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BILLY R. CLEMENTS
VICE PRESIDENT

October 28, 1982

Mr. Harold R. Denton
Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
SUBMITTAL OF PROPOSED UNIT 1
TECHNICAL SPECIFICATIONS

Dear Mr. Denton:

In accordance with the foreward to NUREG-0452, Fall 1981 and your Mr. D. B. Vassallo's letter dated November 16, 1978, we here-by submit proposed Technical Specifications, Radiological Effluent Technical Specifications (RETS) and Off Site Dose Calculation Manual (OCDM). As directed by the above, the Technical Specifications are in the form of a mark-up of NUREG-0452, Rev. 4 and the RETS are in the form of a mark-up of NUREG-0472, Rev. 2.

Enclosed are 15 copies of each item listed. If you have any questions about this matter, please call Richard Werner at (214) 653-4869.

Respectfully submitted,

Billy R Clements

BRC:tls
Enclosures

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Technical Specifications
for
COMANCHE PEAK S.E.S.
UNIT I

INITIAL SUBMITTAL
Draft

10/28/82

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 setpoints such that the setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds with 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 (3.6.4.1).

b. All equipment hatches are closed and sealed,

c. Each air lock is OPERABLE pursuant to 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."~~ Regulatory Guide 1.109, "Calculation of Annual Doses to man from Routine Release of Reactor effluents for the Purpose of evaluating compliance with 10 CFR 50, Appendix I."

$\bar{E}$ - 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 noniodine activity in the coolant.

SEP 1 5 1981

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.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MASTER RELAY TEST

1.15 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.

OPERABLE - OPERABILITY

1.16 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

1.17 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

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

PRESSURE BOUNDARY LEAKAGE

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

QUADRANT POWER TILT RATIO

half
half

1.20 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.

half
halt

RATED THERMAL POWER

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

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE OCCURRENCE

1.23 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

~~SHIELD BUILDING INTEGRITY~~

~~1.24 SHIELD BUILDING INTEGRITY shall exist when:~~

~~a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,~~

~~b. The shield building filtration system is OPERABLE, and~~

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

SHUTDOWN MARGIN

1.25 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

~~a. No change in part length rod position, and~~

b. 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

1.26 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.

STAGGERED TEST BASIS

1.27 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.29 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

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

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.
N.A.	Not applicable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

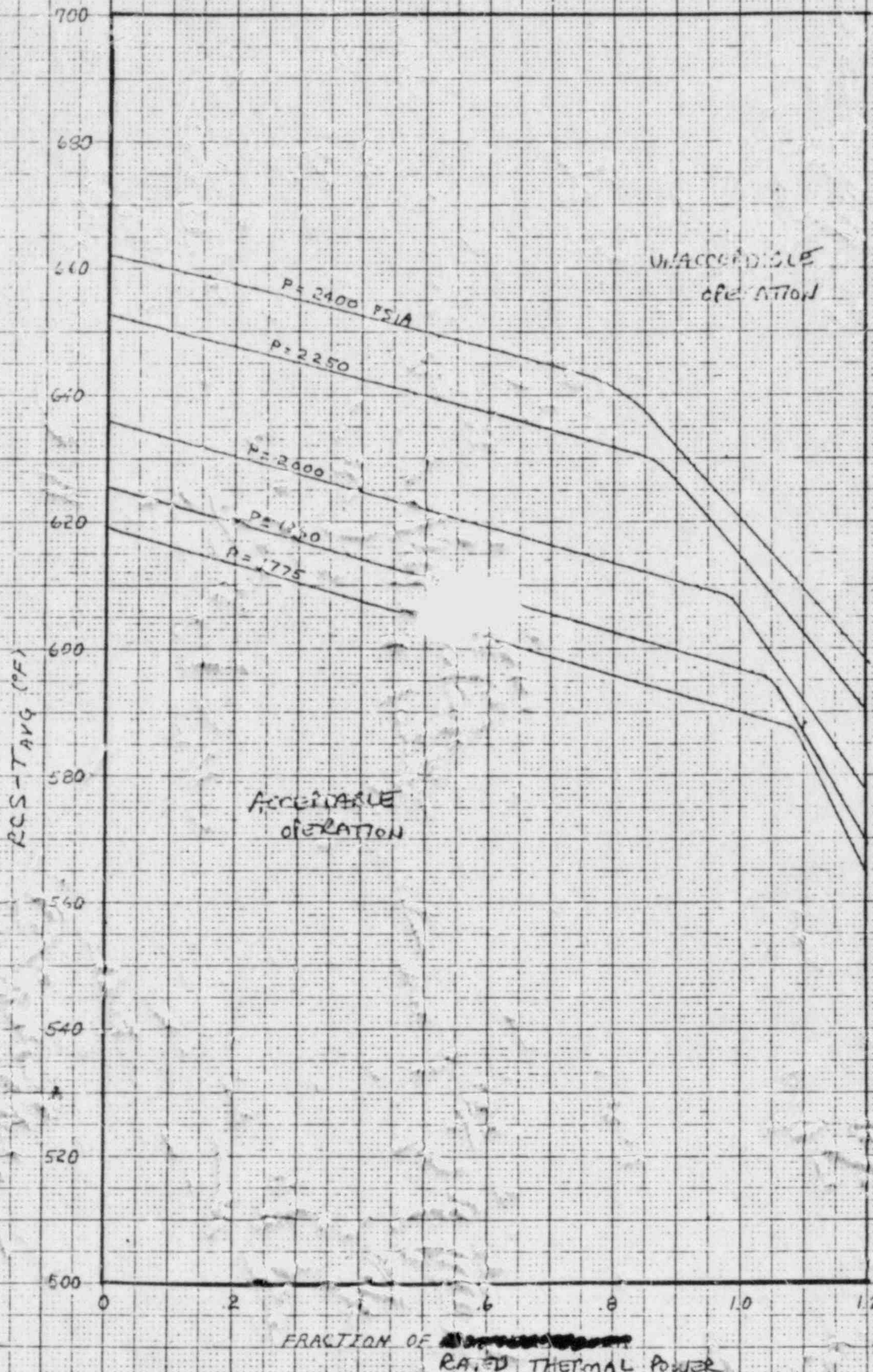
MODES 3, 4 and 5

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 [redacted] 2.1-1

[redacted] REACTOR CORE SAFETY LIMITS

FOUR LOOPS IN OPERATION



NA

(n-1)

FIGURE 2.1-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

With a reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ (25)% of RATED THERMAL POWER High Setpoint - ≤ (109)% of RATED THERMAL POWER	Low Setpoint - ≤ (26)% of RATED THERMAL POWER High Setpoint - ≤ (110)% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ (5)% of RATED THERMAL POWER with a time constant ≥ (1) second	≤ (5.5)% of RATED THERMAL POWER with a time constant ≥ (1) second
4. Power Range, Neutron Flux, High Negative Rate	≤ (~~3~~ 5)% of RATED THERMAL POWER with a time constant ≥ (1) second	≤ (~~3.5~~ 5.5)% of RATED THERMAL POWER with a time constant ≥ (1) second
5. Intermediate Range, Neutron Flux	≤ (25)% of RATED THERMAL POWER	≤ (30)% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ (10^5) counts per second	≤ (1.3×10^5) counts per second
7. Overtemperature ~~ΔT~~ N-16	See Note 1	See Note ~~3~~ 2
8. Overpower ~~ΔT~~ N-16	~~See Note 2~~ 112%	~~See Note 3~~ 114%
9. Pressurizer Pressure--Low	≥ (~~1865~~ 1910) psig	≥ (~~1855~~ 1900) psig
10. Pressurizer Pressure--High	≤ (2385) psig	≤ (2395) psig
11. Pressurizer Water Level--High	≤ (92)% of instrument span	≤ (93)% of instrument span
12. Loss of RCS Flow	≥ (90)% of design flow per loop*	≥ (89)% of design flow per loop*

