7.3 COMPONENT COOLING WATER SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two component cooling water subsystems (shared with Unit 1) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water subsystems 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.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two service water loops (shared with Unit 1) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water loops shall be demonstrated OPERABLE:

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

b. At least once per 6 months by measurement of the movement of the pumphouse and wing walls.

c. At least once per 6 months by measurement of the turbidity and suspended solids in the effluent from the drain system under the service water pump house. If either the turbidity or suspended solids content exceeds 10 ppm a Special Report shall be submitted to the Commission within 30 days outlining the causes and planned corrective action.

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

   1. Verifying that each automatic valve servicing safety related equipment actuates to its correct position on a safety injection signal.

   2. Verifying that each containment isolation valve actuates to its correct position on a containment high-high signal.

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sinks shall be OPERABLE:

a. Service Water Reservoir with:

1. A minimum water level at or above elevation 313 Mean Sea Level, USGS datum, and

2. An average water temperature of less than or equal to 95°F as measured at the service water pump outlet.

b. The North Anna Reservoir with:

1. A minimum water level at or above elevation 244 Mean Sea Level, USGS datum, and

2. An average water temperature of less than or equal to 95°F as measured at the condenser inlet.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHTUDOWN within the following 30 hours.

SURVEILLANCE REQUIRMENTS

4.7.5.1 The ultimate heat sinks shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

4.7.5.2 Data for calculating the leakage from the service water reservoir shall be obtained and recorded at least once per 6 months.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the North Anna Reservoir exceeds 256 feet Mean Sea Level USGS datum, at the main reservoir spillway.

APPLICABILITY: At all times.

ACTION:

With the water level at the main reservoir spillway above elevation 256 feet Mean Sea Level USGS datum:

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

b. Initiate and complete within 36 hours, the following flood protection measures:

    1. Stop the circulating water pumps, and

    2. Close the condenser isolation valves.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the main reservoir spillway shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 255 feet Mean Sea Level USGS datum.

b. Measurement at least once per 2 hours when the water level is equal to or above 255 feet Mean Sea Level USGS datum.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.1 The following control room emergency habitability systems shall be OPERABLE:

a. The emergency ventilation system,

b. The bottled air pressurization system*, and

c. Two air conditioning systems.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either the emergency ventilation system or the bottled air pressurization system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

b. With both the emergency ventilation system and the bottled air pressurization system inoperable, restore at least one of these systems to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

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

d. With both air conditioning systems inoperable, restore at least one system to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

*Shared with Unit 1

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

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

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm $\pm$ 10% (except as shown in Specifications 4.7.7.1e. and f.).

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

   3. Verifying a system flow rate of 1000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

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

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is less than 6 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 84 bottles of air (shared with Unit 1) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

PLANT SYSTEMS

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two safeguards area ventilation systems (SAVS) shall be OPERABLE with:

a. One SAVS exhaust fan, and

b. One auxiliary building HEPA filter and charcoal adsorber assembly (shared with Unit 1).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SAVS system shall be demonstrated OPERABLE:

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

   1. Initiating, from the control room, flow through the auxiliary building HEPA filter and charcoal adsorber assembly and verifying that the SAVS operates for at least 10 hours with the heater on.

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6,300 cfm ± 10% (except as shown in Specifications 4.7.8 le. and f.).

SURVEILLANCE REQUIREMENTS (Cont'd)

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

3. Verifying a system flow rate of 6,300 cfm ± 10% during operation when tested in accordance with ANSI N510-1975.

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

d. At least once per 18 months by:
   1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is less than 6 inches Water Gauge while operating the ventilation system at a flow rate of 6,300 cfm ± 10%.

   2. Verifying that on a Containment Pressure -- High-High Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 6,300 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 6,300 cfm ± 10%.

3/4.7.9 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

RHR - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.9.1 Two residual heat removal subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one residual heat removal subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each residual heat removal ubsystem shall be demonstrated OPERABLE:

a. At least once per 18 months by verifying automatic isolation of the RHR system from the Reactor Coolant System when the RCS pressure is above 660 psig.

b. At least once per 18 months during shutdown by cycling each of the valves in the subsystem flow path not testable during plant operation through one complete cycle of full travel.

c. At least once per 18 months by verifying that each residual heat removal pump develops a differential pressure of greater than or equal to 123 psi.

RHR - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.9.2 As a minimum, one residual heat removal subsystem shall be OPERABLE.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no Residual Heat Removal subsystem OPERABLE, immediately restore at least one RHR subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods. The provisions of Specifications 3.0.3, 3.0.4 and 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.2 Each Residual Heat Removal subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirement of Specification 4.7.9.1, and

a. At least once per 31 days by;

   1. Cycling each testable, remote or automatically operated valve through at least one complete cycle, and

   2. Verifying the correct position of each manual valve not locked, sealed or otherwise secured in position, and

   3. Verifying the correct position of each remote or automatically operated valve.

3/4.7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All snubbers listed in Table 3.7-4a shall be OPERABLE.

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

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.10.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be during the first COLD SHUTDOWN exceeding 24 hours after four months of power operation and shall include all snubbers listed in Table 3.7-4a. If less then two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers 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 snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.
#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing 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 4.7.10.d. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample of at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber which does not meet the functional test acceptance criteria of Specification 4.7.10.d, an additional 10% of that type of snubber shall be functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. As part of this initial sample, at least 25% of the snubbers in each of the following three categories shall be included:

1. The first snubber away from each reactor vessel nozzle

2. Each snubber within five feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Each snubber within ten feet of the discharge from a safety relief valve

Snubbers identified in Table 3.7-4a as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

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*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.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, all snubbers of the same design shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

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

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

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 6.10.2.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Table 3.7-4a shall be reviewed

SURVEILLANCE REQUIREMENTS (Continued)

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 reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 3.7-4a

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIrFICULT TO REMOVE (Yes or No)
556-8	CC-269-C	I	Yes	No
556-10	CC-263-C	I	Yes	No
605-10	CC-269-C	I	Yes	No
548-8	CC-267-A	I	Yes	No
400	CC-244-ABB	A	No	Yes
401	CC-244-ABB	A	No	No
599-21	CC-273-B	I	Yes	Yes
402	CC-244-ABB	A	No	No
725-1	CC-234-RCB	I	Yes	Yes
406	CC-244-PENT.	A	No	Yes
571	CC-251-ABB	A	No	Yes
408	CC-244-PENT.	A	No	No
507	CH-247-RCA	I	No	Yes
401-3	CH-222-RCB	I	No	No
401-4	CH-220-RCB	I	No	No
401-5	CH-221-RCB	I	No	Yes
401-8	CH-221-RCB	I	No	Yes
401-9	CH-225-RCB	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
401-13	CH-230-RCB	I	No	Yes
401-14	CH-236-RCB	I	No	Yes
401-15	CH-236-RCB	I	No	Yes
401-20	CH-236-RCB	I	Yes	Yes
401-23A	CH-248-B	I	Yes	Yes
401-23B	CH-248-B	I	Yes	Yes
401-24A	CH-256-B	I	Yes	Yes
401-24B	CH-256-B	I	Yes	Yes
401-29	CH-234-RCB	I	No	Yes
401-36	CH-238-RCB	I	No	Yes
405-2	CH-250-A	I	Yes	No
406-3A	CH-230-RCB	I	No	Yes
406-3B	CH-230-RCB	I	No	Yes
468-3	CH-232-RCB	I	No	Yes
640-2	CH-217-RCB	I	No	No
640-3	CH-217-RCB	I	No	No
800-1	CH-244-A	I	Yes	No
800-2	CH-244-A	I	Yes	No
800	QS-274-SG	A	No	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
800-6	CH-243-A	I	Yes	No
801-15	CH-238-A	I	Yes	Yes
801-20	CH-243-A	I	Yes	No
801-22	CH-243-A	I	Yes	Yes
466-1	CH-224-RCB	I	No	Yes
482-2	CH-226-RCB	I	No	Yes
483-2	CH-226-RCB	I	No	Yes
633-1A	CH-236-RCB	I	No	Yes
633-1B	CH-236-RCB	I	No	Yes
633-4	CH-236-RCB	I	No	Yes
801-25	CH-237-RCB	I	No	Yes
801-26	CH-237-RCB	I	No	Yes
492-42A	CH-255-C	I	Yes	Yes
492-42B	CH-255-C	I	Yes	Yes
304	QS-255-RCA	I	No	Yes
301	QS-255-RCA	I	No	Yes
706	QS-267-QS	A	No	No
303A	RS-260-RCA	I	No	Yes
303	RS-260-RCA	I	No	Yes
300	RS-333-RCA	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
300A	RS-333-RCA	I	No	Yes
302	RS-333-RCA	I	No	Yes
302A	RS-333-RCA	I	No	Yes
107	RC-237-RCB	I	No	Yes
701	RC-308-P	I	Yes	No
102	RC-246-P	I	Yes	Yes
105	RC-238-RCB	I	No	Yes
121	RC-300-P	I	No	No
112	RC-258-P	I	No	Yes
116	RC-307-P	I	No	No
113	RC-300-P	I	No	No
131	RC-307-P	I	No	No
103	RC-265-P	I	No	No
128	RC-302-P	I	No	No
127	RC-302-P	I	Yes	No
1[illegible]4	RC-308-P	I	Yes	No
145A	RC-308-P	I	Yes	No
145B	RC-308-P	I	Yes	Yes
142A	RC-309-P	I	Yes	No
142B	RC-309-P	I	Yes	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
117	RC-305-P	I	No	No
412-6	RC-264-B	I	Yes	No
412-7	RC-264-B	I	Yes	No
143	RC-308-P	I	Yes	No
111	RC-257-P	I	Yes	Yes
100	RC-244-A	I	Yes	Yes
101	RC-237-A	I	Yes	Yes
115	RC-302-P	I	Yes	No
119	RC-305-P	I	Yes	No
146	RC-308-P	I	Yes	No
147	RC-308-P	I	Yes	No
700	RC-241-C	I	Yes	Yes
148	RC-242-P	I	Yes	Yes
124	RC-307-P	I	No	No
139	RC-308-P	I	[illegible]	No
110	RC-258-P	I	Yes	Yes
118A	RC-309-P	I	Yes	No
120	RC-307-P	I	Yes	No
125	RC-305-P	I	No	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
129	RC-302-P	I	Yes	No
130	RC-302-P	I	Yes	Yes
138	RC-308-P	I	Yes	No
141	RC-309-P	I	Yes	No
140A	RC-303-P	I	Yes	Yes
140B	RC-303-P	I	Yes	Yes
134	RC-304-P	I	Yes	No
135	RC-308-P	I	Yes	No
126	RC-305-P	I	No	No
132	RC-305-P	I	No	No
136	RC-308-P	I	Yes	No
114	RC-302-P	I	Yes	No
137	RC-308-P	I	Yes	No
122	RC-302-P	I	Yes	No
123	RC-302-P	I	Yes	Yes
106	RC-237-RCB	I	No	Yes
133	RC-307-P	I	Yes	No
410-1	RC-264-P	I	No	No
410-2	RC-256-P	I	Yes	Yes
410-3	RC-256-P	I	Yes	Yes
414	RC-270-P	I	Yes	Yes
516-7	RC-259-B	I	Yes	Yes
516-8	RC-260-B	I	Yes	Yes
516-6	RC-253-B	I	Yes	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
518-2	RC-260-B	I	Yes	Yes
523-1	RC-242-B	I	Yes	Yes
444-3	RC-256-C	I	Yes	Yes
445-3	RC-256-A	I	Yes	Yes
446-3	RC-256-B	I	Yes	Yes
447-1	RC-256-A	I	Yes	Yes
447-2	RC-256-A	I	Yes	Yes
448-1	RC-256-B	I	Yes	Yes
448-2	RC-256-B	I	Yes	Yes
449-1	RC-256-C	I	Yes	Yes
449-2	RC-256-C	I	Yes	Yes
453-29	RC-236-A	I	Yes	Yes
453-30A	RC-238-A	I	Yes	No
453-30B	RC-238-A	I	Yes	No
453-31	RC-243-A	I	Yes	No
455-1	RC-243-A	I	Yes	No
458-1	RC-242-C	I	Yes	No
458-2	RC-242-C	I	Yes	No
459-2	RC-243-C	I	Yes	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
459-1	RC-243-C	I	Yes	No
463-1	RC-256-A	I	Yes	No
464-1	RC-256-B	I	Yes	No
457-2B	RC-245-B	I	Yes	No
615-4	RC-243-A	I	Yes	No
618-13	RC-243-B	I	Yes	No
465-1	RC-256-C	I	Yes	No
466-2	RC-268-P	I	No	No
466-3	RC-268-P	I	No	No
467-2	RC-267-P	I	No	No
467-3	RC-267-P	I	Yes	No
617-1	RC-243-B	I	Yes	No
618-12	RC-243-B	I	Yes	No
618-2	RC-243-B	I	Yes	No
616-2	RC-245-A	I	Yes	Yes
531-7	RC-259-A	I	Yes	Yes
531-8	RC-260-A	I	Yes	Yes
537-1	RC-256-A	I	Yes	Yes
534-1	RC-263-A	I	Yes	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
515-1	RC-257-B	I	Yes	Yes
516-2	RC-258-B	I	Yes	Yes
516-3	RC-257-B	I	Yes	Yes
516-5	RC-253-B	I	Yes	Yes
410-0	RC-264-P	I	No	No
547-2	RC-257-C	I	Yes	Yes
547-3	RC-253-C	I	Yes	Yes
553-2	RC-259-C	I	Yes	Yes
529-1	RC-257-A	I	Yes	Yes
531-2	RC-258-A	I	Yes	Yes
531-3	RC-257-A	I	Yes	Yes
531-5	RC-253-A	I	Yes	Yes
531-6	RC-255-A	I	Yes	Yes
619-2	RC-243-C	I	Yes	Yes
619-4	RC-243-C	I	Yes	Yes
411-4	RC-264-A	I	Yes	Yes
411-5	RC-264-A	I	Yes	Yes
413-8	RC-264-C	I	Yes	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
413-9	RC-264-C	I	Yes	Yes
617-3	RC-243-B	I	Yes	Yes
599-3	RC-244-B	I	Yes	No
457-2A	RC-245-B	I	Yes	No
620-19	RC-243-C	I	Yes	No
620-21	RC-243-C	I	Yes	No
620-25	RC-243-C	I	Yes	No
118 B	RC-309-P	I	Yes	No
615-2	RC-243-A	I	Yes	No
547-5	RC-257-C	I	Yes	Yes
547-6A	RC-257-C	I	Yes	Yes
547-6B	RC-258-C	I	Yes	Yes
546-2	RC-257-C	I	Yes	Yes
100A	RH-224-RCB	I	Yes	Yes
103	RH-224-RCB	I	No	No
510	RH-229-RCB	I	Yes	Yes
509	RH-230-RCB	I	No	Yes
105B	RH-228-RCB	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
105A	RH-228-RCB	I	No	Yes
106	RH-240-A	I	No	Yes
102	RH-224-RCB	I	No	Yes
107	RH-239-A	I	No	Yes
117	RH-226-RCB	I	No	No
108A	RH-228-RCB	I	No	No
100B	RH-224-RCB	I	No	Yes
101	RH-237-RCB	I	No	Yes
104	RH-228-RCB	I	No	Yes
109	RH-233-RHR	I	No	No
110	RH-233-RHR	I	No	No
111	RH-234-RHR	I	No	No
112	RH-234-RHR	I	No	No
701	RH-224-RCB	I	Yes	Yes
705	RH-224-RCB	I	No	Yes
707	RH-224-RCB	I	No	No
700	RH-224-RCB	I	No	No
702	RH-224-RCB	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
703	RH-224-RCB	I	No	No
704	RH-224-RCB	I	No	No
706A	RH-224-RCB	I	No	No
706B	RH-224-RCB	I	No	No
708	RH-226-RCB	I	Yes	No
22	RH-216-RCB	A	No	Yes
246	SAE-296-MSVH	A	No	Yes
245	SAE-296-MSVH	A	No	Yes
247	SAE-296-MSVH	A	No	Yes
700A	SAE-281-AFPH	A	No	No
700B	SAE-281-AFPH	A	No	No
704A	SAE-281-AFPH	A	No	No
704B	SAE-281-AFPH	A	No	No
701	SAE-281-AFPH	A	No	No
702	SAE-281-AFPH	A	No	Yes
706	SAE-281-AFPH	A	No	Yes
703	SAE-283-AFPH	A	No	No
707	SAE-283-AFPH	A	No	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
705	SAE-281-AFPH	A	No	No
233	SDHV-308-MSVH	A	No	No
232	SDHV-308-MSVH	A	No	No
234	SDHV-307-MSVH	A	No	No
235	SDHV-307-MSVH	A	No	No
404-1	SGD-304-RCA	I	No	No
405-2	SGD-302-RCA	I	No	Yes
405-3	SGD-302-RCA	I	No	Yes
405-5	SGD-299-RCA	I	No	No
406-1	SGD-305-RCA	I	No	Yes
406-2	SGD-305-RCA	I	No	Yes
406-3	SGD-305-RCA	I	No	Yes
406-4A	SGD-305-RCA	I	No	Yes
406-4B	SGD-305-RCA	I	No	Yes
407-1A	SGD-298-RCA	I	No	No
407-1B	SGD-298-RCA	I	No	No
408-1	SGD-298-RCA	I	No	No
205	SHP-320-RCA	I	No	Yes
207	SHP-321-RCA	I	No	Yes
208	SHP-321-RCA	I	No	Yes
23A	SHP-293-MSVH	A	No	Yes
229	SHP-300-SB	A	No	Yes
206	SHP-305-RCA	I	No	Yes
220A	SHP-286-SB	A	No	Yes
220B	SHP-286-SB	A	No	Yes
221A	SHP-286-SB	A	No	Yes
211A	SHP-300-MSVH	A	No	Yes
211B	SHP-300-MSVH	A	No	Yes
212A	SHP-300-MSVH	A	No	Yes
212B	SHP-300-MSVH	A	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
213A	SHP-300-MSVH	A	No	Yes
23B	SHP-293-MSVH	A	No	Yes
30A	SHP-293-MSVH	A	No	Yes
30B	SHP-293-MSVH	A	No	Yes
38A	SHP-293-MSVH	A	No	Yes
38B	SHP-293-MSVH	A	No	Yes
217A	SHP-300-MSVH	A	No	Yes
217B	SHP-300-MSVH	A	No	Yes
218A	SHP-300-MSVH	A	No	Yes
218B	SHP-300-MSVH	A	No	Yes
219A	SHP-300-MSVH	A	No	Yes
219B	SHP-300-MSVH	A	No	Yes
213B	SHP-300-MSVH	A	No	Yes
243	SHP-307-MSVH	A	No	No
244	SHP-307-MSVH	A	No	No
221B	SHP-286-SB	A	No	Yes
222A	SHP-286-SB	A	No	Yes
223A	SHP-329-RCA	I	Yes	Yes
222B	SHP-286-SB	A	No	Yes
223B	SHP-329-RCA	I	Yes	Yes
225	SHP-329-RCA	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
230B	SHP-298-SB	A	No	Yes
239	SHP-307-MSVH	A	No	No
226	SHP-300-SB	A	No	Yes
223B	SHP-329-RCA	I	No	Yes
209A	SHP-329-RCA	I	No	Yes
209B	SHP-329-RCA	I	No	Yes
242	SHP-307-MSVH	A	No	No
227A	SHP-300-SB	A	No	Yes
230A	SHP-302-SB	A	No	Yes
228	SHP-300-SB	A	No	Yes
227B	SHP-300-SB	A	No	Yes
231	SHP-300-SB	A	No	Yes
223A	SHP-329-RCA	I	No	Yes
257	SHP-267-TBB	A	No	Yes
224	SHP-329-RCA	I	No	Yes
202A	SHP-329-RCA	I	No	Yes
202B	SHP-329-RCA	I	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
203	SHP-321-RCA	I	No	Yes
204	SHP-321-RCA	I	No	Yes
240	SHP-307-MSVH	A	No	No
241	SHP-307-MSVH	A	No	No
546-1	SHP-309-MSVH	A	No	Yes
546-3	SHP-306-MSVH	A	No	No
248	SHP-279-TBM	A	No	Yes
249	SHP-268-TBB	A	No	Yes
250	SHP-273-TBB	A	No	Yes
251A	SHP-291-TBM	A	No	Yes
251B	SHP-291-TBM	A	No	Yes
253	SHP-291-TBM	A	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
254A	SHP-291-TBM	A	No	Yes
254B	SHP-302-TBM	A	No	Yes
255	SHP-274-TBB	A	No	Yes
256	SHP-277-TBB	A	No	Yes
258	SHP-307-TBM	A	No	Yes
259	SHP-267-TBB	A	No	Yes
260A	SHP-279-TBM	A	No	Yes
260B	SHP-275-TBB	A	No	Yes
261	SHP-277-TBB	A	No	Yes
252	SHP-291-TBM	A	No	Yes
201A	SHP-329-RCA	I	No	Yes
201B	SHP-329-RCA	I	No	Yes
461-1	SHP-295-MSVH	A	No	Yes
562-6	SHP-305-MSVH	A	No	Yes
547-1	SHP-309-MSVH	A	No	Yes
547-3	SHP-306-MSVH	A	No	No
563-6	SHP-306-MSVH	A	No	Yes
548-1	SHP-309-MSVH	A	No	Yes
548-3	SHP-306-MSVH	A	No	No
564-6	SHP-305-MSVH	A	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
562-2	SHP-297-MSVH	A	No	Yes
562-3A	SHP-297-MSVH	A	No	No
562-3B	SHP-297-MSVH	A	No	No
562-4	SHP-297-MSVH	A	No	No
563-2	SHP-296-MSVH	A	No	Yes
563-3B	SHP-294-MSVH	A	No	Yes
563-4	SHP-306-MSVH	A	No	No
564-2	SHP-294-MSVH	A	No	Yes
564-3	SHP-295-MSVH	A	No	Yes
564-4	SHP-298-MSVH	A	No	No
563-3A	SHP-294-MSVH	A	No	Yes
462-1	SHP-295-MSVH	A	No	Yes
460-1	SHP-295-MSVH	A	No	Yes
100A	SI-256-B	I	Yes	Yes
103B	SI-243-C	I	Yes	No
105A	SI-242-A	I	Yes	No
102A	SI-221-RCB	I	No	No
104A	SI-221-RCB	I	No	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
106A	SI-221-RCB	I	No	No
505	SI-239-RCA	I	No	Yes
502	SI-247-RCA	I	No	No
503	SI-247-RCA	I	No	No
501	SI-247-RCA	I	No	No
500	SI-247-RCA	I	No	No
105B	SI-242-A	I	Yes	No
416-4	SI-253-C	I	Yes	Yes
504	SI-239-RCA	I	Yes	Yes
103A	SI-243-C	I	Yes	No
101B	SI-244-B	I	Yes	No
100	SI-256-SG	A	No	No
101	SI-262-SG	A	No	No
419-2	SI-251-B	I	Yes	Yes
532-4	SI-257-C	I	Yes	Yes
421-2	SI-251-A	I	Yes	No
421-3	SI-253-A	I	Yes	No
533-5	SI-257-B	I	Yes	Yes
511	SI-230-RCB	I	No	Yes
107	SI-237-RCB	I	Yes	No

