NUREG-0932

Technical Specifications

Virgil C. Summer Nuclear Station, Unit No. 1

Docket No. 50-395

Appendix "A" to
License No. NPF-12

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

August 1982

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ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (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.

MASTER RELAY TEST

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

1.20 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.

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE OCCURRENCE

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

SHUTDOWN MARGIN

1.28 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.

SLAVE RELAY TEST

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.32 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.

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

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

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 and 2.1-2 for 3 and 2 loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

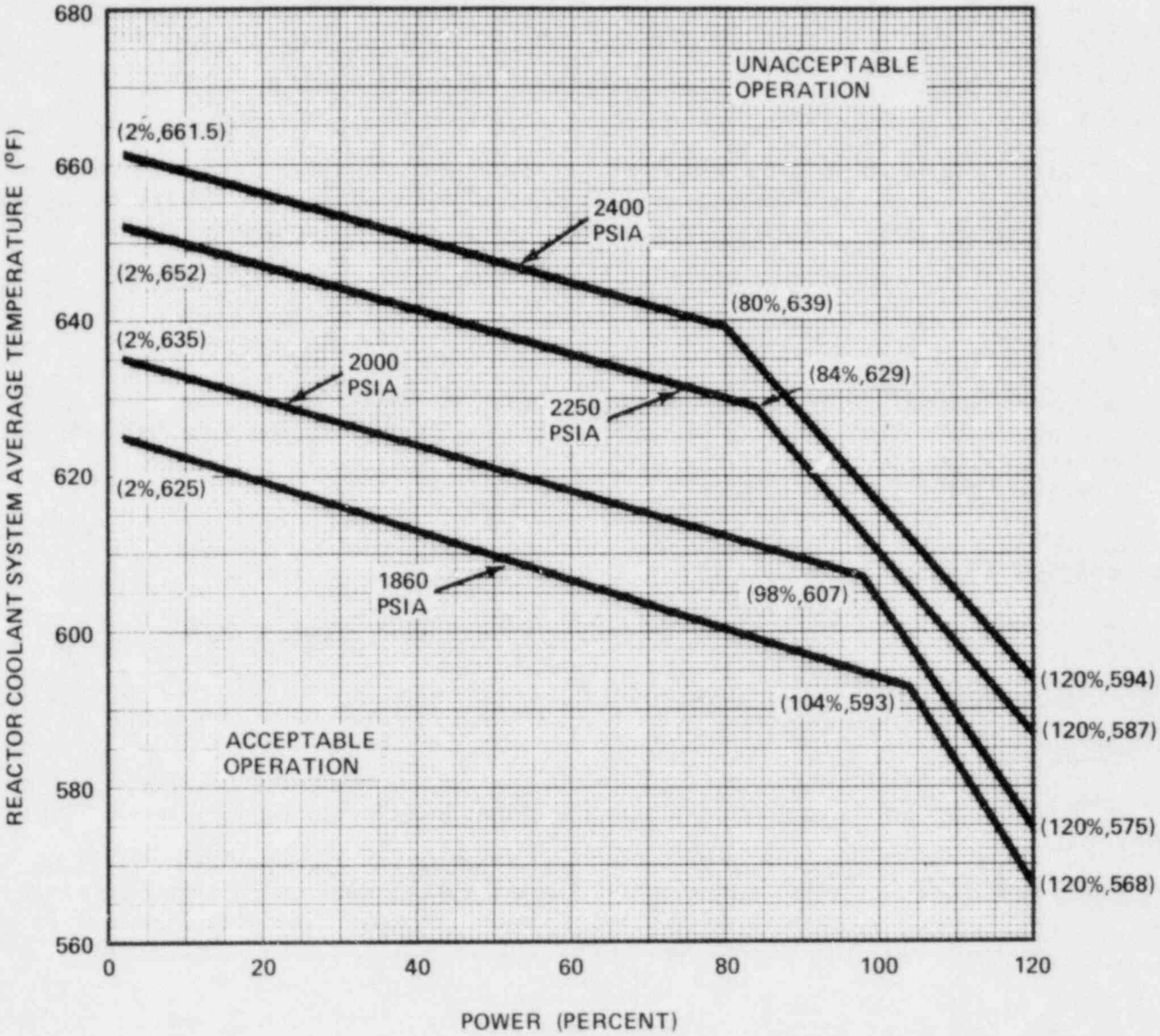

Figure 2.1-1
Reactor Core Safety Limit - Three Loops in Operation

Figure 2.1-2 left blank pending NRC approval of two-loop operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

a. With a reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2-1 adjust the setpoint consistent with the Trip Setpoint value.

b. With the reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, place the channel in the tripped condition within 1 hour, and within the following 12 hours either:

   1. Determine that Equation 2.2-1 was satisfied for the affected channel and adjust the setpoint consistent with the Trip Setpoint value of Table 2.2-1, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

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

where:

Z = the value for column Z of Table 2.2-1 for the affected channel,

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

S = either the "as measured" value (in percent span) of the sensor error, or the value is column S of Table 2.2-1 for the affected channel, and

TA = the value from column TA of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	Not Applicable	NA	NA	NA	NA
2. Power Range, Neutron Flux High Setpoint	7.5	4.56	0	≤109% of RTP	≤111.2% of RTP
Low Setpoint	8.3	4.56	0	≤25% of RTP	≤27.2% of RTP
3. Power Range, Neutron Flux High Positive Rate	1.6	0.5	0	≤5% of RTP with a time constant ≥2 seconds	≤6.3% of RTP with a time constant ≥2 seconds
4. Power Range, Neutron Flux High Negative Rate	1.6	0.5	0	≤5% of RTP with a time constant ≥2 seconds	≤6.3% of RTP with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP	≤31% of RTP
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.1	2.94	1.8	See note 1	See note 2
8. Overpower ΔT	4.5	1.4	1.2	See note 3	See note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.5	≥1870 psig	≥1859 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2380 psig	≤2391 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Loss of Flow	2.5	1.0	1.5	≥90% of loop design flow*	≥89.2% of loop design flow*

*Loop design flow = 98,000 gpm
RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
13. Steam Generator Water Level Low-Low	12.0	9.18	1.5	≥12% of span from 0 to 30% RTP increasing linearly to ≥54.9% of span from 30% to 100% RTP	≥10.2% of span from 0 to 30% RTP increasing linearly to ≥53.1% of span from 30% to 100% RTP
14. Steam/Feedwater Flow Mis-Match Coincident With	16.0	13.24	1.5/ 1.5	≤40% of full steam flow at RTP	≤42.5% of full steam flow at RTP
Steam Generator Water Level Low-Low	12.0	9.18	1.5	≥12% of span from 0 to 30% RTP increasing linearly to ≥54.9% of span from 30% to 100% RTP	≥10.2% of span from 0 to 30% RTP increasing linearly to ≥53.1% of span from 30% to 100% RTP
15. Undervoltage - Reactor Coolant Pump	2.1	1.28	0.23	≥4830 volts	≥4760
16. Underfrequency - Reactor Coolant Pumps	7.5	0	0.1	≥57.5 Hz	≥57.1 Hz
17. Turbine Trip					
A. Low Trip System Pressure	NA	NA	NA	≥800 psig	≥750 psig
B. Turbine Stop Valve Closure	NA	NA	NA	≥1% open	≥1% open

RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
18. Safety Injection Input from ESF	NA	NA	NA	NA	NA
19. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	NA	NA	NA	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
B. Low Power Reactor Trips Block, P-7					
a. P-10 input	7.5	4.56	0	$\leq$10% of RTP	$\leq$12.2% of RTP
b. P-13 input	7.5	4.56	0	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% of turbine impulse pressure equivalent
C. Power Range Neutron Flux P-8	7.5	4.56	0	$\leq$38% of RTP	$\leq$40.2% of RTP
D. Low Setpoint Power Range Neutron Flux, P-10	7.5	4.56	0	$\geq$10% of RTP	$\geq$7.8% of RTP
E. Turbine Impulse Chamber Pressure, P-13	7.5	4.56	0	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% turbine pressure equivalent
20. Reactor Trip Breakers	NA	NA	NA	NA	NA
21. Automatic Actuation Logic	NA	NA	NA	NA	NA

RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT
τ_1, τ_2	=	Time constants utilized in lead-lag controller for ΔT, τ_1 = 8 sec., τ_2 = 3 sec.
$\frac{1}{1 + \tau_3}$	=	Lag compensator on measured ΔT
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 secs.
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
K_1	=	1.090
K_2	=	0.01450
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_4, & τ_5	=	Time constants utilized in the lead-lag controller for T_{avg}, τ_4 = 33 secs., τ_5 = 4 secs.
T	=	Average temperature °F
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg}
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 secs.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

T'	$\leq$	587.4°F Reference T_{avg} at RATED THERMAL POWER
K_3	=	.0006728
P	=	Pressurizer pressure, psig
P'	=	2235 psig, Nominal RCS operating pressure
S	=	Laplace transform operator, sec^{-1}.

and $f_1(\Delta I)$ 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 + 8 percent $f_1(\Delta I) = 0$ 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 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +8 percent, the ΔT trip setpoint shall be automatically reduced by 1.11 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 3.6 percent ΔT span.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:	ΔT	=	as defined in Note 1
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	as defined in Note 1
	τ_1, τ_2	=	as defined in Note 1
	$\frac{1}{1 + \tau_3 S}$	=	as defined in Note 1
	τ_3	=	as defined in Note 1
	ΔT_o	=	as defined in Note 1
	K_4	-	1.091
	K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature
	$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation
	τ_7	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 secs.
	$\frac{1}{1 + \tau_6 S}$	=	as defined in Note 1
	τ_6	=	as defined in Note 1

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3 (continued)

K_6	=	0.001190/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	$\leq$	587.4°F Reference T_{avg} at RATED THERMAL POWER
S	=	as defined in Note 1
$f_2(\Delta I)$	=	0 for all ΔI

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.7 percent ΔT Span.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

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

These curves are based on 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_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System 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 Plants, 1971 Edition which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are also designed to Section III of the ASME Code for Nuclear Power Plants, 1971 Edition 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.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

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

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

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various reactor trip circuits automatica -, open the reactor trip breakers whenever a condition monitored by the Reactor Protection System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Protection System which monitors numerous system variables, therefore, providing protection system functional diversity. The Reactor Protection System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive reactor system cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Protection System includes manual reactor trip capability.

Power Range, Neutron Flux

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

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

Power Range, Neutron Flux, High Rates

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

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

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an

Intermediate and Source Range, Nuclear Flux (Continued)

uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a reactor trip at a current level equivalent to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

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 Pressurizer high and low pressure trips. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, 2) pressurizer pressure, and 3) axial power distribution. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower delta T trip provides assurance of fuel integrity (e.g., no fuel melting and less than 1 percent cladding strain) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and 2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

Pressurizer Pressure

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

Pressurizer Pressure (Continued)

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

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

Pressurizer Water Level

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

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent of full power equivalent), an automatic reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 38 percent of RATED THERMAL POWER) an automatic reactor trip will occur if the flow in any single loop drops below 90 percent of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic reactor trip will occur on loss of flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The steam generator water level low-low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The steam/feedwater flow mismatch in coincidence with a steam generator 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 or equal to 1.63×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below the programmed low

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level (Continued)

level setpoint, 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 Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide reactor core protection against DNB as a result of complete loss of forced coolant flow. 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 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.6 seconds. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power the turbine trip is automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. On decreasing power, Source Range level trips are automatically reactivated and high voltage restored.

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

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

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. On decreasing power the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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 interval, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

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

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

This specification is not applicable in MODES 5 and 6.

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

APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

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

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:

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
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 - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% delta k/k for 3 loop operation.

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or 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 verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, 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.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Surveillance Requirement 4.1.1.1.2 with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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 - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

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

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

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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

b. Less negative than -4.2×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

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

ACTION:

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

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

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

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 3.1.1.3.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

SURVEILLANCE REQUIREMENTS

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

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3.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 -3.3×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.3×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.3.b, at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.0.
*See Special Test Exception 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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 or a gravity feed connection 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 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 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 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

#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 300°F.

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 and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, a differential pressure across the pump of greater than or equal to 2472 psig is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable, at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, a differential pressure across each pump of greater than or equal to 2472 psig is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable, at least once per 31 days, whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers have been secured in the open 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 300°F.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 37,900 gallons,

2. A minimum boron concentration of 2000 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

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

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 13,200 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 453,800 gallons,
2. Between 2000 and 2100 ppm of boron, and
3. A minimum solution temperature of 40°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 2 percent 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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

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

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

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

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

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

c. With one full length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 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 remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of 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, or

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

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

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and 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 shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system at least once per 4 hours.

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of an ANALOG CHANNEL OPERATIONAL 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 2.3 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 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 bank position using the above figures, or

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions 3.10.2 and 3.10.3
#With K_{eff} greater than or equal to 1.0.

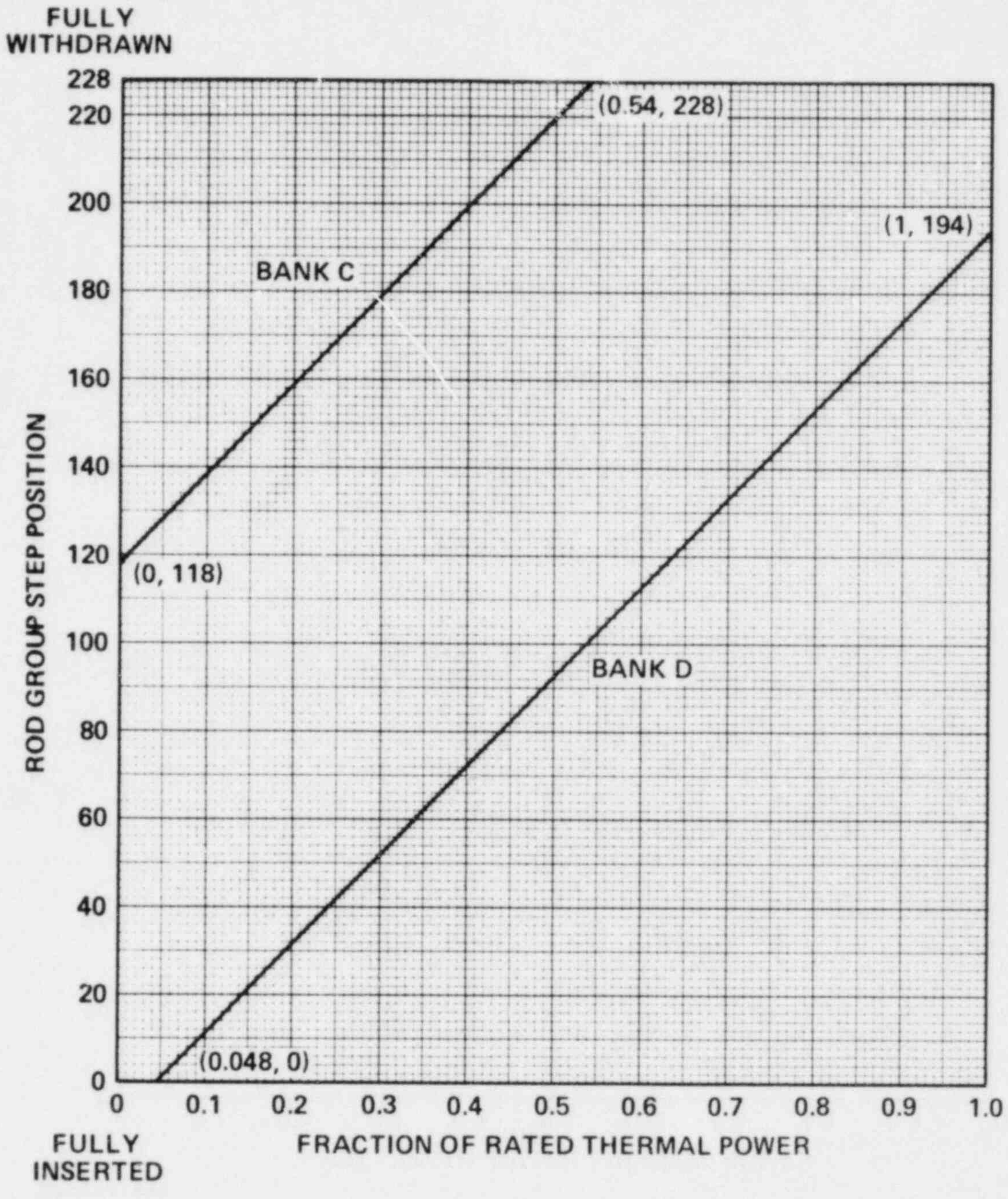

FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
THREE LOOP OPERATION

Figure 3.1-2 left blank pending NRC approval
of two-loop operation

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 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% of 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 90% of RATED THERMAL POWER, within 15 minutes either:

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

      b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

   2. Between 50% and 90% 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 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ±5% target band and ACTION a.2.a) 1), above has been satisfied.

*See Special Test Exception 3.10.2

ACTION (Continued)

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. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

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 2 or more 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.

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.

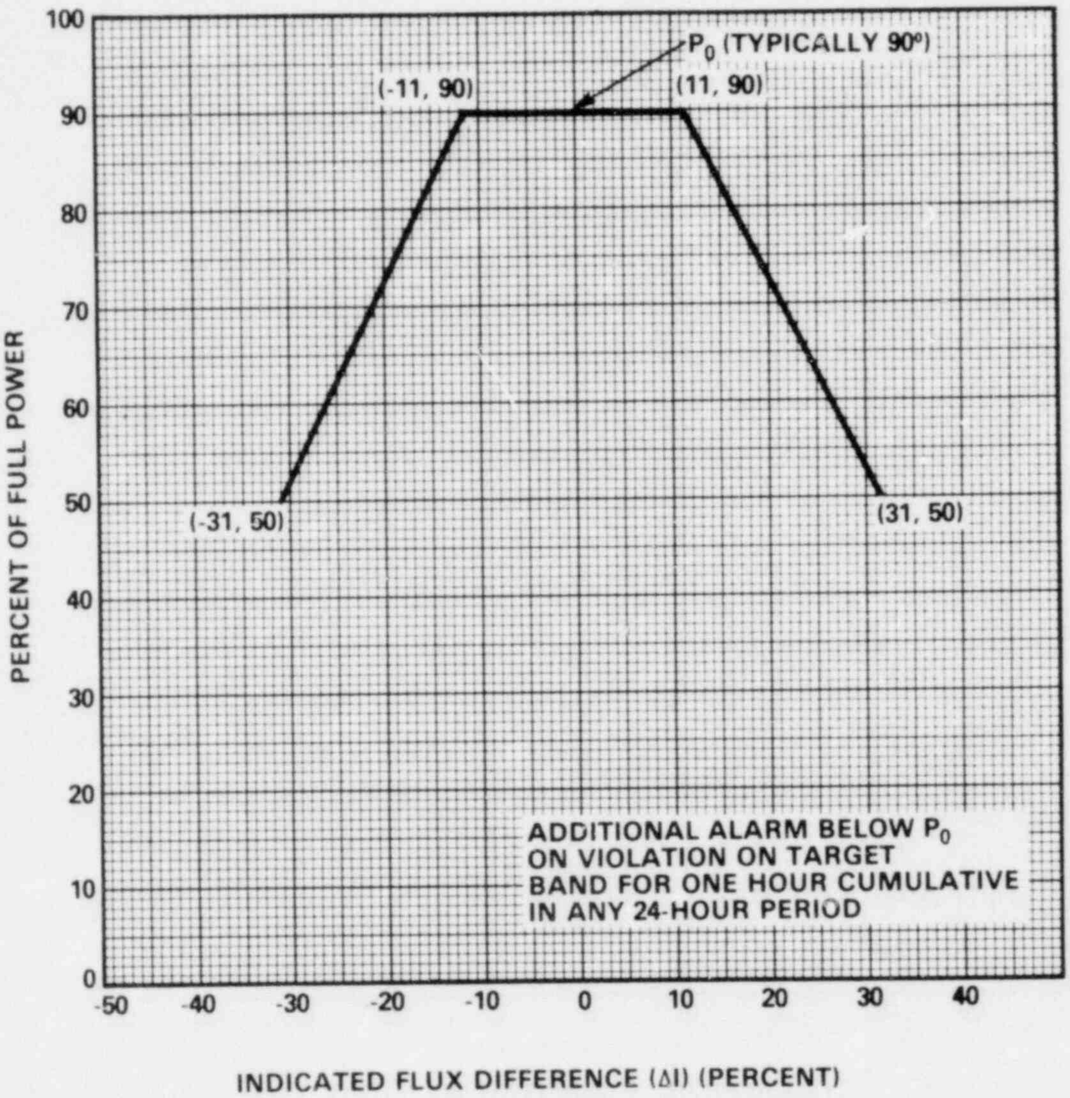

Figure 3.2-1 Axial Flux Difference Limits as a Function of Rated Thermal Power

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

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.14.

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

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

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

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive. (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} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

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

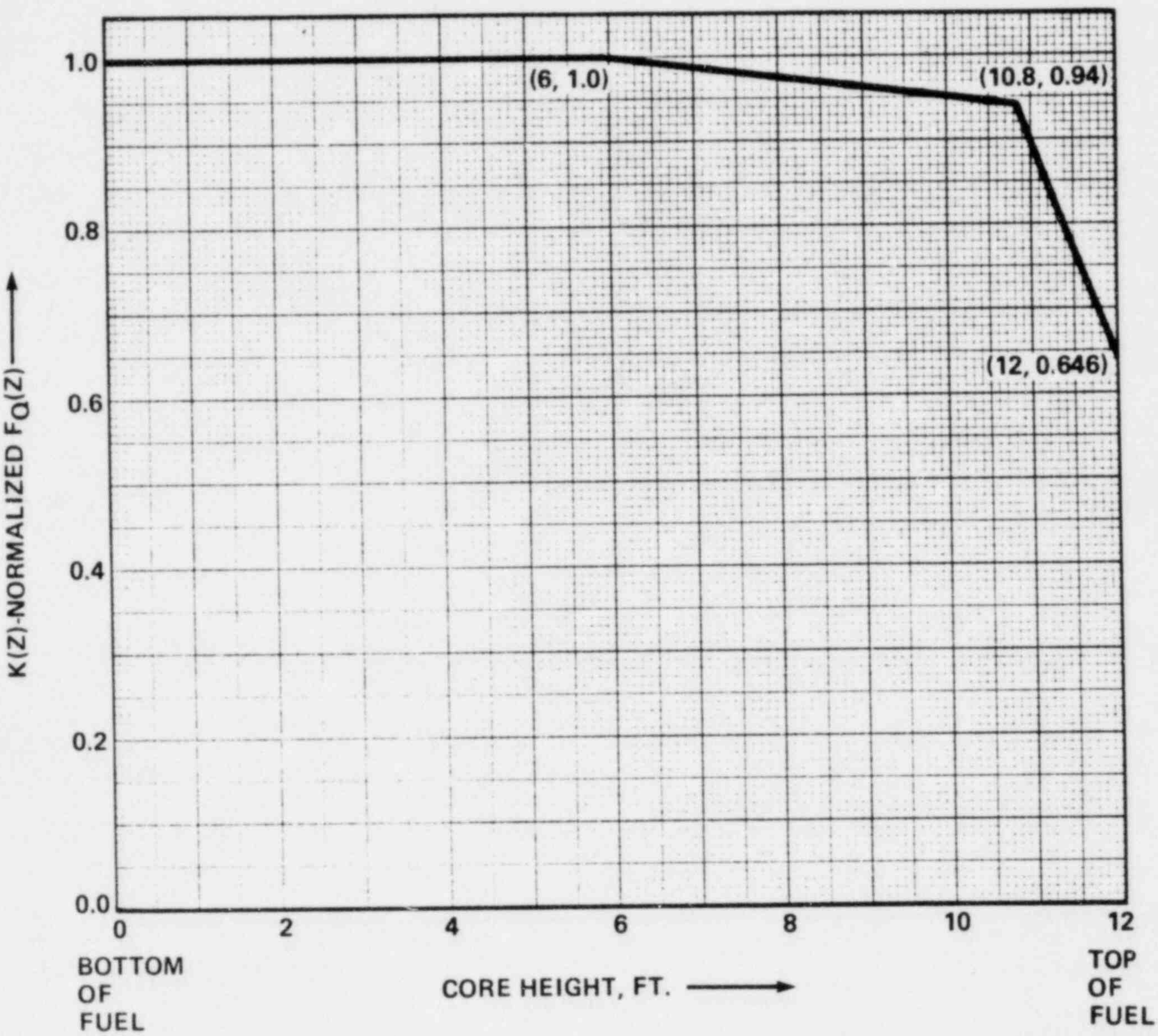

FIGURE 3.2-2

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

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

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R_1, R_2 shall be maintained within the region of allowable operation shown on Figure 3.2-3 for 3 loop operation.

Where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $R_2 = \dfrac{R_1}{[1\text{-}RBP(BU)]}$,

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

d. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$, and

e. RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core).

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

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

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3:

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

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

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

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

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

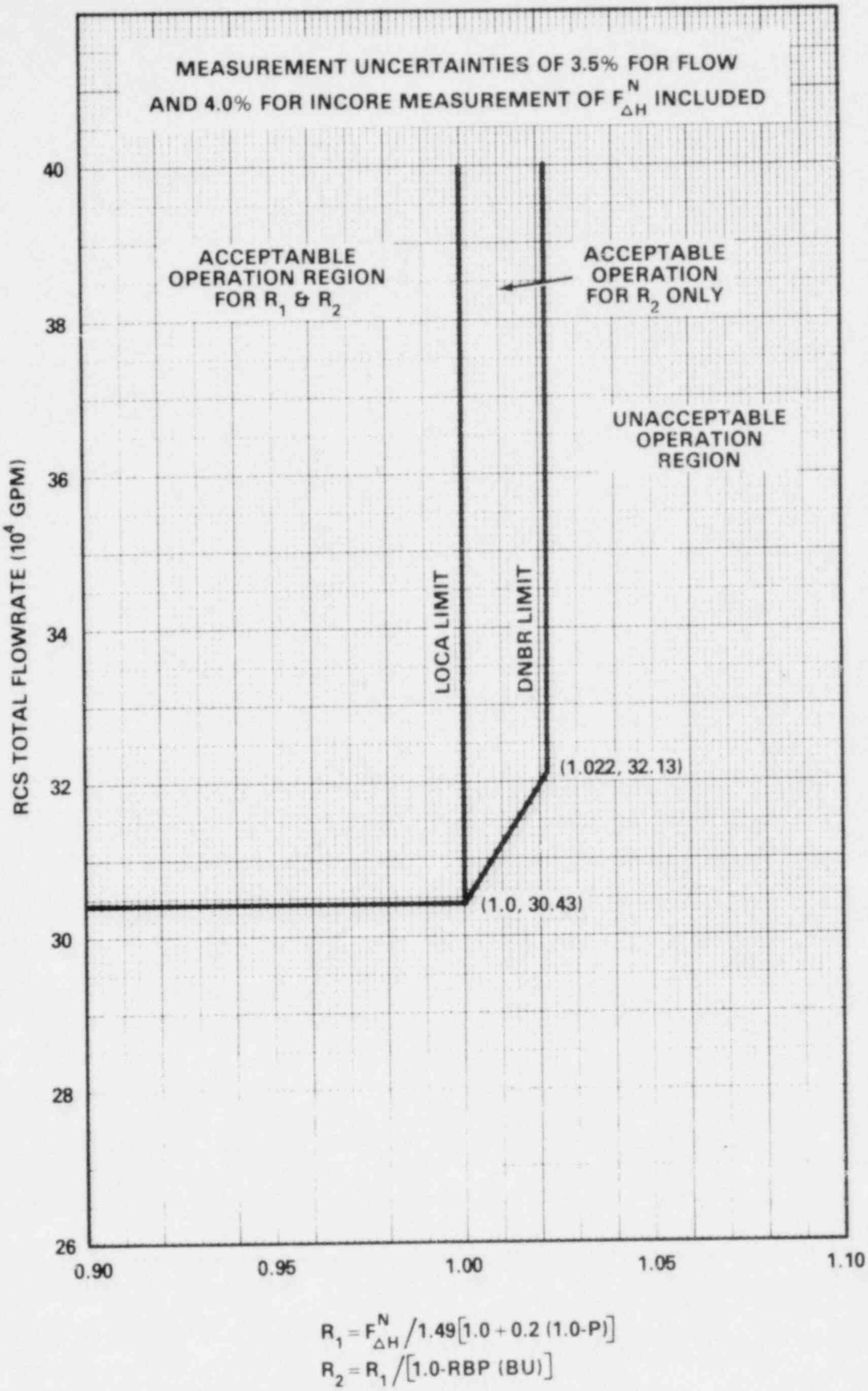

Figure 3.2-3 RCS FLOW RATE VERSUS R

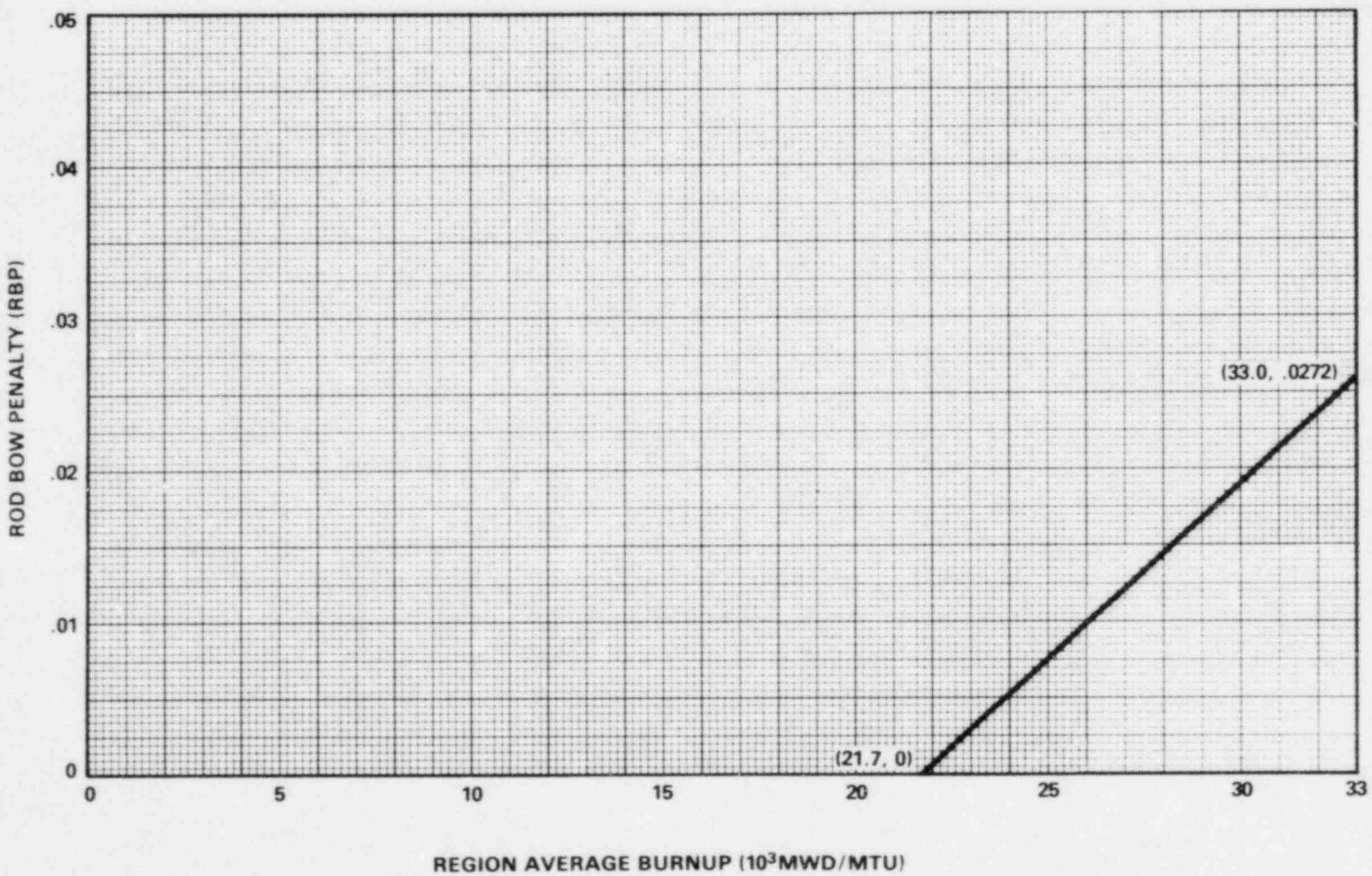

Figure 3.2-4 Rod Bow Penalty as a Function of Burnup

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

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

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

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

   2. Within 2 hours either:

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

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 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 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.