*Design flow is (95,700) gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	47% ≥ ~~(10%)~~ of narrow range instrument span-each steam generator	46% ≥ ~~(9%)~~ of narrow range instrument span-each steam generator
~~14. Steam Generator Water Level Low Coincident With Steam/Feedwater Flow Mismatch~~	~~≥ (25%) of narrow range instrument span--each steam generator~~ ~~≤ (40%) of full steam flow at RATED THERMAL POWER~~	~~≥ (24%) of narrow range instrument span--each steam generator~~ ~~≤ (42.5%) of full steam flow at RATED THERMAL POWER~~
15. Undervoltage-Reactor Coolant Pumps	4620 ≥ ~~(2750)~~ volts-each bus	4554 ≥ ~~(2710)~~ volts-each bus
16. Underfrequency-Reactor Coolant Pumps	57.2 ≥ ~~(57.5)~~ Hz - each bus	57.1 ≥ ~~(57.4)~~ Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	45 ≥ ~~(900)~~ psig	43 ≥ ~~(800)~~ psig
B. Turbine Stop Valve Closure	≥ ~~(~~1%~~)~~ open	≥ ~~(~~1%~~)~~ open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. ~~Reactor Coolant Pump Breaker Position Trip~~	~~Not Applicable~~	~~Not Applicable~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
20. Reactor Trip System Interlocks		
A. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
B. Low Power Reactor Trips Block, P-7		
a. P-10 Input	10% of RATED THERMAL POWER	> 9%, < 11% of RATED THERMAL POWER
b. P-13 Input	< 10% RTP Turbine Impulse Pressure Equivalent	< 11% RTP Turbine Impulse Pressure Equivalent
C. Power Range Neutron Flux, P-8	< ~~35%~~ 48% of RATED THERMAL POWER	< ~~36%~~ 49% of RATED THERMAL POWER
Reactor Trip on Turbine Trip, P-9	≤ 50% of RATED Thermal POWER	≤ 52.2% of RATED THERMAL POWER
D. Low Setpoint Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	> 9%, < 11% of RATED THERMAL POWER
E. Turbine Impulse Chamber Pressure, P-13	< 10% RTP Turbine Impulse Pressure Equivalent	< 11% RTP Turbine Impulse Pressure Equivalent
21. Reactor Trip Breakers	Not Applicable	Not Applicable
22. Automatic Trip Logic	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature ^{16}N trip Setpoint $= K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S} T_c - T_c^0\right) + K_3 (P - P^0) - f_1(\Delta q)$

Where:

T_c = Cold leg temperature, °F

T_c^0 = 558.4°F

P = Pressurizer pressure, psig

P^0 = 2235 psig (indicated RCS nominal operating pressure)

K_1 = 1.11

K_2 = 0.01085

K_3 = 0.000578

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-Lag compensator on measured T_c

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_c, $\tau_1 = 10$ secs, $\tau_2 = 3$ secs.

S = Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

~~P = Pressurizer pressure, psig~~

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

~~S = Laplace transform operator, sec^{-1}~~

and f_1(~~ΔI~~ Δq) is a function of the indicated difference between top and bottom halves 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 (~~-30~~ -35) percent and (~~+5~~ +10) percent, f_1(~~ΔI~~ Δq) = 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 (~~-30~~ -35) percent, the ~~ΔT~~ N-16 trip setpoint shall be automatically reduced by (~~0.89~~ 1.25) percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds (~~14~~ +10) percent, the ~~ΔT~~ N-16 trip setpoint shall be automatically reduced by (~~0.80~~ 1.55) percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

~~NOTE 2:~~ Overpower $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right) \left(\frac{1}{1 + \tau_4 S}\right) T - K_6\left[T\left(\frac{1}{1 + \tau_4 S}\right) - T''\right] - f_2(\Delta I)\right\}$

Where:

$\frac{1}{1 + \tau_1 S}$	=	as defined in Note 1
τ_1	=	as defined in Note 1
ΔT_o	=	as defined in Note 1
K_4	$\leq$	(1.087)
K_5	=	(0.02/°F) for increasing average temperature and (0) for decreasing average temperature
$\frac{\tau_5 S}{1 + \tau_5 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation
τ_5	=	Time constant utilized in the rate-lag controller for T_{avg}, τ_5 = (10) secs.
$\frac{1}{1 + \tau_4 S}$	=	as defined in Note 1
τ_4	=	as defined in Note 1
K_6	=	(0.0012) for $T > T''$ and K_6 = (0) for $T \leq T''$
T	=	as defined in Note 1
T''	=	($\leq$ 578.2°F) Reference T_{avg} at RATED THERMAL POWER
S	=	as defined in Note 1
$f_2(\Delta I)$	=	0 for all ΔI

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: The channel's maximum trip point shall not exceed its computed trip point by more than 1.2 percent.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB 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. R-Grid

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~~ N-16 trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System 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 and are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings~~, are designed to ANSI B 31.1 ______ 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 various reactor trip circuits automatically 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.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

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.

Power Range, Neutron Flux (Continued)

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

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a reactor trip at a current level equivalent to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ~~ΔT~~ N-16

The Overtemperature ~~delta T~~ N-16 trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the 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.

~~OPTIONAL FOR PLANTS PERMITTED N-1 LOOP OPERATION~~

~~Operation with a reactor coolant loop out of service below the (n) loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during (n-1) loop operation exclusive of the Overtemperature delta T setpoint. (n-1) loop operation above the (n) loop P-8 setpoint is permissible after resetting the K1 input to the Overtemperature delta T channels and raising the P-8 setpoint to its (n-1) loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.~~

Overpower ~~ΔT~~ N-16

The Overpower ~~delta T~~ N-16 reactor trip provides assurance of fuel integrity, e.g., no fuel pellet cracking or melting, under all possible overpower conditions, limits the required range for Overtemperature ~~delta T~~ N-16 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, (and 3)~~ axial power distribution, N-16 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.

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 ~~30~~ 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.

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Loss of Flow (Continued)

~~OPTIONAL FOR PLANTS PERMITTED N-1 LOOP OPERATION~~

The P-8 setpoint trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when (n-1) loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for (n-1) loop operation, the P-8 trip at (76%) RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with (n-1) loops in operation.

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 (1.42×10^6) lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below (25) percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The 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 setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage,

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed (1.2) seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed (0.3) 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~~ P-9 (a power level of approximately ~~10~~ 50 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.~~ P-9

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 Coolant Pump Breaker Position Trip~~

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10 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 more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 30 percent of RATER THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7 an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

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-9 On increasing power P-9 defeats the automatic block of reactor trip on turbine trip. On decreasing power P-9 automatically blocks the above trip.

P-10 On increasing power P-10 allows the manual block of the Intermediate Range reactor trip and the flow 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 Turbine impulse chamber pressure, Provides input to P-7.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one 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 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for 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 met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

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

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

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

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

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

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable 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).