TABLE 3.7-4 (Continued)
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
800	SI-258-SG	A	No	No
801	SI-258-SG	A	No	Yes
802A	SI-258-SG	A	No	No
803	SI-258-SG	A	No	No
804	SI-258-SG	A	No	No
806	SI-259-SG	A	No	Yes
807A	SI-259-SG	A	No	Yes
807B	SI-259-SG	A	No	Yes
808	SI-253-SG	A	No	Yes
809A	SI-253-SG	A	No	No
809B	SI-253-SG	A	No	No
810A	SI-263-SG	A	No	Yes
810B	SI-263-SG	A	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
23	SI-262-SG	A	No	Yes
512	SI-226-RCB	I	No	Yes
461-16	SI-250-A	I	No	Yes
802	SI-247-PENT.	A	No	No
419-4	SI-251-B	I	Yes	Yes
424	WCPD-287-TBM	A	No	Yes
601	WDRD-266-TBB	A	No	Yes
603	WDRD-268-TBB	A	No	Yes
605	WDRD-268-TBB	A	No	Yes
202	WFPD-302-RCA	I	No	Yes
212	WFPD-303-RCA	I	No	Yes
231	WFPD-302-RCA	I	No	Yes
232	WFPD-302-RCA	I	No	Yes
233	WFPD-302-RCA	I	No	Yes
234	WFPD-302-RCA	I	No	Yes
235	WFPD-301-SB	A	No	Yes
236	WFPD-295-SB	A	No	Yes
237	WFPD-301-SB	A	No	Yes

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
220	WFPD-294-MSVH	A	No	Yes
221	WFPD-294-MSVH	A	No	Yes
203	WFPD-302-RCA	A	No	Yes
225	WFPD-303-RCA	A	No	Yes
219	WFPD-294-MSVH	A	No	Yes
200	WFPD-302-RCA	I	No	Yes
222	WFPD-277-MSVH	A	No	No
215	WFPD-294-SB	A	No	No
213	WFPD-294-SB	A	No	No
223	WFPD-277-MSVH	A	No	No
201	WFPD-302-RCA	I	No	Yes
217	WFPD-294-MSVH	A	No	Yes
216	WFPD-294-MSVH	A	No	Yes
224	WFPD-277-MSVH	A	No	No
214	WFPD-294-SB	A	No	No
226A	WFPD-294-SB	A	No	Yes
226B	WFPD-294-SB	A	No	Yes
227B	WFPD-294-SB	A	No	No

TABLE 3.7-4a (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION (ELEVATION-AREA)#	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION DURING SHUTDOWN ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
227A	WFPD-294-SB	A	No	No
218	WFPD-294-MSVH	A	No	Yes
230	WFPD-295-SB	A	No	Yes
228	WFPD-295-SB	A	No	Yes
204	WFPD-302-RCA	I	No	Yes
205	WFPD-302-RCA	I	No	Yes
210	WFPD-302-RCA	I	No	Yes
211	WFPD-302-RCA	I	No	Yes
229A	WFPD-295-SB	A	No	Yes
229B	WFPD-295-SB	A	No	Yes
3A	WGCB-246-PENT.	A	No	No
3B	WGCB-246-PENT.	A	No	No
701A	WS-264-QS	A	No	No
701B	WS-264-QS	A	No	No

TABLE 3.7-4a (Continued)

TABLE NOTATIONS

#LOCATION ABBREVIATIONS

Abbreviations	Area
A	Cubicle A
B	Cubicle B
C	Cubicle C
Pent.	Penetration Area Aux. Bldg.
P	Pressurizer Cubicle
RCA	Reactor Containment Annulus
RCB	Reactor Containment Basement
RCP	Reactor Containment Penetration Area
MSVH	Main Steam Valve House
AFPH	Aux. Feedwater Pump House
MSH	Main Steam Header - Turb. Bldg.
TBM	Turbine Bldg. Mezzanine
TBB	Turbine Bldg. Basement
SB	Service Bldg.
SG	Safeguards Bldg.
QS	Quench Spray Area
FWH	Feedwater Header - Turb. Bldg.
ABB	Auxiliary Bldg. Basement
FBB	Fuel Bldg. Basement
RHR	Residual Heat Removal Mezzanine

NOTE: Numbers indicate radial locations in reactor containment.

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

**Modifications to this table due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4a is included with the next License Amendment request.

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3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and:

   1. Either 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 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.11.1.2 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 six months for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

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

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

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

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES

LIMITING CONDITION FOR OPERATION

3.7.12.1 The total settlement of each Class 1 structure or the differential settlement between Class 1 structures shall not exceed the allowable values of Table 3.7-5.

APPLICABILITY: All MODES

ACTION:

a. With either the total settlement of any structure or the differential settlement of any structures exceeding 75 percent of the allowable settlement, conduct an engineering review of field conditions and evaluate the consequences of additional settlement. Submit a special report to the Commission pursuant to Specification 6.9.2 within 60 days, containing the results of the investigation, the evaluation of existing and possible continued settlement and the remedial action to be taken if any, including the date of the next survey.

b. With the total settlement of any structure or the differential settlement of any two structures exceeding the allowable settlement value of Table 3.7-5, be in at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.12.1 The total settlement of each Class 1 structure or the differential settlement between Class 1 structures listed in Table 3.7-5 shall be determined to the nearest 0.01 foot by measurement and calculation at least once per 6 months. Measurements on settlement points SM-7, 8, 9, 10, 15, 16, 17, 18, H-569 and H-584 shall be made at least once per 31 days for the time period following 5 years from the date of issuance of Operating License NPF-4 to Unit 1.