*See Special Test Exception 3.10.2.

ACTION: (Continued)

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

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

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

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

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, 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.

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

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

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

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

ACTION: (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days when the 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 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops In Operation	 2 Loops in Operation
Reactor Coolant System T_{avg}	$\leq$ 592°F	**
Pressurizer Pressure	$\geq$ 2220 psia*	**

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

**These values left blank pending NRC approval of two-loop operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	9
2. Power Range, Neutron Flux					
A. High Setpoint	4	2	3	1, 2	$2^{\#}$
B. Low Setpoint	4	2	3	$1^{\#\#\#}$, 2	$2^{\#}$
3. 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	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	$2^{\#\#}$	4
B. Shutdown	2	0	1	3, 4 and 5	5
C. Shutdown	2	1	2	3*, 4*, 5*	9
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	$6^{\#}$
Two Loop Operation	****	****	****	****	****
8. Overpower ΔT					
Three Loop Operation	3	2	2	1, 2	$6^{\#}$
Two Loop Operation	****	****	****	****	****
9. Pressurizer Pressure-Low	3	2	2	1	$6^{\#}$
10. Pressurizer Pressure--High	3	2	2	1, 2	$6^{\#}$

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	6[#]
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6[#]
B. 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	6[#]
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6[#]
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in each loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	6[#]

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
15. Undervoltage-Reactor Coolant Pumps	3-1/bus	2	2	1	6#
16. Underfrequency-Reactor Coolant Pumps	3-1/bus	2	2	1	6#
17. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	6#
B. Turbine Stop Valve Closure	4	4	1	1	10#
18. Safety Injection Input from ESF	2	1	2	1, 2	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6		2	1	2	2#	7
B. Low Power Reactor Trips Block, P-7	P-10 Input	4	2	3	1	7
	P-13 Input	2	1	2	1	7
C. Power Range Neutron Flux, P-8		4	2	3	1	7
D. Power Range Neutron Flux, P-10		4	2	3	1, 2	7
E. Turbine First Stage Pressure, P-13		2	1	2	1	7
20. Reactor Trip Breakers		2	1	2	1, 2	8
		2	1	2	3*, 4*, 5*	9
21. Automatic Trip Logic		2	1	2	1, 2	8
		2	1	2	3*, 4*, 5*	9

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.

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

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

****Values left blank pending NRC approval of 2 loop operation.

ACTION STATEMENTS

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

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

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

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

ACTION 10- With the number of OPERABLE Channels less than the Total Number of Channels operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

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 time of the neutron flux signal portion of the channel 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. A. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
13. Steam Generator Water Level--Low-Low	$\leq$ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.5 seconds
16. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.9 seconds
17. Turbine Trip	
A. Low Fluid Oil Pressure	Not Applicable
B. Turbine Stop Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Trip System Interlocks	Not Applicable
20. Reactor Trip Breakers	Not Applicable
21. Automatic Trip Logic	Not Applicable

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Steam Generator Water Level - Low Coincident with Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A	1, 2
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
B. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M (8)	N.A.	N.A.	1
C. Power Range Neutron Flux, P-8	N.A.	R(4)	M (8)	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M (8)	N.A.	N.A.	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M (8)	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 11)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

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

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

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

(3) - Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

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

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

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

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

(8) - With power greater than or equal to the interlock setpoint the required OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) - Monthly Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not required.

(11) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include an independent verification of the undervoltage and shunt trips.

LIMITING CONDITION FOR OPERATION

3.3.2 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 or interlock setpoint trip less conservative than the value shown in the Trip Setpoint column of Table 3.3-4 adjust the setpoint consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, place the channel in the tripped condition within 1 hour, and within the following 12 hours either:

   1. Determine that Equation 2.2-1 was satisfied for the affected channel and adjust the setpoint consistent with the Trip Setpoint value of Table 3.3-4, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

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

   where:

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

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

   S = either the "as measured" value (in percent span) of the sensor error, or the value in column S of Table 3.3-4 for the affected channel, and

   TA = the value from column TA of Table 3.3-4 for the affected channel.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure-High	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	15*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line	1, 2, 3	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 Line Pressure-Low	1 pressure/ loop	1 pressure and 2 loops	1 pressure and 2 loops	1, 2, 3##	19*
2. REACTOR BUILDING SPRAY					
a. Manual	2 sets - 2 switches/set	1 set	2 sets	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure--High-3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	4	2	3	1, 2, 3	16

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
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Phase "B" Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
2) Reactor Building Pressure--High-3	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
2) Containment Radio-activity- High	4	2	3	1, 2, 3, 4	17
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual					
i. One Switch/line	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
ii. One Switch/all lines	1	1	1	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	19
c. Reactor Building Pressure--High-2	3	2	2	1, 2, 3	15*
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3	15*
COINCIDENT WITH T_{avg}--Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops	1, 2, 3	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
e. Steam Line Pressure-Low	1 pressure/loop	1 pressure any 2 loops	1 pressure any 2 loops	1, 2, 3##	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	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
6. EMERGENCY FEEDWATER					
a. Manual Initiation	1 per pump	1 per pump	1 per pump	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
i. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any stm gen.	2/stm. gen.	1, 2, 3	15*
ii. Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen	1, 2, 3	15*
d. Undervoltage-both ESF Busses Start Turbine-Driven Pump	2-1/bus	2	2	1, 2, 3	19
e. S.I. Start Motor-Driven Pumps	See 1 above (all S.I. initiating functions and requirements)				
f. Undervoltage-one ESF bus Start Motor-Driven Pumps	2-1/bus	1	2	1, 2	22
g. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	3-1/pump	3-1/pump	3-1/pump	1, 2	19
h. Suction Transfer on Low Pressure	4	2	3	1, 2, 3	19

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
7. LOSS OF POWER					
a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP					
a. RWST level low-low	4	2	3	1, 2, 3	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	19
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) setpoint.

##Trip function may be blocked in this MODE below the P-12 (Low-Low Tavg Interlock) setpoint.

*The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

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

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

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 bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 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, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	NA	NA	NA	NA	NA
b. Automatic Actuation Logic	NA	NA	NA	NA	NA
c. Reactor Building Pressure-High 1	3.0	0.71	1.5	$\leq$3.6 psig	$\leq$3.86 psig
d. Pressurizer Pressure--Low	13.1	10.71	1.5	$\geq$1850 psig	$\geq$1839 psig
e. Differential Pressure Between Steamlines--High	3.0	0.87	1.5/ 1.5	$\leq$97 psig	$\leq$106 psi
f. Steamline Pressure--Low	20.0	10.71	1.5	$\geq$675 psig	$\geq$635 psig(1)
2. REACTOR BUILDING SPRAY					
a. Manual Initiation	NA	NA	NA	NA	NA
b. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
c. Reactor Building Pressure-High 3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	3.0	0.71	1.5	$\leq$12.05 psig	$\leq$12.31 psig

(1) Time constants utilized in lead lag controller for steamline pressure-low are as follows
$\tau_1 \geq$ 50 secs. $\tau_2 \leq$ 5 secs.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1. Manual	NA	NA	NA	NA	NA
2. Safety Injection	See 1 above for all safety injection setpoints and allowable values				
3. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
b. Phase "B" Isolation					
1. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
2. Reactor Building Pressure-High 3	3.0	0.71	1.5	$\leq$12.05 psig	$\leq$12.31 psig
c. Purge and Exhaust Isolation					
1. Safety Injection	See 1 above for all safety injection setpoints and allowable values				
2. Containment Radioactivity High	NA	NA	NA	2X Background	2X Background
3. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
4. STEAM LINE ISOLATION					
a. Manual	NA	NA	NA	NA	NA
b. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA
c. Reactor Building Pressure-High 2	3.0	0.71	1.5	$\leq$6.35	$\leq$6.61
d. Steam Flow in Two Steamlines-High, Coincident with	20.0	13.16	1.5/ 1.5	$\leq$ 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 114.0% of full steam flow at full load.
T_{avg} - Low-Low	4.0	1.12	1.2	$\geq$553°F	$\geq$550.6°F
e. Steamline Pressure - Low	20.0	10.71	1.5	$\geq$675 psig	$\geq$635 psig(1)

(1) Time constants utilized in lead lag controller for steamline pressure low are as follows: $\tau_1 \geq 50$ secs. $\tau_2 \leq 5$ secs.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
5. TURBINE TRIP AND FEEDWATER ISOLATION					
a. Steam Generator Water Level - High-High	5.0	2.18	1.5	≤82.4% of narrow range instrument span	≤84.2% of narrow range instrument span
6. EMERGENCY FEEDWATER					
a. Manual	NA	NA	NA	NA	NA
b. Automatic Actuation Logic	NA	NA	NA	NA	NA
c. Steam Generator Water Level - Low-Low	12.0	9.18	1.5	≥12% of span from 0% to 30% RTP increasing linearly to ≥54.9 of span from 30% to 100% RTP	≥10.2% of span from 0% and 30% RTP increasing linearly to ≥53.1 of span from 30% to 100% RTP
d & f. Undervoltage - ESF Bus				≥5760 volts with a ≤0.25 second time delay	≥5652 volts with a ≤0.275 second time delay
				≥6576 volts with a ≤3.0 second time delay	≥6511 volts with a ≤3.3 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
e. Safety Injection	See 1 above (all SI Setpoints)				
g. Trips of Main Feedwater Pumps	NA	NA	NA	NA	NA
h. Suction transfer on Low Pressure	NA	NA	NA	$\geq$442 ft. 4in.(2)	$\geq$441 ft. 3 in.
7. LOSS OF POWER					
a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	NA	NA	$\geq$5760 volts with a $\leq$0.25 second time delay	$\geq$5652 volts with a $\leq$0.275 second time delay
b. 7.2 kv Emergency Bus Undervoltage	NA	NA	NA	$\geq$6576 volts with a $\leq$3.0 second time delay	$\geq$6511 volts with a $\leq$3.3 second time delay
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP					
a. RWST Level Low-Low	NA	NA	NA	$\geq$18%	$\geq$15%
b. Automatic Actuation Logic and Actuation Relays	NA	NA	NA	NA	NA

(2) Pump suction head at which transfer is initiated is stated in effective water elevation in the condensate storage tank.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	S	Trip Setpoint	Allowable Value
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS					
INTERLOCKS					
a. Pressurizer Pressure, P-11	3.1	.71	1.5	1985 psig	≥1974 psig & ≤1996 psig
b. T_{avg} Low-Low, P-12	4.0	1.12	1.2	553°F	≥550.6°F & ≤555.4°F
c. Reactor Trip, P-4	NA	NA	NA	NA	NA

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
b. Reactor Building Spray	Not Applicable
c. Containment Isolation	
Phase "A" Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
e. Feedwater Isolation	Not Applicable
f. Emergency Feedwater	Not Applicable
g. Essential Service Water	Not Applicable
h. Reactor Building Cooling Fans	Not Applicable
i. Control Room Isolation	Not Applicable
2. Reactor Building Pressure-High	
a. Safety Injection (ECCS)	$\leq 12^{(2)}/27^{(1)}$
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
e. Reactor Building Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 45.0^{(4)}/55.0^{(5)}$
h. Reactor Building Cooling Units	$\leq 33.0^{(4)}/43.0^{(5)}$
i. Control Room Isolation	Not Applicable
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$
e. Reactor Building Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 45.0^{(4)}/55.0^{(5)}$
h. Reactor Building Cooling Units	$\leq 33.0^{(4)}/43.0^{(5)}$
i. Control Room Isolation	Not Applicable
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
e. Reactor Building Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 45.0^{(4)}/55.0^{(5)}$
h. Reactor Building Cooling Units	$\leq 33.0^{(4)}/43.0^{(5)}$
i. Control Room Isolation	Not Applicable
5. Steam Line Pressure-Low	
a. Safety Injection - ECCS	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation - Phase "A"	$\leq 45.0^{(4)}/55.0^{(5)}$
e. Reactor Building and Purge and Exhaust Isolation	Not Applicable
f. Emergency Feedwater Pumps	Not Applicable
g. Service Water System	$\leq 45.0^{(4)}/55.0^{(5)}$
h. Reactor Building Cooling Units	$\leq 33.0^{(4)}/43.0^{(5)}$
i. Steam Line Isolation	≤ 10.0
j. Control Room Isolation	Not Applicable
6. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Steam Line Isolation	≤ 12.0
7. Reactor Building Pressure-High-2	
a. Steam Line Isolation	≤ 9.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Reactor Building Pressure--High-3	
a. Reactor Building Spray	$\leq$ 42.0(4)/52.0(5)
b. Containment Isolation-Phase "B"	Not Applicable
9. Steam Generator Water Level--High-High	
a. Turbine Trip	Not Applicable
b. Feedwater Isolation	$\leq$ 13.0
10. Steam Generator Water Level - Low-Low	
a. Motor-driven Emergency Feedwater Pumps	$\leq$ 60.0
b. Turbine-driven Emergency Feedwater Pumps	$\leq$ 60.0
11. Undervoltage - Both ESF Busses	
a. Turbine-driven Emergency Feedwater Pumps	$\leq$ 60.0
12. Undervoltage-one ESF Bus	
a. Motor-driven Emergency Feedwater Pumps	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

13. Trip of Main Feedwater Pumps	
a. Motor-driven Emergency Feedwater Pumps	Not Applicable
14. Loss of Power	
a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$ 10.3
b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$ 13.3
15. Containment Radioactivity - High	
a. Purge and Exhaust Isolation	Not Applicable
16. RWST level low-low	
a. Automatic Switchover to Containment Sump	Not Applicable
17. AUX FEED SUCTION PRESSURE LOW	
a. Suction transfer	Not Applicable
Note: Response time for Motor-driven Emergency Feedwater Pumps on all S.I. signal starts	$\leq$ 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

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

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

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

(4) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available.

(5) Diesel generator starting and sequence loading delays from undervoltage included.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP FEEDWATER ISOLATION, CONTROL ROOM ISOLATION START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure-High	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection	See 1 above for all Safety Injection Surveillance Requirements							
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. Phase "B" Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
2) Reactor Building Pressure--High-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
2) Containment Radio-activity-High	S	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
3) Safety Injection	See 1 above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	NA.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Reactor Building Pressure--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
Coincident With T_{avg}--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
AUXILIARY FEEDWATER (Continued)								
d. Undervoltage - ESF	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See 1 above for all Safety Injection Surveillance Requirements							
f. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
g. Suction transfer on low pressure	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	R.	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low, Low T_{avg}, P-12	N.A.	R.	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

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 ANALOG CHANNEL OPERATIONAL TEST operations for 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. Spent Fuel Pool Area (RM-G8)	1	*	$\leq$ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	$\leq$ 1 R/hr	$1 - 10^{5}$ mr/hr	28
c. Reactor Building Area					
i. High Range RM-G7 and High Range RM-G18	2	1, 2, 3 & 4	N/A	$10 - 10^{7}$ R/hr $1 - 10^{7}$ R/hr	30
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	$\leq 1 \times 10^{-5}$ μCi/cc (Kr-85)	$10 - 10^{6}$ cpm	27
ii. Particulate Activity	1	**	N/A	$10 - 10^{6}$ cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	$\leq$ 2 x background***	$10 - 10^{6}$ cpm	28
ii. Particulate Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	$10 - 10^{6}$ cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	$\leq$ 2 x background	$10 - 10^{6}$ cpm	29

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
d. Noble Gas Effluent Monitors (High Range)					
i. Main Plant Vent (RM-A13)	1	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30
ii. Main Steam Line (RM-G19A, B, C	1/steam line	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	1	1, 2, 3 & 4	N/A	0.1 - 10^7 mR/hr	30

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - 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 26 - 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 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.11.

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.8.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the emergency mode of operation.

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE Status within 72 hours, or:

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

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

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	S	R	M	6
c. Reactor Building Area				
i. High Range (RM-G7)	S	R***	M	1, 2, 3 & 4
ii. High Range (RM-G18)	S	R***	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	S	R	M	6
ii. Particulate Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

*With fuel in the storage pool or building

**With irradiated fuel in the storage pool

***Channel Calibration will consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

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

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

,,PLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO using a full-core flux map per Specification 4.2.4.2, or

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE at least once per 24 hours, by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

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

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs System, including the following components:		
a. Reactor Building Foundation Mat Accelerometer	0.1 to 40 Hz 0.01 to 1.0g	1
b. Reactor Building Ring Girder Accelerometer	0.1 to 40 Hz 0.01 to 1.0g	1
c. Reactor Building Foundation Mat Trigger	1 to 10 Hz 0.005 to 0.02g	1*
2. Triaxial Peak Accelographs		
a. Side of Steam Generator	0-32 Hz -5g to +5g	1
b. Pressurizer Surge Line	0-32 Hz -5 to +5g	1
c. RHR System Heat Exchanger	0-20 Hz -2g to +2g	1
3. Triaxial Seismic Switches		
a. Reactor Building Foundation Mat	0.1 to 30 Hz 0.01 to 0.25 g	1*
4. Triaxial Response-Spectrum Recorders		
a. Reactor Building Foundation Mat	(1)	1*
b. Steam Generator Support	(1)	1
c. Intermediate Bldg., Elev. 463'	(1)	1
d. Auxiliary Bldg. Foundation	(1)	1

*With control room indication and/or alarm.

(1) Range varies for the multiple elements of the instrument, i.e., 1.6g at 2 Hz, 10g at 5 Hz, 34g at 10 Hz, 12g at 16 Hz.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Time-History Accelerographs, including the following components:			
a. Reactor Building Foundation Mat Accelerometer	M	R	SA
b. Reactor Building Ring Girder Accelerometer	M	R	SA
c. Reactor Building Foundation Mat Trigger*	M	R	SA
2. Triaxial Peak Accelerographs			
a. Side of Steam Generator	NA	R	NA
b. Pressurizer Surge Line	NA	R	NA
c. RHR System Heat Exchanger	NA	R	NA
3. Triaxial Seismic Switches			
a. Reactor Building Foundation Mat*	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. Reactor Building Foundation Mat*	M	R	SA
b. Steam Generator Support	NA	R	NA
c. Intermediate Bldg. Elev. 463'	NA	R	NA
d. Auxiliary Bldg. Foundation	NA	R	NA

*With control room indications and/or alarm.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

CHANNEL	INSTRUMENT DESIGNATION & LOCATION	MINIMUM OPERABLE
1. Wind Speed		
	a. Wind Speed Lower - Primary Met Tower 10m	
	b. Wind Speed Upper - Primary Met Tower 61m	2
2. Wind Direction		
	a. Wind Direction Lower Primary Met Tower 10m	
	b. Wind Direction Upper Primary Met Tower 61m	2
3. Atmospheric Stability		
	a. Delta T 1 Primary Met Tower -10-61m	
	b. Delta T 2 Primary Met Tower -10-40m	2

Elevations nominal above grade elevation

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Wind Speed Lower 10m	D	SA
b. Wind Speed Upper 61m	D	SA
2. Wind Direction		
a. Wind Direction Lower 10m	D	SA
b. Wind Direction Upper 61m	D	SA
3. Atmospheric Stability		
a. Delta T 1 10-61m	D	SA
b. Delta T 2 10-40m	D	SA

Elevations nominal above grade elevation

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	Reactor Trip Switchgear	OPEN-CLOSE	1/trip breaker
2. Pressurizer Pressure	CREP	0-3000 psig	1
3. Pressurizer Level	CREP	0 - 100%	1
4. Steam Generator Pressure	CREP	0 - 1300 psig	1/steam generator
5. Steam Generator Level	CREP	0 - 100%	1/steam generator
6. Condensate Storage Tank Level	CREP	0 - 40 feet	1
7. Reactor Coolant System Hot Leg Temperature	CREP	0 - 700°F	1/loop
8. Reactor Coolant System Cold Leg Temperature	CREP	0 - 700°F	1/loop
9. Reactor Coolant System Pressure	CREP	0-3000 psig	1
10. Pressurizer Relief Tank Level	CREP	0-100%	1
11. Reactor Building Temperature	CREP	50°-350°F	1
12. Boric Acid Tank Level	CREP	0-100%	1/boric acid tank

CREP - Control Room Evacuation Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Level	M	R
6. Condensate Storage Tank Level	M	R
7. Reactor Coolant System Hot Leg Temperature	M	R
8. Reactor Coolant System Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure	M	R
10. Pressurizer Relief Tank Level	M	R
11. Reactor Building Temperature	M	R
12. Boric Acid Tank Level	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 channels less than the Required Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channels to OPERABLE status witin 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

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Building Wide Range 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	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/stm. gen.	1/steam generator
7. Steam Generator Water Level - Wide Range	1/stm. gen.	1/steam generator
8. Emergency Feedwater Flow	1/stm. gen.	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Boric Acid Tank Water Level	2/tank	1/tank
11. Reactor Building Spray Pump Discharge Flow	2	1
12. Reactor Building Temperature	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL OF CHANNELS	MINIMUM CHANNELS OPERABLE
13. Reactor Building RHR Sump Level	2	1
14. Reactor Building Level	2	1
15. Condensate Storage Tank Level	2	1
16. Reactor Building Cooling Unit Service Water Flow	2	1
17. Service Water Temperature-Reactor Building Cooling Unit (Inlet and Discharge)	2 pairs	1 pair
18. NaOH Storage Tank Level	2	1
19. Reactor Coolant System Subcooling Margin Monitor	2	1
20. Pressurizer PORV Position Indicator	2/valve*	1/valve*
21. Pressurizer PORV Block Valve Position Indicator	1/valve	1/valve
22. Pressurizer Safety Valve Position Indicator	2/valve	1/valve
23. In-Core Thermocouples	4/core quadrant	2/core quadrant
24. Reactor Vessel Level	2	1

*Not required when the associated block valve is closed per Specification 3.4.4.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Building Wide Range 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 - Wide Range	M	R
8. Emergency Feedwater Flow	M	R
9. RWST Water Level	M	R
10. Boric Acid Tank Solution Level	M	R
11. Reactor Building Spray Pump Discharge Flow	M	R

TABLE 4.3-7 (continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
12. Reactor Building Temperature	M	R
13. Reactor Building RHR Sump Level	M	R
14. Reactor Building Level	M	R
15. Condensate Storage Tank Level	M	R
16. Reactor Building Cooling Unit Service Water Flow	M	R
17. Service Water Temperature - Reactor Building Cooling Unit (Inlet and Discharge)	M	R
18. NaOH Storage Tank Level	M	R
19. Reactor Coolant System Subcooling Margin Monitor	M	R
20. Pressurizer PORV Position Indicator	M*	R*
21. Pressurizer PORV Block Valve Position Indicator	M	R
22. Pressurizer Safety Valve Position Indicator	M	R
23. In-Core Thermocouples	M	R
24. Reactor Vessel Level	M	R

*Not required when the associated block valve is closed per Specification 3.4.4.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11 or with two adjacent fire detection instruments inoperable in those areas identified in Table 3.3-11 by a # symbol:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
1. REACTOR BUILDING				
Zone 1 Elev. 412'				
Room 12-01 NE		2		3
Room 12-01 SE		2		3
Room 12-01 SW		2		3
Room 12-01 NW		2		3
Zone 2 Elev. 412'				
Room 12-03		1		2
Room 12-07		1		2
Room 12-08		1		2
Zone 3 Elev. 436'				
Room 36-01 NE		2		3
Room 36-01 SE		2		3
Room 36-01 NW		2		3
Room 36-01 SW		2		3
Zone 4 Elev. 462'				
Room 12-03		1		2
Room 12-07		1		2
Room 12-08		1		2
Zone OOO Elev. 412'				
Room 12-08		1		2
Zone OOO Elev. 463'				
Room 63-01 SE		2		3
Room 63-01 SW		1		2
Zone PPP Elev. 412'				
Room 12-03		1		2
Room 12-07		1		2
Zone PPP Elev. 463'				
Room 63-01 NE		1		2
Room 63-01 NW		3#		5#
2. CONTROL BUILDING				
Zone A Elev. 463'				
Room 63-02		1**		1**
Room 63-03		2**		3**
Room 63-04		1**		2**

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
2. CONTROL BUILDING (Cont.)				
Zone A Elev. 448'				
Room 48-02 (Cable Spreading Room-Upper)		17**		18**
Zone A Elev. 436'				
Room 36-02		1**		1**
Room 36-03		2**		3**
Room 36-04		1**		2**
Zone B Elev. 425'				
Room 25-02 (Cable Spreading Room-Lower)		10**		11**
Zone B Elev. 412'				
Room 12-04		2**		3**
Zone B Elev. 400'				
Below Room 12-05		1**		2**
Zone J Elev. 436'				
Room 36-01	2**		3**	
Zone J Elev. 448'				
Room 48-01	21**		22**	
Zone K Elev. 412'				
Room 12-02	2**		3**	
Room 12-03	3**		4**	
Zone K Elev. 425'				
Room 25-01	21**		22**	
Zone L Elev. 436'				
Room 36-11	23***	4#	24***	7#
Zone M Elev. 436'				
Room 36-10	14***	1	15***	2
Zone N Elev. 448'				
Room 48-02 (Cable Spreading Area For HVAC Cabinets)		11#		21#
Zone P Elev. 463'				
Room 63-05 (main Control Board)		3#		6#
Zone R Elev. 463'				
Room 63-05 (HVAC Control Board)		1		1

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
2. CONTROL BUILDING (Cont.)				
Zone S Elev. 448'				
Room 48-02 (Cable Spreading Area for HVAC Cabinets)		9#		17#
Zone VV Elev. 436'				
Room 36-09		1		2
Zone WW Elev. 463'				
Room 63-05		4#		8#
Room 63-06		1		1
Room 63-07		1		2
Room 63-09		1		1
Room 63-10		1		1
Room 63-11		1		1
Room 63-12		1		1
Room 63-13		1		1
Zone XX Elev. 463'				
Room 63-14		1		1
Room 63-15		1		1
Room 63-16		1		1
Room 63-17		1		1
Room 63-18		1		2
Room 63-19		1		1
Room 63-20		1		1
Room 63-21		1		1
Room 63-22		1		2
Room 63-23		1		1
Zone NNN Elev. 482'				
Room 82-01		4#		8#
Room 82-02		4#		7#
Room 82-03		2		3
Room 82-04		1		2
Zone TTT Elev. 412'				
Room 12-11		1		2
3. NORTH PENETRATION ACCESS AREA				
Zone EE Elev. 436'				
Room 36-01		2		3
Zone YY Elev. 412'				
Room 12-01		2#		4#

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE* HEAT	MINIMUM INSTRUMENTS OPERABLE* SMOKE	TOTAL NUMBER OF INSTRUMENTS HEAT	TOTAL NUMBER OF INSTRUMENTS SMOKE
INTERMEDIATE BUILDING (Cont.)				
4. EAST PENETRATION ACCESS AREA				
Zone JJ Elev. 412' Room 12-02		$5^{\#}$		$10^{\#}$
Zone LL Elev. 436' Room 36-02		$5^{\#}$		$10^{\#}$
5. WEST PENETRATION AREA				
Zone Y Elev. 412' Room 12-01		$4^{\#}$		$8^{\#}$
Zone DD Elev. 436' Room 36-01		$4^{\#}$		$8^{\#}$
Zone II Elev. 463'				
Room 63-01		$3^{\#}$		$6^{\#}$
Room 63-03		2		3
6. INTERMEDIATE BUILDING				
Zone AA Elev. 412'				
Room 12-02 W		22**		23**
Room 12-13 A		1**		1**
Room 12-13 B		1**		1**
Room 12-13 C		1**		1**
Zone BB Elev. 412'				
Room 12-02 E		17**		18**
Room 12-10		1**		1**
Zone CC Elev. 436' Room 36-02 W		3**		4**
Zone FF Elev. 423'-6" Room 236-01		$3^{\#}$		$5^{\#}$
Zone GG Elev. 412'				
Room 12-03		1		1
Room 12-04		1		1
Room 12-05		1		2
Room 12-06		1		2
Room 12-07		1		1
Room 12-08		1		1
Room 12-12		1		1
Room 12-14		1		1
Room 12-15		1		1

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
6. INTERMEDIATE BUILDING (Cont.)				
Zone KK Elev. 412'				
Room 12-09		1		1
Zone KK Elev. 423'				
Room 23-01		2#		4#
Room 23-02		2#		4#
Zone KK Elev. 436'				
Room 36-01		2		3
Zone MM Elev. 463'				
Room 63-01		4#		8#
Room 63-02		3#		6#
Room 63-03		1		2
Zone PP Elev. 436'				
Room 36-02		8#		16#
Room 36-03 (CREP)		1		2
Room 36-03 A (CREP)		1		2
Room 36-03 B		1		2
Room 36-04		1		1
Zone MMM Elev. 451'				
Room 51-01		3#		6#
Room 51-02		3#		6#
Room 51-03		1		2
Room 51-04		1		2
Zone FFF Elev. 436'				
Room 36-02		8#		16#
Room 36-02 E		1		1
Zone UUU Elev. 426'				
Room 26-01		2#		4#
Room 26-02		2#		4#
7. AUXILIARY BUIDLING				
Zone Q Elev. 436'				
Room 36-18		2#		4#
Zone HH Elev. 463'				
Room 63-01		1		2
Room 63-04		1		1
Room 63-06		1		2
Room 63-17		1		1

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
7. AUXILIARY BUILDING (Cont.)				
Zone QQ Elev. 452'				
Room 52-01		1		1
Room 52-02		1		1
Zone RR Elev. 463'				
Room 63-16		2**		3**
Room 63-19		1**		1**
Zone ZZ Elev. 374'				
Room 74-12		1		1
Room 74-16		2		3
Room 74-17		2		3
Zone AAA Elev. 388'				
Room 88-23		1		1
Room 88-24		1		1
Room 88-25		1		1
Zone AAA Elev. 397'				
Room 97-02		3#		6#
Room 97-02 N		1		1
Room 97-02 S		1		2
Zone BBB Elev. 388'				
Room 88-05		1		2
Room 88-13 N		2		3
Room 88-13 NE		1		1
Room 88-13 S		1		1
Zone BBB Elev. 397'				
Room 97-01		1		2
Zone BBB Elev. 400'				
Room 00-01		1		1
Room 00-02		2		3
Zone CCC Elev. 463'				
Room 63-09		2		3
Room 63-14		1		2
Zone EEE Elev. 374'				
Room 74-01		1		2
Room 74-07		1		2
Room 74-08		2		3
Room 74-09		3#		5#
Room 74-18		1		1
Zone GGG Elev. 400'				
Room 00-01		3#		6#

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
7. AUXILIARY BUILDING (Cont.)				
Zone HHH Elev. 412'				
Room 12-05		1		1
Room 12-06		1		1
Room 12-11		4#		7#
Room 12-11 N		1		1
Room 12-27		1		1
Zone HHH Elev. 426'				
Room 26-02 W		1		1
Room 26-02 E		1		1
Room 26-02 S		1		1
Zone III Elev. 412'				
Room 12-02		1		1
Room 12-03 A		1		1
Room 12-09		1		1
Room 12-11 N		1		1
Room 12-15		1		1
Room 12-18		1		1
Room 12-28		1		1
Room 12-31		1		1
Zone JJJ Elev. 412'				
Room 12-02		1		1
Room 12-11		4#		7#
Room 12-11 N		1		1
Zone LLL Elev. 436'				
Room 36-33 N		3#		6#
Zone RRR Elev. 388'				
Room 88-05		1		1
Room 88-13		1		2
Room 88-13 S		1		1
Room 88-16		1		1
Zone WWW Elev. 436'				
Room 36-33		3#		5#
Zone WWW Elev. 446'				
Room 46-01		1		1

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
7. AUXILIARY BUILDING (Cont.)				
Zone YYY Elev. 412'				
Room 12-05		1		1
Room 12-27		1		1
Room 12-30		1		1
Zone ZZZ Elev. 412'				
Room 12-01		1		1
Zone ZZZ Elev. 436'				
Room 36-01		1		1
Room 36-03		1		2
Room 36-31		1		1
Room 36-33 S		1		2
8. FUEL HANDLING BUILDING				
Zone EE Elev. 436'				
Room 36-01 W		2		3
Zone EE Elev. 443'-6"				
Room 436-01		1		1
Zone TT Elev. 463'				
Room 36-01		10#		20#
Zone UU Elev. 463'				
Room 36-01		10#		19#
9. SERVICE WATER PUMPHOUSE				
Zone NN Elev. 425'				
Room 25-04		1		2
Room 25-05		3#		6#
Zone OO Elev. 441'				
Room 41-02		1**		1**
Zone OO Elev. 436'				
Room 36-02		8**		9**
Zone DDD Elev. 441'				
Room 41-01		3#		5#
Zone VVV Elev. 425'				
Room 25-03		1		1

TABLE 3.3-11 (Cont.)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
10. DIESEL GENERATOR BUILDING				
Zone DG Elev. 436'				
Room 36-01	1**		2**	
Room 36-02	1**		2**	
Room 36-03	1**		1**	
Room 36-04	1**		1**	
Zone DG Elev. 427'				
Room 27-03	1**		1**	
Room 27-04	1**		1**	
Zone KKK Elev. 400'				
Room 00-01		1		1
Room 00-02		1		1
Zone KKK Elev. 427'				
Room 27-01		1		1
Room 27-02		1		1
Room 27-03		2		3
Room 27-04		2		3

*The fire detection instruments located within the Reactor Building are not required to be operable during performance of Type A Containment Leakage Rate Tests.