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

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

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

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to (1.6%) delta k/k for ~~(n)~~ four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than (1.6%) 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 and continue until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to (1.6%) 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 e below, 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)

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

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

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

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1 0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% 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 1.0% 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 ~~(3.9)~~ −4.0 x 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
-3.1 ~~-(3.0)~~ x 10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-(3.0)~~ x 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.

-3.1

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to (~~541~~)°F.

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APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than
551 (~~541~~)°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to (~~541~~)°F:

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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 (~~551~~)°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

REACTIVITY CONTROL SYSTEMS

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:

a. At least once per 7 days by verifying that the temperature of the ~~heat traced portion of the flow path~~ is greater than or equal to (65)°F when a flow path from the boric acid tanks is used.

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

areas in which the boric acid tanks and associated piping are located

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the ~~heat traced portion of the flow path from the boric acid tanks~~ areas in which the boric acid tanks and associated piping are located is greater than or equal to ~~(65)~~°F when it is a required water source.

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

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

d. 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 ~~(275)~~ 295°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, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2480 psig 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 12 hours, except when the reactor vessel head is removed, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2480 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~(275)~~ 295°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

#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 ~~(275)~~°F.

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

1. A minimum contained borated water volume of 920 gallons,
2. Between (~~20,000~~) 7000 and (~~22,500~~) 7700 ppm of boron, and
3. A minimum solution temperature of (~~145~~) 65 °F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 7120 gallons,
2. A minimum boron concentration of ~~(2000)~~ ppm, and
3. A minimum solution temperature of (~~35~~) 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 (~~35~~) 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 and at least one associated heat tracing system with:

   1. A minimum contained borated water volume of 17,000 gallons,

   2. Between (~~20,000~~ 7000) and (~~22,500~~ 7700) ppm of boron, and

   3. A minimum solution temperature of (~~145~~ 65)°F.

b. The refueling water storage tank with:

   1. A contained borated water volume of between 479,900 and 526,300 gallons,

   2. Between (2000) and (2100) ppm of boron, and

   3. A minimum solution temperature of (~~35~~ 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 1% 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 (~~35~~)°F.
40

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, ~~and all part length 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 or part 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.~~ INSERT A

c. With one full ~~or part~~ 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 Figure~~s~~ (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:

      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.~~ INSERT B

*See Special Test Exceptions 3.10.2 and 3.10.3.

INSERT FOR P 3/4 1-14

INSERT A

b. With one or more rods trippable but inoperable due to causes other than those addressed by ACTION a., POWER OPERATION may continue provided the remainder of the rod(s) in the group are aligned within ± 12 steps (indicated position) of the inoperable rod(s) while maintaining the rod sequence and insertion limits of Figure 3.1-1.

INSERT B

b) The Rod Insertion Limits of Specification 3.1.3.1 b are adjusted within 2 hours as required to ensure a shutdown margin of at least 1.6% ΔK/K when all operable rods are above that limit.

ACTION (Continued)

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

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 ~~and part~~ 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 ~~and each part length rod which is 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 OR PART 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)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown, ~~control~~ and ~~part length~~ 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 ~~or part length~~ 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 a CHANNEL ~~FUNCTIONAL TEST~~ CALIBRATION at least once per 18 months.

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

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to (2.2) seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to ~~(541)~~ 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With the rod drop times within limits but determined with ~~n-1~~ 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

1. Less than or equal to ~~(66)~~ 70% of RATED THERMAL POWER ~~when the reactor coolant stop valves in the nonoperating loop are open~~, ~~or~~

~~2. Less than or equal to (76)% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.~~

SURVEILLANCE REQUIREMENTS

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

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

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

~~c. At least once per 18 months.~~

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SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

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

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure~~s~~ (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 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group 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.

Rod Group Insertion Limits
VERSUS Thermal Power
Four L - Operation

Figure 3.1-1

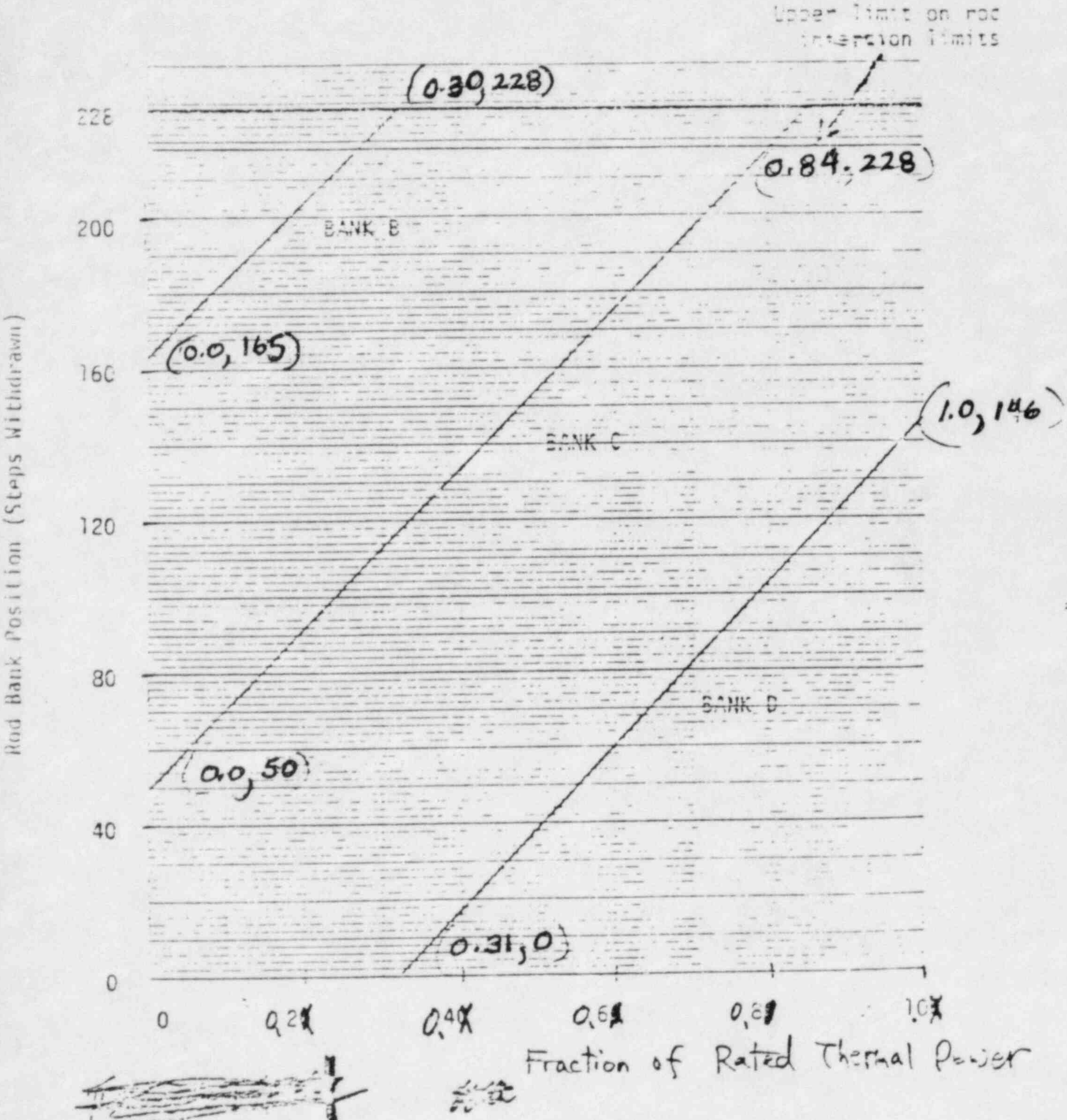

Figure 3.1-2

NOT USED

PART LENGTH ROD INSERTION LIMITS ~~(OPTIONAL)~~ NOT USED

LIMITING CONDITION FOR OPERATION

3.1.3.7 The part length control rod bank shall be:

a. Limited in physical insertion as shown on Figure (3.1-3), and

b. Limited from covering any axial segment of the fuel assemblies for a period in excess of (18) out of any 30 Equivalent Full Power Days.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With the part length control rod bank inserted beyond the insertion limit of Figure (3.1-3), either:

1. Withdraw the part length control rod bank to within the limit within 2 hours, or

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

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

b. With the neutron absorber section of the part length control rod bank covering any axial segment of the fuel assemblies for a period exceeding 18 out of any 30 consecutive EFPD period, either:

1. Reposition the part length control rod group to satisfy the above limit within 2 hours, or

2. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length control rod bank shall be determined at least once per 12 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.