TABLE 3.7-5

ALLOWABLE TOTAL SETTLEMENT OR DIFFERENTIAL SETTLEMENT FOR CLASS 1 STRUCTURES

SETTLEMENT POINT	STRUCTURE	SETTLEMENT POINT	STRUCTURE/COMPONENT	ALLOWABLE TOTAL SETTLEMENT* (FEET)	ALLOWABLE DIFFERENTIAL SETTLEMENT* (FEET)
131	Containment Unit 2	224	Fuel Building	N/A	0.12
131	Containment Unit 2	123	Auxiliary Building	N/A	0.05
106	Containment Unit 2	105	Unit 2 Safeguards Area	N/A	0.07
107	Containment Unit 2	108	Unit 2 Safeguards Area	N/A	0.07
131	Containment Unit 2	124	Unit 2 Main Steam Valve House	N/A	0.03
107	Containment Unit 2	116	Service Building (E-15)	N/A	0.12
111	Safeguards Area Unit 2	124	Unit 2 Main Steam Valve House	N/A	0.12
122	Auxiliary Building	120	Unit 2 Main Steam Valve House	N/A	0.04
123	Auxiliary Building	124	Unit 2 Main Steam Valve House	N/A	0.04
123	Auxiliary Building	224	Fuel Building	N/A	0.05
129	Auxiliary Building	223	Fuel Building	N/A	0.05
122	Auxiliary Building	119	Service Building Tunnel	N/A	0.07
243, 132	Service Building (E-5, E-6)	238	Unit 1 Main Steam Valve House	N/A	0.04
117	**Service Building (E-14)	113	Unit 2 Main Steam Valve House	N/A	0.03 from 7/77
231	Auxiliary Feedwater Pump House Unit 2	249	Pipe Tunnel	N/A	0.12
228	Decontamination Building	250	Pipe Tunnel	N/A	0.06

TABLE 3.7-5 (Continued)

ALLOWABLE TOTAL SETTLEMENT OR DIFFERENTIAL SETTLEMENT FOR CLASS 1 STRUCTURES

SETTLEMENT POINT	STRUCTURE	SETTLEMENT POINT	STRUCTURE/COMPONENT	ALLOWABLE TOTAL SETTLEMENT* (FEET)	ALLOWABLE DIFFERENTIAL SETTLEMENT* (FEET)
226	Fuel Building	251	Waste Gas Decay Tank Enclosure	N/A	0.06
104	Safeguards Area Unit 2	254	Unit 2 Casing Cooling Building	N/A	0.12 from 2/79
7, 10	Service Water Pump House	15, 16, 17, or 18	Service Water Piping at SWPH North Side of Expansion Joint	N/A	0.22 from 7/77
8	Service Water Pump House	H-569 H-584	Pipe Hanger in Reservoir	N/A	0.17
15,16, 17, 18	Service Water Piping at SWPH North Side of Expansion Joint			0.22 from 8/78	N/A
204	Circulating Water Intake Structure			0.15	N/A
158	***Turbine Building (B-9 1/2)			0.06	N/A
114	Service Building (E-17)			0.09	N/A
245, 246	Fuel Oil Pump House			0.03	N/A
206 207, 208, 209	Boron Recovery Tank Dike			0.03	N/A

TABLE 3.7-5 (Continued)

ALLOWABLE TOTAL SETTLEMENT OR DIFFERENTIAL SETTLEMENT FOR CLASS 1 STRUCTURES

SETTLEMENT POINT	STRUCTURE	SETTLEMENT POINT	STRUCTURE/COMPONENT	ALLOWABLE OUT-OF-PLANE DISTORTION
7, 8, 9, 10	Service Water Pump House	7, 8, 9, 10	Service Water Pump House	0.06 feet - any settlement point

* Unless otherwise indicated, allowable settlements are from base-line elevations established in May 1976 or reference elevations corrected to the May 1976 survey.

** Critical differential settlement is downward movement of Point 117 with respect to Point 113.

***Not Class 1 structure, but settlement affects Class 1 pipeline.

PLANT SYSTEMS

3/4.7.13 GROUNDWATER LEVEL-SERVICE WATER RESERVOIR

LIMITING CONDITION FOR OPERATION

3.7.13 The groundwater level of the service water reservoir (common to Units 1 and 2) shall not exceed the elevation at the locations listed in Table 3.7-6. The flow of groundwater from the drains beneath the pumphouse shall not exceed the values given in Table 3.7-6.

APPLICABILITY: All MODES.

ACTION:

With the groundwater level of the service water reservoir or the ground water flow rate exceeding any of the limits of Table 3.7-6, an engineering evaluation shall be performed by a Licensed Civil Engineer to determine the cause of the high ground water or flow rates and the influence on the stability of the service water reservoir and pumphouse. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days, containing the results of the evaluation and any corrective action determined to be necessary. In addition, at the end of the 5 year surveillance period, a summary report will be prepared and submitted to the Commission, within 90 days, illustrating the results of the ground water monitoring program. Based on this report, a determination will be made as to the need for further measurements of ground water conditions.

SURVEILLANCE REQUIREMENTS

4.7.13.1 The groundwater level of the service water reservoir shall be determined to be within the limits by piezometer readings from at least 7 of the locations shown on Table 3.7-6. Readings shall be taken at least once per 31 days for 5 years following the date of issuance of Operating License NPF-4 to North Anna Power Station Unit 1. The groundwater flow rates shall be determined by measurements at the drain outlet gallery. Readings shall be taken at least once per 31 days. The need for further surveillance will be determined at the end of the 5-year period from the date of issuance of Operating License NPF-4. Monitoring information collected since the issuance of the Operating License regarding the groundwater levels and flow rates shall be retained for use in preparing the summary report required by Technical Specification 3.7.13.

4.7.13.2 Piezometer readings shall be taken from piezometers 10 thru 14, inclusive, at least once per 12 months for the time period following 5 years from the date of issuance of Operating License NPF-4 to North Anna Power Station Unit 1. The need for further surveillance will be determined at the end of the 5 year period.

TABLE 3.7-6

ALLOWABLE GROUNDWATER LEVELS - SERVICE WATER RESERVOIR

PIEZOMETER NO.	PIEZOMETER LOCATION	ALLOWABLE GROUNDWATER ELEVATION Mean Sea Level (feet)
10	SE, toe	277
11	SWPH, (Units 1 & 2) crest	280
12	SwPH, (Units 1 & 2) toe	285
13	SWPH, (Units 1 & 2) crest	280
14	SWPH, (Units 1 & 2) crest	280
15	SE, crest	280
16	SE, crest	280
17	SE, crest	280
18	SWPH (Units 3 & 4)	295

DRAIN OUTLETS	LOCATION	ALLOWABLE DRAIN FLOWRATE (gallons per minute)
1 through 6	Drainage Gallery	Flow rate for any month shall not exceed 3 times the average annual flow rate.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.14.1 The fire suppression water system shall be OPERABLE with;

a. Two fire suppression pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies from the North Anna Reservoir and the Service Water Reservoir, and

c. An OPERABLE flow path capable of taking suction from the North Anna Reservoir and the Service Water Reservoir and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the valve at each hose standpipe as required to be OPERABLE per Specification 3.7.14.5.

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, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. Submit a Special Report in accordance with Specification 6.9.2;

      a) By telephone within 24 hours,

      b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

LIMITING CONDITION FOR OPERATION (Continued)

c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.14.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. By verifying the contained water supply volumes pursuant to Specification 4.7.5.1.

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

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

d. By performance of a system flush as necessary to maintain the system water chemistry within acceptable limits.

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 each pump develops at least 2500 gpm at a system head of greater than or equal to 250 feet for 1-FP-P-1 and greater than or equal to 187 feet for 1-FP-P-2.

   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 high pressure pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 80 psig in the main fire loop.

SURVEILLANCE REQUIREMENTS (Continued)

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

4.7.14.1.2 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 220 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 is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscocity, water and sediment.

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

4.7.14.1.3 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

PLANT SYSTEMS

LOW PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following low pressure CO_2 systems shall be OPERABLE with a minimum of 3.5 tons in the storage tank (common to Units 1 and 2) at a minimum pressure of 275 psig:

a. Cable tunnels and vaults

b. Charcoal filters

c. Emergency diesel generator rooms

APPLICABILITY: Whenever equipment protected by the low pressure CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more of the above required low pressure CO_2 systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.2. Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by verifying CO_2 storage tank level and pressure.

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

c. At least once per 18 months by verifying:

   1. The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

   2. Flow from each nozzle during a "Puff Test."

PLANT SYSTEMS

HIGH PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.3 The following high pressure CO^2 systems shall be OPERABLE with the storage tanks having at least 90% of full charge weight:

a. Fuel oil pump rooms

APPLICABILITY: Whenever equipment protected by the high pressure CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more of the above required high pressure CO_2 systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 6 months by verifying CO_2 storage tank weight.

b. 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.

c. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure:

a. Control Room

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.4 Each of the above required Halon 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 6 months by verifying Halon storage tank weight and pressure.

c. At least once per 18 months by:

   1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

   2. Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.14.5. The fire hose stations shown in Table 3.7-7 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-7 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the fire hose station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.14.5 Each of the fire hose stations shown in Table 3.7-7 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

TABLE 3.7-7

FIRE HOSE STATIONS

HOSE RACK IDENTIFICATION

AB-H-1
AB-H-4
AB-H-6
AB-H-8
AB-H-12
AB-H-13
AB-H-15
AB-H-18A
AB-H-19
AB-H-22
AB-H-24
AB-H-27
AB-H-29
AB-H-30
AB-H-32

F-H-1
F-H-3
T-H-7
T-H-25
T-H-21
T-H-22D
T-H-33
T-H-34
HP-H-5
BLR-H-2

3/4.7.15 PENETRATION FIRE BARRIERS

LIMITING CONDITIONS FOR OPERATION

3.7.15 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers), in fire zone boundaries, protecting safety related areas shall be functional.

APPLICABLITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour, either establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

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

SURVEILLANCE REQUIREMENTS

4.7.15 Each of the above required penetration fire barriers shall be verified to be functional:

a. At least once per 18 months, by a visual inspection, and

b. Prior to declaring a penetration fire barrier functional following repairs or maintenance by a visual inspection of the affected penetration fire barrier(s).

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system, and

b. Two separate and independent diesel generators:

1. Each with a separate day tank containing a minimum of 750 gallons of fuel, and

2. A fuel storage system containing a minimum of 45,000 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION (Continued):

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignment indicating power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring the onsite Class 1E power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.2 on a STAGGERED TEST BASIS by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying the fuel level in the day tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals with startup on each signal verified at least once per 124 days.

   a) Manual.
   b) Simulated loss of offsite power by itself.
   c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 2750 kw in less than or equal to 60 seconds, and operates for greater than or equal to 60 minutes.

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

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained as a DRAIN Sample in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying that, on rejection of a load of greater than or equal to 610 kw the voltage and frequency are maintained within 4160 ± 420 volts and 60 ± 1.2 Hz.

3. Verifying that the load sequencing timers are OPERABLE with times within the tolerances shown in Table 4.8-1.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the sequencing timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady state voltage and frequency shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.

5. Verifying that on an ESF actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes.

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

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

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

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the sequencing times and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads and maintains the steady state voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz.

   c) Verifying that all diesel generator trips, except engine overspeed, generator differential and breaker over current are automatically bypassed upon loss of voltage on the emergency bus and/or a safety injection actuation signal.

SURVEILLANCE REQUIREMENTS (Continued)

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 3025 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2750 kw. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2.c.4.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3000 kw.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
    b) Transfer its loads to the offsite power source, and
    c) Proceed through its shutdown sequence.

11. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Remote Local Selection Switch
    b) Emergency Stop Switch

d. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

4.8.1.1.3 Each diesel generator 125-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 pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to 1.200,

3. The pilot cell voltage is greater than or equal to 2.08 volts, and

SURVEILLANCE REQUIREMENTS (Continued)

4. The overall battery voltage is greater than or equal to 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.08 volts under float charge and has not decreased more than 0.05 volts from the value observed during the previous test,

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.08 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

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

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

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

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 0.01 ohms.

4. The battery charger will supply at least ten amperes at 125 volts for at least 4 hours.

d. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8-1

LIST OF LOAD SEQUENCING TIMERS AND DESIGN SETPOINTS
"H" BUS

TIMER IDENTIFICATION	SET POINT (SECONDS)	INITIATING SIGNAL[1]	TOLERANCE (SECONDS)
2FWEA01-62	20	SI	±1.00
2FWEA01-62A	25	LOP	±1.25
2SWEA03-62	10	LOP	±0.50
2RSOA01-62B	35	LOP	±1.75
2RSOA01-62A	210	CDA	±21.0
2CCPA01-62Y	15	LOP	±0.75
2CCPA01-62X	20	LOP	±1.00
2RSIA01-62A	20	LOP	±1.00
2RSIA01-62	195	CDA	±9.75
2QSSA01-62A	15	LOP	±0.75
2HVRA03-62	30	LOP	±1.50
2HVRA04-62	10	LOP	±0.50
2HVRB04-62	10	LOP	±0.50
2HVRC04-62	10	LOP	±0.50
2ENSH06-62A	15	LOP	±0.75

TABLE 4.8-1 (Continued)

LIST OF LOAD SEQUENCING TIMERS AND DESIGN SETPOINTS
"J" BUS

TIMER IDENTIFICATION	SET POINT (SECONDS)	INITIATING[(1)] SIGNAL	TOLERANCE (SECONDS)
2FWEB01-62	20	SI	±1.00
2FWEB01-62A	25	LOP	±1.25
2SWEB03-62	10	LOP	±0.50
2RSOB01-62B	35	LOP	±1.75
2RSOB01-62A	210	CDA	±21.0
2CCPB01-62Y	15	LOP	±0.75
2CCPB01-62X	20	LOP	±1.00
2RSIB01-62A	20	LOP	±1.00
2RSIB01-62	195	CDA	±9.75
2QSSB01-62A	15	LOP	±0.75
2HVRB03-62	30	LOP	±1.50
2HVRD04-62	10	LOP	±0.50
2HVRE04-62	10	LOP	±0.50
2HVRF04-62	10	LOP	±0.50
2ENSJ06-62A	15	LOP	±0.75

(1) SI - Safety Injection
LOP- Loss of Offsite Power
CDA- Containment Depressurization Actuation

TABLE 4.8-2

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. A day tank containing a minimum volume of 750 gallons of fuel,

2. A fuel storage system containing a minimum volume of 45,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5), 4.8.1.1.3 and 4.8.1.1.4.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

4160	volt Emergency Bus # 2H
4160	volt Emergency Bus # 2J
480	volt Emergency Bus # 2H, 2H1
480	volt Emergency Bus # 2J, 2J1
120	volt A.C. Vital Bus # 2-I
120	volt A.C. Vital Bus # 2-II
120	volt A.C. Vital Bus # 2-III
120	volt A.C. Vital Bus # 2-IV

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE.

1 - 4160 volt Emergency Bus, (2H or 2J)

1 - 480 volt Emergency Bus, (2H, 2H1) or (2J, 2J1)

2 - 120 volt A.C. Vital Busses, (2-I, 2-II) or (2-III, 2-IV)

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN "A" consisting of 125-volt D.C. bus No. 2-I and 2-II, 125-volt D.C. battery bank No. 2-I and 2-II and a full capacity charger.