**Automatically Actuates Preaction Sprinkler System.

***Automatically Actuates Low Pressure CO_2 System.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally if this condition prevails for more than 30 days, in the next semiannual effluent report, explain why this condition was not corrected in a timely manner.

c. The provisions of Specifications 6.9.1.12.b, 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - RM-L5 or RM L9	1	36
b. Nuclear (Processed Steam Generator) Blowdown Effluent Line RM-L7 or RM-L9	1	32
c. Steam Generator Blowdown Effluent Line		
1. Unprocessed during Power Operation - RM-L10 or RM-L3	1	32
2. Unprocessed during Startup - RM-L3	1	32
d. Turbine Building Sump Effluent Line - RM-L8	1	33
2. FLOW RATE MEASUREMENT DEVICES*		
a. Liquid Radwaste Effluent Line - Tanks 1 and 2	1/tank	34
b. Penstocks Minimum Flow Interlock**	1	34
c. Nuclear Blowdown Effluent Line	1	34
d. Steam Generator (Unprocessed) Blowdown Effluent Line	1	34
3. TANK LEVEL INDICATING DEVICES		
a. Condensate Storage Tank	1	35

*Flow rate for the monitor RM-L9 is determined by adding flow rates for monitors RM-L5 and RM-L7.

**Minimum dilution flow is assured by an interlock terminating liquid waste releases if minimum dilution flow is not available.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross radioactivity (beta and gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

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

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

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta and gamma) at a limit of detection of at least 10^{-7} microcuries/gram.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue for up to 30 days provided the tank liquid level is estimated during all liquid additions to the tank to prevent overflow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line - RM-L5, RM-L9	D	P	R(3)	Q(1)
b. Nuclear Blowdown Effluent Line - RM-L7	D	P	R(3)	Q(1)
c. Steam Generator Blowdown Effluent Line - RM-L3, RM-L10	D	M	R(3)	Q(1)
d. Turbine Building Sump Effluent Line - RM-L8	D	M	R(3)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Penstocks Minimum Flow Interlock	D(4)	N.A.	R	Q
c. Nuclear Blowdown Effluent Line	D(4)	N.A.	R	Q
d. Steam Generator Blowdown Effluent Line	D(4)	N.A.	R	Q
3. TANK LEVEL INDICATING DEVICES				
a. Condensate Storage Tanks	D	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Loss of Power (alarm only).

3. Instrument indicates a downscale failure (alarm only).

4. Instrument controls not set in operate mode.

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

1. Instrument indicates measured levels above the alarm setpoint.

2. Loss of Power.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

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

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally if this condition prevails for more than 30 days, in the next semiannual effluent report, explain why this condition was not corrected in a timely manner.

c. The provisions of Specifications 6.9.1.12.b, 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release RM-A10 or RM-A3	1	*	38
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Oxygen Monitor	2	**	44
b. Hydrogen Monitor	1	**	42
3. MAIN PLANT VENT EXHAUST SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release from Waste Gas Holdup System RM-A3	1	*	40
b. Iodine Sampler	1	*	43
c. Particulate Sampler	1	*	43
d. Flow Rate Measuring Device	1	*	39
e. Sampler Flow Rate Measuring Device	1	*	39

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4.	REACTOR BUILDING PURGE SYSTEM			
	a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release - RM-A4	1	*	41
	b. Iodine Sampler	1	*	43
	c. Particulate Sampler	1	*	43
	d. Flow Rate Measuring Device	1	*	39
	e. Sampler Flow Rate Monitor	1	*	39

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times during releases via this pathway.

** During waste gas holdup system operation (treatment for primary system offgases).

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days provided one hydrogen analyzer upstream and one hydrogen analyzer downstream are OPERABLE or grab samples are taken and analyzed at least once per 4 hours at the location of the inoperable hydrogen analyzer. With both the channels inoperable, be in at least HOT STANDBY within 6 hours.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - RM-A10, RM-A3	P	P	R(3)	Q(1)	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Oxygen Monitor	D	N.A.	Q(5)	M	**

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. MAIN PLANT VENT EXHAUST SYSTEM					
a. Noble Gas Activity Monitor - RM-A3	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Measuring Device	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. REACTOR BUILDING PURGE SYSTEM					
a. Noble Gas Activity Monitor - RM-A4	D	P,M	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Measuring Device	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.
** During waste gas holdup system operation (treatment for primary system offgases).

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Loss of Power (alarm only).
3. Instrument indicates a downscale failure (alarm only).
4. Instrument controls not set in operate mode.

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

1. Instrument indicates measured levels above the alarm setpoint.
2. Loss of Power.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

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

1. Fifteen hundred ppm hydrogen, balance nitrogen, for the outlet hydrogen monitor and
2. Four volume percent hydrogen, balance nitrogen for the inlet hydrogen monitor.

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

1. Seventy-five ppm oxygen, balance nitrogen, for the outlet oxygen monitor and
2. Three and one-half volume percent oxygen, balance nitrogen for the inlet oxygen monitor.

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

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

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

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

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All Reactor Coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

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.

*See Special Test Exception 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the Reactor Coolant loops listed below shall be OPERABLE and at least one of these Reactor Coolant loops shall be in operation.*

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

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

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

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 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 operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% of wide range indication 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.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the Reactor Coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE and at least one of these Reactor Coolant and/or RHR loops shall be in operation.**

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

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

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

d. Residual Heat Removal Loop A,

e. Residual Heat Removal Loop B,

APPLICABILITY: MODE 4

ACTION:

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

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

---

*A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

**All Reactor Coolant pumps and decay 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 Reactor Coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

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

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

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

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

b. The secondary side water level of at least two steam generators shall be greater than 10 percent of wide range indication.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With less than the above required loops OPERABLE and/or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no residual heat removal 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 residual heat removal loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

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

##A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

*The RHR pump may be 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.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 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.

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional 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.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kw.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

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

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

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, 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.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Performance of a CHANNEL CALIBRATION and

b. Operating the valve through one complete cycle of full travel.

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

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

3. A tube inspection (pursuant to Specification 4.4.5.4.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.

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months 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 in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.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 main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

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

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

5. Defect means an imperfection of such severity that it exceeds the plugging lim.t. 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 (hot leg side) completely around the U-bend to the top support of the cold leg.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.4.5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

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

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Number of Steam Generators per Unit	Three
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One*

*The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the 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.

TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C–1	None	N/A	N/A	N/A	N/A
	C–2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N/A	N/A
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N/A	N/A
	C–3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to specification 6.9.1	All other S. G.s are C–1	None	N/A	N/A
			Some S. G.s C–2 but no additional S. G. are C–3	Perform action for C–2 result of second sample	N/A	N/A
			Additional S. G. is C–3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. A reactor building atmosphere particulate radioactivity monitoring system,

b. The reactor building sump level, and

c. Either the reactor building cooling unit condensate flow rate or a reactor building atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Reactor building sump level-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified is Table 4.3-3

d. Reactor building cooling unit condensate flow detector-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, and

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory 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 3 or 4.

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

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

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

a. During startup following each refueling outage.

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

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NO.	DESCRIPTION
8993 A,B,C	SI to Hot Legs
8992 A,B,C	SI High Head to Hot Legs
8990 A,B,C	SI High Head to Hot Legs
8988 A,B	SI Low Head to Hot Legs
8997 A,B,C	Primary SI High Head to Cold Legs
8995 A,B,C	Alternate SI High Head to Cold Legs
8998 A,B,C	SI to Cold Legs
8973 A,B,C	RHR Low Head to Cold Legs
*8948 A,B,C	Accumulators to Cold Legs
*8956 A,B,C	Accumulators to Cold Legs
8701 A,B	RHR Suction from Hot Legs
8702 A,B	RHR Suction from Hot Legs
8974 A,B	RHR Low Head to Cold Legs

*See Specification 4.4.6.2.2.c.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

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 $T_{avg} \leq 250°F$.

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

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

*Not required with $T_{avg} \leq 250°F$

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 microcurie per gram DOSE EQUIVALENT I-131, and

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

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

ACTION:

MODES 1, 2 and 3*:

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

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

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcurie per 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.

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

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

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

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

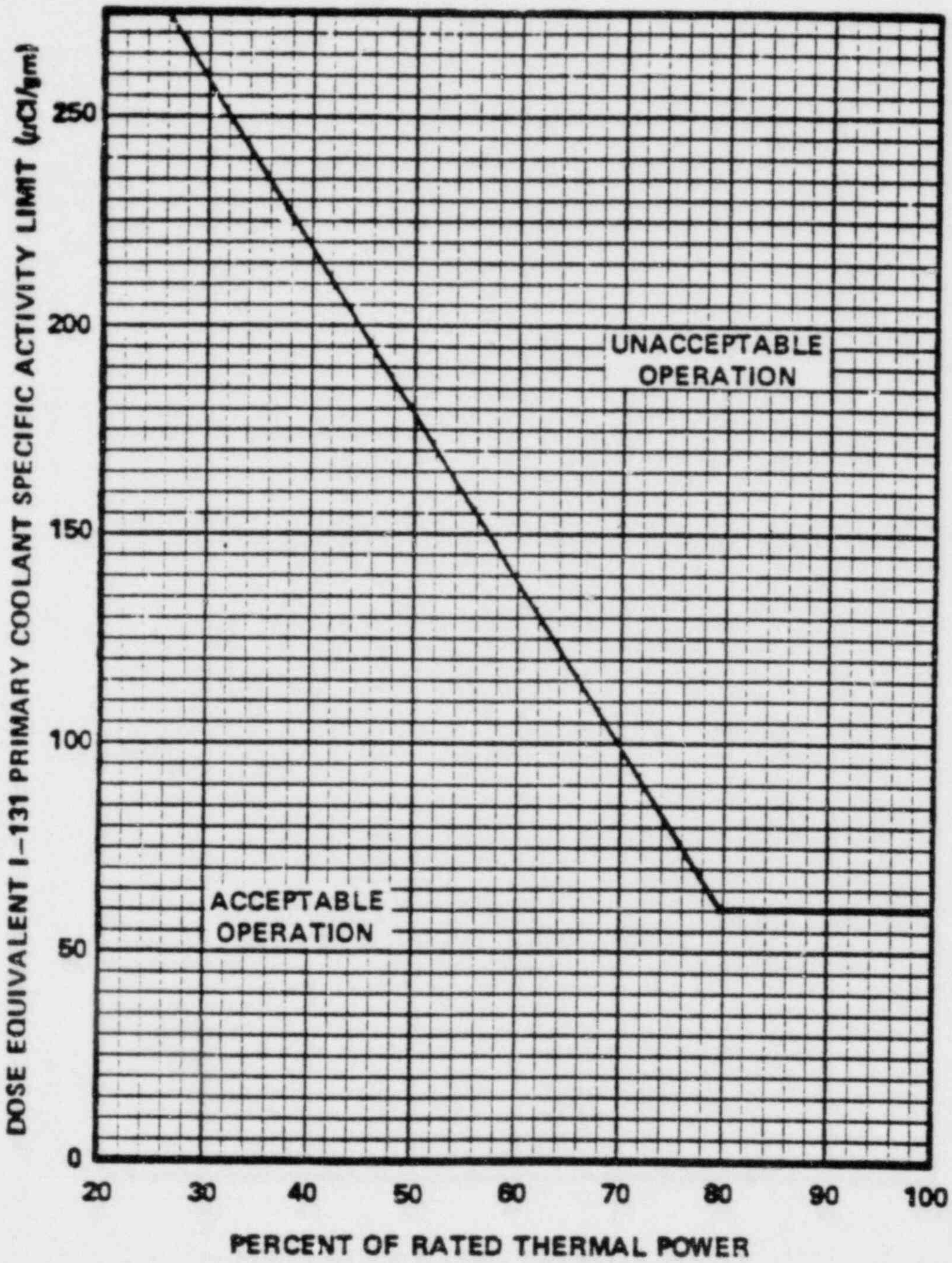

FIGURE 3.4-1

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

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, and

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 fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE IDENTIFICATION	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME, EFPY
U	343°	3.7	1st Refueling
W	110°	3.1	6
Y	290°	3.1	10
Z	340°	3.1	15
V	107°	3.7	STANDBY
X	287°	3.7	STANDBY

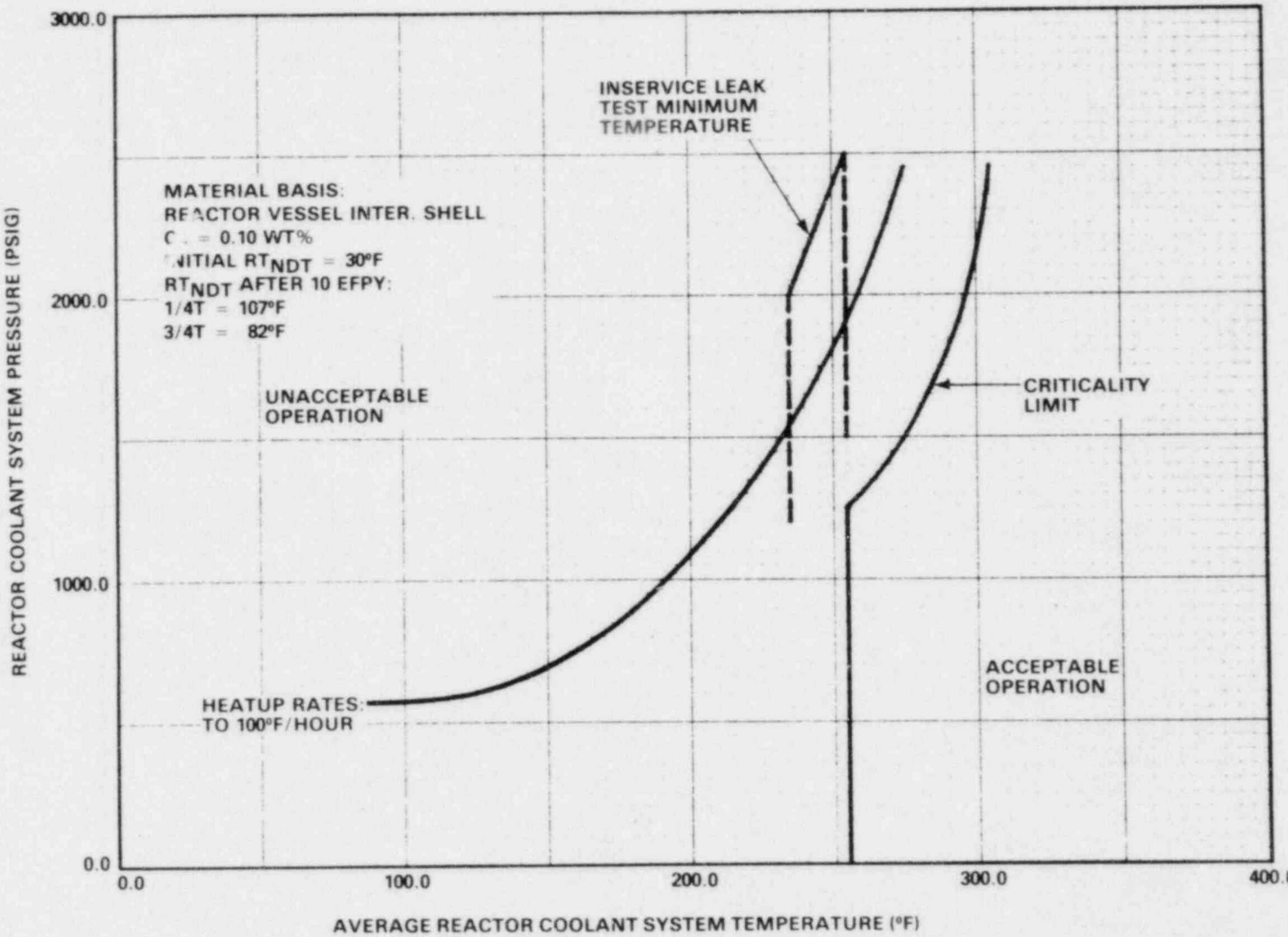

Figure 3.4-2 Reactor Coolant System Pressure - Temperature Limits Versus 100°F/Hour Heatup Rate - Criticality Limit and Inservice Leak Test Limit

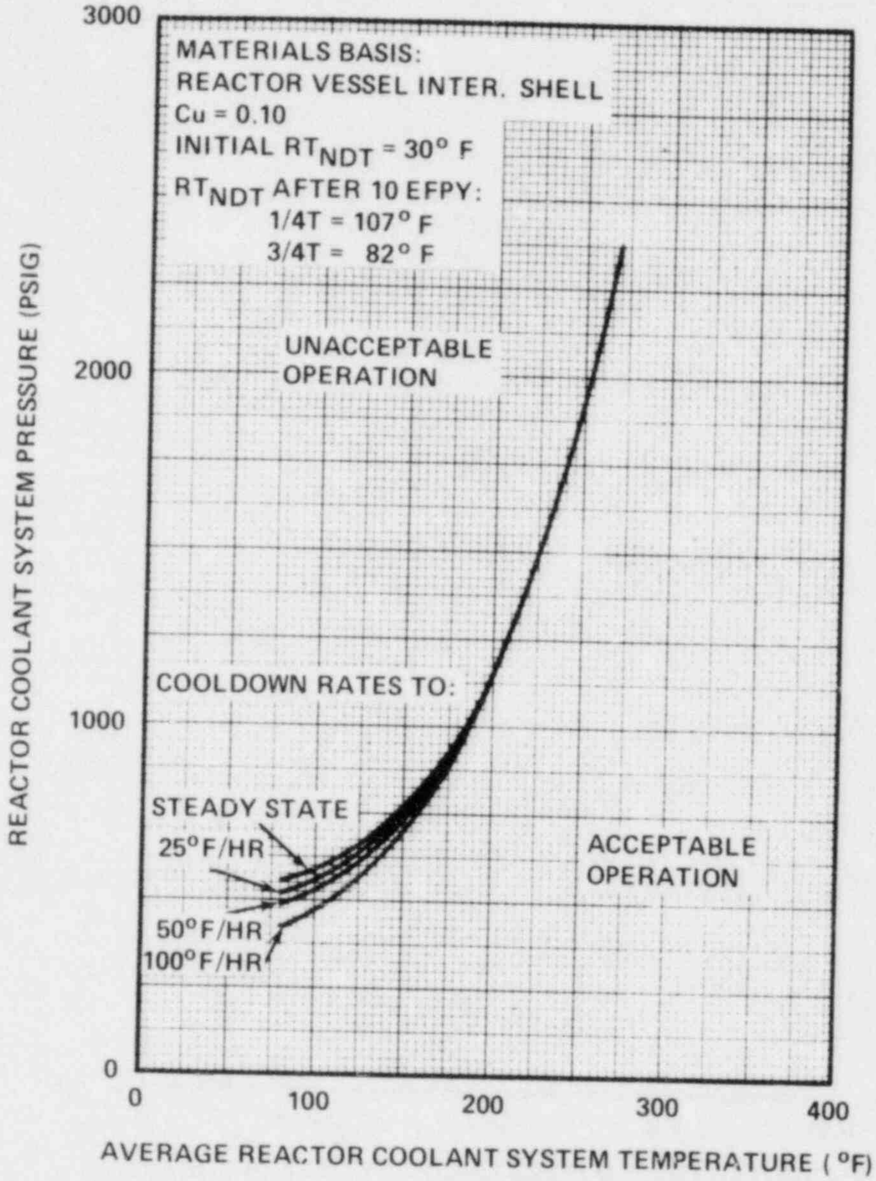

Figure 3.4-3
Reactor Coolant System Pressure - Temperature Limits
Versus Cooldown Rates

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

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

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties 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 less than or equal to the maximum setpoint defined by Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.7 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. In the event either PORV becomes inoperable notify the Commission within 7 days. In the event both PORVs are inoperable, notify the Commission within 24 hours. In both cases 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 cause of the inoperability, plans for restoring the valves to OPERABLE status and any corrective action necessary to prevent recurrence.

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

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

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

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

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

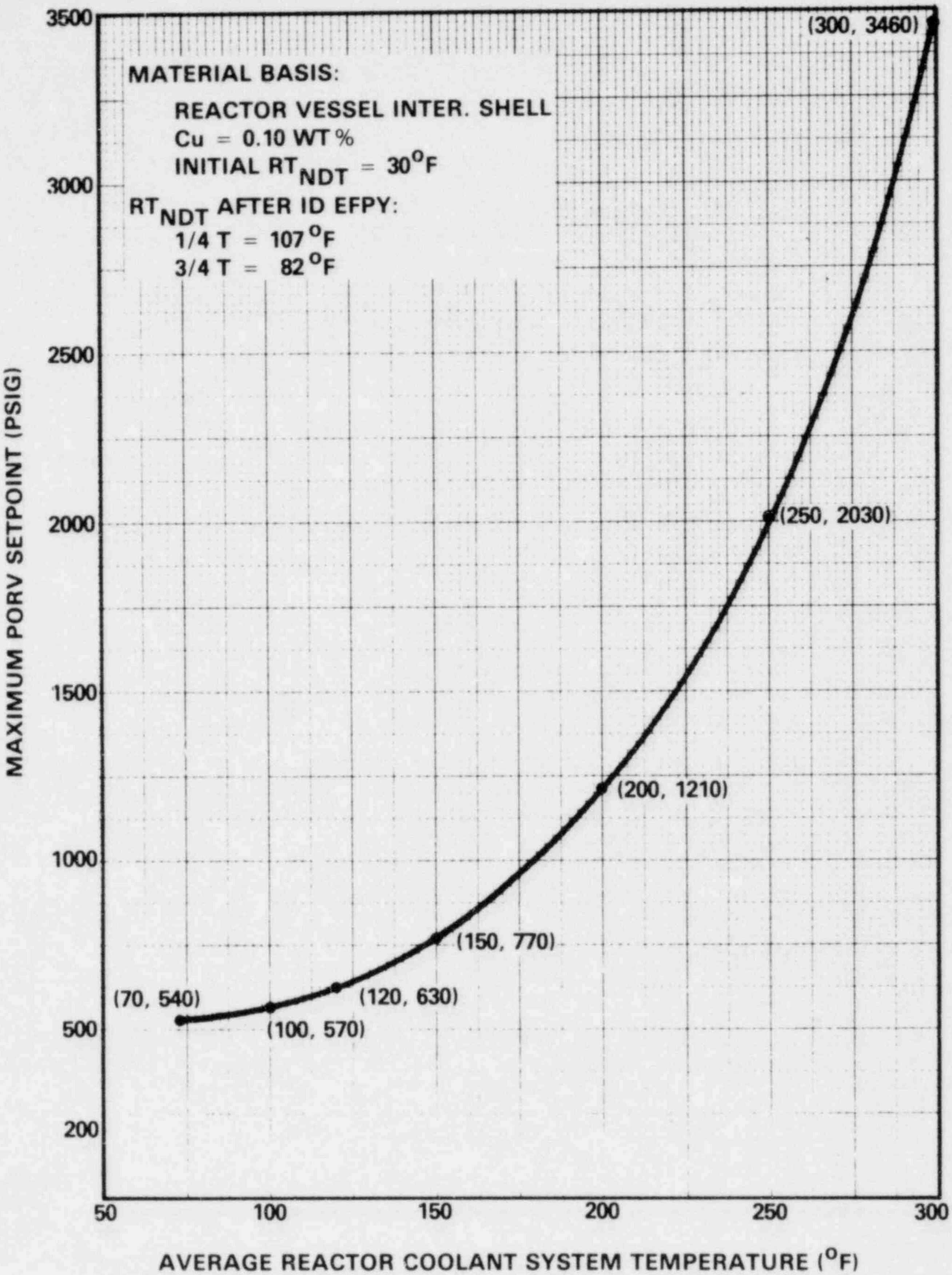

Figure 3.4-4 RCS Cold Overpressurization Protection

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 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 7368 and 7594 gallons,

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

d. A nitrogen cover-pressure of between 600 and 656 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 at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open 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 an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) setpoint,

   2. Upon receipt of a safety injection test signal.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the residual heat removal sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
1. 8884	HHSI Hot Leg Injection	Closed
2. 8886	HHSI Hot Leg Injection	Closed
3. 8888A	LHSI Cold Leg Injection	Open
4. 8888B	LHSI Cold Leg Injection	Open
5. 8889	LHSI Hot Leg Injection	Closed
6. 8701A	RHR Inlet	Closed
7. 8701B	RHR Inlet	Closed
8. 8702A	RHR Inlet	Closed
9. 8702B	RHR Inlet	Closed

b. At least once per 31 days by:

1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and
2. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points.

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

1. For all accessible areas of the reactor building prior to establishing CONTAINMENT INTEGRITY, and
2. Of the areas affected within the reactor building at the completion of each reactor building entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

a) Centrifugal charging pump

b) Residual heat removal pump

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

1. Centrifugal charging pump $\geq$ 2472 psi

2. Residual heat removal pump $\geq$ 128 psi

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

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

2. At least once per 18 months.

HPSI System
Valve Number

a. 8996A
b. 8996B
c. 8996C
d. 8994A
e. 8994B
f. 8994C
g. 8989A
h. 8989B
i. 8989C
j. 8991A
k. 8991B
l. 8991C

SURVEILLANCE REQUIREMENTS (Continued)

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

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

      a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 374 gpm, and

      b) The total pump flow rate is less than or equal to 680 gpm.

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

   1) For residual heat removal pump lines, with a single pump running the sum of the injection line flow rates is greater than or equal to 3663 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} < 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 residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal 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 shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°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 except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F by verifying that the motor circuit breakers have been secured in the open position.

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 minimum contained borated water volume of 900 gallons,

b. A boron concentration of between 20,000 and 22,500 ppm, 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 2% 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 at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 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.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum contained borated water volume of 453,800 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron, and

c. A minimum water temperature of 40°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 witnin 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 when the outside air temperature is less than 40°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.4.

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

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a (47.1 psig) and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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

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, 0.20 percent by weight of the containment air per 24 hours at P_a, 47.1 psig, or

2. Less than or equal to L_t, 0.10 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 23.6 psig.

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (47.1 psig) or at P_t (23.6 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 either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed.

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

   1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a, or 0.25 L_t.

   2. Has a duration sufficient to establish accurately the change in leakage rate 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 at P_a (47.1 psig) or P_t (23.6 psig).

d. Type B and C tests shall be conducted with gas at P_a (47.1 psig) at intervals no greater than 24 months except for tests involving:

   1. Air locks.

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

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

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

g. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each reactor building 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 0.10 L_a at P_a, 47.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one reactor building 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 six hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the reactor building 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 six hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage rate is less than or equal to 0.01 L_a when the volume between the door seals is pressurized to greater than or equal to 8.0 psig for at least 3 minutes.

b. By conducting overall air lock leakage tests at not less than P_a, 47.1 psig, and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months#, and

2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

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

At least once per 6 months#, by verifying that the seal leakage rate is less than or equal to 0.01 L_a when the volume between the handwheel shaft seals is pressurized to greater than or equal to 8.0 psig for at least 3 minutes.

#The provisions of Specification 4.0.2 are not applicable.

* Exemption to Appendix J of 10 CFR 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Reactor building internal pressure shall be maintained between -0.1 and 1.5 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With the structural integrity of the containment not conforming to the requirements of Specification 4.6.1.6.1.b, perform an engineering evaluation of the containment to demonstrate the acceptability of containment tendons within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity of the containment otherwise not conforming to the requirements of Specification 4.6.1.6, 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 30 days after completion of the inspection describing the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective actions taken.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment tendons shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The structural integrity of the tendons shall be demonstrated by:

a. Determining that a representative sample* of at least 15 tendons (4 dome, 5 vertical, and 6 hoop) each has a lift off force of greater than or equal to 95% of its Base Value indicated in Table 4.6-1a. If the lift off force of a selected tendon in a group lies between the 95% Base Value and 90% of the Base Value, one tendon on each side of this tendon shall be checked for its lift off force. If the lift off forces of the adjacent tendons are greater than or equal to 95% of their Base Values in Table 4.6-1b, the single deficiency shall be considered unique and acceptable. For tendon(s) not conforming to

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.

these requirements, a determination shall be made as to the cause of the occurrence and the tendon(s) shall be restored to the required level of integrity.

If the lift-off force of the selected tendon lies below 90% of its Base Value, the tendon shall be completely detensioned and a determination made as to the cause of the occurrence.

b. Determining that the average of the Normalized Lift Off Forces for each tendon group (vertical, dome and hoop) is greater than or equal to the minimum required average tendon force for the group. The minimum required average tendon force is 1195 kips for vertical tendons, 1115 kips for dome tendons, and 1181 kips for hoop tendons. The Normalized Lift Off Force for a tendon is obtained by adding the Normalizing Factor appearing in Table 4.6-2 to the lift off force. Failure to comply with this requirement may be evidence of abnormal degradation of the containment structure.