PART LENGTH ROD INSERTION LIMITS (if required by DNB considerations)

LIMITING CONDITION FOR OPERATION

3.1.3.7 All part length rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one part length rod not fully withdrawn, within 1 hour either:

a. Fully withdraw the rod, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 Each part length rod shall be determined to be fully withdrawn by:

a. Verifying the position of the part length rod prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

b. Verifying, at least once per 31 days, that electric power has been disconnected from its drive mechanism by physical removal of a breaker from the circuit.

* See Special Test Exceptions 3.10.2. and 3.10.3.

PART LENGTH ROD GROUP INSERTION
LIMIT VERSUS THERMAL POWER

FIGURE 3.1-3

NOT USED

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

INSERT

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

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the above required ~~±(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 above required ~~±(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 above required 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 above required ~~±(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)%~~ above required 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)%~~ above required 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 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

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

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. ~~with all part length control rods fully withdrawn.~~ 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.

Insert for Page 3/4 2-1

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

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

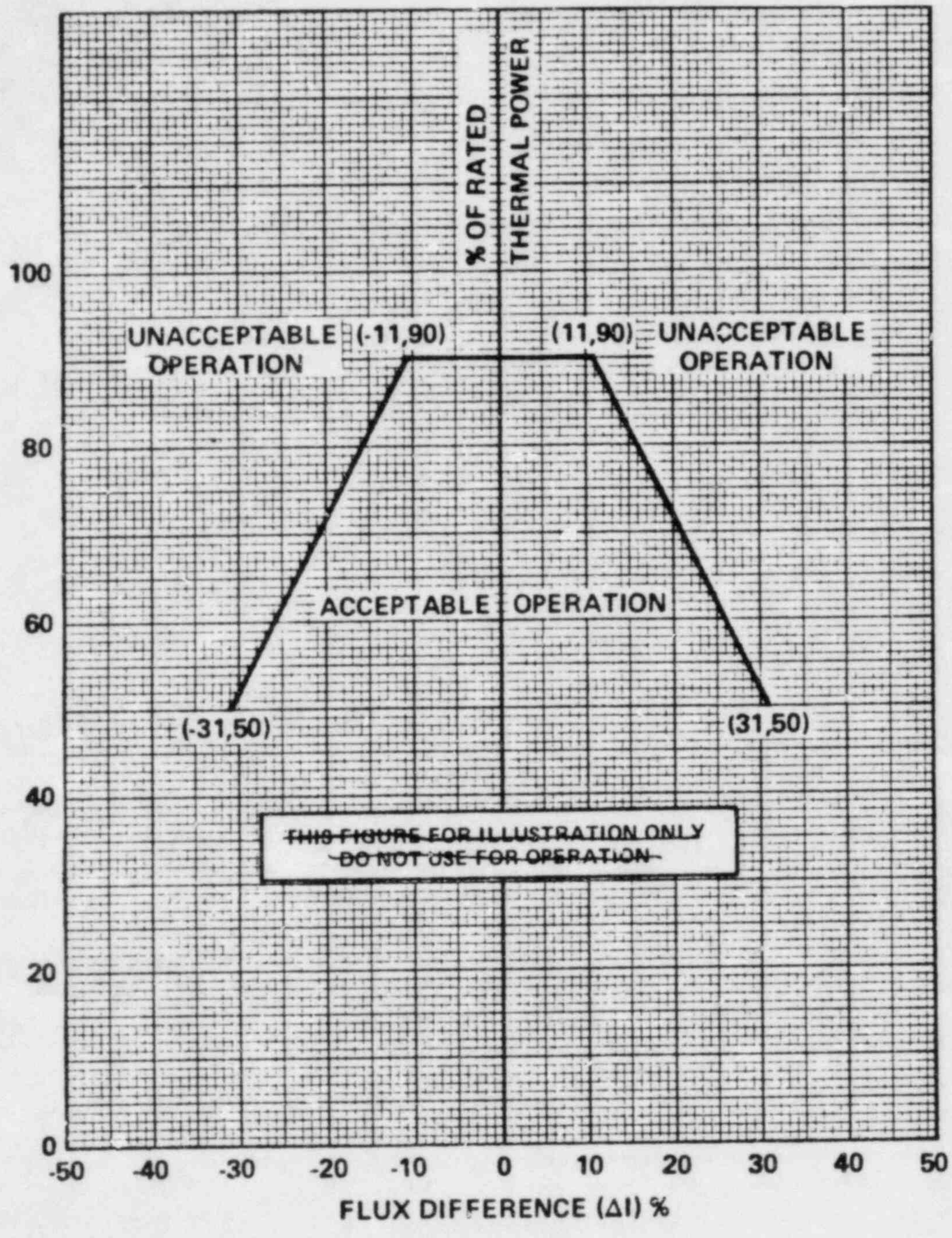

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

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:

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

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

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

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

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. ~~Comply with either of the following ACTIONS:~~

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and 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~~ Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. ~~The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.~~ N-16

2. ~~Reduce THERMAL POWER as necessary to meet the limits of Specification (3.2.6) using the APDMS with the latest incore map and updated $\bar{R}$. (APDMS plants only)~~

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.

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0
1.1
1.25 1.4 1.6

150mm

6"

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

2. The relationship:

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

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

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

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

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

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

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

e. The F_{xy} limits for RATED THERMAL POWER (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 6.9.1.10.

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" ~~or part length~~ control rods.

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

1. ~~The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}, and (for plants with $F_Q(Z)$ less than 2.32 and using APDMS)~~

2. 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.