TRAIN "B" consisting of 125-volt D.C. bus No. 2-III and 2-IV, 125-volt D.C. battery bank No. 2-III and 2-IV and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 125-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 pilot cell is between the minimum and maximum level indication marks,

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to 1.200,

3. The pilot cell voltage is is greater than or equal to 2.08 volts, and

4. The overall battery voltage is greater than or equal to 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.08 volts under float charge and has not decreased more than 0.05 volts from the value observed during the original acceptance test.

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.08 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

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

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

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

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 0.01 ohms.

4. The battery charger will supply at least 200 amperes at 125 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

2 125-volt D.C. busses, 2-I or 2-III and 2-II or 2-IV

2 125-volt battery bank and charger associated with the above D.C. busses.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 Hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.2.5 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) shown in Table 3.8-1 inoperable either:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated circuit breaker within 72 hours and verify the circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their circuit breakers tripped, or

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

SURVEILLANCE REQUIREMENTS

4.8.2.5 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that, on a rotating basis at least one 4.16 KV circuit breaker is OPERABLE by performing the following:

      (a) A CHANNEL CALIBRATION of the associated protective relays, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1.

2. By verifying the OPERABILITY of molded case circuit breakers, by selecting and functionally testing a representative sample of at least 10% of all the circuit breakers of that type. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functonally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By verifying the OPERABILITY of fuses, by selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

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

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number and Location (BKR NO)	Nominal Trip Setpoint (Amperes)	Nominal Response Time (Sec)	System Powered (Mark No)
1. 4160VAC			
25A3	Variable	0.030*	2-RC-P-01A
25B3	Variable	0.030*	2-RC-P-01B
25C3	Variable	0.030*	2-RC-P-01C
25H14	Variable	0.030*	2-RH-P-01A
25J14	Variable	0.030*	2-RH-P-01B
2. 480VAC from load centers			
24A1-2	9000	0.12	PZR Htrs.
24A2-11	3000	0.075	Stm Gen Supp Htrs
24B1-2	5250	0.10	PZR Htrs
24B2-9	4500	0.05	Stm Gen Supp Htrs
24C1-2	5250	0.055	PZR Htrs
24C2-16	3000	0.075	Refuel & Maint. Power
24H1-2	7500	0.06	2-RS-P-01A
24H1-5	5250	0.057	2-HV-F-01A
24H1-6	5250	0.10	PZR Htrs
24H-7	5250	0.10	2-HV-F-01C
24J1-2	7500	0.06	2-RS-P-01B
24J1-5	5250	0.057	2-HV-F-01B
24J1-6	5250	0.10	PZR. Htrs.
3. 480VAC from MCC			
2A1-1A1L	800	0.060	Loop 1 Pwr Recepts.
2A1-1C2	250	0.060	MOV-CC-200A
2A1-1C3	250	0.060	MOV-HV-200A
2A1-1C4	250	0.060	MOV-HV-200C
2A1-1D3L	500	0.060	Incore Inst Drive A
2A1-1E3	250	0.060	2-HV-F-03A
2A1-1F1	250	0.060	2-RC-P-01A1
2A1-1F3	500	0.060	2-RC-P-01A2
2A1-1F5	250	0.060	2-DA-P-04A
2B1-1A2	500	0.060	2-HV-F-92C
2B1-1B4L	250	0.060	Manip. Crane
2B1-1B4R	500	0.060	Fuel Transfer Cab.
2B1-1C1	500	0.060	2-HV-F-92B
2B1-1C2R	250	0.060	RCC Change Fixture
2B1-1C3	500	0.060	2-HV-F-92A

*Circuit Breaker Opening Time

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number and Location (BK NO)	Nominal Trip Setpoint (Amperes)	Nominal Response Time (Sec	System Powered (Mark No)
2B1-2A1L	800	0.060	RHR Cub. Pwr. Recepts.
2B1-2A2R	800	0.060	Loop 2 Pwr. Recepts.
2B1-2A2L	500	0.060	Incore Inst. Drive B
2B1-2A3	2000	0.060	React. Cont. Crane
2B1-2C2	250	0.060	MOV-CC-200B
2B1-2E3	250	0.060	2-HV-F-03B
2B1-2E4	250	0.060	2-RC-P-01B1
2B1-2E5	250	0.060	2-DA-P-04B
2B1-2F2	500	0.060	2-RC-P-01B2
2B1-2F4	250	0.060	2-DG-P-01A
2C1-1A3L	800	0.060	Cont. Elevator
2C1-1B3R	500	0.060	Incore Inst. Drive C
2C1-1B3L	500	0.060	Monorail Fdr.
2C1-1C2	500	0.060	2-RC-P-01C2
2C1-1D3	250	0.060	2-RC-P-01C1
2C1-1E3	250	0.060	2-DA-P-05
2C1-1E4	250	0.060	2-DA-P-01B
2C1-1A2L	800	0.060	Loop 3 Pwr. Recepts.
2C1-4A1	250	0.060	2-NS-P-01A
2C1-4B4L	1250	0.060	Stm. Gen. Supp. Htrs.
2C2-1A2R	800	0.060	Port Crane Recepts.
2C2-1A2L	1250	0.060	Stm. Gen. Supp. Htrs.
2C2-1B2R	1250	0.060	Stm. Gen. Supp. Hrs.
2C2-1C1	250	0.060	2-NS-P-01B
2H1-2NA1	2000	0.060	2-HV-F-37A
2H1-2NA3	500	0.060	2-IA-C-02A
2H1-2NB1	2000	0.060	2-HV-F-37B
2H1-2NC4	2000	0.060	2-HV-F-37C
2H1-2NH4	500	0.060	MOV-2865B
2H1-2NK1	500	0.060	2-IA-C-02A
2H1-2NL3	500	0.060	MOV-2865A
2H1-2SB4	250	0.060	MOV-2536
2H1-2SC1	500	0.060	MOV-2720A
2H1-2SD1	250	0.060	MOV-2700

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number and Location (BKR NO)	Nominal Trip Setpoint (Amperes)	Nominal Response Time (Sec)	System Powered (Mark No)
2H1-2SF1	250	0.060	MOV-2585
2H1-2SF2	500	0.060	MOV-2590
2H1-2SF3	250	0.060	MOV-2586
2H1-2SG1	250	0.060	MOV-2587
2H1-2SG2	500	0.060	MOV-2594
2H1-2SG3	500	0.060	MOV-2591
2H1-2SH1	250	0.060	MOV-2380
2H1-2SH2	500	0.060	MOV-2595
2H1-2SH3	500	0.060	MOV-2592
2H1-2SJ1	500	0.060	MOV-2593
2H1-2SK2L	500	0.060	Incore Inst. Drive D
2J1-2NA1	2000	0.060	2-HV-F-37F
2J1-2NA3	500	0.060	2-IA-C-02B
2J1-2NB1	500	0.060	2-IA-C-02B
2J1-2NK4	500	0.060	MOV-2865C
2J1-2S C1	500	0.060	MOV-2720B
2J1-2S F2	250	0.060	MOV-2535
2J1-2S F3	250	0.060	MOV-2701
2J1-2S G4	2000	0.060	2-HV-F-37E
2J1-2S H2L	500	0.060	Incore Inst. Drive E
2J1-2SJ1	2000	0.060	2-HV-F-37D

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.2.6 The thermal overload protection and/or bypass devices, integral with the motor starter, of each valve listed in Table 3.8.2 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.2.6 The above required thermal overload protection and/or bypass devices shall be demonstrated OPERABLE;

a. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

   1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2. Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBR ATION of a representative sample of at least 25% of all thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

TABLE 3.8-2

MOTOR-OPERATED VALVES-THERMAL OVERLOAD PROTECTION
AND/OR BYPASS DEVICES

Valve Number	Function	Bypass Device
MOV-QS200A	Quench Spray Pump Suction Valve	No
MOV-QS200B	Quench Spray Pump Suction Valve	No
MOV-QS201A	Quench Spray Pump Discharge Valve	No
MOV-QS201B	Quench Spray Pump Discharge Valve	No
MOV-QS202A	Chemical Addition to Refueling Water Storage Tank Valve	No
MOV-QS202B	Chemical Addition to Refueling Water Storage Tank Valve	No
MOV-RS200A	Casing Cooling to Outside Recirculation Spray Pump	No
MOV-RS200B	Casing Cooling to Outside Recirculation Srpay Pump	No
MOV-RS201A	Casing Cooling to Outside Recirculation Spray Pump	No
MOV-RS201B	Casing Cooling to Outside Recirculation Spray Pump	No
MOV-RS255A	Outside Recirculation Spray Pump Suction Valve	No
MOV-RS255B	Outside Recirculation Spray Pump Suction Valve	No
MOV-RS256A	Outside Recirculation Spray Pump Discharge Valve	No
MOV-RS256B	Outside Recirculation Spray Pump Discharge Valve	No
MOV-SW200A	Outlet to Service Water Reservior	No
MOV-SW200B	Outlet to Service Water Reservoir	No
MOV-SW201A	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW201B	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW201C	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW201D	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW203A	Recriculation Spray Heat Exchanger Supply Shutoff	No
MOV-SW203B	Recirculation Spray Heat Exchanger Supply Shutoff	No
MOV-SW203C	Recirculation Spray Heat Exchanger Supply Shutoff	No
MOV-SW203D	Recirculation Spray Heat Exchanger Supply Shutoff	No
MOV-SW202A	Recirculation Spray Heat Exchanger Supply Header Crossover Isolation Valve	No
MOV-SW206A	Recirculation Spray Heat Exchanger Supply Header Crossover Isolation Valve	No
MOV-SW202B	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW206B	Recirculation Spray Heat Exchanger Supply Header Isolation	No
MOV-SW204A	Return from Recirculation Spray Heat Exchanger	No
MOV-SW204B	Return from Recirculation Spray Heat Exchanger	No
MOV-SW204C	Return from Recirculation Spray Heat Exchanger	No
MOV-SW204D	Return from Recirculation Spray Heat Exchanger	No
MOV-SW205A	Recirculation Spray Heat Exchanger Return Header Isolation	No
MOV-SW205B	Recirculation Spray Heat Exchanger Return Header Isolation	No

TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES-THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve Number	Function	Bypass Device
MOV-SW205C	Recirculation Spray Heat Exchanger Return Header Isolation	No
MOV-SW205D	Recirculation Spray Heat Exchanger Return Header Isolation	No
MOV-SW208A	Component Cooling Heat Exchanger Supply Valve	No
MOV-SW208B	Component Cooling Heat Exchanger Supply Valve	No
MOV-SW210A	Supply to Air Cooling Coils	No
MOV-SW210B	Supply to Air Cooling Coils	No
MOV-SW214A	Return from Air Cooling Coils	No
MOV-SW214B	Return from Air Cooling Coils	No
MOV-SW215A	Auxiliary S.W. Pump Discharge Header Isolation Valve	No
MOV-SW215B	Auxiliary S.W. Pump Discharge Header Isolation Valve	No
MOV-SW217	Auxiliary S.W. Pump Discharge Header Isolation Valve	No
MOV-SW213A	Fuel Pit Coolers Supply Header	No
MOV-SW213B	Fuel Pit Coolers Supply Header	No
MOV-SW219	Cir. Water Intake Service Water Pump Make-up Isolation Valve	No
MOV-SW220A	S.W. Discharge to Circ. Tunnel	No
MOV-SW220B	S.W. Discharge to Circ. Tunnel	No
MOV-FW200A	Aux. Steam Generator Feedpump Discharge Valve	No
MOV-FW200B	Aux. Steam Generator Feedpump Discharge Valve	No
MOV-FW200C	Aux. Steam Generator Feedpump Discharge Valve	No
MOV-FW200D	Aux. Steam Generator Feedpump Discharge Valve	No
MOV-HV204-1	Emergency Control Room Vent	No
MOV-HV204-2	Emergency Control Room Vent	No
MOV-HV211A	A/C Control Room Chiller Outlet	No
MOV-HV211B	A/C Control Room Chiller Outlet	No
MOV-HV211C	A/C Control Room Chiller Outlet	No
MOV-HV213A	A/C Cond. Water Discharge	No
MOV-HV213B	A/C Cond. Water Discharge	No
MOV-HV213C	A/C Cond. Water Discharge	No
MOV-HV218-1	Emergency Air Supply	No
MOV-HV218-2	Emergency Air Supply	No
MOV-2700	Residual Heat Removal Inlet Isolation Valve	No
MOV-2701	Residual Heat Removal Inlet Isolation Valve	No
MOV-2720A	Residual Heat Removal Outlet Isolation	No
MOV-2720B	Residual Heat Removal Outlet Isolation	No
MOV-2590	Reactor Coolant System Loop A Hot Leg Stop Valve	No
MOV-2592	Reactor Coolant System Loop B Hot Leg Stop Valve	No
MOV-2594	Reactor Coolant System Loop C Hot Leg Stop Valve	No
MOV-2591	Reactor Coolant System Loop A Cold Leg Stop Valve	No
MOV-2593	Reactor Coolant System Loop B Cold Leg Stop Valve	No
MOV-2595	Reactor Coolant System Loop C Cold Leg Stop Valve	No
MOV-2585	Reactor Coolant System Loop A Bypass	No
MOV-2586	Reactor Coolant System Loop B Bypass	No
MOV-2587	Reactor Coolant System Loop C Bypass	No
MOV-2381	RCP Seal Leakoff Isolation Valve	No
MOV-2380	RCP Seal Leakoff Isolation Valve	No

TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES-THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve Number	Function	Bypass Device
MOV-2535	Reactor Coolant System Pressurizer Relief Isolation Valve	No
MOV-2536	Reactor Coolant System Pressurizer Relief Isolation Valve	No
MOV-2115B	Charging Pump Suction from Refueling Water Storage Tank	No
MOV-2115C	Charging Pump Suction from Volume Control Tank	No
MOV-2115D	Charging Pump Suction from Refueling Water Storage Tank	No
MOV-2115E	Charging Pump Suction from Volume Control Tank	No
MOV-2286A	Charging Pump A Discharge	No
MOV-2287A	Charging Pump A Alternate Discharge	No
MOV-2286B	Charging Pump B Discharge	No
MOV-2287B	Charging Pump B Alternate Discharge	No
MOV-2286C	Charging Pump C Discharge	No
MOV-2287C	Charging Pump C Alternate Discharge	No
MOV-2289A	Charging Pumps Discharge Line Stop Valve	No
MOV-2289B	Charging Pumps Discharge Line Stop Valve	No
MOV-2373	Charging Pumps Recirculation Stop Valve	No
MOV-2275A	Charging Pump A Recirc. Stop Valve	No
MOV-2275B	Charging Pump B Recirc. Stop Valve	No
MOV-2275C	Charging Pump C Recirc. Stop Valve	No
MOV-2350	Emergency Borate Valve	No
MOV-2370	Charging Pump to Seal Water Recirc. Stop Valve	No
MOV-2836	High Head Cold Leg Safety Injection Isolation Valve	No
MOV-2869A	Charging Pump Discharge Header Safety Injection Stop Valve	No
MOV-2869B	Charging Pump Discharge Header Safety Injection Stop Valve	No
MOV-2267A	Charging Pump A Suction Valve (VCT)	No
MOV-2267B	Charging Pump A Suction Valve (LHSI)	No
MOV-2269A	Charging Pump B Suction Valve (VCT)	No
MOV-2269B	Charging Pump B Suction Valve (LHSI)	No
MOV-2270A	Charging Pump C Suction Valve (VCT)	No
MOV-2270B	Charging Pump C Suction Valve (LHSI)	No
MOV-2860A	Low Head Safety Injection Pump A Suction Valve	No
MOV-2860B	Low Head Safety Injection Pump B Suction Valve	No
MOV-2862A	Low Head Safety Injection Pump A Suction Valve	No
MOV-2862B	Low Head Safety Injection Pump B Suction Valve	No
MOV-2863A	Low Head Safety Injection Pump A Discharge to Charging Pumps	No
MOV-2863B	Low Head Safety Injection Pump B Discharge to Charging Pumps	No
MOV-2864A	Low Head Safety Injection Pump A Discharge Valve	No
MOV-2864B	Low Head Safety Injection Pump B Discharge Valve	No