   If the Normalized Lift-Off Force of any tendon is less than the applicable minimum required average tendon force, an investigation shall be conducted to determine the cause and extent of occurrence. This investigation shall include as a minimum the measurement of lift-off forces of tendons adjacent to the deficient tendon to determine if the average of the tendon lift-off forces in this region of the containment is equal to or greater than the minimum required average tendon force. Failure to comply with this requirement may be evidence of abnormal degradation of the containment structure.

c. Detensioning one tendon in each group (dome, vertical and hoop) from the representative sample. One wire shall be removed from each detensioned tendon and examined to determine:

   1. That over the entire length of the tendon wire, the wire has not undergone corrosion, cracks or damage to the extent that an abnormal condition is indicated.

   2. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

d. Determining for each tendon in the above representative tendon sample, that an analysis of a sample of the sheathing filler grease is within the following limits:

1.	Grease Voids	$\leq$ 5% of net duct volume
2.	Chlorides	$\leq$ 10 PPM
3.	Sulphides	$\leq$ 10 PPM
4.	Nitrates	$\leq$ 10 PPM
5.	Water Content	$\leq$ 10% by weight

SURVEILLANCE REQUIREMENTS (Continued)

If the inspections performed at 1, 3, and 5 years indicate no abnormal degradation of the tendon system, the number of sample tendons may be reduced to 3 dome, 3 vertical, and 3 hoop for subsequent inspections. Upon the completion of the five year inspection, the results of the first three inspections shall be evaluated to determine if an abnormal condition is evident for the tendon system. Based on the conclusions of this evaluation, the sample tendons with their Base Values and Normalizing Factors will be specified for all subsequent inspections.

4.6.1.6.2 At the same inspection frequency as the tendons, the structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be determined by a visual inspection and verifying that no abnormal material or structural behavior is evident

4.6.1.6.3 At the same inspection frequency as the Type A containment leakage rate test, the structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined prior to each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying that no abnormal material or structural behavior is evident.

Table 4.6-1a. Base value of tendon force

SURVEILLANCE TENDONS					
INSPECTION PERIOD					
1		2		3	
Tendon	Base value (kips)	Tendon	Base value (kips)	Tendon	Base value (kips)
D-104	1275	D-125	1230	D-108	1287
D-129	1245	D-219	1258	D-121	1225
D-219	1265	D-228	1275	D-219	1254
D-328	1287	D-324	1273	D-312	1271
V-23	1328	V-23	1319	V-23	1313
V-46	1309	V-30	1287	V-37	1299
V-67	1332	V-53	1316	V-60	1294
V-92	1299	V-76	1315	V-83	1314
V-115	1322	V-99	1309	V-106	1296
3AC	1324	3AC	1313	3AC	1307
8BA	1272	13BA	1283	8CB	1254
13CB	1284	18CB	1264	18BA	1254
28CB	1263	28BA	1264	28AC	1261
38AC	1256	33CB	1282	33BA	1277
38BA	1253	36AC	1278	38CB	1230

Table 4.6-1b. Base value of tendon force

ADJACENT TENDONS					
INSPECTION PERIOD					
1		2		3	
Tendon	Base value (kips)	Tendon	Base value (kips)	Tendon	Base value (kips)
D-103	1245	D-124	1284	D-107	1241
D-105	1250	D-126	1262	D-109	1222
D-128	1296	D-218	1288	D-120	1284
D-130	1267	D-220	1291	D-122	1264
D-218	1296	D-227	1270	D-218	1281
D-220	1299	D-229	1235	D-220	1286
D-327	1244	D-323	1256	D-311	1266
D-329	1237	D-325	1227	D-313	1237
V-22	1306	V-22	1300	V-22	1295
V-24	1320	V-24	1317	V-24	1313
V-45	1308	V-29	1301	V-36	1284
V-47	1322	V-31	1327	V-38	1293
V-66	1309	V-52	1316	V-59	1308
V-68	1309	V-54	1298	V-61	1309
V-91	1309	V-75	1304	V-82	1297
V-93	1327	V-77	1313	V-84	1311
V-114	1313	V-98	1280	V-105	1297
V-1	1320	V-100	1300	V-107	1307
2AC	1277	2AC	1270	2AC	1264
4AC	1264	4AC	1252	4AC	1245
7BA	1324	12BA	1267	7CB	1303
9BA	1292	14BA	1263	9CB	1284
12CB	1285	17CB	1271	17BA	1261
14CB	1272	19CB	1287	19BA	1289
27CB	1277	27BA	1289	27AC	1297
29CB	1280	29BA	1272	29AC	1254
37AC	1283	32CB	1262	32BA	1259
39AC	1294	34CB	1232	34BA	1253
37BA	1294	35AC	1297	37CB	1276
39BA	1273	37AC	1275	39CB	1291

Table 4.6-2. Normalizing factors (N.F.)

INSPECTION PERIOD					
1		2		3	
Tendon	N.F. (kips)	Tendon	N.F. (kips)	Tendon	N.F. (kips)
D-104	-24	D-125	36	D-108	-42
D-129	33	D-219	10	D-121	40
D-219	10	D-228	-28	D-219	10
D-328	-21	D-324	-12	D-312	-20
V-23	-15	V-23	-15	V-23	-15
V-46	11	V-30	31	V-37	-5
V-67	-21	V-53	-24	V-60	11
V-92	25	V-76	-11	V-83	-15
V-115	-10	V-99	5	V-106	7
3AC	-56	3AC	-56	3AC	-56
8BA	18	13BA	-26	8CB	26
13CB	-23	18CB	29	18BA	34
28CB	26	28BA	17	28AC	10
38AC	40	33CB	-17	33BA	-16
38BA	40	36AC	0	38CB	54

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 6-inch containment purge supply and exhaust isolation valves may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed close, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 6-inch containment purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close the open 6-inch valve(s) or isolate the penetration within 4 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment purge supply and exhaust isolation valve shall be verified to be:

a. Closed at least once per 24 hours,

b. Sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 6-inch purge supply and exhaust isolation valves have been open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and automatically transferring suction to the spray sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building spray system shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 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 each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

   2. Verifying that each spray pump starts automatically on a Reactor Building Spray Actuation test signal.

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

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 3140 and 3230 gallons of between 20.0 and 22.0 percent by weight NaOH solution, and

b. A flow path capable of adding NaOH solution from the spray additive tank to the suction of each reactor building spray pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

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

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

d. At least once per 5 years by verifying each solution flow rate from the following drain connections in the spray additive system:

1. NaOH Tank to Loop A $\geq$ 15 gpm
2. NaOH Tank to Loop B $\geq$ 15 gpm

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 Two independent groups of reactor building cooling units shall be OPERABLE with at least one of two cooling units OPERABLE in slow speed in each group.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required reactor building cooling units inoperable and both reactor building spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required reactor building cooling units inoperable, and both reactor building spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required reactor building cooling units inoperable and one reactor building spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of reactor building cooling units shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each cooling unit group from the control room, and verifying that each cooling unit group operates for at least 15 minutes in the slow speed mode.

b. At least once per 18 months by:

   1. Verifying that each fan group starts automatically on a safety injection test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 4,000 gpm to each cooling unit group.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent groups of HEPA filter banks (associated with the OPERABLE reactor building cooling units of Specification 3.6.2.3) with at least one filter bank in each group, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one group of HEPA filter banks OPERABLE, restore one of the inoperable banks in the other group to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 The two groups of HEPA filter banks shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter 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 and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,270 cfm ± 10%.

    2. Verifying a system flow rate of 60,270 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Verifying that the pressure drop across the HEPA filters is less than 3 inches Water Gauge while operating the system at a flow rate of 60,270 cfm ± 10%.

2. Verifying that the filter bypass damper can be opened by operator action.

3. Verifying that the filter bypass damper closes on a Safety Injection Test Signal.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ASNI N510-1975 while operating the system at a flow rate of 60,270 cfm ± 10%.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 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

4.6.4.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

4.6.4.2 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 Reactor Building Purge and Exhaust isolation test signal, each Purge and Exhaust valve actuates to its isolation position.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.3 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 A. PHASE "A" ISOLATION	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
1. 7501-AC	CRDM Coolant Water Inlet Line	40
2. 7502-AC	CRDM Coolant Water Inlet Line	40
3. 7503-AC	CRDM Coolant Water Outlet Line	40
4. 7504-AC	CRDM Coolant Water Outlet Line	40
5. 503A-BD #	Steam Generator A Blowdown Line	40
6. 503B-BD #	Steam Generator B Blowdown Line	40
7. 503C-BD #	Steam Generator C Blowdown Line	40
8. 8100-CS	Reactor Coolant Pump Seal Water Return	40
9. 8112-CS	Reactor Coolant Pump Seal Water Return	40
10. 8149A-CS	Reactor Coolant To Letdown Heat Exchanger	40
11. 8149B-CS	Reactor Coolant To Letdown Heat Exchanger	40
12. 8149C-CS	Reactor Coolant To Letdown Heat Exchanger	40
13. 8152-CS	Reactor Coolant To Letdown Heat Exchanger	40
14. 6797-FS	Fire Service Deluge To Charcoal Filters	40
15. 6050A-HR	Normal Reactor Building Pressure Line	40
16. 6054-HR	Normal Reactor Building Pressure Line	40
17. 2660-IA	Reactor Building Instrument Air Inlet Line	40
18. 2662A-IA	Reactor Building Instrument Air Suction Line	40
19. 2662B-IA	Reactor Building Instrument Air Suction Line	40
20. 6242A-ND	Reactor Building Sump Drain	40
21. 6242B-ND	Reactor Building Sump Drain	40
22. 8028-RC	Pressurizer Relief Tank Makeup Water Line	40
23. 8033-RC	Pressurizer Relief Tank N_2 Supply-Return Line	40
24. 8047-RC	Pressurizer Relief Tank N_2 Supply-Return Line	40
25. 8860-SI	Full Line To Accumulators	40
26. 8880-SI	Accumulator Nitrogen Supply	40
27. 8871-SI	Accumulator Test Line	40
28. 8961-SI	Accumulator Test Line	40
29. 9311A-SS	Sampling Line Supply To Radiation Monitor	40
30. 9311B-SS	Sampling Line Supply To Radiation Monitor	40

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER A. PHASE "A" ISOLATION (Continued)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
31. 9312A-SS	Sampling Line Supply Return From Radiation Monitor	40
32. 9312B-SS	Sampling Line Return Return From Radiation Monitor	40
33. 9339-SS	Sample Return Line To PRT	40
34. 9341-SS	Sample Return Line To PRT	40
35. 9356A-SS	Sampling Line From Pressurizer	40
36. 9356B-SS	Sampling Line From Pressurizer	40
37. 9357-SS	Sampling Line From Pressurizer	40
38. 9364B-SS	Sampling Lines From Reactor Coolant Loop B	40
39. 9365B-SS	Sampling Lines From Reactor Coolant Loop B	40
40. 9364C-SS	Sampling Lines From Reactor Coolant Loop C	40
41. 9365C-SS	Sampling Lines From Reactor Coolant Loop C	40
42. 9387-SS	Sampling Line From Accumulators	40
43. 9398A-SS #	Sampling Line From Steam Generator A Blowdown	40
44. 9398B-SS #	Sampling Line From Steam Generator B Blowdown	40
45. 9398C-SS #	Sampling Line From Steam Generator C Blowdown	40
46. 7126-WL	Reactor Coolant Drain Tank Vent Header	40
47. 7150-WL	Reactor Coolant Drain Tank Vent Header	40
48. 1003-WL	Reactor Coolant Drain Tank Discharge To Waste	40
49. 7136-WL	Reactor Coolant Drain Tank Discharge To Waste	40
B. PHASE "B" ISOLATION		
1. 9568-CC	Component Cooling To R. C. Pumps Bearings	60
2. 9600-CC	Component Cooling To R. C. Pumps	60
3. 9605-C	Component Cooling From R. C. Pumps Bearings	60
4. 9606-CC	Component Cooling From R. C. Pumps Bearings	60
5. 1633A-FW #	Chemical Feed Line To Feedwater Loop A	60
6. 1633B-FW #	Chemical Feed Line To Feedwater Loop B	60
7. 1633C-FW #	Chemical Feed Line To Feedwater Loop C	60
C. REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION		
1. 0001A-AH	Reactor Building Purge Supply	5
2. 0001B-AH	Reactor Building Purge Supply	5
3. 0002A-AH	Reactor Building Purge Exhaust	5

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION (Continued)			
4.	0002B-AH	Reactor Building Purge Exhaust	5
5.	6056-HR	Alternate Reactor Building Purge Supply Line	5
6.	6057-HR	Alternate Reactor Building Purge Supply Line	5
7.	6066-HR	Alternate Reactor Building Purge Exhaust Line	5
8.	6067-HR	Alternate Reactor Building Purge Exhaust Line	5
D. MANUAL (1)			
1.	8767-DN	Demineralized Water Line	N/A
2.	8768-DN	Demineralized Water Line	N/A
3.	6772-FS	Fire Service Hose Reel Supply	N/A
4.	6773-FS	Fire Service Hose Reel Supply	N/A
5.	2679-IA	Breathing Air Supply Line	N/A
6.	2680-IA	Breathing Air Supply Line	N/A
7.	6587-NG	Nitrogen Supply To Steam Generators	N/A
8.	8090A-RC	Dead Weight Tester	N/A
9.	8090B-RC	Dead Weight Tester	N/A
10.	2912-SA	Reactor Building Service Air	N/A
11.	6671-SF	Refueling Cavity Drain Line	N/A
12.	6672-SF	Refueling Cavity Drain Line	N/A
13.	6697-SF	Refueling Cavity Fill Line	N/A
14.	6698-SF	Refueling Cavity Fill Line	N/A
15.	7135-WL	Reactor Coolant Drain Tank Discharge To Waste	N/A
E. REMOTE MANUAL (2)			
1.	9602-CC	Component Cooling To R. C. Pumps	N/A
2.	8102A-CS	Seal Injection To Reactor Coolant Pump A	N/A
3.	8102B-CS	Seal Injection To Reactor Coolant Pump B	N/A
4.	8102C-CS	Seal Injection To Reactor Coolant Pump C	N/A

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER E. REMOTE MANUAL (Continued)		FUNCTION	MAXIMUM ISOLATION TIME (SEC)
5.	8107-CS	Charging Line To Regenerative Heat Exchange	N/A
6.	6050B-HR	Hydrogen Analyzer Return Line	N/A
7.	6051A-HR	Hydrogen Analyzer Supply Line	N/A
8.	6051B-HR	Hydrogen Analyzer Supply Line	N/A
9.	6051C-HR	Hydrogen Analyzer Supply Line	N/A
10.	6052A-HR	Hydrogen Analyzer Return Line	N/A
11.	6052B-HR	Hydrogen Analyzer Return Line	N/A
12.	6053A-HR	Hydrogen Analyzer Supply Line	N/A
13.	6053B-HR	Hydrogen Analyzer Supply Line	N/A
14.	8701A-RH	RHR Pump Suction From Reactor Coolant Loop A	N/A
15.	8701B-RH	RHR Pump Suction From Reactor Coolant Loop C	N/A
16.	8801A-SI	Boran Injection Tank To Reactor Coolant Loops	N/A
17.	8801B-SI	Boran Injection Tank To Reactor Coolant Loops	N/A
18.	8811A-SI	RHR Pump A Suction From Recirculation Sump	N/A
19.	8811B-SI	RHR Pump B Suction From Recirculation Sump	N/A
20.	8884-SI	High Head Safety Injection To Reactor Coolant Loops	N/A
21.	8885-SI	High Head Safety Injection To Reactor Coolant Loops	N/A
22.	8886-SI	High Head Safety Injection To Reactor Coolant Loops	N/A
23.	8888A-SI	Low Head Safety Injection To Reactor Coolant Loops	N/A
24.	8888B-SI	Low Head Safety Injection To Reactor Coolant Loops	N/A
25.	8889-SI	Low Head Safety Injection To Reactor Coolant Loops	N/A
26.	3003A-SP	Supply To Reactor Building Spray Nozzles	N/A
27.	3003B-SP	Supply To Reactor Building Spray Nozzles	N/A
28.	3004A-SP	Spray Pump A Suction From Recirculation Sump	N/A
29.	3004B-SP	Spray Pump B Suction From Recirculation Sump	N/A
30.	3103A-SW	Service Water From Reactor Building Cooling Unit A	N/A
31.	3103B-SW	Service Water From Reactor Building Cooling Unit B	N/A
32.	3106A-SW	Service Water To Reactor Building Cooling Unit A	N/A
33.	3106B-SW	Service Water To Reactor Building Cooling Unit B	N/A
34.	3110A-SW	Service Water To Reactor Building Cooling Unit A	N/A
35.	3110B-SW	Service Water To Reactor Building Cooling Unit B	N/A

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER F. CHECK		FUNCTION	MAXIMUM ISOLATION TIME (SEC)
1.	7541-AC	CRDM Coolant Water Inlet Line	N/A
2.	7544-AC	CRDM Coolant Water Outlet Line	N/A
3.	9570-CC	Component Cooling To R. C. Pump Bearings	N/A
4.	9689-CC	Component Cooling From R. C. Pump Bearings	N/A
5.	8103-CS	Reactor Coolant Pump Seal Water Return	N/A
6.	8368A-CS	Seal Injection To R. C. Pump A	N/A
7.	8368B-CS	Seal Injection To R. C. Pump B	N/A
8.	8368C-CS	Seal Injection To R. C. Pump C	N/A
9.	8381-CS	Charging Line To Regenerative Heat Exchanger	N/A
10.	6799-FS	Fire Service Deluge To Charcoal Filters	N/A
11.	2661-IA	Instrument Air Supply To Reactor Building	N/A
12.	6588-NG	Nitrogen Supply To Steam Generators	N/A
13.	8046-RC	Pressurizer Relief Tank Makeup Water Line	N/A
14.	2913-SA	Service Air Supply To Reactor Building	N/A
15.	3009A-SP	Supply To Reactor Building Spray Nozzles	N/A
16.	3009B-SP	Supply To Reactor Building Spray Nozzles	N/A
17.	8947-SI	Accumlator Nitrogen Supply	N/A
18.	8861-SI	Fill Line To Accumulators	N/A

#Valve not subject to Type "C" leakage test.

(1) Manual valves may be opened on an intermittent basis under administrative control.

(2) Remote manual valve positions are maintained by administrative control.

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing 19.8 volume percent hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 Two independent post accident hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

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

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

   3. Verifying the integrity of 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.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

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

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

b. 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 Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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	43

TABLE 3.7-2

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)
1	**
2	**
3	**

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

**These values left blank pending NRC approval of two-loop operation.

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (±1%)*	ORIFICE SIZE
S/G A		
XVS-2806 A	1176 psig	4.515 In dia/16 sq in
XVS-2806 B	1190 psig	4.515 In dia/16 sq in
XVS-2806 C	1205 psig	4.515 In dia/16 sq in
XVS-2806 D	1220 psig	4.515 In dia/16 sq in
XVS-2806 E	1235 psig	4.515 In dia/16 sq in
S/G B		
XVS-2806 F	1176 psig	4.515 In dia/16 sq in
XVS-2806 G	1190 psig	4.515 In dia/16 sq in
XVS-2806 H	1205 psig	4.515 In dia/16 sq in
XVS-2806 I	1220 psig	4.515 In dia/16 sq in
XVS-2806 J	1235 psig	4.515 In dia/16 sq in
S/G C		
XVS-2806 K	1176 psig	4.515 In dia/16 sq in
XVS-2806 L	1190 psig	4.515 In dia/16 sq in
XVS-2806 M	1205 psig	4.515 In dia/16 sq in
XVS-2806 N	1220 psig	4.515 In dia/16 sq in
XVS-2806 P	1235 psig	4.515 In dia/16 sq in

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

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency 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 1350 psig at greater than or equal to 90 gpm flow.

2. Verifying that the steam turbine driven pump develops a discharge pressure of greater than or equal to 1330 psig at a flow of greater than or equal to 97 gpm when the secondary steam supply pressure is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable.

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that each automatic valve in the flow path from the condensate storage tank to the steam generators is in the fully open position whenever the emergency feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

5. Verifying that valves 1010-EF and 1007-EF are locked in the open position.

b. At least once per 3 months by verifying that the check valve in the instrument air supply line to the six emergency feedwater control valve air accumulators closes when the normal instrument air supply is not available.

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

1. Each emergency feed pump starts as designed automatically upon receipt of an emergency feedwater actuation test signal.

2. The six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available.

3. The turbine driven emergency feedwater pump can be manually stopped from the main control board by closing the steam supply valve with air from the accumulator when the normal instrument air supply is not available.

4. Each automatic valve in the flow path actuates to its correct position on receipt of an emergency feedwater actuation test signal.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 172,700 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The 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 emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE at least once per 12 hours by verifying service water system pressure whenever the service water system is the supply source for the emergency 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 microcuries/gram DOSE EQUIVALENT I-131.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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 LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours;

Otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

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

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

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

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

SURVEILLANCE REQUIREMENTS

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

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1. 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a simulated SI test signal or on an ESFLS, as applicable.

c. At least once per 18 months, by verifying that each service water system booster pump starts automatically on a safety injection signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 415 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 95°F at the discharge of the service water pumps.

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 SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIRMENTS

4.7.5 The service water pond shall be determined OPERABLE at least once per 24 hours by verifying the water temperature and water level to be within their limits.

3/4.7.6 CONTROL ROOM NORMAL AND EMERGENCY AIR HANDLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent control room normal and emergency air handling systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one control room normal and emergency air handling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room normal and emergency air handling system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air cleanup system in the recirculation mode.

b. With both control room emergency air cleanup systems inoperable, or with the OPERABLE control room emergency air cleanup system, required to be in the recirculation mode by ACTION (a), not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.6 Each control room normal and emergency air handling system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

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 20,000 cfm ± 10%.
2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, 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 20,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 20,000 cfm ± 10%.
2. Verifying that on a simulated SI or high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.
3. Verifying that on a simulated SI or high radiation test signal the system starts the normal and emergency air handling systems which pressurize the control room to a positive pressure of greater than or equal to 1/8 inch W.G. relative to the outside atmosphere and maintains the 1/8 inch W.G. positive pressure with a maximum of 1000 cfm of outside air during system operation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 20,000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 20,000 cfm ± 10%.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

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

b. Visual Inspections

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

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

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

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

c. Refueling Outage Inspections

At least once per 18 months an inspection shall be performed of all the snubbers listed in Tables 3.7-4a and 3.7-4b attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using at least one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; or (iii) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.7.f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of either: (1) At least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.7.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested, or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1, "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.7.f. The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in

e. Functional Tests (Continued)

the "Accept" region testing of that type of snubber may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested.

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

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

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

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

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

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

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

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

g. Functional Test Failure Analysis

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

g. Functional Test Failure Analysis (Continued)

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

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

h. Functional Testing of Repaired and Replaced Snubbers

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

i. Snubber Seal Replacement Program

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

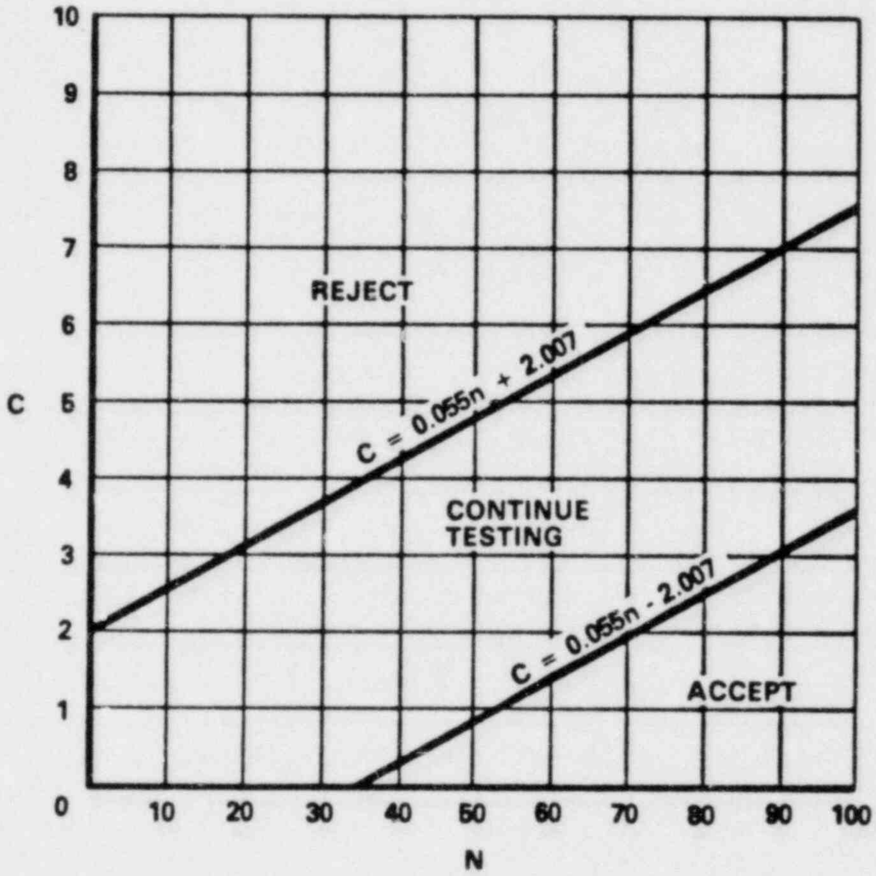

FIGURE 4.7-1 SAMPLING PLAN FOR SUNBBER FUNCTIONAL TEST

Table 3.7-4a

Safety-Related Hydraulic Snubbers*

Paul-Monroe

System	Size (Kips) Small	Medium	Large
	N/A	N/A	1000
FW			15
Subtotal-1			15
Subtotal-2			15
TOTAL	15		

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

Table 3.7-4b

Safety-Related Mechanical Snubbers*

Pacific-Scientific

System				Size (Kips)			
	Small			Medium		Large	
	1/4	1/2	1	3	10	35	100
AS			1	1			
CC	2	1	3	19	20	11	
CS	7	9	18	9			
DG				2	16		
EF			2		1		
FW			1	18	34	32	1
MB				7	2	1	
MS			5	8	9	44	3
RC	6	19	80	49	1	10	
RH	1		21	89	40	3	
SF		1	6	8	3		
SI		7	14	44	36	5	
SP		1	2	53	7		
SW				12	11	3	
Subtotal-1	16	38	153	319	180	109	4
Subtotal-2	54			652		113	
TOTAL	819						

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

3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.8.1 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.

4.7.8.2 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 materials:

   1. With a half-life greater than 30 days (excluding Hydrogen 3), and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.8.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9.1 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, and

b. An OPERABLE flow path capable of taking suction from the circulating water intake structure and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valve, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.9.2, 3.7.9.4, and 3.7.9.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump inoperable, restore the inoperable pump to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:

      a) By telephone within 24 hours,

      b) Confirmed by telegraph, mailgram, or facsimile transmission no later than the first working day following the event, and

      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

4.7.9.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 31 days by starting the electric motor driven pump and operating it for at least 15 minutes.

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 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

d. 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 pump develops at least 2500 gpm at a system head of 125 psig

   2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 85 psig.

e. At least once per 3 years by performing a flow test of the system in accordance with Section 11, Chapter 5 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.9.1.2 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 150 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.9.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

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

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

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

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

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.9.2 The following spray and/or sprinkler systems shall be OPERABLE:

a. Fuel Handling Building Charcoal Filter Plenums - Aux. Bldg. Elev. 463'

b. Control Room Emergency Charcoal Filter Plenums - Control Bldg. Elev. 485'

c. Diesel Fire Pump Room Wet Pipe Sprinkler System - Circulating Water Intake Structure Elev. 436'.

d. Diesel Generator Building Preaction Sprinkler System - Diesel Generator Building Elev. 436' and 427'.

e. Cable Spreading Rooms and Cable Chases Preaction Sprinkler System - Control Building Elev. 463', 448', 436', 425', 412' and 400'.

f. Service Water Pump House Preaction Sprinkler System - Service Water Pump House Elev. 436 and 441.

g. Intermediate Building Preaction Sprinkler System - Intermediate Building Elev. 412.

h. Auxiliary Building Preaction Sprinkler System - 463' Elev.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas 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

4.7.9.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the preaction sprinkler system, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a preaction system test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2. By a visual inspection of the dry pipe, spray and sprinkler headers to verify their integrity, and

   3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

PLANT SYSTEMS

CO_2 SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9.3 The Relay Room - Control Building Elevation - 436' low pressure CO_2 system shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the CO_2 system is required to be OPERABLE.

ACTION:

a. With the above required CO_2 system inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment. 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

4.7.9.3.1 The above required CO_2 system shall be demonstrated OPERABLE 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.

4.7.9.3.2 The above required low pressure CO_2 system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 85% and pressure to be greater than 275 psig, and

b. At least once per 18 months by verifying:

    1. The system valves and associated ventilation dampers and fire door release mechanisms actuate manually and automatically, upon receipt of a simulated actuation signal, and

    2. Flow from each nozzle during a "Puff Test."

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.9.4 The fire hose stations shown in Table 3.7-5 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 3.7-5 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye connected to the length of hose at the station, the other outlet of the wye connected to a hose of sufficient length to provide coverage for the area unprotected by the inoperable hose station. This shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and plans and schedule for restoring the station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.9.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation, 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. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

   3. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK IDENTIFICATION**
Auxiliary Building	374	4138, 4140, 4142
"	388	4144, 6764
"	412	6766, 6761
"	436	6783, 6769, 4148, 4147
"	463	6784, 6755, 4149
"	397	4143
Reactor Building	412	6778, 6780, 6776
"	436	6777, 6782
"	463	6781, 6779
Fuel Handling Building	436	6802
"	463	6804, 6807
Control Building	463	6809
"	482	6815, 6810
Intermediate Building	412	4128, 4121, 4122
"	436	4129, 4124, 4123
"	463	4130

*List all Fire Hose Stations required to ensure the OPERABILITY of safety related equipment.

**Identified by isolation valve number

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.9.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-6 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours. Restore the hydrant or hose house 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 hydrant or hose house to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.9.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

   2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

   3. Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-6

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER*
Service Water Pumphouse	3

*These hydrant numbers are the numbers physically indicated on the hydrant houses.

3/4.7.10 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.10 All fire barrier assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable and piping penetration seals and ventilation seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour either, establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol. Restore the inoperable fire rated assembly and sealing device to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperable fire rated assembly and/or sealing device and the plans and schedule for restoring the fire rated assembly and sealing device to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.10.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample with no apparent changes in appearance or abnormal degradation is found.

4.7.10.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The position of each closed fire door at least once per 24 hours.

b. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. The position of each locked closed fire door at least once per 7 days.

d. The OPERABILITY of the fire door supervision system by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days.