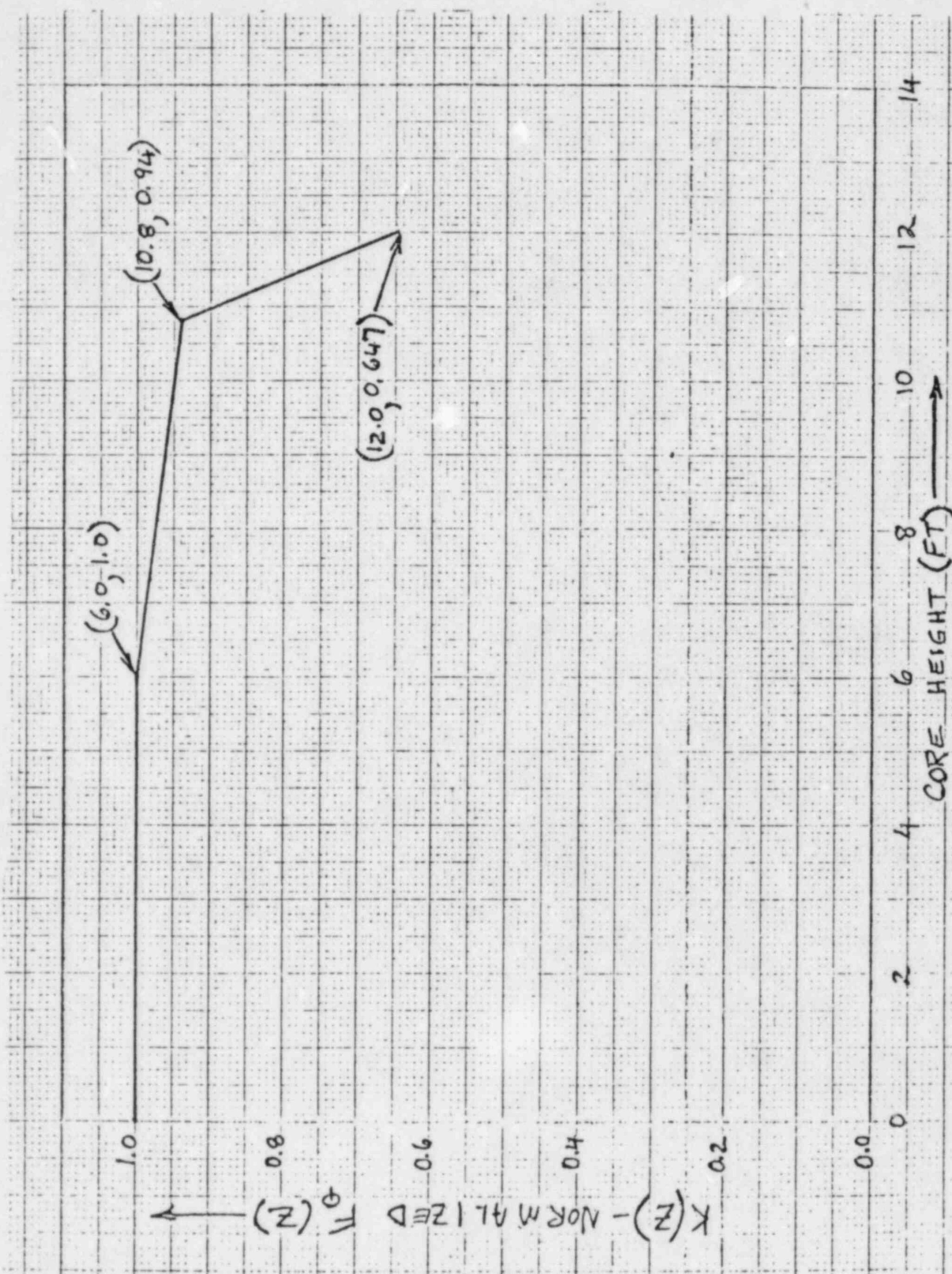

FIGURE 3.2-2. K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALY 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 4 loop operation.

Where:

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

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

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.

RCS TOTAL FLOW RATE (10^4 GPM)

LATER

FIGURE 3.2-3 RCS TOTAL FLOWRATE VERSUS R — FOUR LOOPS IN OPERATION

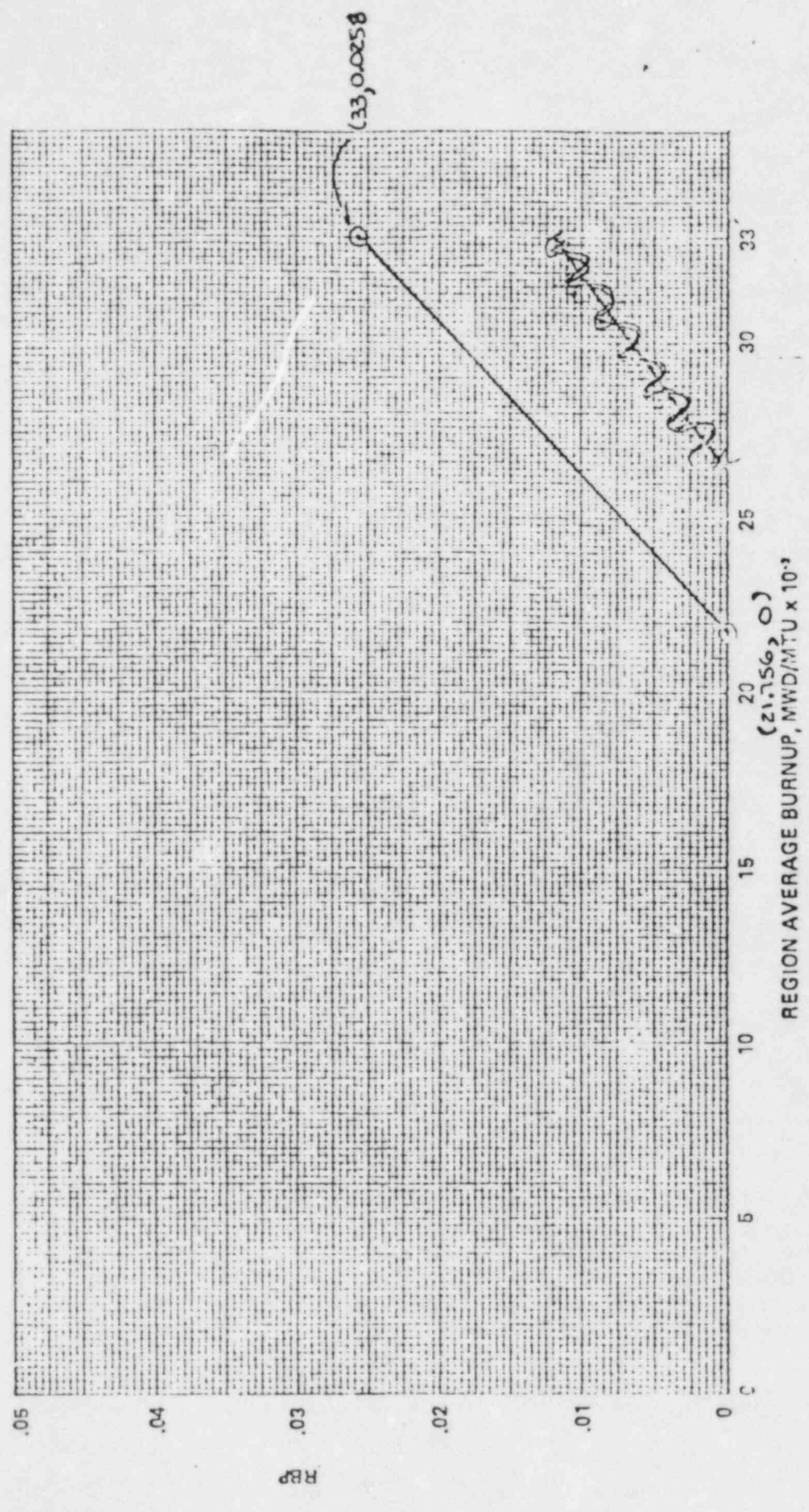

FIGURE 3.2-4 ROD BOW PENALTY VERSUS REGION AVERAGE BURNUP

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 QUARANT 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 ~~or part length~~ 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, control or ~~part length~~ 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 the 4 pairs of symmetric thimble locations, 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		
	~~N~~ 4 Loops In Operation	~~N-1 Loops In Operation & Loop Stop Valves Open~~	~~N-1 Loops In Operation & Loop Stop Valves Closed~~
Reactor Coolant System T_{avg}	≤ ~~(581)~~ 594°F	~~≤ (569)°F~~	~~≤ (570)°F~~
Pressurizer Pressure	≥ 2220 psia*	~~≥ (2220) psia*~~	~~≥ (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.