TABLE 3.8-2 (Continued)

MOTOR-OPERATED VALVES-THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve Number	Function	Bypass Device
MOV-2885A	Low Head Safety Injection Pump A Rec. Valve	No
MOV-2885B	Low Head Safety Injection Pump B Rec. Valve	No
MOV-2885C	Low Head Safety Injection Pump A Recirc. Valve	No
MOV-2885D	Low Head Safety Injection Pump B Recirc. Valve	No
MOV-2890A	Low Head Safety Injection Pump A Discharge Line Stop Valve	No
MOV-2890B	Low Head Safety Injection Pump B Discharge Line Stop Valve	No
MOV-2890C	Low Head Safety Injection Pump Discharge Line Stop Valve	No
MOV-2890D	Low Head Safety Injection Pump Discharge Line Stop Valve	No
MOV-2867A	Boron Injection Tank Inlet Valve	No
MOV-2867B	Boron Injection Tank Inlet Valve	No
MOV-2867C	Boron Injection Tank Inlet Valve	No
MOV-2867D	Boron Injection Tank Inlet Valve	No
MOV-2865A	Accumulator Tank Discharge Valve	No
MOV-2865B	Accumulator Tank Discharge Valve	No
MOV-2865C	Accumulator Tank Discharge Valve	No

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod located within the reactor pressure vessel, in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

* The reactor shall be maintained in MODE 6 whenever the reactor vessel head is unbolted or removed and fuel is in the reactor vessel.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

SURVEILLANCE REQUIREMENTS (Continued)

b. Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9.

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS. Written documentation of the 12 hour checks is not required.

REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 3250 pounds, and
   2. An overload cut off limit less than or equal to 2850 pounds.

b. The auxiliary hoist used for movement of control rods having:

   1. A minimum capacity of 700 pounds, and
   2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

CRANE TRAVEL - SPENT FUEL PIT

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2500 pounds shall be prohibited from travel over irradiated fuel assemblies in the spent fuel pit.

APPLICABILITY: With irradiated fuel assemblies in the spent fuel pit.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Loads shall verified to be less than 2500 pounds prior to movement over irradiated fuel assemblies in the spent fuel pit.

RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 4 hours.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.
The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from the containment gaseous and particulate radiation monitoring instrumentation channels.

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least, 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During CORE ALTERATIONS while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the startup of and at least once per 24 hours thereafter during CORE ALTERATIONS.

REFUELING OPERATIONS

SPENT FUEL PIT WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

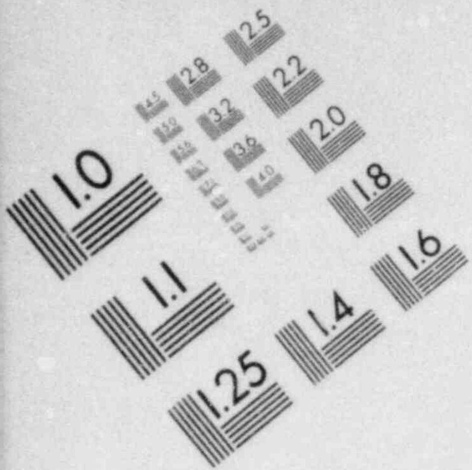
APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pit.

ACTION:

With the requirements of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pit areas and place the load in a safe condition. Restore water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pit shall be determined to be at least at the minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pit.



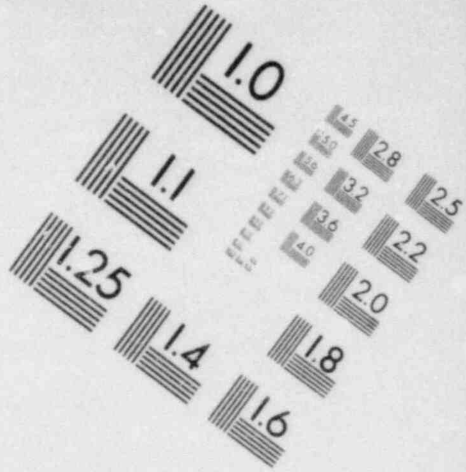
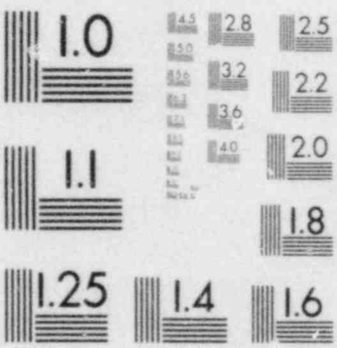


IMAGE EVALUATION
TEST TARGET (MT-3)



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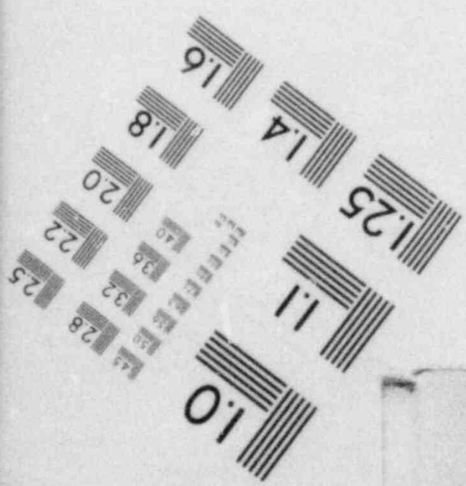
MICROCOPY RESOLUTION TEST CHART



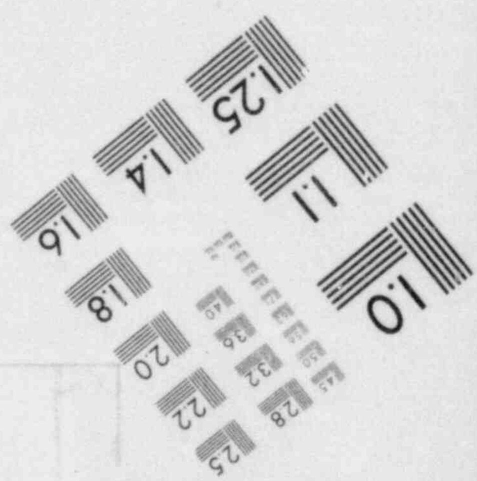


IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

FUEL BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 A fuel building ventilation system shall be OPERABLE and discharging through at least one auxiliary building HEPA filter and charcoal adsorber assembly.

APPLICABILITY:

a. During irradiated fuel movement within the spent fuel pit, or

b. During crane operation with loads over irradiated fuel in the spent fuel pit.

ACTION:

a. With a fuel building ventilation system inoperable, irradiated fuel movement within the storage pool or crane operation with loads over the spent fuel pit may proceed provided the fuel building ventilation system is in operation and discharging through at least one train of HEPA filters and charcoal adsorber assemblies.

b. With no fuel building ventilation system OPERABLE, suspend all operations involving movement of irradiated fuel within the spent fuel pit or crane operation with loads over the spent fuel pit until at least one fuel building ventilation system is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3, 3.0.4 and 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel building ventilation system shall be demonstrated OPERABLE and discharging through at least one auxiliary building HEPA filter and charcoal adsorber assembly;

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber assembly for 15 minutes,

b. At least once per 18 months during system operation, by verifying a 1/8 inch vacuum, water gauge, relative to the outside atmosphere, and

c. By performance of the Surveillance Requirements of Specification 4.7.8.1 b, c, d, e and f.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate and continue boration at greater than or equal to 10 gpm of a solution containing at least 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing at least 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

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

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications shall be performed at least once per 12 hours during PHYSICS TESTS.

a. Specification 4.2.2.2 and 4.2.2.3.

b. Specification 4.2.3.1 and 4.2.3.2.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.1.5, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F at least once per 30 minutes durng PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTION

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indicator channels inoperable, or more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indicator channels shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indicator channels agree:

a. Within 12 steps when the rods are stationary, and
b. Within 24 steps during rod motion.

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* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one control rod bank is withdrawn from the fully inserted position at one time.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these technical specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 requires each Reactor Coolant System accumulator to be OPERABLE and provides explicit ACTION requirements if one accumulator is inoperable. Under the terms of Specification 3.0.3, if more than one accumulator is inoperable, the unit is required to be in at least HOT STANDBY within 1 hour and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Quench Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required Quench Spray Systems are inoperable, the unit is required to be in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the following 6 hours, and in at least COLD SHUTDOWN in the next 30 hours. It is assumed that the unit is brought to the required MODE within the required times by promptly initiating and carrying out the appropriate ACTION statement.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for a 72 hour out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.8.1.1 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. The ACTION statement provides a 24-hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits, would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

In MODES 5 or 6, Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the [illegible]illance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to me[illegible] the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability con[illegible]on takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM, as provided by either one RCP or one RHR pump as required by Specification 3.4.1.1, provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9957 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control. The requirement that certain valves remain closed at all times except during planned boron dilution or makeup, activities provides assurance that an inadvertent boron dilution will not occur.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.0×10^{-4} delta k/k/°F. The MTC value of -3.1×10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value -4.0×10^{-4} delta k/k/°F.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 4450 gallons of 20,000 ppm borated water from the boric acid storage tanks or 70,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 340°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.2 BORATION SYSTEMS (Continued)

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

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

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} greater than or equal to 500°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.10 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other

THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 81% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 81% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 81% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

POOR ORIGINAL

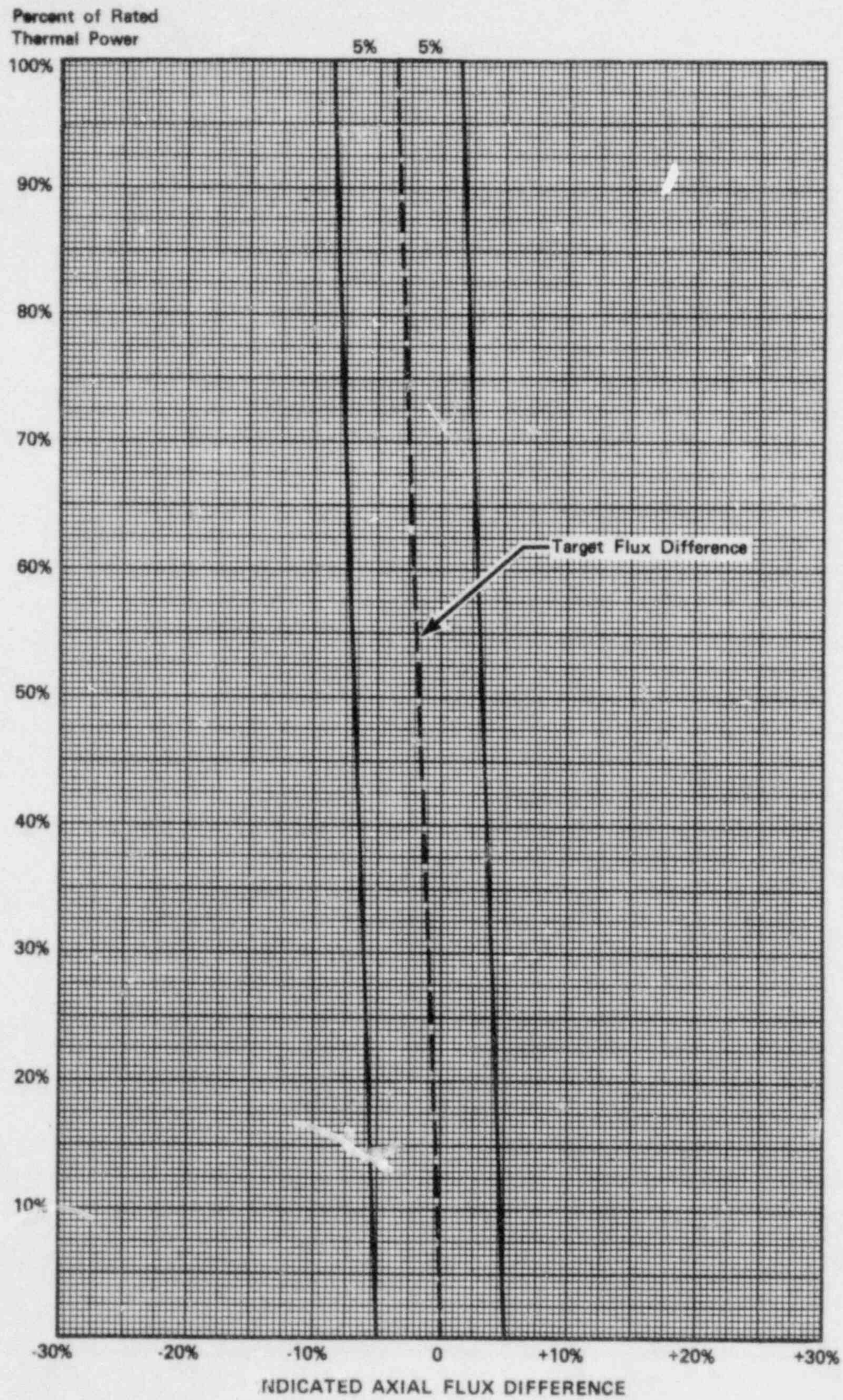


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS - $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable and will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, 4% is the appropriate experimental error allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which means that normal operation will result in $F^N_{\Delta H}$ less than or equal to 1.55/1.08. The 8% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above 90% of RATED THERMAL POWER. This additional limitation on F_Q is necessary in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

Not Applicable.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

Not Applicable

3/4.3.3.5 AUXILIARY SHUTDOWN PANEL MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to (1) monitor the core flux patterns that are representative of the peak core power density, and (2) limit the core average axial power profile such that the total power peaking factor F_Q is maintained within acceptable limits.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

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

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 340°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting from the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 and 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

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

The power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE ensures that the plant will be able to establish natural circulation.

3/4.4.5 STEAM GENERATORS

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

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the orignal tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are generally consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 30 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDRY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the North Anna site such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the first full-power service period.