3/4.7.11 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.11 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

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

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.11 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-7

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Charging-SI Pump Room #1	102
2.	Charging-SI Pump Room #2 (swing)	102
3.	Charging-SI Pump Room #3	102
4.	RHR-Spray Pump Room #1	102
5.	RHR-Spray Pump Room #2	102
6.	MCC 1DA2Y Room	102
7.	Switchgear 1DB1 and MCC 1DB2Y Room	102
8.	Switchgear 1DA Room	102
9.	Switchgear 1DB Room	102
10.	Battery 1A Room	88
11.	Battery 1B Room	88
12.	Charger 1A Room	102
13.	Charger 1B Room	102
14.	Charger 1A/1B Room	102
15.	Relay Room	83
16.	Component Cooling pump "A" Speed Switch Room	102
17.	Component Cooling pump "B" Speed Switch Room	102
18.	Component Cooling Pump "C" Speed Switch Room	102
19.	Evacuation Panel "A" Room	83
20.	Evacuation Panel "B" Room	83
21.	Service Water Booster Pumps Area	102
22.	Emergency Feedwater Pumps Area	102
23.	Diesel Generator 1A Room	120
24.	Diesel Generator 1B Room	120
25.	Service Water Pump/Screen Room	102
26.	Service Water Switchgear Room "A"	102
27.	Service Water Switchgear Room "B"	102
28.	Service Water Switchgear Room "C"	102
29.	Diesel Generator Exciter Cabinet Room "A"	102
30.	Diesel Generator Exciter Cabinet Room "B"	102

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 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. A separate day fuel tank containing a minimum volume of 300 gallons of fuel,

   2. A separate fuel storage system containing a minimum volume of 42,500 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 one diesel generator inoperable in addition to ACTION a or b above, verify that:

   (1) All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   (2) When in MODE 1, 2, or 3, the steam-driven auxiliary feed pump is OPERABLE.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day tank,

SURVEILLANCE REQUIREMENTS (Continued)

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 normal standby condition and accelerates to at least 504 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 7200 $\pm$ 720 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.

   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.

   e) Simulated degraded offsite power by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 4250 kW in less than or equal to 60 seconds, and operates with a load greater than or equal to 4250 kW for at least an additional 60 minutes,

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

c. At least once per 92 days and from new fuel prior to addition to the storage tanks, by obtaining a sample of fuel oil in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken (or prior to adding new fuel to the storage tank) verify in accordance with the tests specified in ASTM-D975-77 that the sample has:

   a) A water and sediment content of less than or equal to 0.05 volume percent.

   b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

   c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity @ 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 1 week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70.

3. Within 2 weeks of obtaining the sample verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137 Position 2.a are met when tested in accordance with ASTM-D975-77.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of greater than or equal to 830 kw while maintaining voltage at 7200 $\pm$ 720 volts and frequency at 60 $\pm$ 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kw without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 7200 $\pm$ 720 volts and 60 $\pm$ 1.2 Hz during this test.

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. The generator voltage and frequency shall be 7200 $\pm$ 720 volts and 60 $\pm$ 1.2 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. After 5 minutes of standby operation verify that on a simulated loss of offsite power,

a) the loads are shed from the emergency busses

b) the diesel generator does not connect to the bus for at least 5 seconds, and

c) that subsequent loading of the diesel generator is in accordance with design requirements.

6. 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 in the emergency mode, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4676 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4250 kw. The generator voltage and frequency shall be 7200 ± 720 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.8.1.1.2.d.4.b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 4548 kw.

9. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

SURVEILLANCE REQUIREMENTS (Continued)

c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

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

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a. Barring Device

    b. Remote-Local-Maintenance Switch

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

f. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the 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.

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 fuel tank containing a minimum volume of 300 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 30,000 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5) and 4.8.1.1.3.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

   1. The parameters in Table 4.8-2 meet the Category B limits,

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3. The average electrolyte temperature of 10 of the connected cells is above 60°F.

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

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

   2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

   4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[b]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank and its associated full capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; and initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses:

a. Train A A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DA and 1EA.
   2. 480 volt Emergency Busses # 1DA1, 1DA2 and 1EA1.

b. Train B A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DB and 1EB.
   2. 480 volt Emergency Busses # 1DB1, 1DB2, and 1EB1.

c. 120 volt A.C. Vital Busses # 5902 and 5901 energized from an associated inverter connected to D.C. Bus # 1HA*.

d. 120 volt A.C. Vital Busses # 5904 and 5903 energized from an associated inverter connected to D.C. Bus # 1HB*.

e. 120 volt A.C. Vital Bus #5907 energized.

f. 120 volt A.C. Vital Bus #5908 energized.

g. 125 volt D.C. Bus 1HA energized from Battery Bank #1A.

h. 125 volt D.C. Bus 1HB energized from Battery Bank #1B.

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

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Vital Bus not energized, re-energize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one of A.C. Vital Busses #5901, 5902, 5903, or 5904 either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus re-energize the A.C. Vital Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*The inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABITLITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. 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 DEVICE TEST SETPOINT CRITERIA

	EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
	7.2 KV Swgr.					
1)	XPP0030A-RC Reactor Coolant Pump A	PRIMARY	XSW1A #9	LONG TIME INSTANT GROUND INST.	3960 Amps 5805 Amps 11 Amps	≤ 15.75 Sec. N/A N/A
	BUS1A Normal Feed	BACKUP	XSW1A #5	LONG TIME	5544 Amps	≤ 15.33 Sec.
	BUS1A Emergency Feed	BACKUP	XSW1A #3	LONG TIME	5544 Amps	≤ 15.33 Sec.
2)	XPP0030B-RC Reactor Coolant Pump B	PRIMARY	XSW1B #7	LONG TIME INSTANT GROUND INST.	3690 Amps 5808 Amps 11 Amps	≤ 15.75 Sec. N/A N/A
	BUS1B Normal Feed	BACKUP	XSW1B #5	LONG TIME	5544 Amps	≤ 15.33 Sec.
	BUS1B Emergency Feed	BACKUP	XSW1B #3	LONG TIME	5544 Amps	≤ 15.33 Sec.
3)	XPP0030C-RC Reactor Coolant Pump C	PRIMARY	XSW1C #3	LONG TIME INSTANT GROUND INST.	3960 Amps 5808 Amps 11 Amps	≤ 15.75 Sec. N/A N/A
	BUS1C Normal Feed	BACKUP	XSW1C #9	LONG TIME	5544 Amps	≤ 15.33 Sec.
	BUS1C Emergency Feed	BACKUP	XSW1C #13	LONG TIME	5544 Amps	≤ 15.33 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

	EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
	480 V SWGR.					
1)	XFN0067A-AH CRDM CLNG. SYS. FAN A	PRIMARY	XSW1A3/1C	LONG TIME SHORT TIME INSTANT	540 Amps 2700 Amps 2025 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
	XSWIA3 MAIN INCOMING	BACKUP	XSW1A3/4B	LONG TIME SHORT TIME INSTANT	4800 Amps 7200 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
	BUS TIE TO XSWIC3	BACKUP	XSW1A3/4C	LONG TIME SHORT TIME INSTANT	3000 Amps 4500 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
2)	XFN0067D-AH CRDM CLNG. SYS. FAN D	PRIMARY	XSW1A3/BA	LONG TIME SHORT TIME INSTANT	540 Amps 2700 Amps 2025 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
	XSWIA3 MAIN INCOMING	BACKUP	XSW1A3/4B	LONG TIME SHORT TIME INSTANT	4800 Amps 7200 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
	BUS TIE TO XSWIC3	BACKUP	XSW1A3/4C	LONG TIME SHORT TIME INSTANT	3000 Amps 4500 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
3)	XCR0004-FH REACTOR BLDG POLAR CRANE	PRIMARY	XSW1A3/2C	LONG TIME SHORT TIME INSTANT	744 Amps N/A 4050 Amps	$\leq$ 98 Sec. N/A $\leq$ 0.09 Sec.
	XSWIA3 MAIN INCOMING	BACKUP	XSW1A3/4B	LONG TIME SHORT TIME INSTANT	4800 Amps 7200 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
3) CONTINUED:					
BUS TIE TO XSWIC3	BACKUP	XSW1A3/4C	LONG TIME SHORT TIME INSTANT	3000 Amps 4500 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
4) XFN0009A-AH R.B., REACTOR COMPART. CLNG FAN A	PRIMARY	XSW1A3/2A	LONG TIME SHORT TIME INSTANT	360 Amps 1350 Amps 1350 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
XSWIA3 MAIN INCOMING	BACKUP	XSW1A3/4B	LONG TIME SHORT TIME INSTANT	4800 Amps 7200 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
BUS TIE TO XSWIC3	BACKUP	XSW1A3/4C	LONG TIME SHORT TIME INSTANT	3000 Amps 4500 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
5) XFN0067B-AH CRDM CLNG. SYS. FAN B	PRIMARY	XSW1B3/2D	LONG TIME SHORT TIME INSTANT	525 Amps 2250 Amps 2250 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
XSWIB3 MAIN INCOMING	BACKUP	XSW1B3/4B	LONG TIME SHORT TIME INSTANT	9000 Amps 9000 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
EMERGENCY FEED FROM XSW1DB1	BACKUP	XSW1B3/3B	LONG TIME SHORT TIME INSTANT	4800 Amps 6000 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
6) XFN0009B-AH R.B., REACTOR COMPART. CLNG FAN B	PRIMARY	XSW1B3/3A	LONG TIME SHORT TIME INSTANT	525 Amps 1125 Amps. 1500 Amps	$\leq$ 12 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
6) CONTINUED:					
200 AMP FUSE	BACKUP	XPN5471	$\geq$ 3.75 Milliohms		N/A
7) XFN00067C-AH CRDM CLNG. SYSTEM FAN C	PRIMARY	XSW1C3/2D	LONG TIME SHORT TIME INSTANT	540 Amps 2700 Amps 2025 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
XSW1C3 MAIN INCOMING	BACKUP	XSW1C3/3B	LONG TIME SHORT TIME INSTANT	4800 Amps 7200 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
BUS [illegible] XSWIA3	BACKUP	XSW1A3/4C	LONG TIME SHORT TIME INSTANT	3000 Amps 4500 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.32 Sec. N/A
8) MFN0097B-AH R.B. CLNG. UNIT FAN XFN64B EMERGENCY MOTOR	PRIMARY	XSW1DB1/6D	LONG TIME SHORT TIME INSTANT	525 Amps 1500 Amps 2250 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.
XSWIDB1 MAIN INCOMING	BACKUP	XSW1DB1/4B	LONG TIME SHORT TIME INSTANT	6300 Amps 9000 Amps N/A	$\leq$ 12 Sec. $\leq$ 0.50 Sec. N/A
9) MFN0096B-AH R.B. CLNG. UNIT FAN XFN64B NORMAL MOTOR	PRIMARY	XSW1DB1/7B	LONG TIME SHORT TIME INSTANT	1260 Amps 5400 Amps 5400 Amps	$\leq$ 30 Sec. $\leq$ 0.17 Sec. $\leq$ 0.09 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
9) CONTINUED:					
XSWIDB1 MAIN INCOMING	BACKUP	XSW1DB1/4B	LONG TIME	6300 Amps	≤ 12 Sec.
			SHORT TIME	9000 Amps	≤ 0.50 Sec.
			INSTANT	N/A	N/A
10) MFN0096C-AH R.B. CLNG. UNIT FAN XFN65A NORMAL MOTOR	PRIMARY	XSW1DA1/5B	LONG TIME	1260 Amps	≤ 30 Sec.
			SHORT TIME	5400 Amps	≤ 0.17 Sec.
			INSTANT	5400 Amps	≤ 0.09 Sec.
XSWIDA1 MAIN INCOMING	BACKUP	XSW1DA1/4B	LONG TIME	6300 Amps	≤ 12 Sec.
			SHORT TIME	9000 Amps	≤ 0.50 Sec.
			INSTANT	N/A	N/A
11) MFN0097C-AH R.B. CLNG. UNIT FAN XFN65A EMERGENCY MOTOR	PRIMARY	XSW1DA1/6C	LONG TIME	525 Amps	≤ 30 Sec.
			SHORT TIME	1500 Amps	≤ 0.17 Sec.
			INSTANT	2250 Amps	≤ 0.09 Sec.
XSWIDA1 MAIN INCOMING	BACKUP	XSW1DA1/4B	LONG TIME	6300 Amps	≤ 12 Sec.
			SHORT TIME	9000 Amps	≤ 0.50 Sec.
			INSTANT	N/A	N/A
12) MFN0096A-AH R.B. CLNG. UNIT FAN XFN64A NORMAL MOTOR	PRIMARY	XSW1DA1/6B	LONG TIME	1260 Amps	≤ 30 Sec.
			SHORT TIME	5400 Amps	≤ 0.17 Sec.
			INSTANT	5400 Amps	≤ 0.09 Sec.
XSWIDA1 MAIN INCOMING	BACKUP	XSW1DA1/4B	LONG TIME	6300 Amps	≤ 12 Sec.
			SHORT TIME	9000 Amps	≤ 0.17 Sec.
			INSTANT	N/A	N/A
13) MFN0097A-AH R.B. CLNG. UNIT FAN XFN64A EMERGENCY MOTOR	PRIMARY	XSW1DA1/5C	LONG TIME	525 Amps	≤ 30 Sec.
			SHORT TIME	1500 Amps	≤ 0.17 Sec.
			INSTANT	2250 Amps	≤ 0.09 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
13) CONTINUED:					
XSWIDA1 MAIN INCOMING	BACKUP	XSW1DA1/4B	LONG TIME	6300 Amps	$\leq$ 12 Sec.
			SHORT TIME	9000 Amps	$\leq$ 0.50 Sec.
			INSTANT	N/A	N/A
14) MFN0096D-AH	PRIMARY	XSW1DB1/7C	LONG TIME	1260 Amps	$\leq$ 30 Sec.
R.B. CLNG. UNIT FAN			SHORT TIME	5400 Amps	$\leq$ 0.17 Sec.
XFN65B NORMAL MOTOR			INSTANT	5400 Amps	$\leq$ 0.09 Sec.
XSWIDB1 MAIN INCOMING	BACKUP	XSW1DB1/4B	LONG TIME	6300 Amps	$\leq$ 12 Sec.
			SHORT TIME	9000 Amps	$\leq$ 0.50 Sec.
			INSTANT	N/A	N/A
15) MFN0097D-AH	PRIMARY	XSW1DB1/6C	LONG TIME	525 Amps	$\leq$ 30 Sec.
R.B. CLNG. UNIT FAN			SHORT TIME	1500 Amps	$\leq$ 0.17 Sec.
XFN65B EMERGENCY MOTOR			INSTANT	2250 Amps	$\leq$ 0.09 Sec.
XSWIDB1 MAIN INCOMING	BACKUP	XSW1DB1/4B	LONG TIME	6300 Amps	$\leq$ 12 Sec.
			SHORT TIME	9000 Amps	$\leq$ 0.50 Sec.
			INSTANT	N/A	N/A
16) XHR0004A-HR	PRIMARY	XSW1DA2/5C	LONG TIME	315 Amps	$\leq$ 12 Sec.
H_2 RECOMBINER PWR. PNL.			SHORT TIME	N/A	N/A
TO RECOMBINER FEED			INSTANT	900 Amps	$\leq$ 0.09 Sec.
XSWIDA2 MAIN INCOMING	BACKUP	XSW1DA2/4B	LONG TIME	4800 Amps	$\leq$ 12 Sec.
			SHORT TIME	7200 Amps	$\leq$ 0.50 Sec.
			INSTANT	N/A	N/A
17) XHR0004B-HR	PRIMARY	XSW1DB2/5C	LONG TIME	315 Amps	$\leq$ 12 Sec.
H_2 RECOMBINER PWR. PNL.			SHORT TIME	N/A	N/A
TO RECOMBINER FEED			INSTANT	900 Amps	$\leq$ 0.09 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
17) CONTINUED:						
XSWIDB2 MAIN INCOMING		BACKUP	XSW1DB2/4B	LONG TIME	4800 Amps	≤ 12 Sec.
				SHORT TIME	7200 Amps	≤ 0.50 Sec.
				INSTANT	N/A	N/A
440 Vac CRDM PWR. CAB. 1AC, CONTROL BANK A,						
18) MECHANISM 1 -						
XCA1A-CR	A59-Fu13	PRIMARY	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A59-Fu17	BACKUP	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A51-Fu1	PRIMARY	XCA1A	≥ 1.4 Milliohms		N/A
XCA1A-CR	A57-Fu1	BACKUP	XCA1A	≥ 1.4 Milliohms		N/A
XCA1A-CR	A59-Fu21	PRIMARY	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A61-Fu45	BACKUP	XCA1A	≥ 6 Milliohms		N/A
19) MECHANISM 2 -						
XCA1A-CR	A59-Fu14	PRIMARY	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A59-Fu18	BACKUP	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A51-Fu2	PRIMARY	XCA1A	≥ 1.4 Milliohms		N/A
XCA1A-CR	A57-Fu2	BACKUP	XCA1A	≥ 1.4 Milliohms		N/A
XCA1A-CR	A59-Fu22	PRIMARY	XCA1A	≥ 6 Milliohms		N/A
XCA1A-CR	A61-Fu46	BACKUP	XCA1A	≥ 6 Milliohms		N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1AC, CONTINUED:					
20) MECHANISM 3 -					
XCA1A-CR	A59-Fu15	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A59-Fu19	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A52-Fu1	PRIMARY	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR	A58-Fu1	BACKUP	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR	A59-Fu23	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A61-Fu47	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
21) MECHANISM 4 -					
XCA1A-CR	A59-Fu16	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A59-Fu20	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A52-Fu2	PRIMARY	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR	A58-Fu2	BACKUP	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR	A59-Fu24	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A61-Fu48	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
440 Vac CRDM POWER CABINET 1AC, CONTROL BANK C, GROUP 1					
22) MECHANISM 1 -					
XCA1A-CR	A60-Fu25	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR	A60-Fu29	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1AC, CONTINUED:				
XCA1A-CR A53-Fu1	PRIMARY	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A57-Fu1	BACKUP	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A60-Fu33	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A61-Fu45	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
23) MECHANISM 2 -				
XCA1A-CR A60-Fu26	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A60-Fu30	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A53-Fu2	PRIMARY	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A57-Fu2	BACKUP	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A60-Fu34	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A61-Fu46	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
24) MECHANISM 3 -				
XCA1A-CR A60-Fu27	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A60-Fu31	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A54-Fu1	PRIMARY	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A58-Fu1	BACKUP	XCA1A	$\geq$ 1.4 Milliohms	N/A
XCA1A-CR A60-Fu35	PRIMARY	XCA1A	$\geq$ 6 Milliohms	N/A
XCA1A-CR A61-Fu47	BACKUP	XCA1A	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1AC, CONTINUED:					
25) MECHANISM 4 -					
XCA1A-CR	A60-Fu28	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A60-Fu32	BACKUP	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A54-Fu2	PRIMARY	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A58-Fu2	BACKUP	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A60-Fu36	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A61-Fu48	BACKUP	XCA1A	≥ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 1AC, SHUTDOWN BANK A, GROUP 1					
26) MECHANISM 1 -					
XCA1A-CR	A61-Fu41	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A60-Fu37	BACKUP	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A55-Fu1	PRIMARY	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A57-Fu1	BACKUP	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A61-Fu49	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A61-Fu45	BACKUP	XCA1A	≥ 6 Milliohms	N/A
27) MECHANISM 2 -					
XCA1A-CR	A61-Fu42	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A60-Fu38	BACKUP	XCA1A	≥ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1AC, CONTINUED:					
XCA1A-CR	A55-Fu2	PRIMARY	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A57-Fu2	BACKUP	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A61-Fu50	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A61-Fu46	BACKUP	XCA1A	≥ 6 Milliohms	N/A
28) MECHANISM 3 -					
XCA1A-CR	A61-Fu43	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A60-Fu39	BACKUP	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A56-Fu1	PRIMARY	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A58-Fu1	BACKUP	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A61-Fu51	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A61-Fu47	BACKUP	XCA1A	≥ 6 Milliohms	N/A
29) MECHANISM 4 -					
XCA1A-CR	A61-Fu44	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A60-Fu40	BACKUP	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A56-Fu2	PRIMARY	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A58-Fu2	BACKUP	XCA1A	≥ 1.4 Milliohms	N/A
XCA1A-CR	A61-Fu52	PRIMARY	XCA1A	≥ 6 Milliohms	N/A
XCA1A-CR	A61-Fu48	BACKUP	XCA1A	≥ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
440 VAC CRDM POWER CABINET 2AC, SHUTDOWN BANK A, GROUP 2					
30) MECHANISM 1 -					
XCA2A-CR	A59-Fu13	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A59-Fu17	BACKUP	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A51-Fu1	PRIMARY	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A57-Fu1	BACKUP	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A59-Fu21	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A61-Fu45	BACKUP	XCA2A	≥ 6 Milliohms	N/A
31) MECHANISM 2 -					
XCA2A-CR	A59-Fu14	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A59-Fu18	BACKUP	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A51-Fu2	PRIMARY	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A57-Fu2	BACKUP	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A59-Fu15	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A59-Fu19	BACKUP	XCA2A	≥ 6 Milliohms	N/A
32) MECHANISM 3 -					
XCA2A-CR	A59-Fu15	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A59-Fu19	BACKUP	XCA2A	≥ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 AC, CONTINUED:					
XCA2A-CR	A52-Fu11	PRIMARY	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A58-Fu1	BACKUP	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A59-Fu23	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A61-Fu47	BACKUP	XCA2A	≥ 6 Milliohms	N/A
33) MECHANISM 4 -					
XCA2A-CR	A59-Fu16	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A59-Fu20	BACKUP	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A52-Fu2	PRIMARY	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A58-Fu2	BACKUP	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A59-Fu24	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A61-Fu45	BACKUP	XCA2A	≥ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 2 AC, CONTROL BANK C, GROUP 2					
34) MECHANISM 1-					
XCA2A-CR	A60-F25	PRIMARY	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A60-Fu29	BACKUP	XCA2A	≥ 6 Milliohms	N/A
XCA2A-CR	A53-Fu1	PRIMARY	XCA2A	≥ 1.4 Milliohms	N/A
XCA2A-CR	A57-Fu1	BACKUP	XCA2A	≥ 1.4 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 AC, CONTINUED:				
XCA2A-CR A60-Fu33	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A61-Fu45	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
35) MECHANISM 2 -				
XCA2A-CR A60-Fu26	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A60-Fu30	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A53-Fu2	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A57-Fu2	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A60-Fu34	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A60-Fu46	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
36) MECHANISM 3 -				
XCA2A-CR A60-Fu27	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A60-Fu31	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A54-Fu1	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A58-Fu1	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A60-Fu35	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A61-Fu47	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM POW. CAB. 2AC, CONTINUED: 37) MECHANISM 4 -					
XCA2A-CR	A60-Fu28	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A60-Fu32	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A54-Fu2	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR	A58-Fu2	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR	A60-Fu36	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A61-Fu48	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 2AC, SHUTDOWN BANK A, GROUP 2 38) MECHANISM 1 -					
XCA2A-CR	A61-Fu41	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A60-Fu37	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A55-Fu1	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR	A57-Fu2	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR	A61-Fu49	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A61-Fu45	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
39) MECHANISM 2 -					
XCA2A-CR	A61-Fu42	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR	A60-Fu38	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 AC, CONTINUED:				
XCA2A-CR A55-Fu2	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A57-Fu2	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A61-Fu50	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A61-Fu46	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
40) MECHANISM 3 -				
XCA2A-CR A61-Fu43	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A60-Fu39	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A56-Fu1	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A58-Fu1	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A61-Fu51	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A61-Fu47	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
41) MECHANISM 4-				
XCA2A-CR A61-Fu44	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A60-Fu40	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A56-Fu2	PRIMARY	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A58-Fu2	BACKUP	XCA2A	$\geq$ 1.4 Milliohms	N/A
XCA2A-CR A61-Fu52	PRIMARY	XCA2A	$\geq$ 6 Milliohms	N/A
XCA2A-CR A61-Fu48	BACKUP	XCA2A	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
440 VAC CRDM POWER CABINET 1BD, SHUTDOWN BANK B, GROUP 1					
42) MECHANISM 1 -					
XCA1B-CR	A59-Fu13	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A59-Fu17	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A51-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR	A57-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR	A59-Fu21	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A61-Fu45	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
43) MECHANISM 2 -					
XCA1B-CR	A59-Fu14	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A59-Fu18	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A51-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR	A57-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR	A59-Fu22	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A61-Fu46	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
44) MECHANISM 3 -					
XCA1B-CR	A59-Fu15	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR	A59-Fu19	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1BD, CONTINUED:				
XCA1B-CR A52-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A59-Fu23	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu47	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
45) MECHANISM 4 -				
XCA1B-CR A59-Fu16	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A59-Fu20	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A52-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A59-Fu24	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu48	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET, CONTROL BANK D, GROUP 1				
46) MECHANISM 1 -				
XCA1B-CR A60-Fu25	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu29	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A53-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A57-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1BD, CONTINUED:				
XCA1B-CR A60-Fu33	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu45	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
47) MECHANISM 2 -				
XCA1B-CR A60-Fu26	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu30	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A53-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A57-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A60-Fu34	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu46	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
48) MECHANISM 3 -				
XCA1B-CR A60-Fu27	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu31	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A54-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A60-Fu35	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu47	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1BD, CONTINUED:				
49) MECHANISM 4 -				
XCA1B-CR A60-Fu28	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu32	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A54-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A60-Fu36	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu48	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 1BD, SHUTDOWN BANK B, GROUP 1				
50) MECHANISM 1 -				
XCA1B-CR A61-Fu41	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu37	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A55-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A57-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A61-Fu49	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu45	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
51) MECHANISM 2 -				
XCA1B-CR A61-Fu42	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu38	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 1BD, CONTINUED:				
XCA1B-CR A55-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A57-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A61-Fu50	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu46	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
52) MECHANISM 3 -				
XCA1B-CR A61-Fu43	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu39	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A56-Fu1	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu1	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A61-Fu51	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu47	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
53) MECHANISM 4 -				
XCA1B-CR A61-Fu4[illegible]	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A60-Fu40	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A56-Fu2	PRIMARY	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A58-Fu2	BACKUP	XCA1B	$\geq$ 1.4 Milliohms	N/A
XCA1B-CR A61-Fu52	PRIMARY	XCA1B	$\geq$ 6 Milliohms	N/A
XCA1B-CR A61-Fu48	BACKUP	XCA1B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
440 VAC CRDM POWER CABINET 2 BD, CONTROL BANK B, GROUP 2					
54) MECHANISM 1 -					
XCA2B-CR	A59-Fu13	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A59-Fu17	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A51-Fu1	PRIMARY	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR	A57-Fu1	BACKUP	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR	A59-Fu21	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A61-Fu45	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
55) MECHANISM 2 -					
XCA2B-CR	A59-Fu14	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A59-Fu18	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A51-Fu2	PRIMARY	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR	A57-Fu2	BACKUP	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR	A59-Fu22	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A61-Fu46	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
56) MECHANISM 3 -					
XCA2B-CR	A59-Fu15	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR	A59-Fu19	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 BD, CONTINUED:					
XCA2B-CR	A52-Fu1	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A58-Fu1	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A59-Fu23	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu47	BACKUP	XCA2B	≥ 6 Milliohms	N/A
57) MECHANISM 4 -					
XCA2B-CR	A59-Fu16	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A59-Fu20	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A52-Fu2	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A58-Fu2	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A59-Fu24	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu48	BACKUP	XCA2B	≥ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 2BD, CONTROL BANK D, GROUP 2					
58) MECHANISM 1 -					
XCA2B-CR	A60-Fu25	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu29	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A53-Fu1	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A57-Fu1	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 BD, CONTINUED:					
XCA2B-CR	A60-Fu33	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu45	BACKUP	XCA2B	≥ 6 Milliohms	N/A
59) MECHANISM 2 -					
XCA2B-CR	A60-Fu26	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu30	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A53-Fu2	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A57-Fu2	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A60-Fu34	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu46	BACKUP	XCA2B	≥ 6 Milliohms	N/A
60) MECHANISM 3 -					
XCA2B-CR	A60-Fu27	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu31	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A54-Fu1	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A58-Fu1	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A60-Fu35	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu47	BACKUP	XCA2B	≥ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION		DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 BD, CONTINUED:					
61) MECHANISM 4 -					
XCA2B-CR	A60-Fu28	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu32	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A54-Fu2	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A58-Fu2	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A60-Fu36	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu48	BACKUP	XCA2B	≥ 6 Milliohms	N/A
440 VAC CRDM POWER CABINET 2BD, SHUTDOWN BANK D, GROUP 2					
62) MECHANISM 1-					
XCA2B-CR	A61-Fu41	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu37	BACKUP	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A55-Ful	PRIMARY	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A57-Ful	BACKUP	XCA2B	≥ 1.4 Milliohms	N/A
XCA2B-CR	A61-Fu49	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A61-Fu45	BACKUP	XCA2B	≥ 6 Milliohms	N/A
63) MECHANISM 2 -					
XCA2B-CR	A61-Fu42	PRIMARY	XCA2B	≥ 6 Milliohms	N/A
XCA2B-CR	A60-Fu38	BACKUP	XCA2B	≥ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
CRDM PWR. CAB. 2 BD, CONTINUED:				
XCA2B-CR A55-Fu2	PRIMARY	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A57-Fu2	BACKUP	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A61-Fu50	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A61-Fu46	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
64) MECHANISM 3 -				
XCA2B-CR A61-Fu43	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A60-Fu39	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A56-Fu1	PRIMARY	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A58-Fu1	BACKUP	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A61-Fu51	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A61-Fu47	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
65) MECHANISM 4 -				
XCA2B-CR A61-Fu44	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A60-Fu40	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A56-Fu2	PRIMARY	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A58-Fu2	BACKUP	XCA2B	$\geq$ 1.4 Milliohms	N/A
XCA2B-CR A61-Fu48	PRIMARY	XCA2B	$\geq$ 6 Milliohms	N/A
XCA2B-CR A61-Fu48	BACKUP	XCA2B	$\geq$ 6 Milliohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
125 VDC 66) DPN8007C-ED Emergency LTG. PNL. 7	PRIMARY	DPN1HX/14	120 Amps	$\leq$ 100 Sec.
	BACKUP	XPN5262 (FUSE)	$\geq$ 1.4 Milliohms	N/A
120 VAC MISC. 67) XBJ0002-IC/INCORE THERMOCOUPLE REF. JUNCT. BOX 2	PRIMARY	APN1FX1/25	45 Amps	$\leq$ 100 Sec.
	BACKUP	XPN5261 (FUSE)	$\geq$ 4 Milliohms	N/A
68) XBJ0001-IC/INCORE THERMOCOUPLE REF. JUNCT. BOX 1	PRIMARY	APN1FX1/24	45 Amps	$\leq$ 100 Sec.
	BACKUP	XPN5261 (FUSE)	$\geq$ 4 Milliohms	N/A
69) XPN7060-CR/ROD POSITION INDICATION PNL. 1	PRIMARY	APN1FC1/2	120 Amps	$\leq$ 100 Sec.
	BACKUP	XPN5272 (FUSE)	$\geq$ 1.4 Millohms	N/A
70) XPN7061-CR/ROD POSITION INDICATION PNL. 2	PRIMARY	APN1FC1/4	120 Amps	$\leq$ 100 Sec.
	BACKUP	XPN5272 (FUSE)	$\geq$ 1.4 Milliohms	N/A
71) APN5915-EV/TRANSMITTER PWR. SUPPLY CAB. NO. 3	PRIMARY	APN5906/25	60 Amps	$\leq$ 100 Sec.
	BACKUP	APN5915 (FUSE)	$\geq$ 4 Millohms	N/A