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 the performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

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

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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*	13
2. Power Range, Neutron Flux - High Setpoint	4	2	3	1, 2	2#
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	1	2	3*, 4*, 5*	13
C. Shutdown	2	0	1	3, 4, and 5	5
7. Overtemperature ~~ΔT~~ N-16					
~~A. Four Loop Plant~~					
~~Four Loop Operation~~	4	2	3	1, 2	6#
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
~~B. Three Loop Plant~~					
~~Three Loop Operation~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ~~ΔT~~ N-16					
~~A. Four Loop Plant~~					
~~Four Loop Operation~~	4	2	3	1, 2	6#
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
~~B. Three Loop Plant~~					
~~Three Loop Operation~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~
9. Pressurizer Pressure-Low					
~~A. Four Loop Plant~~	4	2	3	1	6#
~~B. Three Loop Plant~~	~~3~~	~~2~~	~~2~~	~~1~~	~~7#~~
10. Pressurizer Pressure--High					
~~A. Four Loop Plant~~	4	2	3	1, 2	6#
~~B. Three Loop Plant~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
11. Pressurizer Water Level--High	3	2	2	1	7#
12. Loss of Flow					
A. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
B. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Steam Generator Water Level--Low-Low	4 ~~3~~/stm. gen.	2/stm. gen. in any operating stm. gen.	3 ~~2~~/stm. gen. each operating stm. gen.	1, 2	~~*#~~ 6#
14. ~~Steam Generator Water Level - Low Coincident With Steam/Feedwater Flow Mismatch~~	~~2 stm. gen. level and 2 stm/feed-flow mismatch in each stm. gen.~~	~~1 stm. gen. level coincident with 1 stm./feed-flow mismatch in same stm. gen.~~	~~1 stm. gen. level and 2 stm/geed-flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm/feed-flow mismatch in same steam gen.~~	~~1, 2~~	~~7#~~
15. Undervoltage-Reactor Coolant Pumps					
~~A. Four Loop Plant~~	4-1/bus	2	3	1	6#
~~B. Three Loop Plant~~	~~3-1/bus~~	~~2~~	~~2~~	~~1~~	~~7#~~
16. Underfrequency-Reactor Coolant Pumps					
~~A. Four Loop Plant~~	4-1/bus	2	3	1	6#
~~B. Three Loop Plant~~	~~3-1/bus~~	~~2~~	~~2~~	~~1~~	~~7#~~
17. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	4	1	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESF	2	1	2	1, 2	12
19. ~~Reactor Coolant Pump Breaker Position Trip~~					
~~A. Above P-8~~	~~1/breaker~~	~~1~~	~~1/breaker~~	~~1~~	~~10#~~
~~B. Above P-7 and below P-8~~	~~1/breaker~~	~~2~~	~~1/breaker per operating loop~~	~~1~~	~~11#~~
20. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
B. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
C. Power Range Neutron Flux, P-8	4	2	3	1	8
D. Reactor Trip on Turbine Trip P-9	4	2	3	1	8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
D. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
E. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
21. Reactor Trip Breakers	2	1	2	1, 2	12
	2	1	2	3*, 4*, 5*	13
22. Automatic Trip Logic	2	1	2	1, 2	12
	2	1	2	3*, 4*, 5*	13

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position, 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.

ACTION STATEMENTS

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

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

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

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

ACTION 7 - 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 8 - 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 9~~ - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

~~ACTION 10~~ - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below the P-8 (Power Range Neutron Flux Interlock) setpoint within the next 2 hours. Operation below the P-8 setpoint may continue pursuant to ACTION 11.

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

ACTION 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; 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 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPEARABLE status within 48 hours or open the reactor trip breakers within the next hour.

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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~~ N-16	$\leq$ (4.0) seconds*
8. Overpower ~~ΔT~~ N-16	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. ~~(This provision is not applicable to CP's docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)~~

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow	
A. Single Loop (Above P-8)	$\leq$ (1.0) seconds
B. 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 Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch~~	~~Not Applicable~~
15. Undervoltage-Reactor Coolant Pumps	< (1.5) seconds
16. Underfrequency-Reactor Coolant Pumps	< (0.6) seconds
17. Turbine Trip	
A. Low Fluid Oil Pressure	Not Applicable
B. Turbine Stop Valve	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. ~~Reactor Coolant Pump Breaker Position Trip~~	~~Not Applicable~~
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	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(9)	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
	Low Setpoint	S(9)	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(9)	R(4, 5)	S/U(1),M	N.A.	N.A.	$1^{\#\#\#}$, 2
6.	Source Range, Neutron Flux	S(9)	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	$2^{\#\#}$, 3, 4, 5
7.	Overtemperature ~~ΔT~~ N-16	S	R	M	N.A.	N.A.	1, 2
8.	Overpower ~~ΔT~~ N-16	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.	N.A.	N.A.	S/U(1, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	N.A.	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 Coolant Pump Breaker Position Trip~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~1~~
20. 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
D Reactor Trip on Turbine Trip P-9	N.A.	R	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
E ~~D~~. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M (8)	N.A.	N.A.	1, 2
F ~~E~~. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M (8)	N.A.	N.A.	1
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7)	N.A.	1, 2, 3*, 4*, 5*
22. 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) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Compare 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.

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

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

(6) - Incore - Excore Calibration.

(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 applicable.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the 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~~ STATION SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3#	20*
e. ~~Differential Pressure Between Steam Lines - High~~				~~1, 2, 3##~~	
~~i) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15*~~
~~Three Loops Operating~~	~~3/operating steam line~~	~~1###/steam line any operating steam line~~	~~2/operating steam line~~		~~16~~
e Steamline Pressure Low	3/steam line	2/steamline in any steamline	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
SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION CONTROL ROOM ISOLATION, START DIESEL GENERATORS CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER (Continued)					
ii) Three Loop Plant					
Three Loops Operating	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line		15*
Two Loops Operating	3/operating steam line	2###/steam line twice in either operating steam line	2/operating steam line		16
f. Steam Flow in Two Steam Lines-High				1, 2, 3##	
i) Four Loop Plant					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		15*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		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
SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER (Continued)					
ii) Three Loop Plant					
Three Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		15*
Two Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		16
Coincident With Either					
T_{avg}--Low-Low				1, 2, 3##	
i) Four Loop Plant					
Four Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops		15*
Three Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		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
SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER (Continued)					
ii) Three Loop Plant					
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		15*
Two Loops Operating	1 T_{avg}/ operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any operating loop		16
Or, Coincident With					
Steam Line Pressure-Low				1, 2, 3##	
i) Four Loop Plant					
Four Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops		15*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		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
SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER (Continued)					
ii) Three Loop Plant					
Three Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops		15*
Two Loops Operating	1 pressure/ loop	1### pressure in any operating loop	1 pressure any operating loop		16
2. CONTAINMENT SPRAY					
a. Manual	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-- ~~High-High~~ High-3	4	2	3	1, 2, 3	17
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	19
2) Safety Injection	See 1 above for all Safety Injection initiating functions and requirements				