POOR ORIGINAL

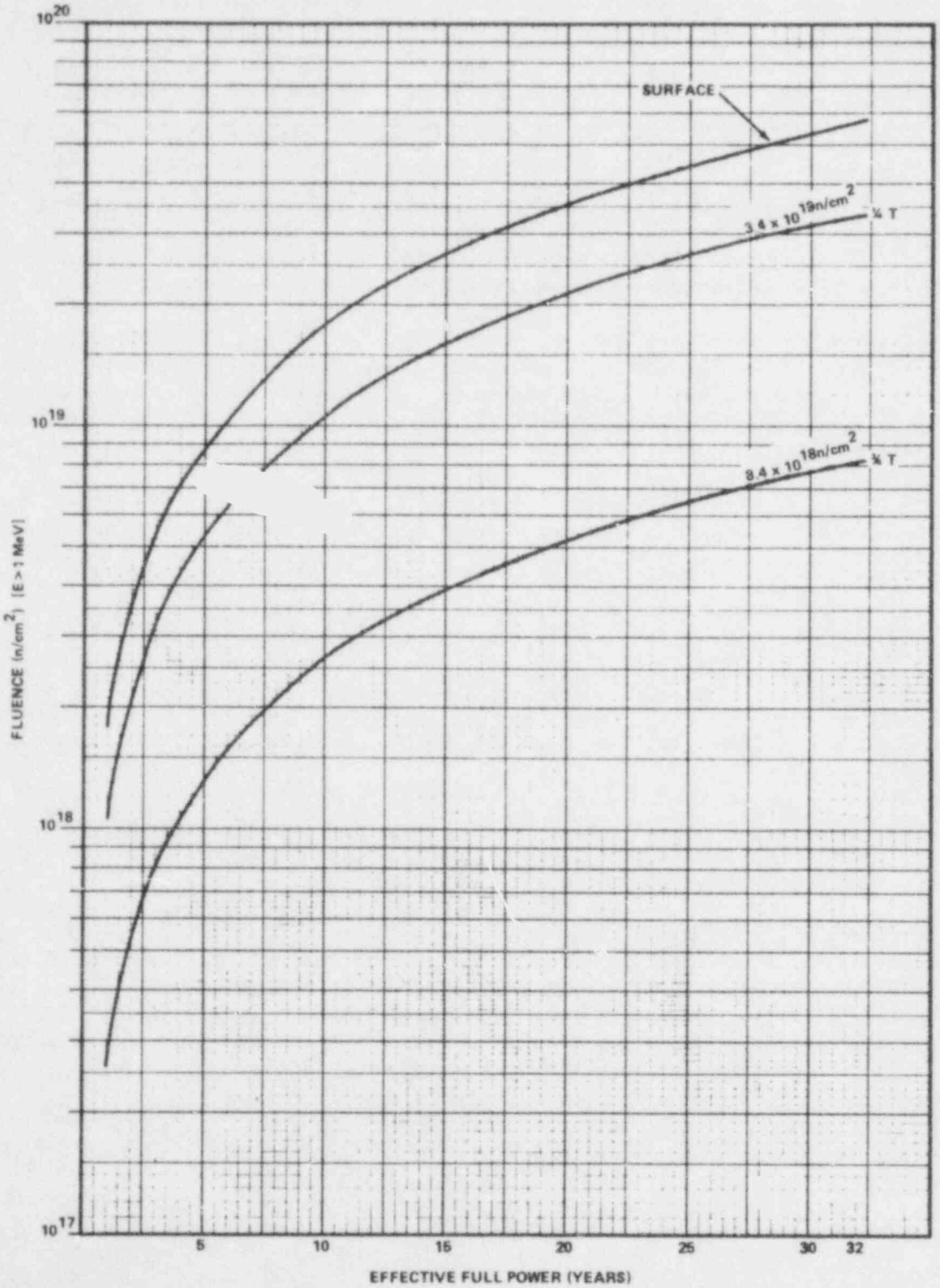


FIGURE B 3/4 4-1 FAST NEUTRON FLUENCE (E >1mev) AS A FUNCTION OF FULL POWER SERVICE LIFE

a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

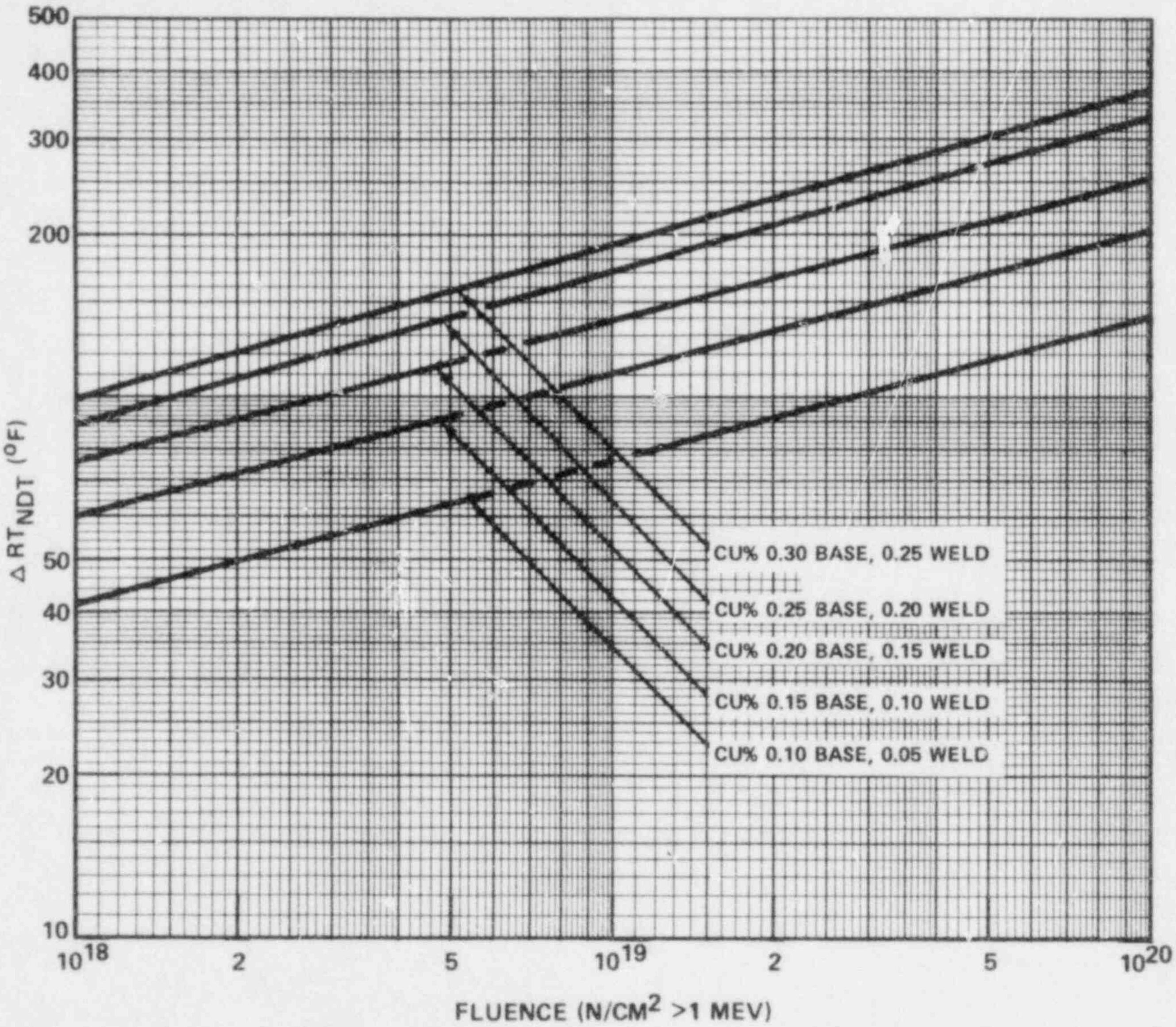


Figure B 3/4.4-2

Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550^o F Temperature

NORTH ANNA - UNIT 2

TABLE B 3/4 4-1

REACTOR VESSEL TOUGHNESS TABLE (UNIT 2)

Component	Heat No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/ 35 mil Temp. (°F) Parallel to Major Working Direction	Minimum 50 ft-lb/ 35 mil Temp. (°F) Normal to Major Working Direction	RT_{NDT} (°F)	Average Upper Shelf (ft-lb) Parallel to Major Working Direction	Average Upper Shelf (ft-lb) Normal to Major Working Direction
Cl. Hd. Dome	53650-1	A533,B,Cl.1			-22	10	30*	-22	79(c)	
Cl. Hd. Flange	523122	A508,Cl.2			-31	4	24*	-31	131(c)	
Ves. Flange	523000	A508,Cl.2			-22	-41	-21*	-22	146(c)	
Inlet Nozzle	990426	A508,Cl.2			-40	60	80*	20	72(c)	
Inlet Nozzle	54567-2	A508,Cl.2			-31	53	73*	13	118(c)	
Intet Nozzle	54590-2	A508,Cl.2			-31	19	39*	-21	92(c)	
Outlet Nozzle	990426-22	A508,Cl.2			-13	41	61*	1	80(c)	
Outlet Nozzle	990426-3	A508,Cl.2			-31	43	63*	3	62(c)	
Outlet Nozzle	791291	A508,Cl.2			-22	21	41*	-19	115(c)	
Upper Shell	990598									
	291396	A508,Cl.2	0.08	0.010	+5	49	69*	9	86(c)	
Inter. Shell	990496									
	292429	A508,Cl.2	0.09	0.010	-49	49	135(s)	75(s)	85(c)	74(s)
Lower Shell	990533									
	297355	A508,Cl.2	0.13	0.013	-13	31	116(s)	56(s)	92(c)	80(s)
Bot. Hd. Seg.	53648-1	A533,B,Cl.1			-22	-9	11*	-22	94(c)	
Bot. Hd. Seg.	53648-4	A533,B,Cl.1			-13	31	51*	-6	77(c)	
Bot. Hd. Dome	53695-1	A533,B,Cl.1			-40	21	41*	-19	87(c)	
Weld		Weld	0.088	0.017	-67		12	-48		107
Haz		Haz			-49		-20	-49		125

* Estimated temperature based on NRC Standard Review Plan 5.3.2

(c) Average of highest impact values reported - % shear not reported

(s) Average tranverse data obtained fro- surveillance program.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 8 effective full power years of service life. The 8 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 8 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. The first capsule will be removed at the end of the first core cycle. Successive capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure -temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressuretemperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 340°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a charging pump and its injection into a water solid RCS.

When the temperature of the RCS cold legs is between 320°F and 340°F, overpressure protection can also be provided by a bubble in the pressurizer. In such a case, a maximum pressurizer water volume of 457 cu. ft. has been selected to provide at least 10 minutes for operator response in the event of a malfunction resulting in maximum flow from one charging pump.

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.10.2 STEAM GENERATOR SUPPORTS

For the A572 material, operation above 225° provides a conservative temperature limit and thus toughness level in the steel. This assures the safety of the A572 material even under the worst postulated accident conditions. The points to be monitored were determined during hot functional testing, which indicated the top level corner lags the middle level corner during heatup; however, once the material achieved 225°F the top level corner exceeded the temperature of the middle level corner. The latter thus becomes the controlling zone during operation.

For the monitored top level corner of the steam generator supports, operation below 355°F provides assurance that no other region of the supports will exceed this temperature. The monitored top level corner is the highest temperature region in the supports. With the temperature of the supports less than 355°F all materials will be within allowable stress limits even, under the worst postulated accident conditions.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 340°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 137°F at 22,500 ppm boron.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

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

3/4.6.1.2 CONTAINMENT LEAKAGE

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

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

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 and 3/4.6.1.5 INTERNAL PRESSURE AND TEMPERATURE

The limitations on containment internal pressure and average air temperature ensure that

1) The containment pressure is prevented from reaching the containment lower design pressure of 5.5 psia for an inadvertent containment spray actuation,

2) That the peak clad fuel temperature will remain less than 2200°F for a LOCA and

3) That for either a LOCA or MSLB:

   a) The peak containment pressure will be limited to the upper containment design pressure of 45 psig,

   b) The containment internal pressure can be returned subatmospheric within 60 minutes, and

   c) Safety related equipment within the containment will not experience temperatures greater than those to which they have previously been qualified.

   d) It is a design criteria that the containment internal pressure remain subatmospheric after 60 minutes.

The limits shown in Figure 3.6-1 and Specification 3.6.1.5 are consistant with the assumptions of the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 40.6 psig in the event of a LOCA. The visual examination of the concrete and liner and the Type A leakage tests are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH AND RECIRCULATION SPRAY SYSTEMS

The OPERABILITY of the containment spray systems ensures that containment depressurization and subsequent return to subatmospheric pressure will occur in the event of a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses.

3/4.6.2.3 CHEMICAL ADDITION SYSTEM

The OPERABILITY of the chemical addition system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH minimum volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usuable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM

3/4.6.5.1 STEAM JET AIR EJECTOR

The closure of the isolation valves in the suction of the steam jet air ejector ensures that 1) the containment internal pressure may be maintained within its operation limits by the mechanical vacuum pumps and 2) the containment atmosphere is isolated from the outside environment in the event of a LOCA. These valves are required to be closed for containment intergrity.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.8×10^6 lbs/hr which is 105 percent of the total secondary steam flow of 12.2×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

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

For 3 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

For 2 loop operation with stop valves closed

$$SP = \frac{(X) - (Y)(U)}{X} \times 71$$

For 2 loop operations with
stop valves open

$$SP = \frac{(X) - (Y)(U)}{X} \times 66$$

Where:

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

V = maximum number of inoperable safety valves per steam line

U = maximum number of inoperable safety valves per operating steam line

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation

71 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation with stop valves closed.

66 = Maximum percent of RATED THERMAL POWER permissible by P-8 setpoint for 2 loop operation with stop valves open.

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 4,275,420

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 855,084

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

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

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 340 gpm at a pressure of 1064 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1064 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 EMERGENCY CONDENSATE STORAGE TANK

The OPERABILITY of the emergency condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usuable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM TRIP VALVES

The OPERABILITY of the main steam trip valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam trip valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 and 3/4.7.1.7 STEAM TURBINE and OVERSPEED PROTECTION

The turbine generator at the North Anna facility is arranged in a non-peninsular orientation. Analysis has shown that this arrangement is such that if a turbine failure occurs as a result of destructive overspeed, potentially damaging missles could impact the auxiliary building, containment, control room and other structures housing safety related equipment. The requirements of these two specifications provide additional assurance that the facility will not be operated with degraded valve performance and/or flawed turbine material which are the major contributors to turbine failures.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SUBSYSTEM

The OPERABILITY of the component cooling water subsystem ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27. "Ultimate Heat Sink for Nuclear Plants", March 1974.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation 256 feet Mean Sea Level USGS datum is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

The OPERABILITY of the control room ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

The OPERABILITY of the safeguards area ventilation system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.9.1 and 3/4.7.9.2 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

The OPERABILITY of the RHR system ensures that residual heat removal capability is available below 350°F following plant shutdown. The RHR system is not part of the ECCS system.