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
460 VAC MISC.				
72) PRESS. HTR. GROUP 23,49,50 -RC	PRIMARY	APN4101/1	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
73) PRESS. HTR. GROUP 28, 55, 56 -RC	PRIMARY	APN4101/2	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
74) PRESS. HTR. GROUP 29, 57, 58 -RC	PRIMARY	APN4101/3	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
75) PRESS. HTR. GROUP 26, 53, 54 -RC	PRIMARY	APN4101/4	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
76) PRESS. HTR, GROUP 21, 47, 48 -RC	PRIMARY	APN4101/5	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
77) PRESS. HTR. GROUP 1, 2, 22 - RC	PRIMARY	APN4101/6	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
78) PRESS. HTR. GROUP 5, 6, 27 -RC	PRIMARY	APN4101/7	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.
79) PRESS. HTR. GROUP 3, 4, 35 -RC	PRIMARY	APN4101/8	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	$\leq$ 1 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
460 VAC MISC. CONTINUED:				
80) PRESS. HTR. GROUP 7, 8, 30 -RC	PRIMARY	APN4101/9	270 Amps	≤ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	≤ 1 Sec.
81) PRESS. HTR. GROUP 24, 51, 52 -RC	PRIMARY	APN4101/10	270 Amps	≤ 75 Sec.
	BACKUP	APN4101/MN.	4000 Amps	≤ 1 Sec.
82) PRESS. HTR. GROUP 17, 18, 42 -RC	PRIMARY	APN4102/1	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
83) PRESS. HTR. GROUP 19, 20, 45 -RC	PRIMARY	APN4102/2	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
84) PRESS. HTR, GROUP 38, 67, 68 -RC	PRIMARY	APN4102/3	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
85) PRESS. HTR. GROUP 39, 69, 70 - RC	PRIMARY	APN4102/4	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
86) PRESS. HTR. GROUP 44, 75, 76 -RC	PRIMARY	APN4102/5	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
87) PRESS. HTR. GROUP 41, 71, 72 -RC	PRIMARY	APN4102/6	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
460 VAC MISC. CONTINUED:				
88) PRESS. HTR. GROUP 43, 73, 74 -RC	PRIMARY	APN4102/7	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
89) PRESS. HTR. GROUP 15, 16, 40 -RC	PRIMARY	APN4102/8	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
90) PRESS. HTR. GROUP 46, 77, 78 -RC	PRIMARY	APN4102/9	270 Amps	≤ 75 Sec.
	BACKUP	APN4102/MN.	4000 Amps	≤ 1 Sec.
91) PRESS. HTR. GROUP 9, 10, 32 -RC	PRIMARY	APN4103/1	270 Amps	≤ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	≤ 2 Sec.
92) PRESS. HTR, GROUP 11, 12, 35 -RC	PRIMARY	APN4103/2	270 Amps	≤ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	≤ 2 Sec.
93) PRESS. HTR. GROUP 31, 59, 60 - RC	PRIMARY	APN4103/3	270 Amps	≤ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	≤ 2 Sec.
94) PRESS. HTR. GROUP 36, 65, 66 -RC	PRIMARY	APN4103/4	270 Amps	≤ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	≤ 2 Sec.
95) PRESS. HTR. GROUP 13, 14, 37 -RC	PRIMARY	APN4103/5	270 Amps	≤ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	≤ 2 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
460 VAC MISC. CONTINUED:				
96) PRESS. HTR. GROUP 33, 61, 62 -RC	PRIMARY	APN4103/6	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	$\leq$ 2 Sec.
97) PRESS. HTR. GROUP 34, 63, 64 -RC	PRIMARY	APN4103/7	270 Amps	$\leq$ 75 Sec.
	BACKUP	APN4103/MN.	3000 Amps	$\leq$ 2 Sec.
480 VAC MOTOR CONTROL CENTERS				
98) XFN0066A-AH/RB CHARCOAL CLEANUP UNIT FAN A	PRIMARY	XMC1A3X/10GK	1500 Amps	N/A
	BACKUP	XMC1A3X/10GK	210 Amps	$\leq$ 200 Sec.
99) XPP0138-ND/LEAK DETECTION SUMP PUMP	PRIMARY	XMC1A3X/41L	225 Amps	N/A
	BACKUP	XMC1A3X/41L	45 Amps	$\leq$ 100 Sec.
100) XDO 0001-IC/TERM. BOX FOR INCORE NEUTRON DETECTOR DRIVES A,B,C,D,E	PRIMARY	XMC1A3X/3EG	45 Amps	$\leq$ 100 Sec.
	BACKUP	XMC1A3X/3EG	45 Amps	$\leq$ 100 Sec.
101) XVG9593A-CC/MOV REACT. COOL Pump A THERMAL Barrier	PRIMARY	XMC1A3X/9IM	33 Amps	N/A
	BACKUP	XMC1A3X/9IM	45 Amps	$\leq$ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MISC. CONTINUED:				
102) XPP0149-SW/DRPI COOLING UNIT BSTR. PUMP	PRIMARY	XMC1A3X/11AD	225 Amps	N/A
	BACKUP	XMC1A3X/11AD	45 Amps	$\leq$ 100 Sec.
103) XFN0068A-AH/SECONDARY COMPT. (LOOP A) CLG. FAN A	PRIMARY	XMC1A3X/11EH	720 Amps	N/A
	BACKUP	XMC1A3X/11EH	150 Amps	$\leq$ 200 Sec.
104) XFN0069A-AH/SECONDARY COMPT. (LOOP B) CLG. FAN A	PRIMARY	XMC1A3X/10CF	720 Amps	N/A
	BACKUP	XMC1A3X/10CF	150 Amps	$\leq$ 200 Sec.
105) XFN0070A-AH/SECONDARY COMPT. (LOOP C) CLG. FAN A	PRIMARY	XMC1A3X/9AD	720 Amps	N/A
	BACKUP	XMC1A3X/9AD	150 Amps	$\leq$ 200 Sec.
106) XPP0051A-WL/R.C. DRAIN TANK - PUMP A	PRIMARY	XMC1A3X/6CG	720 Amps	N/A
	BACKUP	XMC1A3X/6CG	210 Amps	$\leq$ 200 Sec.
107) XPP0059A-ND/INCORE INSTR. CHASE SUMP PUMP A	PRIMARY	XMC1A3X/4AD	225 Amps	N/A
	BACKUP	XMC1A3X/4AD	45 Amps	$\leq$ 100 Sec.
108) XPP0115A-ND/R.B. SUMP PUMP A	PRIMARY	XMC1A3X/4EH	87 Amps	N/A
	BACKUP	XMC1A3X/4EH	45 Amps	$\leq$ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC*, CONTINUED:				
109) XTF9003-EM/RECEPTACLE TRANSFORMER 3	PRIMARY	XMC1A3X/1AC	240 Amps	$\leq$ 200 Sec.
	BACKUP	XMC1A3X/1AC	240 Amps	$\leq$ 200 Sec.
110) XVG9576-CC/MOV, ISOLATION RCDT	PRIMARY	XMC1B2X/5IM	225 Amps	N/A
	BACKUP	XMC1B2X/5IM	60 Amps	$\leq$ 100 Sec.
111) XVG9583-CC/MOV, GATE EXCESS LETDOWN HX	PRIMARY	XMC1B2X/3AE	225 Amps	N/A
	BACKUP	XMC1B2X/3AE	45 Amps	$\leq$ 100 Sec.
112) XFN0007B-AH/REFUELING WATER SURFACE SUPPLY FAN B	PRIMARY	XMC1B3X/9AD	87 Amps	N/A
	BACKUP	XMC1B3X/9AD	45 Amps	$\leq$ 100 Sec.
113) XFN0066B-AH/R.B. CHARCOAL CLEANUP UNIT FAN B	PRIMARY	XMC183X/10FJ	1,500 Amps	N/A
	BACKUP	XMC1B3X/10FJ	210 Amps	$\leq$ 200 Sec.
114) XXP0059B-ND/INCORE INSTR. CHASE SUMP PUMP B	PRIMARY	XMC1B3X/3EH	225 Amps	N/A
	BACKUP	XMC1B3X/3EH	45 Amps	$\leq$ 100 Sec.
115) XPP0115B-ND/R.B. SUMP PUMP B	PRIMARY	XMC1B3X/3IL	87 Amps	N/A
	BACKUP	XMC1B3X/3IL	45 Amps	$\leq$ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
116) APN4005-EM/480 VAC POWER PNL.-FEEDER	PRIMARY	XMC1B3X/5CE	270 Amps	≤ 200 Sec.
	BACKUP	XMC1B3X/5CE	270 Amps	≤ 200 Sec.
117) XTF8009-EM/TRANSFORMER FOR LIGHTING PNL. 9 (NORMAL LTG.)	PRIMARY	XMC1B3X/5HJ	240 Amps	≤ 200 Sec.
	BACKUP	XMC1B3X/5HJ	240 Amps	≤ 200 Sec.
118) XFN0068B-AH/SECONDARY COMPT. (LOOP A)CLG FAN B	PRIMARY	XMC1B3Y/3AD	720 Amps	N/A
	BACKUP	XMC1B3Y/3AD	180 Amps	≤ 200 Sec.
119) XFN0069B-AH/SECONDARY COMPT. (LOOP B) CLG FAN B	PRIMARY	XMC1B3Y/3EH	720 Amps	N/A
	BACKUP	XMC1B3Y/3EH	180 Amps	≤ 200 Sec.
120) XFN0070B-AH/SECONDARY COMPT. (LOOP C) CLG. FAN B	PRIMARY	XMC1B3Y/3IL	720 Amps	N/A
	BACKUP	XMC1B3Y/3IL	180 Amps	≤ 200 Sec.
121) APN4012-EM/WELDING RECEPT. PWR. PNL.	PRIMARY	XMC1B3Y/7GL	600 Amps	≤ 300 Sec.
	BACKUP	XMC1B3Y/7GL	600 Amps	≤ 300 Sec.
122) APN4013-EM/WELDING RECEPT. PWR. PNL.	PRIMARY	XMC1B3Y/9GL	600 Amps	≤ 300 Sec.
	BACKUP	XMC1B3Y/9GL	600 Amps	≤ 300 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
123) R.C. PUMP C HEATER-RC	PRIMARY	XMC1B3Y/5AD	225 Amps	N/A
	BACKUP	XMC1B3Y/5AD	45 Amps	≤ 100 Sec.
124) R.C. PUMP C, OIL LIFT PMP -RC	PRIMARY	XMC1B3Y/5EH	450 Amps	N/A
	BACKUP	XMC1B3Y/5EH	75 Amps	≤ 100 Sec.
125) XPP0051B-WL/R.C. DRAIN TANK PUMP B	PRIMARY	XMC1B3Y/8GK	720 Amps	N/A
	BACKUP	XMC1B3Y/8GK	210 Amps	≤ 200 Sec.
126) XTF8008-EM/TRANSFORMER FOR LTG. PNL. 8, UNDERWATER LIGHTING	PRIMARY	XMC1B3Y/4HJ	240 Amps	≤ 200 Sec.
	BACKUP	XMC1B3Y/4HJ	240 Amps	≤ 200 Sec.
127) XVG9593B-CC/MOV, RC PUMP B THERMAL BARRIER	PRIMARY	XMC1B3Y/4AE	225 Amps	N/A
	BACKUP	XMC1B3Y/4AE	45 Amps	≤ 100 Sec.
128) XFN0068C-AH/SECONDARY COMPT. (LOOP A) CLG FAN C	PRIMARY	XMC1C3X/8AE	1500 Amps	N/A
	BACKUP	XMC1C3X/8AE	210 Amps	≤ 200 Sec.
129) R.C. PUMP B HEATER -RC	PRIMARY	XMC1C3X/5EH	225 Amps	N/A
	BACKUP	XMC1C3X/5EH	45 Amps	≤ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
130) R.C. PUMP B OIL LIFT PUMP -RC	PRIMARY	XMC1C3X/6EH	450 Amps	N/A
	BACKUP	XMC1C3X/6EH	75 Amps	≤ 100 Sec.
131) XFN0069C-AH/SECONDARY COMPT. (LOOP B)CLG. FAN C	PRIMARY	XMC1C3X/2EH	720 Amps	N/A
	BACKUP	XMC1C3X/2EH	150 Amps	≤ 200 Sec.
132) XFN0070C-AH/SECONDARY COMPT. (LOOP C) CLG FAN C	PRIMARY	XMC1C3X/2IL	720 Amps	N/A
	BACKUP	XMC1C3X/2IL	150 Amps	≤ 200 Sec.
133) XTF8006-EM/TRANSFORMER FOR LTG. PNL. 6, NORMAL LIGHTING	PRIMARY	XMC1C3X/4CE	240 Amps	≤ 200 Sec.
	BACKUP	XMC1C3X/4CE	240 Amps	≤ 200 Sec.
134) XVG9593C-CC/MOV, R.C. PUMP C THERMAL BARRIER	PRIMARY	XMC1C3X/4IM	225 Amps	N/A
	BACKUP	XMC1C3X/4IM	45 Amps	≤ 100 Sec.
135) XFN0107-VL/CONTROL ROD POSIT. DATA CAB. CLG FAN	PRIMARY	XMC1C3X/2AD	450 Amps	N/A
	BACKUP	XMC1C3X/2AD	75 Amps	≤ 100 Sec.
136) R.C. PUMP A HEATER -RC	PRIMARY	XMC1C3X/5AD	225 Amps	N/A
	BACKUP	XMC1C3X/5AD	45 Amps	≤ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
137) R.C. PUMP A OIL LIFT PUMP -RC	PRIMARY	XMC1C3X/6AD	450 Amps	N/A
	BACKUP	XMC1C3X/6AD	75 Amps	≤ 100 Sec.
138) XFN0007A-AH/REFUELING WATER SURFACE SUPPLY FAN A	PRIMARY	XMC1C3X/3EH	87 Amps	N/A
	BACKUP	XMC1C3X/3EH	45 Amps	≤ 100 Sec.
139) XFN0008-AH/REFUELING WATER SURFACE EXHAUST FAN	PRIMARY	XMC1C3X/3IL	720 Amps	N/A
	BACKUP	XMC1C3X/3IL	150 Amps	≤ 200 Sec.
140) XVG9605-CC/MOV - R.B.	PRIMARY	XMC1DA2X/3IM	45 Amps	≤ 100 Sec.
	BACKUP	XMC1DA2X/3IM	45 Amps	≤ 100 Sec.
141) XVG8701A-RH/RHR LOOP 1 INLET ISOLATION VLV.	PRIMARY	XMC1DA2X/7FJ	225 Amps	N/A
	BACKUP	XMC1DA2X/7FJ	45 Amps	≤ 100 Sec.
142) XVG8808A-SI/ACCUMULATOR A ISOLATION VLV.	PRIMARY	XMC1DA2X/8AE	300 Amps	≤ 200 Sec.
	BACKUP	XMC1DA2X/8AE	300 Amps	≤ 200 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
143) XVG8808C-SI/ACCUMULATOR C ISOLATION VLV.	PRIMARY	XMC1DA2X/8FJ	300 Amps	≤ 200 sec.
	BACKUP	XMC1DA2X/8FJ	300 Amps	≤ 200 sec.
144) XVG8000B-RC/PRESS. RELIEF ISOLATION VLV.	PRIMARY	XMC1DA2X/6IM	225 Amps	N/A
	BACKUP	XMC1DA2X/6IM	45 Amps	≤ 100 sec.
145) XVG3108A-SW/R.B. Recirc. Unit A -Isolation Vlv.	PRIMARY	XMC1DA2Y/16IM	45 Amps	≤ 100 sec.
	BACKUP	XMC1DA2Y/16IM	45 Amps	≤ 100 sec.
146) XVG3108B-SW/R.B. Recirc. Unit B - Isolation Vlv.	PRIMARY	XMC1DA2Y/15CG	45 Amps	≤ 100 sec.
	BACKUP	XMC1DA2Y/15CG	45 Amps	≤ 100 sec.
147) XVG3109A-SW/R.B. Recirc. Unit A - Isolation Vlv.	PRIMARY	XMCIDA2Y/15HL	45 Amps	≤ 100 sec.
	BACKUP	XMC1DA2Y/15HL	45 Amps	≤ 100 sec.
148) XVG3109B-SW/R.B. Recirc. Unit B - Isolation Vlv.	PRIMARY	XMC1DA2Y/14CG	45 Amps	≤ 100 sec.
	BACKUP	XMC1DA2Y/14CG	45 Amps	≤ 100 sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
149) XVT8112-CS/SEAL WATER RETURN ISOLATION VLV.	PRIMARY	XMC1DA2Y/3IM	45 Amps	≤ 100 Sec.
	BACKUP	XMC1DA2Y/3IM	45 Amps	≤ 100 Sec.
150) XVG8701B-RH/RHR LOOP 3 INLET ISOLATION VLV.	PRIMARY	XMC1DA2Y/18IM	225 Amps	N/A
	BACKUP	XMC1DA2Y/18IM	45 Amps	≤ 100 Sec.
151) XVG8000C-RC/PRESS. RELIEF ISOLATION VLV.	PRIMARY	XMC1DB2X/8DH	225 Amps	N/A
	BACKUP	XMC1DB2X/8DH	45 Amps	≤ 100 Sec.
152) XVG3108C-SW/R.B. RECIRC UNIT C ISOLATION VLV.	PRIMARY	XMC1DB2Y/18IM	45 Amps	≤ 100 Sec.
	BACKUP	XMC1DB2Y/18IM	45 Amps	≤ 100 Sec.
153) XVG3108D-SW/R.B.RECIRC. UNIT D ISOLATION VLV.	PRIMARY	XMC1DB2Y/19IM	45 Amps	≤ 100 Sec.
	BACKUP	XMC1DB2Y/19IM	45 Amps	≤ 100 Sec.
154) XVG3109C-SW/R.B. RECIRC UNIT C ISOLATION VLV.	PRIMARY	XMC1DB2Y/20IM	45 Amps	≤ 100 Sec.
	BACKUP	XMCIDB2Y/20IM	45 Amps	≤ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
155) XVG 3109D-SW/R.B. RECIRC. UNIT D ISOLATION VLV.	PRIMARY	XMC1DB2Y/21IM	45 Amps	≤ 100 Sec.
	BACKUP	XMC1DB2Y/21IM	45 Amps	≤ 100 Sec.
156) XVG 8702A-RH/RHR LOOP 1 INLET ISOLATION VLV.	PRIMARY	XMC1DB2Y/4AE	225 Amps	N/A
	BACKUP	XMC1DB2Y/4AE	45 Amps	≤ 100 Sec.
157) XVG 8702B-RH/RHR LOOP 3 INLET ISOLATION VLV.	PRIMARY	XMC1DB2Y/4FJ	225 Amps	N/A
	BACKUP	XMC1DB2Y/4FJ	45 Amps	≤ 100 Sec.
158) XVG 8808B-SI/ACCUMULATOR B ISOLATION VLV.	PRIMARY	XMC1DB2Y/16IM	300 Amps	≤ 200 Sec.
	BACKUP	XMC1DB2Y/16IM	300 Amps	≤ 200 Sec.
159) XVG 8000A-RC/PRESS. RELIEF ISOLATION VLV.	PRIMARY	XMC1DB2Y/3IM	225 Amps	N/A
	BACKUP	XMC1DB2Y/3IM	45 Amps	≤ 100 Sec.
160) XVG 8095A-RC/REACTOR HEAD VENT VLV. TO PRESS. RELIEF TANK	PRIMARY	XMC1DA2X/5IM	225 Amps	N/A
	BACKUP	XMC1DA2X/5IM	45 Amps	≤ 100 Sec.

TABLE 3.8-1 (continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT	RESPONSE TIME
480 VAC MCC, CONTINUED:				
161) XVG 8095B-RC/REACTOR HEAD VENT VLV. TO PRESS. RELIEF TANK	PRIMARY	XMC1DB2Y/23FJ	225 Amps	N/A
	BACKUP	XMC1DB2Y/23FJ	60 Amps	$\leq$ 100 Sec.
162) XVG 8096A-RC/REACTOR HEAD VENT VLV. TO PRESS. RELIEF TANK	PRIMARY	XMC1DA2X/7AE	225 Amps	N/A
	BACKUP	XMC1DA2X/7AE	60 Amps	$\leq$ 100 Sec.
163) XVG 8096B-RC/REACTOR HEAD VENT VLV. TO PRESS. RELIEF TANK	PRIMARY	XMC1DB2Y/12IM	225 Amps	N/A
	BACKUP	XMC1DB2Y/12IM	60 Amps	$\leq$ 100 Sec.
164) XVG 7503-AC/CRDM COOLING WATER OUTLET VLV.	PRIMARY	XMC1DA2X/11IM	225 Amps	N/A
	BACKUP	XMC1DA2X/11IM	60 Amps	$\leq$ 100 Sec.
165) XVG 7502-AC/CRDM COOLING WATER INLET VLV.	PRIMARY	XMC1DB2X/7IM	225 Amps	N/A
	BACKUP	XMC1DB2X/7IM	60 Amps	$\leq$ 100 Sec.

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection and 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

4.8.4.2 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a TRIP ACTUATING DEVICE OPERATIONAL 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 CALIBRATION of a representative sample of at least 25% of:

   1. All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

   2. All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor operator when the thermal overload is OPERABLE and not bypassed, 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 (YES/NO)
XVB9503A - CC	RHR HX A ISOLATION VALVE	NO
XVB9503B - CC	RHR HX B ISOLATION VALVE	NO
XVG9568 - CC	Comp. Cool To RB MOV	YES
XVG9600 - CC	RC Pump At RB MOV	YES
XVG9605 - CC	RB, CC Return MOV	YES
XVG9606 - CC	RB, CC Return MOV	YES
XVG0115B - CS	RWST To CHARGING Pp. VALVE	YES
XVG0115C - CS	VCT TO CHARGING Pp. ISOL. VALVE	YES
XVG0115D - CS	RWST TO CHARGING Pp. VALVE	YES
XVG0115E - CS	VCT TO CHARGING Pp. ISOL. VALVE	YES
XVG8106 - CS	CHARGING Pump MINIFLOW VALVE	NO
XVG8107 - CS	RCS CHARGING LINE VALVE	YES
XVG8108 - CS	RCS CHARGING LINE VALVE	YES
XVG8130A - CS	CHARGING Pp. SUCTION HEADER ISOL. Vv.	NO
XVG8130B - CS	CHARGING Pp. SUCTION HEADER ISOL. Vv.	NO
XVG8131A - CS	CHARGING Pp. SUCTION HEADER ISOL Vv.	NO
XVG8131B - CS	CHARGING Pp. SUCTION HEADER ISOL. Vv.	NO
XVG8132A - CS	CHARGING Pp. DISCHARGE HEADER ISOL Vv.	NO
XVG8132B - CS	CHARGING Pp. DISCHARGE HEADER ISOL Vv.	NO
XVG8133A - CS	CHARGING Pp. DISCHARGE HEADER ISOL Vv.	NO
XVG8133B - CS	CHARGING Pp. DISCHARGE HEADER ISOL Vv.	NO
XVT8100 - CS	SEAL WATER RETURN ISOL VALVE	YES
XVT8109A - CS	CHARGING Pp. A MINIFLOW ISOL Vv.	NO
XVT8109B - CS	CHARGING Pp. B MINIFLOW ISOL Vv.	NO
XVT8109C - CS	CHARGING Pp. C MINIFLOW ISOL Vv.	NO
XVT8112 - CS	SEAL WATER RETURN ISOL Vv.	YES
XVG6797 - FS	FIRE SERVICE CONTAINMENT ISOL Vv.	YES
XVK1633A - FW	CHEMICAL FEED ISOL VALVE	YES
XVK1633B - FW	CHEMICAL FEED ISOL VALVE	YES
XVK1633C - FW	CHEMICAL FEED ISOL VALVE	YES
XVG2802A - MS	EFWP MAIN STEAM BLOCK	YES
XVG2802B - MS	EFWP MAIN STEAM BLOCK	YES
XVT2813 - MS	MAIN STEAM TO EFWP DRAIN	YES
XVG8706A - RH	RHRS TO CHARGING Pump VALVE	NO
XVG8706B - RH	RHRS TO CHARGING Pump Valve	NO
XVT0602A - RH	RHR Pump A MINIFLOW VALVE	NO
XVT0602B - RH	RHR Pump B MINIFLOW VALVE	NO
XVG8801A - SI	BORON INJ. TANK DISCHARGE VALVE	YES
XVG8801B - SI	BORON INJ. TANK DISCHARGE VALVE	YES
XVG8803A - SI	BORON INJ. TANK INLET ISOL. Vv.	YES
XVG8803B - SI	BORON INJ. TANK INLET ISOL Vv.	YES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
XVG8808A - SI	ACCUMULATOR A ISOL VALVE	YES
XVG8808B - SI	ACCUMULATOR B ISOL VALVE	YES
XVG8808C - SI	ASSUMULATOR C ISOL VALVE	YES
XVG8809A - SI	RWS TO RHR Pp. A ISOL Vv.	NO
XVG8809B - SI	RWS TO RHR Pp. B ISOL Vv.	NO
XVG8811A - SI	RECIRC SUMP TO RHR Pp. A ISOL Vv.	YES
XVG8811B - SI	RECIRC SUMP TO RHR Pp. B ISOL Vv.	YES
XVG8812A - SI	RECIRC SUMP TO RHR Pp. A ISOL Vv.	YES
XVG8812B - SI	RECIRC SUMP TO RHR Pp. B ISOL Vv.	YES
XVG8884 - SI	HIGH HEAD TO HOT LEG INJECTION HEADER ISOL Vv.	NO
XVG8885 - SI	HIGH HEAD TO COLD LEG INJECTION HEADER ISOL Vv.	NO
XVG8886 - SI	HIGH HEAD TO HOT LEG INJECTION HEADER ISOL Vv.	NO
XVG8887A - SI	LOW HEAD INJ. TO HOT LEG RECIRC LINE Vv.	NO
XVG8887B - SI	LOW HEAD INJ. TO HOT LEG RECIRC. LINE Vv.	NO
XVG8888A - SI	LOW HEAD TO COLD LEG INJECTION ISOL Vv., RHR Pump A	NO
XVG8888B - SI	LOW HEAD TO COLD LEG CROSS TIE VALVE	NO
XVG8889 - SI	LOW HEAD TO HOT LEG INJECTION RECIRC LINE ISOL Vv.	NO
XVG3001A - SP	RSWT TO RB SPRAY Pp. A SUCTION Vv.	YES
XVG3001B - SP	RSWT TO RB SPRAY Pp. B SUCTION Vv.	YES
XVG3002A - SP	NOAH TANK TO RB SPRAY Pp. A SUCTION Vv.	YES
XVG3002B - SP	NOAH TANK TO RB SPRAY Pp. B SUCTION Vv.	YES
XVG3003A - SP	SPRAY HEADERS ISOL CIRC. A Vv.	YES
XVG3003B - SP	SPRAY HEADERS ISOL CIRC. B Vv.	YES
XVB3106A - SW	SW BOOSTER PUMP A DISCHARGE	YES
XVB3106B - SW	SW BOOSTER PUMP B DISCHARGE	YES
XVB3110A - SW	IND. COOLING TO A RB COOLERS	YES
XVB3110B - SW	IND. COOLING TO B RB COOLERS	YES
XVG3103A - SW	A COOLERS OUTLET	YES
XVG3103B - SW	B COOLERS OUTLET	YES
XVG3107A - SW	A COOLERS TO SW POND	YES
XVG3107B - SW	B COOLERS TO SW POND	YES
XVG3111A - SW	A RB COOLERS TO IND. COOLING	YES
XVG3111B - SW	B RB COOLERS TO IND. COOLING	YES
XVG3112A - SW	A RB COOLERS TO IND. COOLING	YES
XVG3112B - SW	B RB COOLERS TO IND. COOLING	YES
XVG3108A - SW	RB COOLER 1A INLET	YES
XVG3108B - SW	RB COOLER 2A INLET	YES
XVG3108C - SW	RB COOLER 1B INLET	YES
XVG3108D - SW	RB COOLER 2B INLET	YES
XVG3109A - SW	RB COOLER 1A OUTLET	YES
XVG3109B - SW	RB COOLER 2A OUTLET	YES
XVG3109C - SW	RB COOLER 1B OUTLET	YES
XVG3109D - SW	RB COOLER 2B OUTLET	YES

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6* with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 The following valves shall be verified locked closed** at least once per 72 hours: 8430, 8454, 8441 and 8439.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**Valves may be opened under administrative control to add borated makeup.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor 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.

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.

3/4.9.4 REACTOR BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor 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 reactor building 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 Reactor Building Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the reactor building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the reactor building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required reactor building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Reactor Building 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 reactor building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the Reactor Building Purge and Exhaust isolation valves per the applicable portions of Specification 4.6.4.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive 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 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1. A minimum capacity of 3000 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of drive 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 drive rods and fuel assemblies within the reactor pressure vessel.

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 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive 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 1250 pounds.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no residual heat removal loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*The 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.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor 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 or to establish greater than or equal to 23 feet of water above the reactor pressure flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*Prior to initial criticality 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.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Reactor Building Purge Supply and Exhaust Isolation System inoperable, close each of the Purge and Exhaust penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The Reactor Building Purge Supply 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 Reactor Building Purge Supply and Exhaust isolation occurs on manual initiation, on a high radiation test signal from each of the containment radiation monitoring instrumentation channels, and on a high radiation test signal from the reactor building manipulator crane area channels.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

3/4.9.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-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:

SURVEILLANCE REQUIREMENTS (Continued)

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 30,000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, 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 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 30,000 cfm ± 10%.

   2. Verifying that on a loss of offsite power test signal, the system automatically starts.

   3. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 cfm ± 10%.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s),

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of 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. Specifications 4.2.2.2 and 4.2.2.3

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range 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 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up 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 start up and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating start up and PHYSICS TESTS.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (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 indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems 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 indication systems 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 shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	$5x10^{-7}$
1. Waste Monitor Tanks			I-131	$1x10^{-6}$
2. Condensate Demineral-izer Backwash Receiving Tank	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	$1x10^{-5}$
3. Nuclear Blowdown Monitor Tank	P Each Batch	M Composite[b]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[e]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	$5x10^{-7}$
1. Steam Generator Blowdown			I-131	$1x10^{-6}$
2. Turbine Building Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
3. Service Water Effluent	D Grab Sample	M Composite[c]	H-3	1x10-5
			Gross Alpha	$1x10^{-7}$
	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detactable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be composited in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ODCM, to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to the releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.1.2.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (see Figure 5.1-4) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

*Per reactor

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions or radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 6.9.1.12.b, 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

SETTLING POND

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each settling pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases, where,

A_j = Pond inventory limit for single radionuclide "j", in curie.

C_j = 10 CFR 20, Appendix B, Table II, column 2, concentration for single radionuclide "j", microcuries/ml.

V = design volume of liquid and slurry in the pond, in gallons.

264 = Conversion unit, microcuries/curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in the settling pond exceeding the above limit, immediately suspend all additions of radioactive material to the pond and within 48 hours reduce the pond contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of slurry (used powdex resin) to be transferred to the settling ponds shall be determined to be within the above limit by analyzing a representative sample of the slurry, and batches to be transferred to the settling ponds shall be limited by the expression:

$$\sum \frac{Q_j}{C_j} < 0.6$$

SURVEILLANCE REQUIREMENTS (Continued)

where

Q_j = concentration of radioactive materials in wet, drained slurry (used powdex resin) for radionuclide "j" excluding tritium, dissolved or entrained noble gas and radionuclides with less than 8 day half-life in microcuries per gram. The analysis shall include at least Ce-144, Cs-134, Cs-137, Sr-89, Sr-90, Co-58 and Co-60. Estimates of Sr-89, Sr-90, batch concentrations shall be based on the most recently available quarterly composite analyses.