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
CONTAINMENT ISOLATION (continued)					
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Phase "B" Isolation					
1) Manual	2	1 with 2 coincident switches	2	1, 2, 3, 4	19
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-~~High~~ 3	4	2	3	1, 2, 3	17
c. Purge and Exhaust Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	18
2) Containment Radioactivity-High	~~4~~ 1	~~2~~ 1	~~2~~ 1	1, 2, 3, 4	18
3) Safety Injection	See 1 above for all Safety Injection initiating functions and requirements				

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	1/steam line	1/steam line	1/operating steam line	1, 2, 3	24
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Containment Pressure-- ~~High-High~~ High-2	4	2	3	1, 2, 3	17
~~d. Steam Flow in Two Steam Lines--High~~				~~1, 2, 3~~	
~~i) Four Loop Plant~~					
~~Four Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Three Loops Operating~~	~~2/operating steam line~~	~~1### /any operating steam line~~	~~1/operating steam line~~		~~16~~
~~ii) Three Loop Plant~~					
~~Three Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Two Loops Operating~~	~~2/operating steam line~~	~~1### /any operating steam line~~	~~1/operating steam line~~		~~16~~
d. Steam Line Pressure —Low	3/steam line	2/steam line in any Steam Line	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
STEAM LINE ISOLATION (Continued)					
~~Coincident With Either~~					
T_{avg}--Low-Low				1, 2, 3	
i) Four Loop Plant					
Four Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops		15*
Three Loops Operating	1 T_{avg}/operating loop	1[###] T_{avg} in any operating loop	1 T_{avg} in any two operating loops		16
ii) Three Loop Plant					
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		15*
Two loops Operating	1 T_{avg}/operating loop	1[###] T_{avg} in any operating loop	1 T_{avg} in any operating loop		16
e. Negative Steam Pressure Rate-high	3/steam Line	2/steam Line in any steam Line	2/Steam Line	3[#], 4	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
~~STEAM LINE ISOLATION (Continued)~~					
Or, Coincident With					
Steam Line Pressure-Low				1, 2, 3	
i) Four Loop Plant					
Four Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 3 loops		15*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops		16
ii) Three Loop Plant					
Three Loops Operating	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops		15*
Two Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure any operating loop		16
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	** 3/stm. gen.	2/stm. gen. in any operating stm gem.	2/stm. gen. in each operating stm. gen.	1, 2	15*
b. Automatic Actuation Logic and Actuation Relay	2	1	2	1, 2,	22

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. AUXILIARY FEEDWATER					
a. Manual Initiation	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Stm. Gen. Water Level-Low-Low					
i. Start Motor-Driven Pumps	~~3~~ 4/stm. gen.	2/stm. gen. in any operating stm gen.	~~2~~ 3/stm. gen. in each operating stm. gen.	1, 2, 3	15*
ii. Start Turbine-Driven Pump	~~3~~ 4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	~~2~~ 3/stm. gen in each operating stm. gen	1, 2, 3	15*
d. ~~Undervoltage-RCP Start Turbine-Driven Pump~~	~~4-1/bus~~	~~2~~	~~3~~	~~1, 2~~	~~20*~~
e. Safety Injection Start Motor-Driven Pumps ~~and Turbine-Driven Pump~~	See 1 above for all Safety Injection initiating functions and requirements				
f. Station Blackout Start Motor-Driven Pumps and Turbine-Driven Pump	2	1	2	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
AUXILIARY FEEDWATER (continued)					
g. Trip of Main Feedwater Pumps Start Motor-Driven Pumps ~~and Turbine-Driven Pump~~	2/pump	1/pump	1/pump	1, 2	19
7. AUTOMATIC INITIATION OF SWITCHOVER TO CONTAINMENT SUMP					
a. RWST Level - Low	4	2	3	1, 2, 3, 4	17
Coincident With					
~~Containment Sump Level - High~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3, 4~~	~~17~~
~~And~~					
Safety Injection	See 1 above for Safety Injection initiating functions and requirements				
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
8. ~~LOSS OF POWER~~					
~~a. 4 kv Bus~~ 6.9 Safeguards ~~Loss of Voltage~~	~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3, 4~~	~~20*~~
~~b. Grid Degraded Voltage~~	~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3, 4~~	~~20*~~

INSERT A

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
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23

INSERT A
FOR TABLE 3.3-3

#8 LOSS of POWER

FUNCTIONAL UNIT	TOTAL OF CH		IMUM NNELS RABLE	APPLICABLE MODES	ACTION
8. LOSS OF POWER (6.9KV Safeguards System Undervoltage)					
a. Preferred Offsite Source Undervoltage:					
Undervoltage Relays	2/Bus	2/Bus	2/Bus	1,2,3,4	20*
Diesel Start Timer	1/Bus	1/BUS	1/BUS	1,2,3,4	20*
Source Bkr. Trip Timer	1/BUS	1/Bus	1/BUS	1,2,3,4	20*
b. BUS Undervoltage:					
1). Diesel Start:					
Undervoltage Relays	2/BUS	2/BUS	2/BUS	1,2,3,4	20*
Timer	1/BUS	1/BUS	1/BUS	1,2,3,4	20*
2) Initiation of Solid State Safeguards System Sequencer:					
Undervoltage Relays	4/BUS	2/BUS	3/BUS	1,2,3,4	20*
Timers	4/BUS	2/BUS	3/BUS	1,2,3,4	20*

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 T_{avg} Interlock) setpoint.

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

*The provisions of Specification 3.0.4 are not applicable.

** (INSERT A)

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 hour 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 a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.~~

ACTION 17 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - 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 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

ACTION 21 - 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 22 - 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 23 With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 24 With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification (3.7.1.5).

for Page 3/4 3-29

INSERT A

** The channel which provides a steam generator water level control signal (if one of the 3 specified channels is selected to provide input into steam generator water level control) must be maintained in the tripped condition with the exception that the channel may be taken out of the tripped condition to allow testing of redundant channels.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTANMENT COOLING FANS AND ESSENTIAL SERVICE WATER.		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
c. Containment Pressure--High -1	≤ ~~5~~ 3.7 psig	≤ ~~5.5~~ 4.1 psig
d. Pressurizer Pressure--Low	≥ ~~1765~~ 1810 psig	≥ ~~1755~~ 1800 psig
e. ~~Differential Pressure Between Steam Lines--High~~	~~≤ 100 psi~~	~~≤ 112 psi~~
f. ~~Steam Flow in Two Steam Lines--High~~	~~≤ 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~~	~~≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load~~
~~Coincident With Either~~		
~~1. T_{avg}--Low-Low, or~~	~~≥ (541)°F~~	~~≥ (539)°F~~
~~2. Steam Line Pressure--~~Low	~~≥ (600) psig~~	~~≥ (580) psig~~
e. Steam line Pressure -Low	≥ 585 psig	≥ 565 PSIG (a)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
c. Containment Pressure--High-~~High~~ 3	≤ 18.7 ~~(23)~~ psig	≤ 19.1 ~~(22)~~ psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Safety Injection	See 1 above for all Safety Injection Trip Setpoints/ Allowable Values	
3. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
3. Containment Pressure--High-~~High~~ 3	≤ 18.7 ~~(20)~~ psig	≤ 19.1 ~~(22)~~ psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
CONTAINMENT ISOLATION (continued)		
c. Purge and Exhaust Isolation		
1. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
2. Containment Radioactivity--High	(≤ x background)	(≤ x background)
3. Safety Injection	See 1 above for all Safety Injection Trip Setpoints/ Allowable Values	
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
c. Containment Pressure--High ~~High~~ 2	≤ (~~20~~ 6.7) psig	≤ (~~22~~ 7.1) psig
d. ~~Steam Flow in Two Steam Lines--High~~	~~≤ 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~~	~~≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load~~
d. steam Line pressure - Low	≥ 585 psig	≥ 565 psig
e. negative steam Pressure rate - high	≥ 100 psi/sec	≥ 110 psi/sec