3/4.7.10 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly insalled snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service review not intended to affect plant operation.

3/4.7.11 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES

In order to assure that settlement does not exceed allowable values, a program has been established to conduct a survey of a specified number of points at the site on a semiannual basis. The first survey was conducted in May 1976 to establish baseline elevations for most of the points. Where applicable, the baseline elevations of points established subsequent to the May 1976 survey have been adjusted to the May 1976 survey by an evaluation of the settlement records of settlement points on the same structure or on nearby structures. Baseline elevations for points established on dates other than May 1976 are indicated in Table 3.7-5. Additional surveys will be performed semiannually. The determination of the elevation of the points shall be by precise leveling with second order Class II accuracy as defined by the U.S. Department of Commerce National Oceanic and Atmospheric Administration, National Ocean Survey, 1974. When any settlement point listed in Table 3.7-5 is inaccessible during a survey, comparison to allowable settlement shall be based on settlement of other points on the same structure or on nearby structures having similar foundation conditions. When any settlement point listed in Table 3.7-5 is dislocated or replaced, a documented review of the settlement records of points on the same structure and additionally points on nearby structures having similar foundation conditions shall provide a new reference elevation for the point that reflects the estimated settlement since the baseline survey. If the total settlement or differential settlement exceeds 75 percent of the allowable value, the frequency of surveillance shall be increased as dictated by the engineering review. Measurements of certain points are required to be performed at least once per 31 days for the first five years of facility operation to provide additional settlement information.

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES (Continued)

Allowable differential movement is controlled by pipe deflections permitted by fixation points in buildings. The items limiting differential settlement are as follows:

Reference	Monitoring Points	Limiting Item
Containment Unit 2	Fuel Building	Fuel Transfer Tube
Containment Unit 2	Auxiliary Building	3"-SI-417
Containment Unit 2	Unit 2 Safeguards Area	6"-RH-418
Containment Unit 2	Unit 2 Main Steam Valve House	6"-SI-416
Containment Unit 2	Service Building (E-15)	16"-WFPD-409
Safeguards Area Unit 2	Unit 2 Main Steam Valve House	24"-WS-426
Auxiliary Building	Unit 2 Main Steam Valve House	8"-SI-440
Auxiliary Building	Fuel Building	4"-RP-28
Auxiliary Building	Service Building Tunnel	24"-WS-102
Service Building (E-5, E-6)	Unit 1 Main Steam Valve House	24"-WS-26-151-Q3
Service Building (E-14)	Unit 2 Main Steam VAlve House	24"-WS-426-151-Q3
Auxiliary Feedwater Pump House Unit 2	Pipe Tunnel	6"-WCMU-412-151-Q3
Decontamination Building	Pipe Tunnel	3"-CC-90-151-Q3
Fuel Building	Waste Gas Decay Tank Enclosure	4"-GW-30-154-Q3
Safeguards Area Unit 2	Unit 2 Casing Cooling Building	6"-RS-455-153A-Q3
Service Water Pump House	Service Water Piping @ SWPH	Expansion Joint
Service Water Pump House	Pipe Hanger in Reservoir	24"-WS-11-151-Q3
Service Water Pump House	Service Water Pump House	Mat

The items limiting total settlement of structures are as follows:

Monitoring Points	Limiting Items
Service Water Piping @ SWPH	36"-WS-1-151-Q3
Circulating Water Intake Structure	Service Water Piping Expansion Joint
Turbine Building (B-9 1/2)	24"-WS-425-151-Q3
Service Building (E-17)	36"-WS-1-151-Q3
Fuel Oil Pump House	2 1/2"-FOF-151-S
Boron Recovery Tank Dike	No settlement expected; settlement in excess of 0.03 feet would indicate an abnormality.

3/4.7.13 GROUNDWATER LEVEL-SERVICE WATER RESERVOIR

A program to monitor groundwater levels in the area of the service water reservoir has been established to ensure that the integrity of the service water reservoir embankments and pumphouse is maintained.

Groundwater threshold levels have been established based on historical groundwater data available in 1977. These levels are sufficiently conservative to ensure that the service water reservoir and pumphouse will perform their intended function. An engineering evaluation will be performed if these threshold values are exceeded, to determine if there is any substantive cause to believe that any aspect of the service water reservoir, dike or pumphouse will not perform its intended function. A conclusion to this effect, and the appropriate corrective actions to be performed, will be reported to the Commission.

The groundwater threshold levels are periodically reviewed to determine whether a changing groundwater environment warrants a change in threshold levels.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinquish fires occuring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, CO_2, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.15 PENETRATION FIRE BARRIERS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penentrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either 1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or 2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1 and 3/4.8.2 A.C. and D.C. POWER SOURCES AND DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977.

Containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or be demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance frequency applicable to molded case circuit breakers and/or fuses provides assurance of breaker and/or fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide

3/4.8.1 and 3/4.8.2 A.C. and D.C. POWER SOURCES AND DISTRIBUTION (Continued)

variety exists within any manufacturer's brand of molded case circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuse for surveillance purposes.

The OPERABILITY of the motor-operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, March 1977.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT

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

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

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

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

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

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

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND SPENT FUEL PIT

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

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM

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

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGINS

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is, at times, necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which may in turn cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.5.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATOR CHANNELS-SHUTDOWN

This special test exception permits the position indicator channels to be inoperable during rod drop time measurement. This exception is required since the data necessary to determine the rod drop time is derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and therefore can not be observed if the position indicator channels remain OPERABLE.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape with a dome roof and having the following design features:

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 190 feet, 7 inches.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 10 feet.

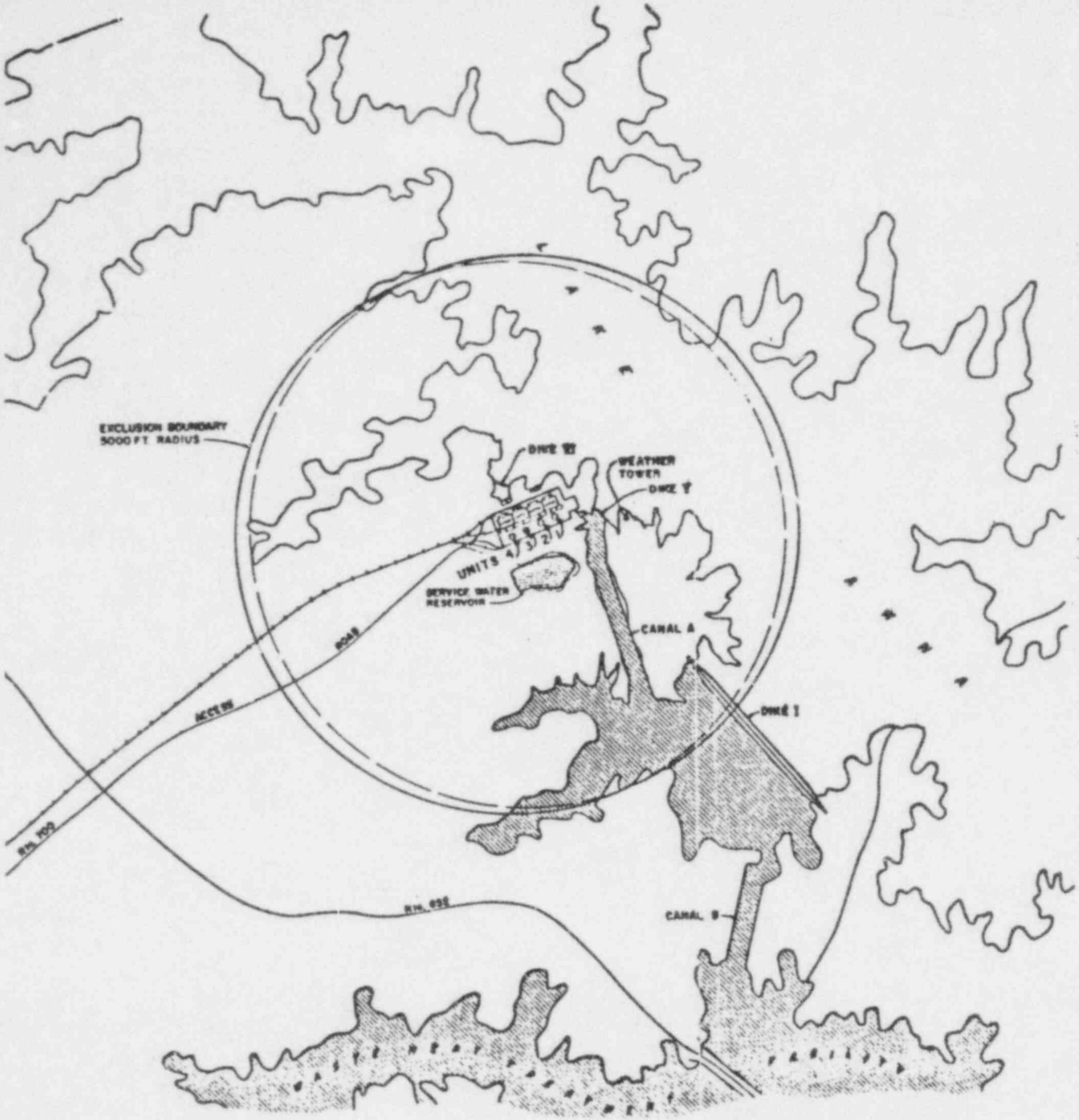
f. Nominal thickness of the cylindrial portion of the steel liner = 3/8 inches.

g. Net free volume = 1.825×10^6 cubic feet.

h. Nominal thickness of hemispherical dome portion of the steel liner = 1/2 inch.

DESIGN PRESSURE AND TEMPERATURE

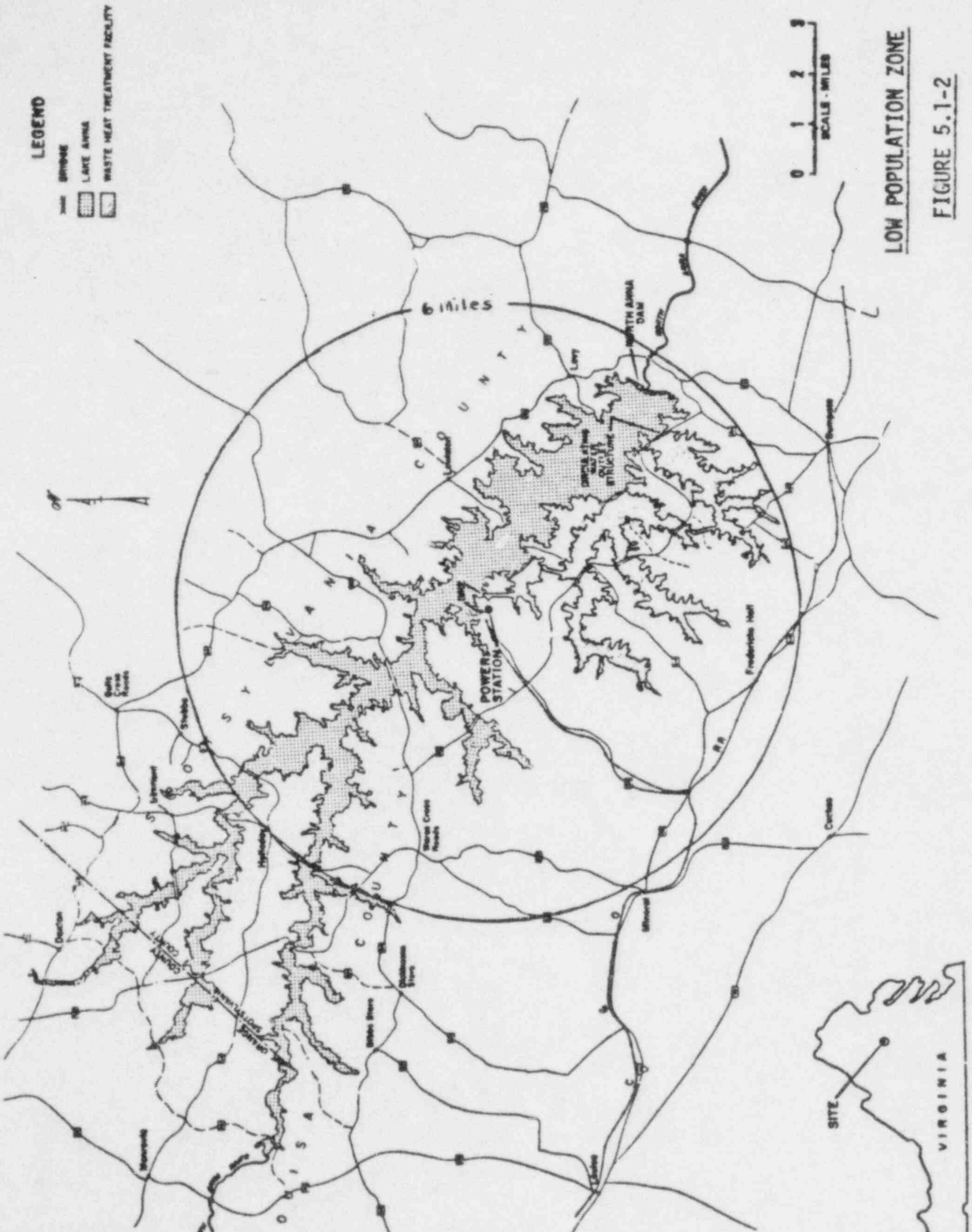
5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.



EXCLUSION AREA

NORTH ANNA POWER STATION

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9957 $\pm$ 10 cubic feet at a nominal Tavg of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 3.7% delta k/k for uncertainties.

b. A nominal 14 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed .98, with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

5.6.1.3 If new fuel for the first core loading is stored dry in the spent fuel storage racks the center-to-center distance between the new fuel assemblies will be administratively limited to 28 inches and the k_{eff} shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 288.83 feet Mean Sea Level, USGS datum.

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 966 fuel assemblies.

5.7 COMPONENT CYCLIC or TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at 100°F/hr and 200 cooldown cycles at 100°F/hr	Heatup cycle -T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle -T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at 200°F/hr	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER. (Full Power Trip)
	10 inadvertent pressurizer auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function and shall be the only individual that may direct the licensed activities of licensed operators. A management directive to this effect, signed by the Executive Vice President, Power, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODES 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include the minimum shift crew shown in Table 6.2-1 or any personnel required for other essential functions during a fire emergency.