C_j = 10 CFR 20, Appendix B, Table II column 2 concentration for single radionuclide "j", in microcuries/milliliter.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in unrestricted areas due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines and for all radioactive materials in particulate form and tritium with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioiodines, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE MONITORING AND SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)(μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emmitters[g]	1×10^{-4}
B1 Reactor Building -36" Purge Line -6" Purge Line	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
B2 Reactor Building -6" Purge Line (if continuous)	M[b] Grab Sample	M[b]	Principal Gamma Emmitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. Main Plant Vent	M[b,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
D.1. Reactor Building Purge 2. Main Plant Vent	Continuous Sampler[f]	W[d] Charcoal Sample	I-131 I-133	1×10^{-12} 1×10^{-10}
	Continuous Sampler[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] I-131, others	1×10^{-11}
	Continuous Sampler[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous Sampler[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous Monitor	Noble Gas Monitor	Noble Gases Gross Beta	2×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detactable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt shall be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

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*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed within 24 hours following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11 2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any otner report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from radioiodines tritium and radioactive materials in particulate form, and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released from each reactor unit (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium radioiodines, and radioactive materials in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.3.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the reactor shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 30 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

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*Per reactor unit

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 1 hour and less than or equal to 2% by volume within 48 hours.

c. The provisions of Specifications 6.9.1.12.b, 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 160,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFCATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, within 30 days, pursuant to Specification 6.9.2 a Special Report which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1 and 3.11.2.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.1.13. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

Table 3.12-1 Radiological environmental monitoring program
Virgil C. Summer Nuclear Station

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
AIRBORNE			
I. Particulates	A 3 Indicator samples to be taken at locations (in different sectors) beyond but as close to the exclusion boundary as practicable where the highest offsite sectoral ground level concentrations are anticipated.(1)	Continuous sampler operation with weekly collection.	Gross beta following filter change; Quarterly composite (by location) for gamma isotopic.
	B 1 Indicator sample to be taken in the sector beyond but as close to the exclusion boundary as practicable corresponding to the residence having the highest anticipated offsite ground level concentration or dose.(1)		
	C 1 Indicator sample to be taken at the location of one of the dairies most likely to be affected.(1)(2)		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
AIRBORNE, (continued)			
	D 2 Control samples to be taken at locations at least 10 air miles from the site and not in the most prevalent wind directions.(1)		
II. Radioiodine	A 3 Indicator samples to be taken at two locations as given in I.A. above.	Continuous sampler operation with weekly cannister collection.	Gamma Isotopic for I-131 weekly.
	B 1 Indicator sample to be taken at the location as given in I.B. above.		
	C 1 Indicator sample to be taken at the location as given in I.C. above.		
	D 2 Control samples to be taken at locations similar in nature to I.D. above.		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
AIRBORNE, (continued)			
III. Direct	A 13 Indicator stations with two or more dosimeters to form an inner ring of stations in the 13 accessible sectors within 1-2 miles of the plant.	Monthly or quarterly.	Gamma dose monthly or quarterly.
	B 16 Indicator stations with two or more dosimeters to form an outer ring of stations in the 16 sectors within 3 to 5 miles of the plant.		
	C 8 Stations with two or more dosimeters to be placed in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
WATERBORNE			
IV. Surface Water	A 1 Indicator sample downstream to be taken at a location which allows for mixing and dilution in the ultimate receiving river.	Time composite samples with collection every month (corresponds to USGS continuous sampling site).(3)	Gamma isotopic monthly with quarterly composite (by location) to be analyzed for tritium.(5)
	B 1 Control sample to be taken at a location on the receiving river, sufficiently far upstream such that no effects of pumped storage operation are anticipated.		
	C 1 Indicator sample from location immediately upstream of the nearest downstream municipal water supply.		
	D 1 Indicator sample to be taken in the upper reservoir of the pumped storage facility.		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
WATERBORNE, (continued)			
	E 1 Indicator sample to be taken in the upper reservoir's non-fluctuating recreational area.	Grab sampling monthly.(3)	As in IV.A above.
	F 1 Control sample to be taken at a location on a separated unaffected watershed reservoir.		
V. Ground Water	A 2 Indicator samples to be taken within the exclusion boundary and in the direction of potentially affected ground water supplies.	Quarterly grab sampling.(5)	Gamma isotopic and tritium analyses quarterly.(5)
	B 1 Control sample from unaffected location.		
VI. Drinking Water	A 1 Indicator sample from nearby public ground water supply.	Monthly grab sampling.(3)	Monthly(3) gamma isotopic and gross Beta analyses and quarterly (5) composite for tritium analyses.
	B 1 Indicator (finished water) sample from the nearest downstream water supply.	Monthly composite sample	

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
INGESTION			
VII. Milk(2)	A Samples from milking animals in 3 locations within 5 km distant having the highest dose potential. If there are none then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per year.	Semi-monthly when animals are on pasture, (6) monthly other times.(3)	Gamma isotopic and I-131 analysis semi-monthly (6) animals are on pasture; monthly (3) at other times.
	B 1 Control sample to be taken at the location of a dairy > 20 miles distant and not in the most prevalent wind direction.(1)		
	C 1 Indicator grass (forage) sample to be taken at one of the locations beyond but as close to the exclusion boundary as practicable where the highest offsite sectoral ground level concentrations are anticipated.(1)	Monthly when available.(3)	Gamma Isotopic.

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
INGESTION, (continued)			
	D 1 Indicator grass (forage) sample to be taken at the location of VII A above when animals are on pasture.	Monthly when available.(3)	Gamma Isotopic.
	E 1 Control grass (forage) sample to be taken at the location of VII B above.		
VIII. Food Products	A 3 Samples of broad leaf vegetation grown in 3 nearest offsite locations of highest calculated annual average ground-level D/Q if milk sampling is not performed within 3 km or if milk sampling is not performed at a location within 5 to 10 km where the doses are calculated to be greater than 1 mrem/yr.	Monthly when available	Gamma isotopic analysis on edible portion.
	B 1 Control sample for the same foods taken at a location at least 10 miles distant and not in the most prevalent wind direction if		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
INGESTION, (continued)			
	B (Cont'd) milk sampling is not performed within 3 km or if milk sampling is not at a location within 5 to 8 km where the doses are calculated to be greater than 1 mrem/yr.[k]		
IX. Fish	A 1 Indicator sample to be taken at a location in the upper reservoir.	Semi-annual(7) collection of the following specie types if available: bass, brea, crappie; catfish, carp; forage fish (shad).	Gamma isotopic on edible portions semi-annually.
	B 1 Indicator sample to be taken at a location in the lower reservoir		
	C 1 Indicator sample to be taken at a location in the upper reservoir's nonfluctuating recreational area.		
	D 1 Control sample to be taken at a location on the receiving river, sufficiently far upstream such that no effects of pumped storage operation are anticipated.		

Table 3.12-1 (continued)

Exposure Pathway and/or Sample	Minimum Number of Sample Locations and Criteria for Selection	Sampling and Collection Frequency	Type and Frequency of Analysis
AQUATIC			
X. Sediment	A 1 Indicator sample to be taken at a location in the upper reservoir.	Semi-annual grab sample.(7)	Gamma isotopic.
	B 1 Indicator sample to be taken in the upper reservoir's non-fluctuating recreational area.		
	C 1 Indicator sample to be taken on the shoreline of the lower reservoir.		
	D 1 Control sample to be taken in receiving river, sufficiently far upstream such that no effects of pumped storage operation are anticipated.		

NOTES

(1) Sample site locations are based on the meteorological analysis for the period of record as presented in Chapters 5 and 6 of the OLER.

(2) Milking animal and garden survey results will be analyzed annually. Should the survey indicate new dairying activity, the owners shall be contacted with regard to a contract for supplying sufficient samples. If contractual arrangements can be made, site(s) will be added for additional milk sampling up to a total of 3 Indicator locations.

(3) Not to exeed 35 days.

(4) Time composite samples are samples which are collected with equipment capable of collecting an aliquot at time intervals which are short (e.g., hourly) relative to the compositing period.

(5) At least once per 100 days.

(6) At least once per 18 days.

(7) At least once per 200 days.

NOTE: Deviations from this sampling schedule may occasionally be necessary if sample media are unobtainable due to hazardous conditions, seasonal unavailability, insufficient sample size, malfunctions of automatic sampling or analysis equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. Deviations from sampling-analysis schedule will be described in the annual report.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Food Products (pCi/Kg, wet)
H-3	2×10^{4}(a)	N.A.	N.A.	N.A.	N.A.
Mn-54	1×10^{3}	N.A.	3×10^{4}	N.A.	N.A.
Fe-59	4×10^{2}	N.A.	1×10^{4}	N.A.	N.A.
Co-58	1×10^{3}	N.A.	3×10^{4}	N.A.	N.A.
Zn-65	3×10^{2}	N.A.	1×10^{4}	N.A.	N.A.
Zr-Nb-95	4×10^{2}	N.A.	2×10^{4}	N.A.	N.A.
I-131	2	0.9	N.A.	3	1×10^{2}
Cs-134	30	10	1×10^{3}	60	1×10^{3}
Cs-137	50	20	2×10^{3}	70	2×10^{3}
Ba-La-140	2×10^{2}	N.A.	N.A.	3×10^{2}	N.A.

(a) For drinking water samples. This is 40 CFR Part 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}	N.A.	N.A.	N.A.	N.A.
H-3	2000	N.A.	N.A.	N.A.	N.A.	N.A.
Mn-54	15	N.A.	130	N.A.	N.A.	N.A.
Fe-59	30	N.A.	260	N.A.	N.A.	N.A.
Co-58, 60	15	N.A.	130	N.A.	N.A.	N.A.
Zn-65	30	N.A.	260	N.A.	N.A.	N.A.
Zr-95	30	N.A.	N.A.	N.A.	N.A.	N.A.
Nb-95	15	N.A.	N.A.	N.A.	N.A.	N.A.
I-131	1[b]	7×10^{-2}	N.A.	1	60	N.A.
Cs-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{-2}	150	18	80	180
Ba-140	60	N.A.	N.A.	60	N.A.	N.A.
La-140	15	N.A.	N.A	15	N.A.	N.A.

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. Table 4.12-1 lists detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of this guide, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). (Current literature defines the LLD as the detection capability for the instrumentation only, and the MDC, minimum detectable concentration, as the detection capability for a given instrument, procedure, and type of sample.)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per disintegration),

V is the sample size (in units of mass or volume),

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting.

The value of s_b used in the calculation of the LLD for a particular measurement system should be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance.

TABLE 4.12-1 (Continued)

TABLE NOTATION

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

b. LLD for drinking water samples.

c. Other peaks potentially due to reactor operations (fission and activation products) which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM (or participants in the EPA crosscheck program shall provide the EPA program code designation for the unit) shall be included in the Annual Radiological Environmental Operating Report.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The Bases contained in this section provide the bases for the specifications in Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

The specification of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required ECCS subsystems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Reactor Building Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the requirements of Specification 3.0.3, if both of the required Reactor Building Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2 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 percent 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. In MODES 3, 4 and 5 the most limiting accident is a boron dilution accident. A 2 percent delta k/k SHUTDOWN MARGIN provides adequate protection in these MODES.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.2×10^{-4} delta k/k/°F. The MTC value of 3.3×10^{-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 of -4.2×10^{-4} k/k/°F.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.77% 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 12475 gallons of 7000 ppm borated water from the boric acid storage tanks or 64,040 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 changes 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 275°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 2 percent delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2000 gallons of 7000 ppm borated water from the boric acid storage tanks or 9690 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

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 and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the ± 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 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 13 steps, indicated, from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

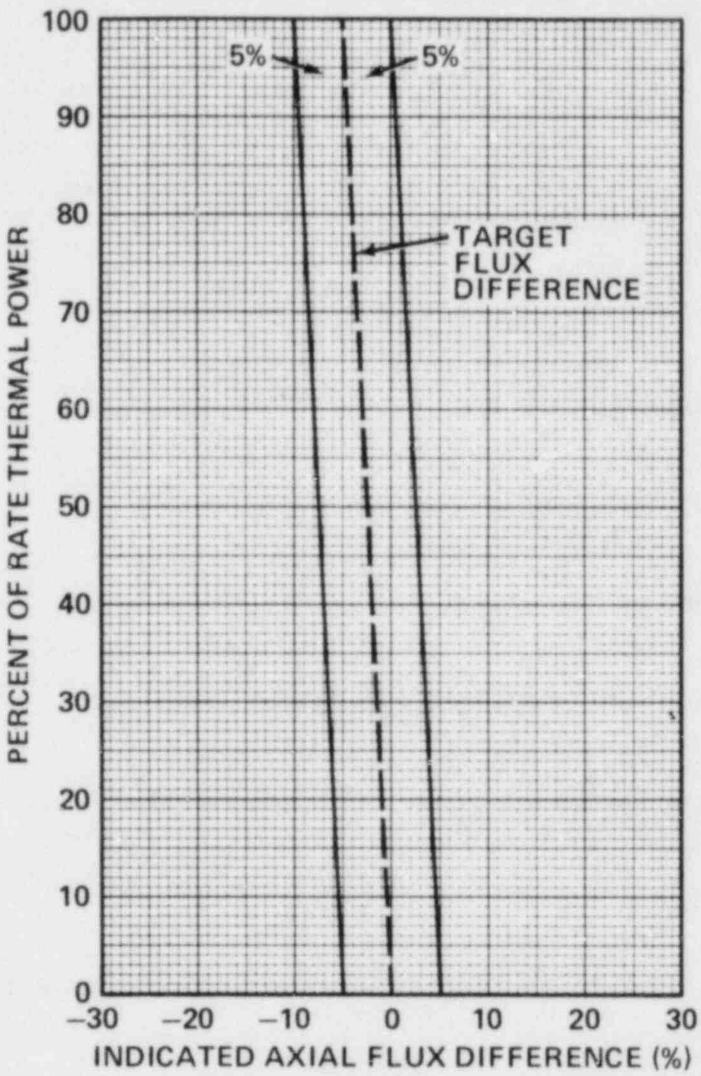


FIGURE B 3/4 2-1
INDICATED AXIAL FLUX DIFFERENCE
VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R_1, as calculated in 3.2.3 and used in Figure 3.2.3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed. R_2, as defined, allows for the inclusion of a penalty for rod bow on DNBR only. Thus knowing this "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "tradeoffs" in excess of R equal to 1.0 for the purpose of offsetting the rod bow DNBR penalty.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for Rated Thermal Power (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the accept 'e region of operation shown on Figure 3.2-3.

Fue, rod bowing reduces the value of DNB ratio. Credit is available to partially offset this reduction. This credit comes from a generic margin which totals 9.1 percent. The penalties applied to $F^N_{\Delta H}$ to account for Rod Bow (Figure 3.2-4) as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 and W 8691, Rev. 1 (partial rod bow test data).

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbels. These locations are C-8, E-5, E-11, H-3, H-13, L-5, 2-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the 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 Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE OCCURRENCES.

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

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite, or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing primary coolant loop temperature, P-12 automatically reinstates safety injection actuation and steam line isolation on low steam line pressure, and removes a blocking signal from the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation and steam line isolation on low steam line pressure and automatically provides a blocking signal to the steam dump system.

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 determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

BASES

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MINITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 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 MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 300°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 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. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

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, 1974 Edition.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. 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 enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES (PORV's)

The power operated relief valves 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 power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 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 proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission 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 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 CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 33 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 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 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.

OPERATIONAL LEAKAGE (Continued)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 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 limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

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.

   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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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 625°F.

5) System 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 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 10 effective full power years of service life. The 10 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and phosphorus content of the material in question, can be predicted using Figures B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse copper trend curves shown by Figure B 3/4 4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 10 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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 the following paragraphs.

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

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	Cu %	P %	NDTT (°F)	MIN. 50 FT-LB 35 MIL TEMP. (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF (FT-LB)
Closure Head	A533-B-Class 1			-20	40	-20	106
Head Flange	SA508 Class 2			10	<60	10	129
Vessel Flange	SA508 Class 2			0	<60	0	172
Inlet Nozzle	SA508 Class 2			-20	<40	-20	130
Inlet Nozzle	SA508 Class 2			0	<60	0	114.5
Inlet Nozzle	SA508 Class 2			-20	<40	-20	135
Outlet Nozzle	SA508 Class 2			-10	<50	-10	146
Outlet Nozzle	SA508 Class 2			-10	<50	-10	165
Outlet Nozzle	SA508 Class 2			0	<50	0	150
Nozzle Shell	A533-B-Class 1	.13	.010	-20	78	18	100.5
Nozzle Shell	A533-B-Class 1	.12	.009	-30	86	26	91
Inter. Shell	A533-B-Class 1	.10	.009	-20	90	30	80.5
Inter. Shell	A533-B-Class 1	.09	.006	-20	40	-20	106.5
Lower Shell	A533-B-Class 1	.08	.005	-10	70	10	91.5
Lower Shell	A533-B-Class 1	.08	.005	-30	70	10	106
Trans. Ring	A533-B-Class 1			-40	23	-37	107
Bottom Head	A533-B-Class 1			-10	42	-10	134
Core Region Weld		.06	.013	-50	16	-44	84
Weld HAZ				-70	-37	-70	130

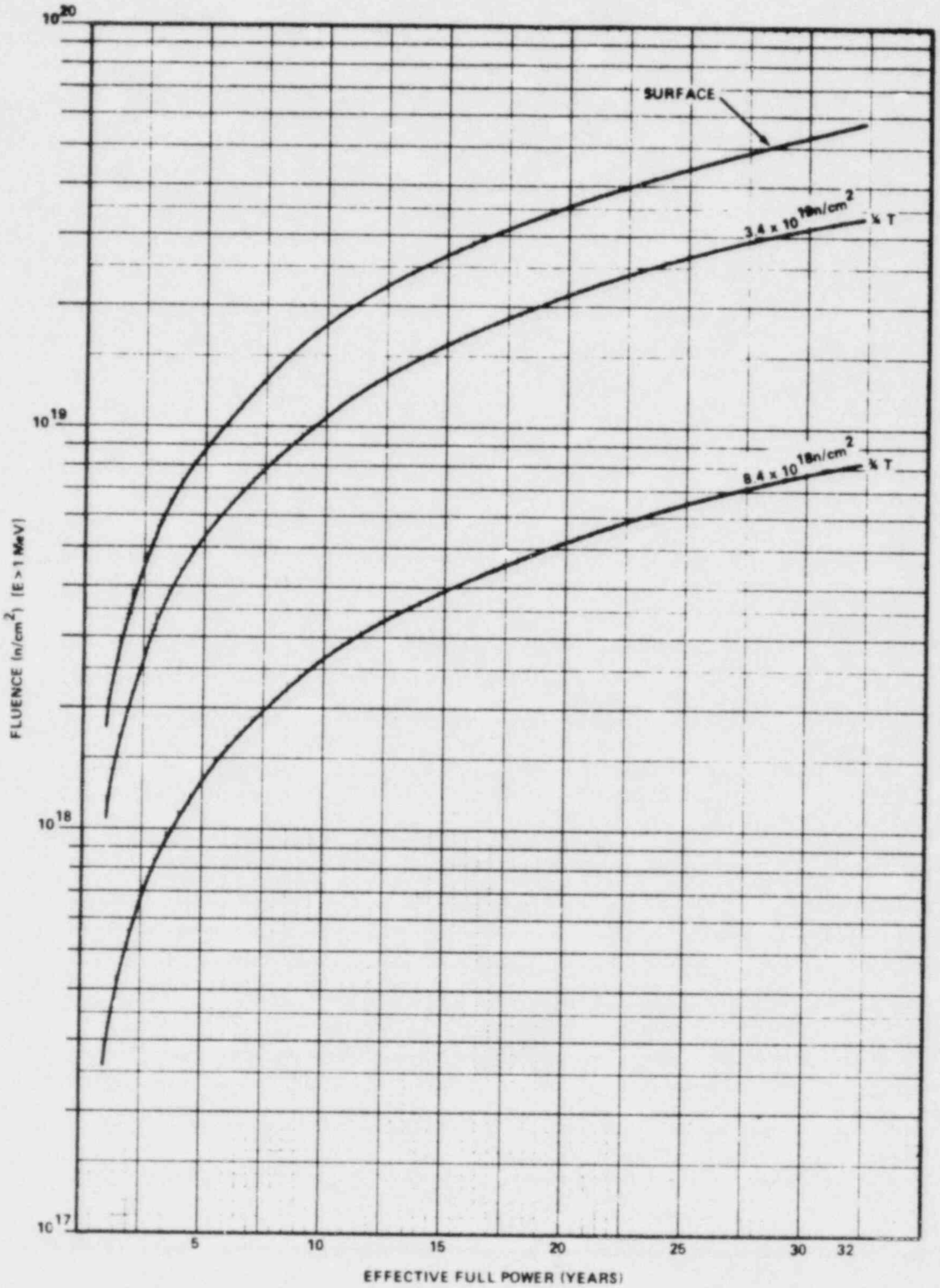


FIGURE B 3/4 4.1 FAST NEUTRON FLUENCE (E >1mev) AS A FUNCTION OF FULL POWER SERVICE LIFE

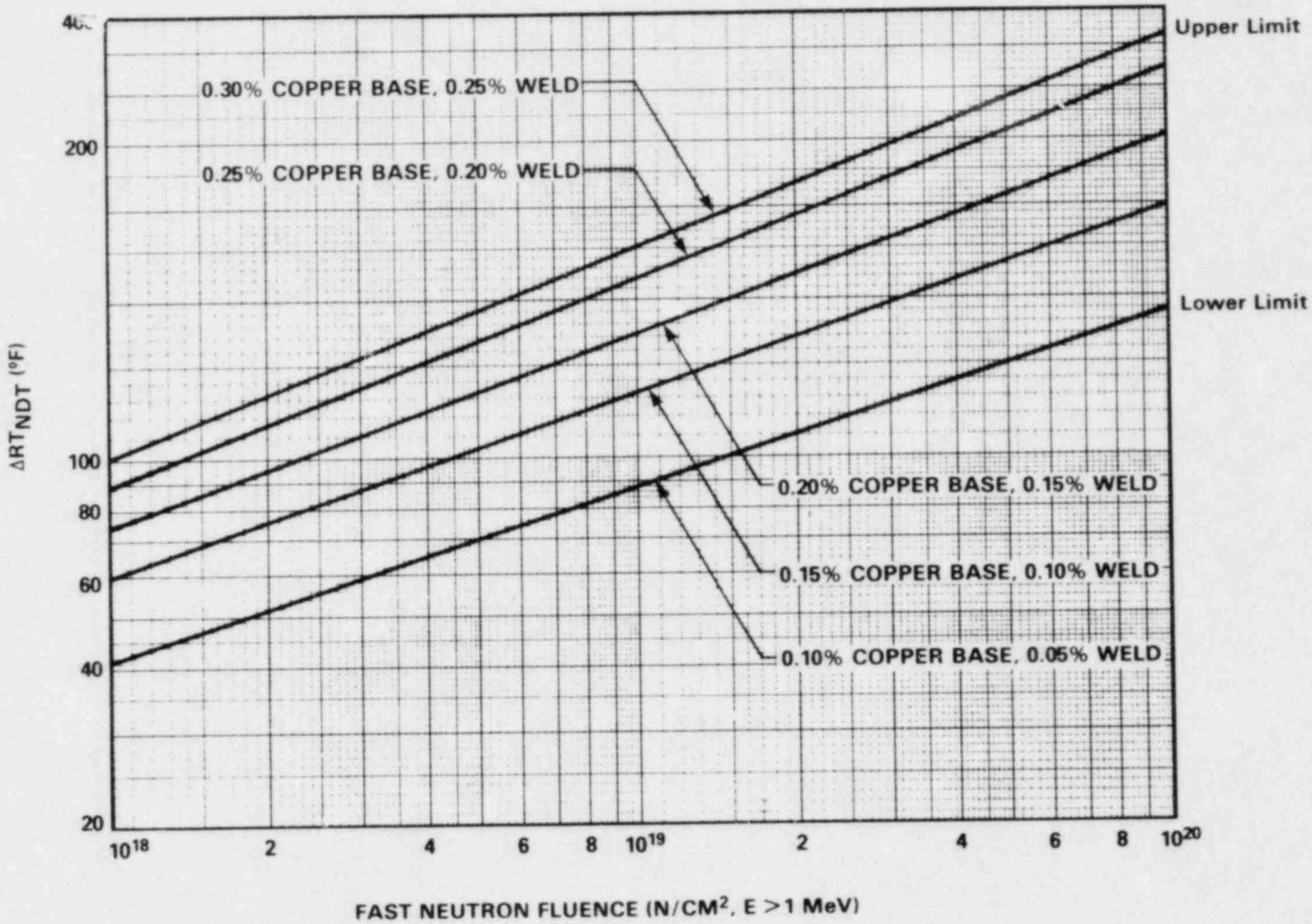


Figure B 3/4.4-2 Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550°

PRESSURE/TEMPERATURE LIMITS (Continued)

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:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where, K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

PRESSURE/TEMPERATURE LIMITS (Continued)

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 heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity 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

PRESSURE/TEMPERATURE LIMITS (Continued)

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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of 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 at least 2.7 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 300°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 HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

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 EMERGENCY CORE COOLING SYSTEM (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.

ECCS SUBSYSTEMS (Continued)

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 charging pump to be inoperable below 300°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 135°F at 22,500 ppm boron.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the

REFUELING WATER STORAGE TANK (Continued)

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 containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 REACTOR BUILDING AIR LOCKS

The limitations on closure and leak rate for the reactor building air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig and 2) the reactor building peak pressure does not exceed the design pressure of 57 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 47.1 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 47.1 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 REACTOR BUILDING 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 47.1 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The tendon lift off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required prestress level exists in the containment. In order to assess the rate of force loss, the lift off force for a tendon is compared with the force predicted for the tendon times a reduction factor of 0.95. This resulting force is referred to as the 95% Base Value. The predicted tendon force is equal to the original stressing force minus losses due to elastic shortening of the tendon, stress relaxation of the tendon wires, and creep and shrinkage of the concrete. The 5% reduction on the predicted force is intended to compensate for both uncertainties in the prediction techniques for the losses and for inaccuracies in the lift-off force measurements.

REACTOR BUILDING STRUCTURAL INTEGRITY (Continued)

In order for the tendon lift off force to be indicative of the level of prestress force in the containment, each measured force must be adjusted for the known differences which exist among the tendons due to original stressing force and elastic shortening loss. This adjustment is accomplished through the use of a Normalizing Factor ($NF_i(t)$). This factor is added to the lift off force, which results in the Normalized Lift Off Force. The Normalizing Factor is given by:

$$NF_i(t) = \{F_{ave}(o) - F_i(o)\} \{1 - \frac{SR(t)}{100}\} + \Delta F^T_{es} \{\frac{N - 2n + 1}{2N}\}$$

$\{F_{ave}(o) - F_i(o)\}$ is the group average lock-off force at original stressing, minus the original stressing force for the specific tendon.

$SR(t)$ is stress relaxation (percent) which occurs at time t after original stressing.

ΔF^T_{es} is the total elastic shortening tendon force loss.

n is the stressing sequen[illegible]rising the specific tendon.

N is the total number of stressing sequences for the group of tendons which comprise the specific tendons.

i refes to the specific tendon.

t refers to the time after original stressing of the current inspection period.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," April 1979, except that in place of the Lower Limit and 90% Lower Limit defined by these Regulatory Guides, the 95% Base Value and 90% Base Value, respectively, are used.

3/4.6.1.7 REACTOR BUILDING VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 36-inch valves cannot be inadvertently opened, they are sealed closed in accordance with the Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being spplied to the valve operator.

The use of the containment purge lines is restricted to the 6 inch purge supply and exhaust isolation valves since unlike the 36 inch valves the 6 inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR 100 would not be exceeded in the event of an accident during purging operations.

Periodic leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves with resilient material seals will provide early indication of seal degradation and will allow the opportunity for repair before gross leakage failures develop. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 REACTOR BUILDING SPRAY SYSTEM

The OPERABILITY of the reactor building spray system ensures that reactor building depressurization and cooling capability will be available in the event of a steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The reactor building spray system and the reactor building cooling system are redundant to each other in providing post accident cooling of the reactor building atmosphere. However, the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the reactor building spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 REACTOR BUILDING COOLING SYSTEM

The OPERABILITY of the reactor building cooling system ensures that 1) the reactor building air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the reactor building spray systems during post-LOCA conditions.

The reactor building cooling system and the reactor building spray system are redundant to each other in providing post accident cooling of the reactor building atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable ESF equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

The OPERABILITY of the containment filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the reactor building atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the reactor building atmosphere or pressurization of the reactor building and, is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.5 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 the reactor building below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 13.76×10^6 lbs/hr which is 110 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 3.7-2.

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

$$SP = \frac{(X) - (Y)(U)}{X} \times (*)$$

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

SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation.

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation. This value left blank pending NRC approval of 2 loop operation.

X = Total relieving capacity of all safety valves per steam line in lbs/hour.

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency 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.

The emergency feedwater system is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of at least two steam generators while allowing for (1) any spillage through the design worst-case break of the emergency feedwater line, (2) the design worst-case single failure, and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 11 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable 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 offsite 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 LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event the steam line rupture occurs within the reactor building. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on the average impact valves of the steam generator material at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 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.

ULTIMATE HEAT SINK (Continued)

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 CONTROL ROOM NORMAL AND EMERGENCY AIR HANDLING SYSTEM

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. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.7 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip and 100 kip capacity manufactured by company "A" are of the same type. The same design mechanical snubber manufactured by company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

SNUBBERS (Continued)

To provide assurance of snubber functional reliability one of two sampling and acceptance criteria methods are used:

1) functionally test 10 percent of a type of snubber with an additional 10 percent tested for each functional testing failure, or

2) functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

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

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

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.

3/4.7.10 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.11 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators", Revision 1, October 1979.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The 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. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

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 REACTOR BUILDING PENETRATIONS

The requirements on reactor 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 reactor building 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

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 and fuel assembly, MANIPULATOR CRANE (Continued)

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 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal 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 thru the reactor core to minimize the effects 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 at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.

3/4.9.9 and 3/4.9.10 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the 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 in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM-SHUTDOWN

This special test exception permits the position indication systems to be inoperable during rod drop time measurements. The 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 indication systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to an individual, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4 11.1.5 SETTLING PONDS

The inventory limits of the settling ponds (SP) are based on limiting the consequences of an uncontrolled release of the pond inventory. The expression in Specification 3.11.1.5 assumes the pond inventory is uniformly mixed, that the pond is located in an uncontrolled area as defined in 10 CFR 20, and that the concentration limit in Note 1 to Appendix B of 10 CFR 20 applies.

The batch limits of slurry to the chemical treatment ponds assure that radioactive material in the slurry transferred to the SP are "as low as is reasonably achievable" in accordance with 10 CFR 50.36a. The expression in Specification 4.11.1.5 assures no batch of slurry will be transferred to the SP unless the sum of the ratios of the activity of the radionuclides to their respective concentration limitation is less than the ratio of the 10 CFR 50, Appendix I, Section II.A, total body level to the 10 CFR 20, 105(a), whole body dose limitation, or that:

$$\sum_j \frac{c_j}{C_j} < \frac{3 \text{ mrem/yr}}{500 \text{ mrem/yr}} = 0.006$$

where

c_j = radioactive slurry concentration for radionuclide "j" entering the unrestricted area SP, in microcuries/milliliter

3/4 11.1.5 SETTLING PONDS (Continued)

C_j = 10 CFR 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microcuries/milliliter.

For the design of filter/demineralizers using powder resin, the slurry wash volume and the weight of resin used per batch is fixed by the cell surface area and the slurry volume to resin weight ratio is constant at 100 milliliters/gram of wet, drained resin with a moisture content of approximately 55 to 60% (bulk density of about 58 pounds per cubic feet). The wet drained slurry density is approximately 1 gr/ml and the absorption characteristic for gamma radiation is essentially that of water. Therefore,

$$\sum_j \frac{c_j}{C_j} = \sum_j \frac{Q_J}{C_j\ (10^2\ \text{ml/gm})} < 0.006\text{, and}$$

$$\sum_j \frac{Q_j}{C_j} < .6\ \frac{\mu\text{Ci/gm}}{\mu\text{Ci/ml}}$$

Where the terms are defined in Specification 4.11.1.5.