#The health physics technician and Fire Brigade composition may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

6.2.3 SAFETY ENGINEERING STAFF (SES)

FUNCTION

6.2.3.1 The SES shall function to examine plant operating characteristics, NRC issuances, industry advisories, License Event Reports, and other sources which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The SES shall be composed of at least five dedicated, full-time engineers located onsite.

RESPONSIBILITIES

6.2.3.3 The SES shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

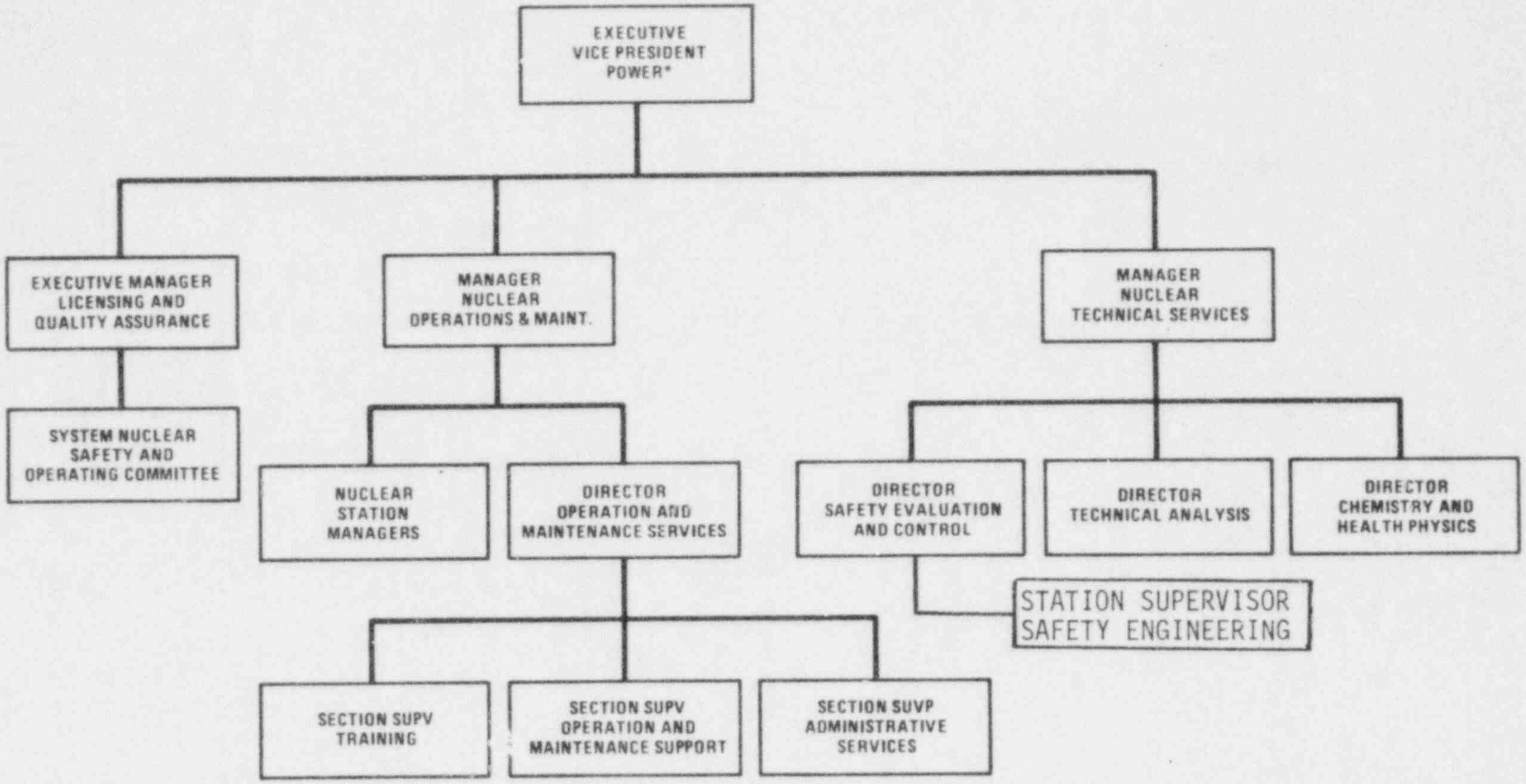
6.2.3.4 The SES shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Director, Safety Evaluation and Control.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to Shift Supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit.

6.2.4.2 The Shift Technical Advisor shall disseminate relevant operational experience identified by the SES.

*Not responsible for sign-off function.



*RESPONSIBLE FOR CORPORATE FIRE PROTECTION PROGRAM

Figure 6.2-1 Offsite Organization for Facility Mana‸ ment and Technical Support

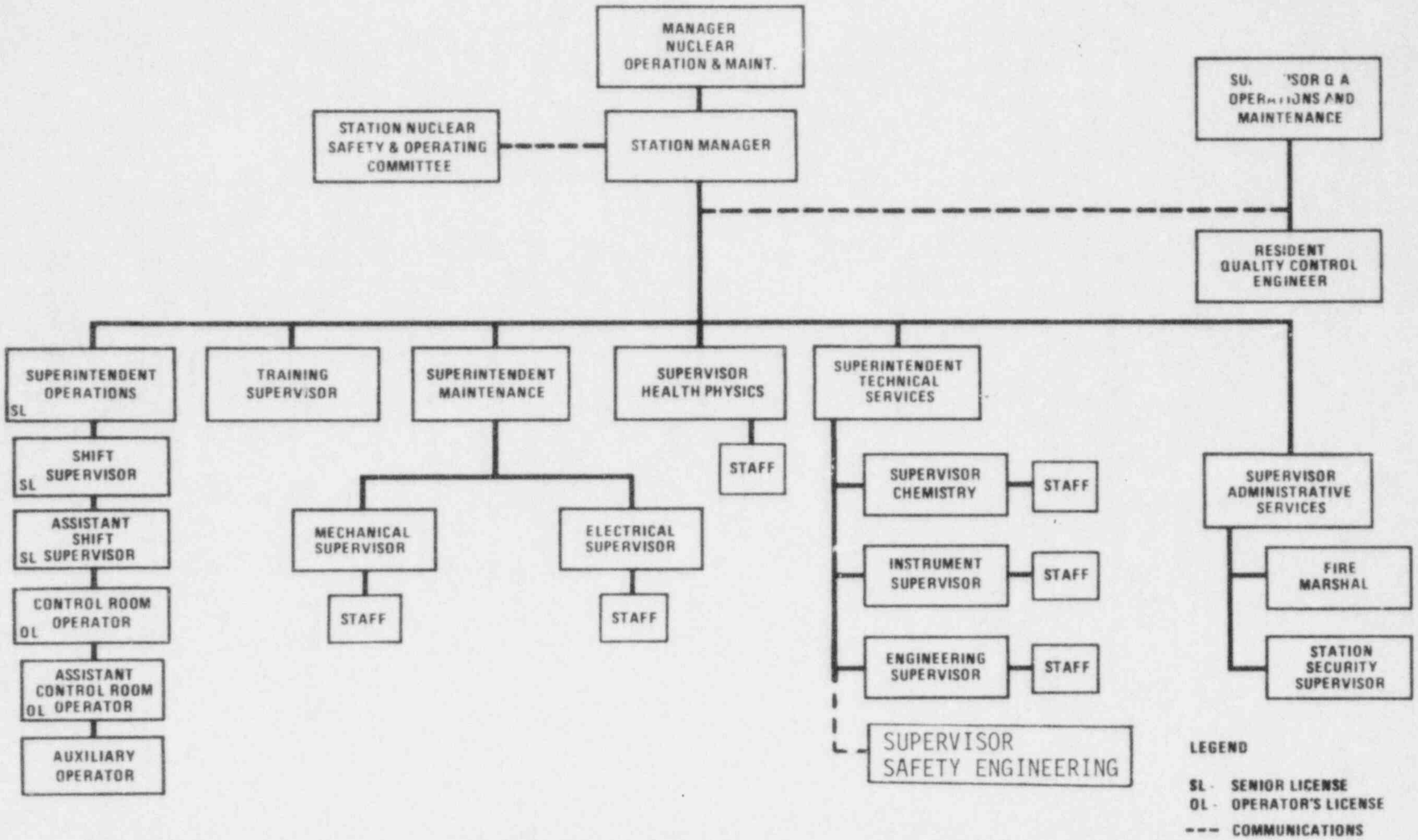


Figure 6.2-2 Facility Organization – North Anna – Units 1 and 2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

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

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

a/ Individual may fill the same position on Unit 1

b/ One of the two required individuals may fill the same position on Unit 1.

TABLE 6.2-1 (Continued)

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 2.
SRO - Individual with a Senior Reactor Operators License on Unit 2.
RO - Individual with a Reactor Operators License on Unit 2.
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command function.

Licensed operators shall:*

1. Not work more than 12 hours straight,
2. Not work more than 24 hours in any 48-hour period,
3. Not work more than 72 hours in any 7-day period,
4. Not work more than 14 consecutive days without having 2 consecutive days off.

*Deviation from these requirements may be authorized by the Station Manager in accordance with established procedures and with documentation of the cause. Overtime limits do not include shift turnover time.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in the March 28, 1980 NRC letter to all licenses, except for the Supervisor-Health Physics who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 The Station Manager is responsible for ensuring that retraining and replacement training programs for the facility staff are maintained and that such programs meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevent industry operational experience identified by the ISEG.

6.5 REVIEW AND AUDIT

6.5.1 STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC)

FUNCTION

6.5.1.1 The SNSOC shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SNSOC shall be composed of the:

Chairman:	Station Manager
Vice-Chairman:	Superintendent - Operations
Member:	Superintendent - Maintenance
Member:	Superintendent - Technical Services

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SNSOC Chairman to serve on a temporary basis; however, no more than one alternates shall participate as a voting member in SNSOC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SNSOC shall meet at least once per calendar month and as convened by the SNSOC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SNSOC shall consist of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8.1 and changes thereto, 2) all programs required by Specification 6.8.4 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director, Nuclear Operations and to the Chairman of the System Nuclear Safety and Operating Committee.

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

AUTHORITY

6.5.1.7 The SNSOC shall:

a. Recommend to the Station Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Manager, Nuclear Operations and Maintenance and the Chairman of the System Nuclear Safety and Operating Committee of disagreement between the SNSOC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Manager, Nuclear Operations and Maintenance and Chairman of the System Nuclear Safety and Operating Committee.

6.5.2 SYSTEM NUCLEAR SAFETY AND OPERATING COMMITTEE (SyNSOC)

FUNCTION

6.5.2.1 The SyNSOC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Quality assurance practices

i. Other appropriate fields associated with the unique characteristics of the nuclear power plant.

COMPOSITION

6.5.2.2 The SyNSOC shall be composed of the Chairman and four other members. Membership shall be composed of the Managers or Directors of the Power Station Engineering, Fuel Resources, Nuclear Operations, and Licensing and Quality Assurance Departments or qualified designees from these departments and a sixth qualified member selected by the five other members. Members of the SyNSOC shall have an academic degree in an engineering or physical science field and, in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in Section 6.5.2.1.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SyNSOC Chairman to serve on a temporary basis.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the SyNSOC Chairman to provide expert advice to the SyNSOC.

MEETING FREQUENCY

6.5.2.5 The SyNSOC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of the SyNSOC shall consist of not less than a majority of the members or duly appointed alternates and shall be subject to the following constraints:

1. The Chairman or Vice Chairman shall be present for all meetings.

2. No more than a minority of the quorum shall have line responsibility for operation of the stations.

3. A motion-carrying vote must consist of no less than three (3) votes.

4. No more than a minority of a quorum may be alternates.

REVIEW

6.5.2.7 The following subjects shall be reviewed by the SyNSOC:

a. Written safety evaluations of changes in the stations as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2) and is accomplished by review of minutes of the Station Nuclear Safety and Operating Committee and the design change program.

b. Proposed changes in procedures, proposed changes in the station, or proposed tests or experiments, any of which may involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Matters of this kind shall be referred to the SyNSOC by the Station Nuclear Safety and Operating Committee following its review prior to implementation.

c. Changes in the technical specifications or license amendments relating to nuclear safety prior to implementation except in those cases where the change is identical to a previously reviewed proposed change.

REVIEW (Cont'd)

d. Violations and reportable occurrences such as:

1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations from normal or expected performance of station safety-related structures, systems, or components; and

3. Reportable occurrences as defined in the station Technical Specifications.

Review of events covered under this paragraph shall include the results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Any other matter involving safe operation of the nuclear power stations which a duly appointed subcommittee or committee member deems appropriate for consideration, or which is referred to the SyNSOC by the Station Nuclear Safety and Operating Committee.

f. Reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

AUDITS

6.5.2.8 Audits of station activities shall be performed under the cognizance of the SyNSOC. These audits shall encompass:

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

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

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

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

e. The Station Emergency Plan and implementing procedures at least once per 24 months.

f. The Station Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the SyNSOC or the Executive Vice President-Power.

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

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUTHORITY

6.5.2.9 The SyNSOC shall report to and advise the Executive Manager - Licensing and Quality Assurance, who shall advise the Executive Vice President - Power on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8 and review and approve the meeting minutes of the SyNSOC.

RECORDS

6.5.2.10 Records of SyNSOC activities shall be prepared, maintained and disseminated as indicated below within 14 working days of each meeting or following completion of the review or audit.

1. Executive Vice President-Power
2. Nuclear Power Station Managers
3. Manager - Nuclear Operations and Maintenance
4. Members of the SyNSOC
5. Others that the Chairman of the SyNSOC may designate.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the SNSOC and submitted to the SyNSOC and the Manager - Nuclear Operations and Maintenance.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager, Nuclear Operations and Maintenance, and the SyNSOC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SyNSOC and the Manager - Nuclear Operations and Maintenance within 14 days of the violation.

6.8 PROCEDURES & PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

6.8.2 Each procedure of 6.8.1 above, and changes hereto, shall be reviewed by the SNSOC and approved by the Station Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all control point chemistry conditions,

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action, and

(vii) Monitoring of the condensate at the discharge of the condensate pumps for evidence of condenser inleakage. When condenser in-leakage is confirmed, the leak shall be repaired, plugged, or isolated within 96 hours.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

MONTHLY OPERATING REPORT

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

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

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

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

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

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

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

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

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

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

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

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

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of the licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation, or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in item 6.9.1.8(c) above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement, Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Each REPORTABLE OCCURRENCE submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material release to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

g. Records of reactor tests and experiments

h. Records of training and qualification for current members of the plant staff.

i. Records of in-service inspections performed pursuant to these Technical Specifications.

j. Records of Quality Assurance activities required by the QA Manual.

k. Records of the service lives of all hydraulic snubbers listed on Table 3.7-4a including the date at which the service life commences. (This requirement commences with the Full Power License.)

l. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

m. Records of meetings of the SNSOC and the SyNSOC.

n. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation mointoring device which continously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in the protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

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*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provide they comply with approved radiation protection procedures for entry in high radiation areas.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.