The batch limits provide assurance that activity input to the SP will be minimized, and a means of identifying radioactive material in the inventory limitation of Specification 3.11.1.5.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrem/year.

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive materials in particulate form and tritium are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carring out any operation which is part of the nuclear fuel cycle.

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as _a posteriori_ (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

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.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 REACTOR BUILDING

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, pre-stressed, post-tensioned 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 = 187 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor pad = 4 feet.

f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1.842×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 57 psig and a temperature of 283°F.

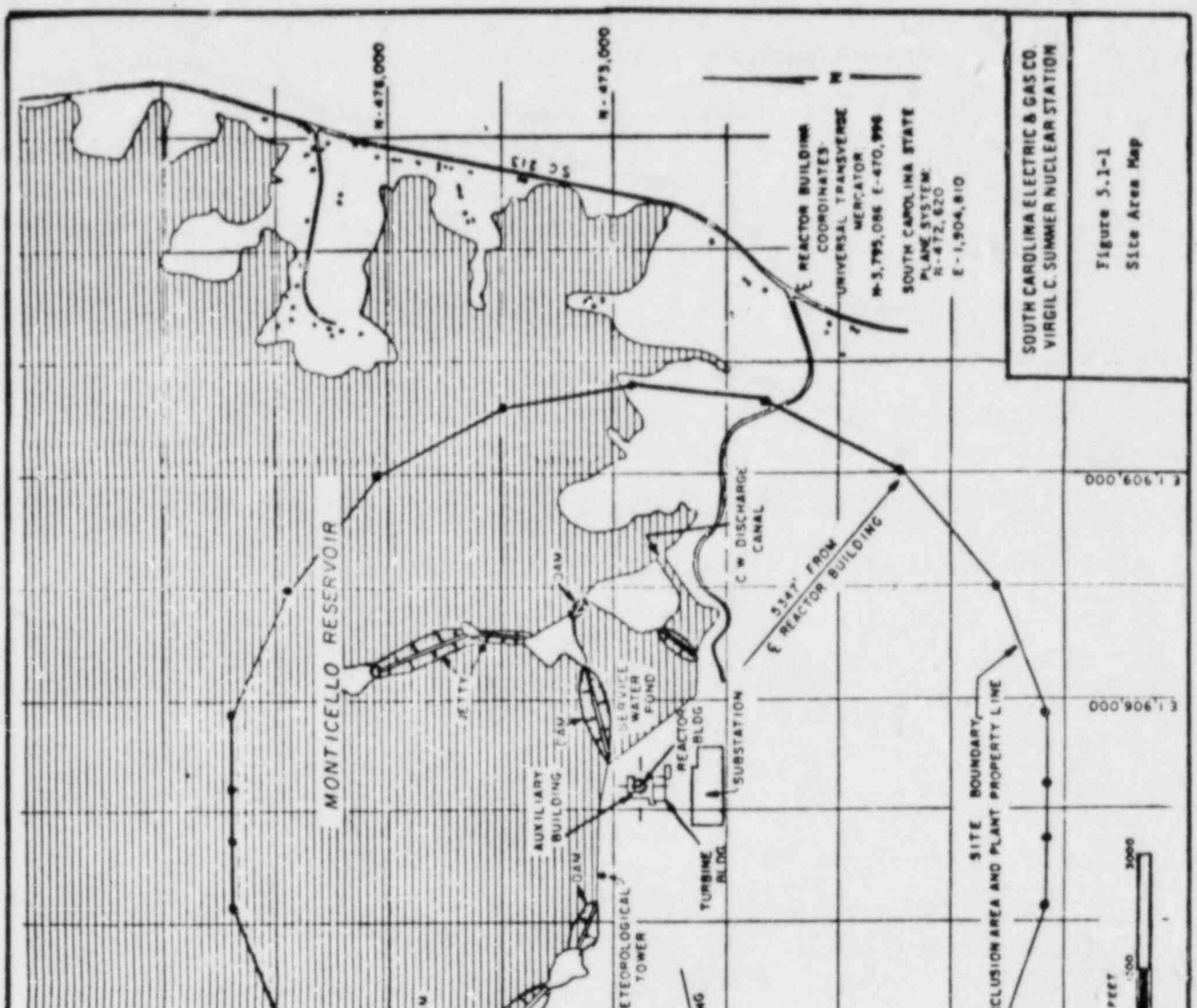


Figure 5.1-1
Site Area Map

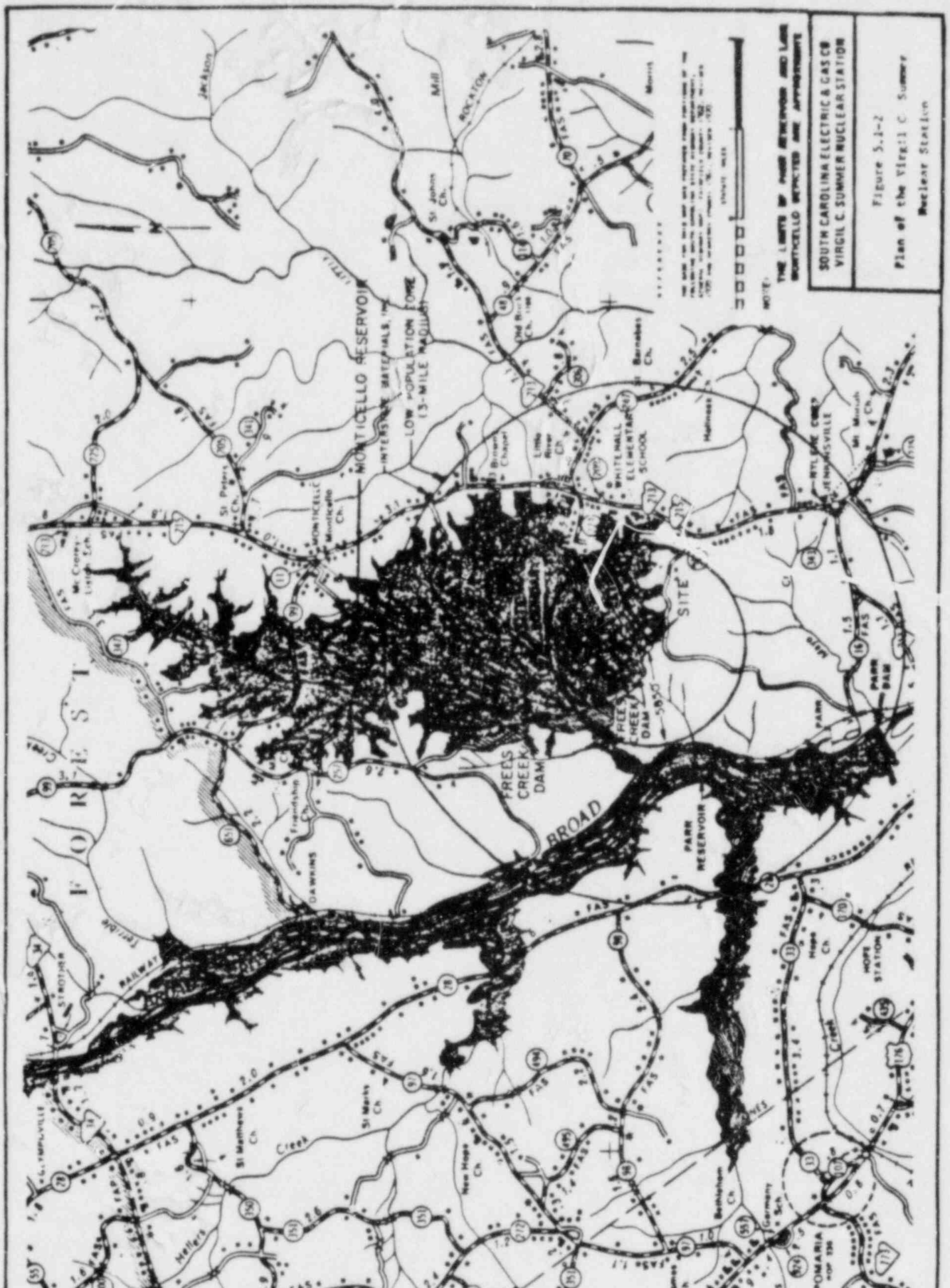
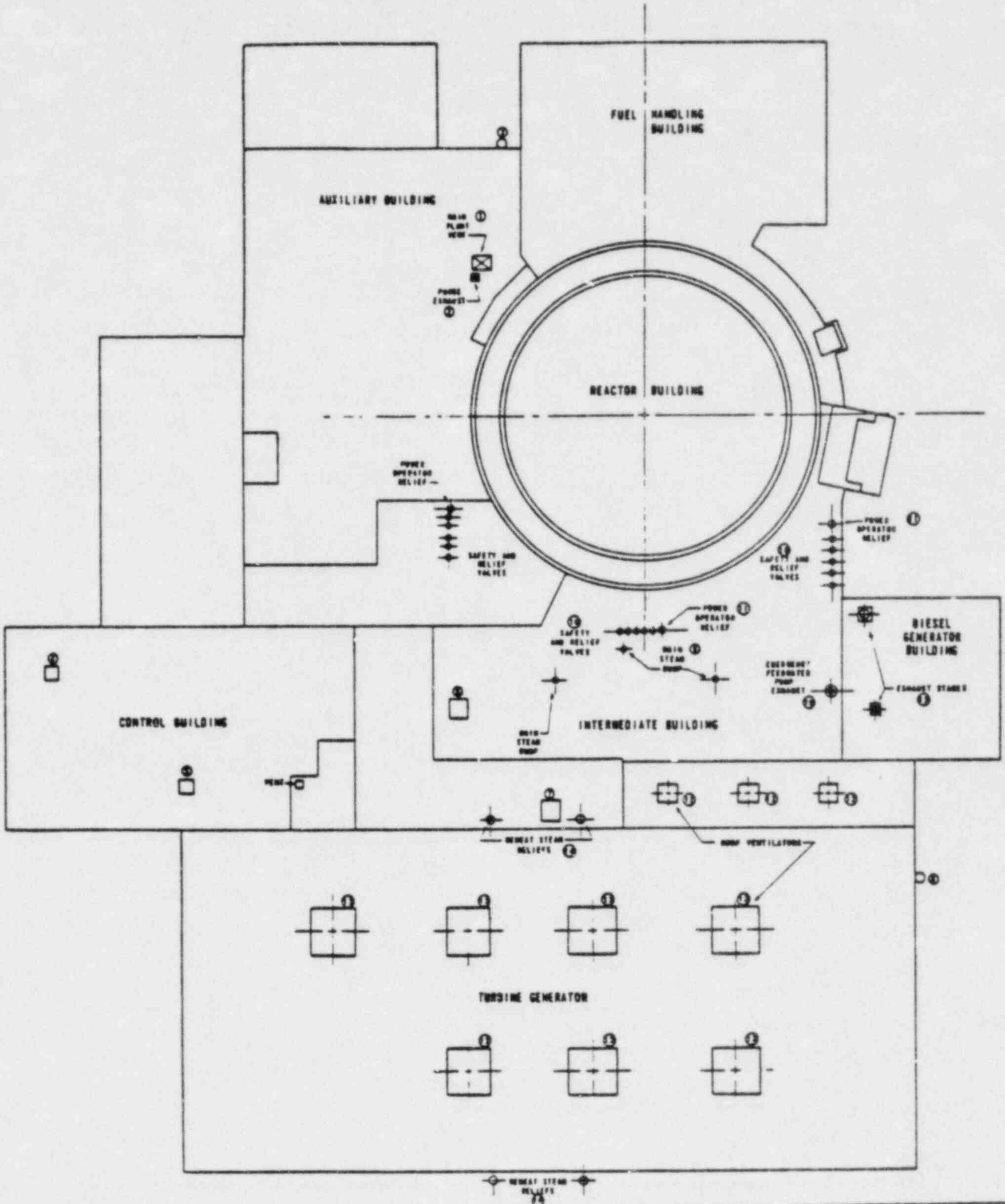


Figure 5.1-2

Plan of the Virgil C. Summer Nuclear Station

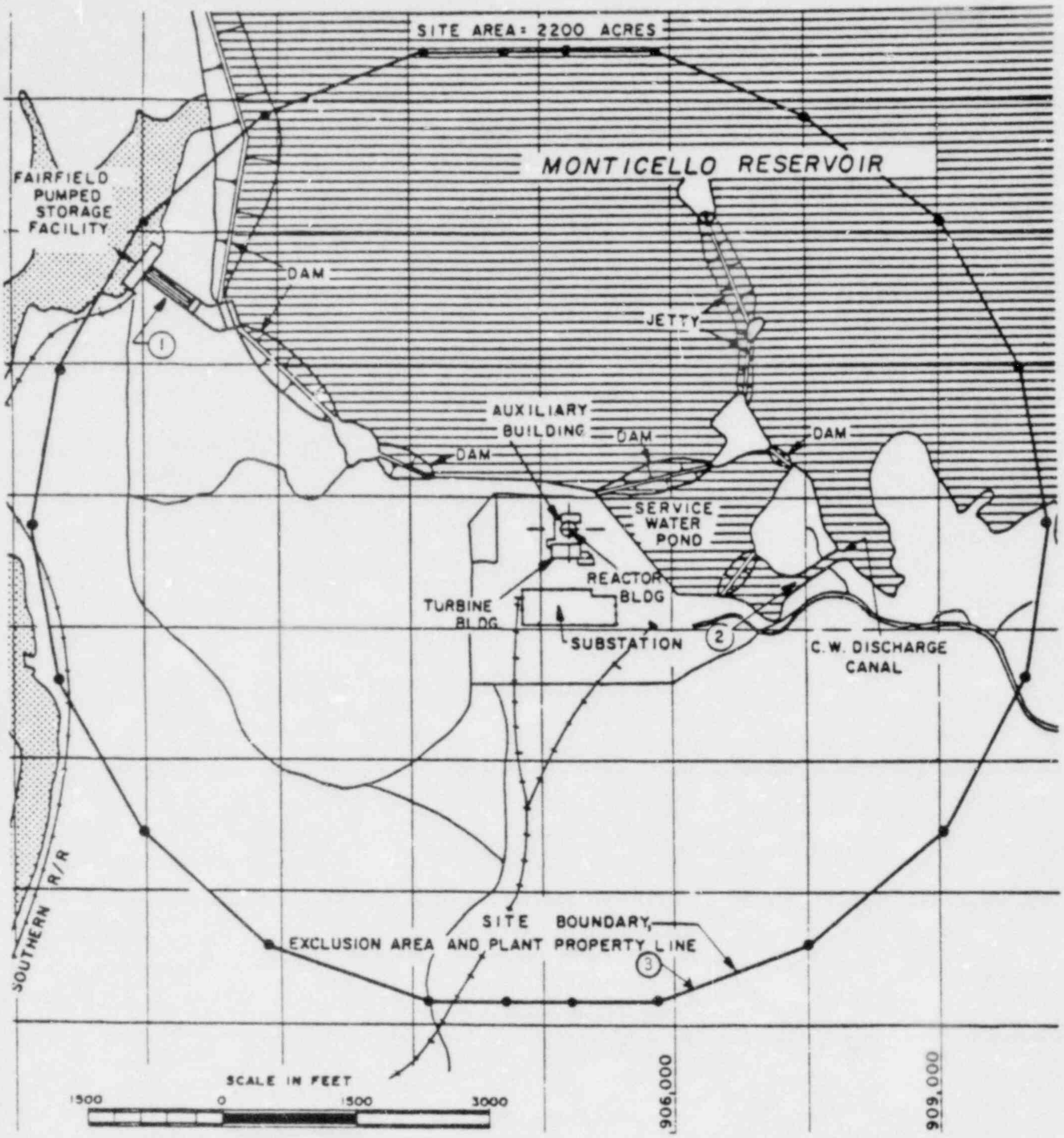


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Potentially Radioactive Gaseous Waste Release Points

FIGURE 5.1-3

NOTE: See Figure 5.1-4 for site boundary for gaseous effluents



LIQUID RELEASES:

1. FAIRFIELD PUMPED STORAGE FACILITY PENSTOCKS
   (A) LIQUID WASTE PROCESSING SYSTEM
   (B) PROCESSED STEAM GENERATOR BLOWDOWN
2. CIRCULATING WATER DISCHARGE CANAL
   (A) UNPROCESSED STEAM GENERATOR BLOWDOWN
   (B) TURBINE BUILDING FLOOR DRAINS

GASEOUS RELEASES:

3. SITE BOUNDARY FOR GASEOUS RELEASES

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Location of Liquid Release Points

FIGURE 5.1-4

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 1766 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 9407 ± 100 cubic feet at a nominal T_{avg} of 586.8°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 for uncertainties as described in Section 4.3 of the FSAR.

b. A nominal 14 inch center-to-center distance between fuel assumblies placed in the storage racks.

5..6.1.2 The K_{eff} for new fuel for the first core loading stored dry in a checkerboard pattern in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 682 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 $\leq$ 100°F/hr and 200 cooldown cycles at $\leq$ 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 $\leq$ 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 ESF Electrical System.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 inadvertent auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to $\geq$ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 3107 psig.
	200 large stepload decrease with steam dump	Load decreases of more than 10% RATED THERMAL POWER occurring in 1 minute or less.
Secondary System	1 steam line break.	Break in a > 6 inch steam line.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 1350 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Manager, Virgil C. Summer Nuclear Station shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for unit operations. A management directive to this effect, signed by the Vice President, Nuclear Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2-1.

UNIT STAFF

6.2.2 The unit 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 MODE 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 Shift Supervisor and the other 2 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

#The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Manager, Virgil C. Summer Nuclear Station or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Manager, Virgil C. Summer Nuclear Station or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

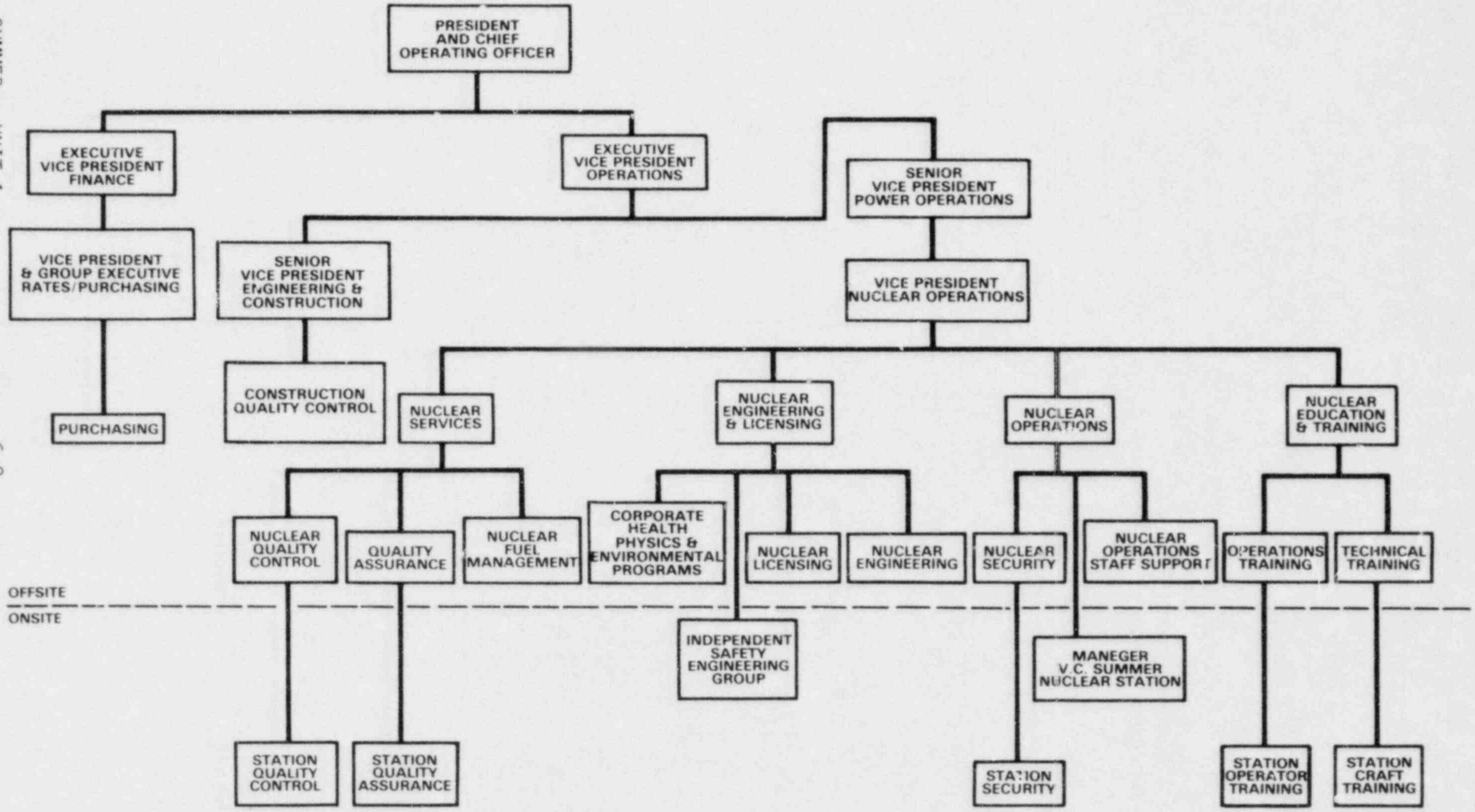


FIGURE 6.2-1

SOUTH CAROLINA ELECTRIC AND GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
OFFSITE ORGANIZATION

Note: This organization may be used for major maintenance and modification.

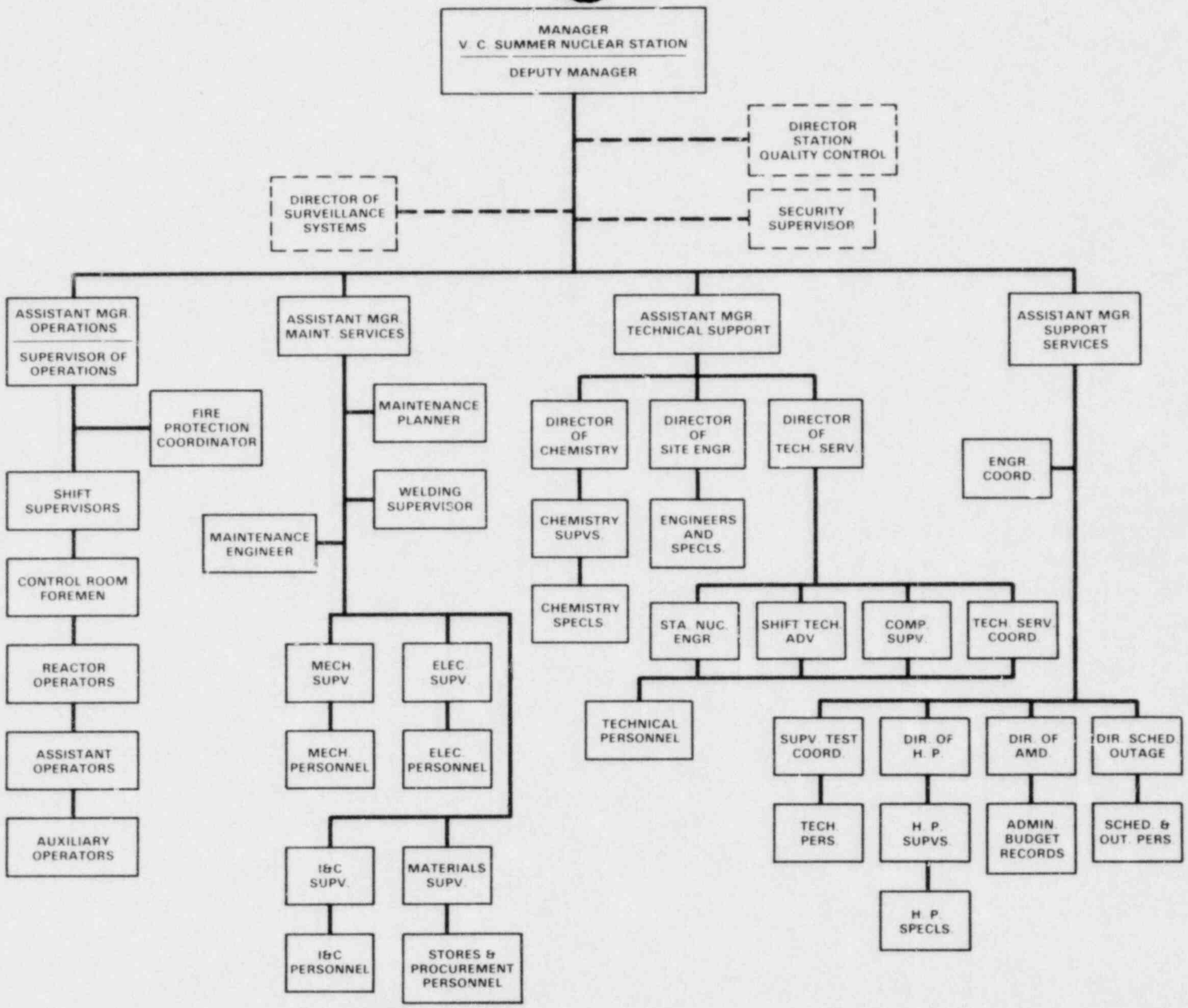


Figure 6.2-2
Virgil C. Summer Nuclear Station
Functional Organization

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SUMMER UNIT 1

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3, & 4	MODES 5 & 6
SS	1	1
CRF	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
CRF - Control Room Foreman with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
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 Control Room Foreman 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 or SRO license shall be designated to assume the Control Room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of a multi-disciplined dedicated onsite group with a minimum assigned complement of five engineers or appropriate specialists.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for procedure revisions, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Group Manager Nuclear Engineering and Licensing.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 letter to all licensees, except for the Director Health Physics who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, the Shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure I of the October 30, 1979 NRC letter to all operating nuclear power plants, and the members of the Independent Safety Engineering Group, each of whom shall have a Bachelor of Science degree or registered Professional Engineer and at least two years experience in their field. At least one year experience shall be in the nuclear field.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained and shall 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 Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with identified relevant industry operational experience.

*Not responsible for sign-off function.

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the Manager, Virgil C. Summer Nuclear Station on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Review Committee shall be composed of the:

Chairman:	Manager or Deputy Manager, Virgil C. Summer Nuclear Station
Member:	Assistant Manager Operations
Member:	Assistant Manager Technical Support
Member:	Assistant Manager Maintenance Services
Member:	Assistant Manager Support Services
Member:	Director of Health Physics or a Health Physics Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, no more than two alternates including the Chairman's alternate, if applicable, shall participate as voting members in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Safety Review Committee shall review:

a. Station administrative procedures and changes thereto,

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question and all programs required by Specification 6.8 and changes thereto.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. All written reports concerning events requiring 24 hour notification to the Commission.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.

n. The unexpected offsite release of radioactive material and the report as described in 6.9.1.13(e).

o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

a. Recommend in writing to the Manager Virgil C. Summer Nuclear Station approval or disapproval of items considered under 6.5.1.6a, c, d, e, j, and k above.

b. Render determinations in writing to the Manager Virgil C. Summer Nuclear Station with regard to whether or not each item considered under 6.5.1.6a, c, and d above constitutes an unreviewed safety question.

c. Make recommendations in writing to the Station Manager that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.

d. Provide written notification within 24 hours to the General Manager-Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the Manager, Virgil C. Summer Nuclear Station however, the Manager, Virgil C. Summer Nuclear Station shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Safety Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the General Manager-Nuclear Operations and the Chairman of the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The Nuclear Safety Review Committee 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

COMPOSITION

6.5.2.2 NSRC shall consist of a Chairman and four or more other members appointed by the Senior Vice President Power Operations. No more than a minority of the members of the NSRC shall have line responsibility for the operation of the unit.

The NSRC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of five years of technical experience of which a minimum of three years shall be in one or more of the disciplines of 6.5.2.1a through h. In the aggregate, the membership of the committee shall provide specific practical experience in the majority of the disciplines of 6.5.2.1a through h.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the Senior Vice President Power Operations; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 3 NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NSRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Plant Safety Review Committee.

AUDITS

6.5.2.8 The NSRC shall have cognizance of the audits listed below. Audits may be performed by using established SCE&G groups such as the ISEG and QA or by outside groups as determined by the NSRC. Audit reports or summaries will be the basis for NSRC action:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. Any other area of unit operation considered appropriate by the NSRC or the Senior Vice President, Power Operations.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or a qualified outside firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

k. The radiological environmental monitoring program and the results thereof, including the performance of activities required by the quality assurance program per R.G. 4.15 Rev. 1, February 1979, at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

AUTHORITY

6.5.2.9 The NSRC shall report to and advise the Senior Vice President, Power Operations on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Senior Vice President Power Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President Power Operations within 14 days following completion of the review.

c. Audit summary reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NSRC and to the Senior Vice President Power Operations. Full audits shall be forwarded to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/ group which prepared the procedure or procedure change, but who may be from the same organization as the individual/ group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved as delineated in writing by the Station Manager. The Station Manager will approve administrative procedures, security implementing procedures and emergency plan implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewd as designated by the Station Manager. Each such modification shall be designed as authorized by Nuclear Engineering and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred in by the Station Manager.

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.

d. Occurrences reportable pursuant to the Technical Specification 6.9 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such report shall be approved by the Station Manager and forwarded to the Chairman of the Nuclear Safety Review Committee.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall be members of the plant staff that meet or exceed the qualification requirements of Section 4.4 of ANSI 18.1, 1971, as previously designated by the Station Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether or not an unreviewed safety question is involved.

RECORDS

6.5.3.2 Records of the above activities shall be provided to the Station Manager, PSRC and/or NSRC as necessary for required reviews.

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 PSRC and submitted to the NSRC and the General Manager - Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The General Manager, Nuclear Operations and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the General Manager, Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
b. Refueling operations.
c. Surveillance and test activities of safety-related equipment.
d. Security Plan.
e. Emergency Plan.
f. Fire Protection Program.
g. PROCESS CONTROL PROGRAM.
h. OFFSITE DOSE CALCULATION MANUAL.
i. Effluent and environmental monitoring program using the guidance in Regulatory Guide 4.15, Revision 1, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in 6.5 above.

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 chemical and volume control, letdown, safety injection, residual heat removal, nuclear sampling, liquid radwaste handling, gas radwaste handling and reactor building spray system. 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, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions,

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training personnel,
(ii) Procedures for sampling and analysis,
(iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORT

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[1/] 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/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmenal operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. Historical annual average meteorology or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's 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.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator 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 unit shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence of development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY-DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the Regional Administrator of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

    1. A description of the event and equipment involved.
    2. Causes(s) for the unplanned release.
    3. Actions taken to prevent recurrence.
    4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

6.9.1.14 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator 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 unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the NSRC.

l. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of analysis required by the radiological environmental monitoring program.

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 (RWP).* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

*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, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area. The maximum allowable stay time for individuals in that area shall be established prior to entry. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP direct or remote (such as use of closed circuit TV cameras) continuous surveillance shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:
   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
   c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.
2. Shall become effective upon review and acceptance as set forth in 6.5 above.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

**Measurement made at 18" from source of radioactivity.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance as set forth in 6.5 above.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Plant Safety Review Committee. The discussion of each change shall contain:

   a. A summary of the evalution that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance as set forth in 6.5 above.

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The Virgil C. Summer Nuclear Station, Unit No. 1 Technical Specifications were prepared by the U. S. Nuclear Regulatory Commission to set forth the limits, operating conditions and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

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