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
~~STEAM LINE ISOLATION (continued)~~		
~~Coincident With Either~~		
~~1. T_{avg}--Low-Low, or~~	~~$\geq$ (541)°F~~	~~$\geq$ (539)°F~~
~~2. Steam Line Pressure--Low~~	~~$\geq$ (600) psig~~	~~$\geq$ (580) psig~~
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water level--High-High	82.4 $\leq$ (~~67~~)% of narrow range instrument span each steam generator	83.4 $\leq$ (~~68~~)% of narrow range instrument span each steam generator
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
6. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
c. Steam Generator Water Level-Low-Low	47% $\geq$ (~~18~~)% of narrow range instrument span each steam generator	46% $\geq$ (~~17~~)% of narrow range instrument span each steam generator
d. ~~Undervoltage - RCP~~	~~$\geq$ (70)% RCP bus voltage~~	~~$\geq$ (69)% RCP bus voltage~~
e. Safety Injection	See 1 above for all Safety Injection Trip Setpoints/ Allowable Values	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
AUXILIARY FEEDWATER (continued)		
f. Station Blackout	See 8.b.2) below ~~≥ ( )% Transfer Bus Voltage~~	~~≥ ( )% Transfer Bus Voltage~~
g. Trip of Main Feedwater Pumps	Not Applicable	Not Applicable
7. AUTOMATIC INITIATION OF SWITCHOVER TO CONTAINMENT SUMP		
a. RWST Level - Low Coincident With	≥ (~~130"~~ 18'-8") from tank base	≥ (~~126"~~ 17'-9") from tank base
~~Containment Sump Level - High and~~	~~≤ (30") above elev. (680')~~	~~≤ (32.5") above elev. (680')~~
Safety Injection	See 1 above for all Safety Injection Trip Setpoints/ Allowable Values)	
b. Automatic Actuation Logic and Actuation Relays	Not Applicable	Not Applicable
~~8. LOSS OF POWER~~		
~~a. 4.16~~ 6.9 ~~kV Emergency Bus Undervoltage (Loss of Voltage)~~	~~(±) volts with a (±) second time delay~~	~~(±) volts with a (±) second time delay~~
~~b. 4.16~~ 6.9 ~~kV Emergency Bus Undervoltage (Degraded Voltage)~~	~~(±) volts with a (±) second time delay~~	~~(±) volts with a (±) second time delay~~

INSERT B

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, P-11	≤ ~~(2000)~~ 1960 psig	≤ ~~(2010)~~ 1970 psig
~~b. Low-Low T_{avg}, P-12~~	~~(543)°F~~	~~≥ (541)°F and ≤ (545)°F~~
c. Reactor Trip, P-4	Not Applicable	Not Applicable

(a) Time constants utilized in the lead-lag controller for steam pressure-Low are $\tau_1 \geq 50$ seconds and $\tau \leq 5$ seconds.

(b) The time constant utilized in the rate-lag controller for negative steam pressure rate-high = 50 seconds.

INSERT B
FOR TABLE 3.3-4 #8 LOSS OF POWER

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. LOSS OF POWER (6.9KV Safeguards System Undervoltage)		
a. Preferred Offsite Source Undervoltage:		
Undervoltage Relays	4800V	≥ 4692V
Diesel Start Timer	0.75 Sec.	≤ 0.825 S
Source Bkr. Trip Timer	0.5 Sec.	≤ 0.55 S
b. Bus Undervoltage:		
1). Diesel Start:		
Undervoltage Relays	2100V	≥ 1992V
Timer	0.75 Sec.	≤ 0.825 S
2) Initiation of Solid State Safeguards System Sequencer:		
Undervoltage Relays	4800V	≥ 4692V
Timers		≤ 0.55 Sec.

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. Containment Spray	Not Applicable
c. Containment Isolation	
Phase "A" Isolation	Not Applicable
Phase "B" Isolation	Not Applicble
Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
e. Feedwater Isolation	Not Applicable
f. Auxiliary Feedwater	Not Applicable
g. ~~Essential~~ Station Service Water	Not Applicable
h ~~Containment Cooling Fans~~	~~Not Applicable~~
i. Control Room Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27.0[(1)]/12[(5)]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0[(3)]
d. Containment Isolation-Phase "A"	≤ 17.0[(2)]/27.0[(1)]
e. Containment Vent and Purge Isolation	≤ 25.0[(1)]/10.0[(2)]
f. Auxiliary Feedwater Pumps	≤ 60.0
g. ~~Essential~~ Station Service Water System	≤ 32.0[(2)]/47.0[(1)]
h. ~~Containment Cooling Fans~~	~~≤ (55.0)[(1)]/(40.0)[(2)]~~
i. Control Room Isolation	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0[1]/12.0[5]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0[3]
d. Containment Isolation-Phase "A"	≤ 17.0[2]/27.0[1]
e. Containment Vent and Purge Isolation	≤ 25.0[1]/10.0[2]
f. Auxiliary Feedwater Pumps	≤ 60.0
g. ~~Essential~~ Station Service Water System	≤ 47.0[1]/32.0[2]
~~h. Containment Cooling Fans~~	~~≤ (55.0)[1]/(40.0)[2]~~
i. Control Room Isolation	Not Applicable
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 22.0[4]/12.0[5]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0[3]
d. Containment Isolation-Phase "A"	≤ 17.0[2]/27.0[1]
e. Containment Vent and Purge Isolation	≤ 25.0[1]/10.0[2]
f. Auxiliary Feedwater Pumps	≤ 60.0
g. ~~Essential~~ Station Service Water System	≤ 32.0[2]/47.0[1]
~~h. Containment Cooling Fans~~	~~≤ (55.0)[1]/(40.0)[2]~~
i. Control Room Isolation	Not Applicable
5. Negative Steam Pressure Rate - High ~~Steam Flow in Two Steam Lines - High Coincident with T_{avg} - Low-Low~~	
~~a. Safety Injection (ECCS)~~	~~≤ (24.0)[4]/(14.0)(5)~~
~~b. Reactor Trip (from SI)~~	~~≤ (4.0)~~
~~c. Feedwater Isolation~~	~~≤ (9.0)[3]~~
~~d. Containment Isolation-Phase "A"~~	~~≤ (19.0)[2]/(29.0)[1]~~
~~e. Containment Vent and Purge Isolation~~	~~≤ (27.0)[1]/(12.0)[2]~~
~~f. Auxiliary Feedwater Pumps~~	~~≤ (60.0)~~
~~g. Essential Service Water System~~	~~≤ (34.0)[2]/(49.0)[1]~~
h. Steam Line Isolation	≤ ~~(9.0)~~[3] 7.0
~~i. Containment Cooling Fans~~	~~≤ (57.0)[1]/(42.0)[2]~~
~~j. Control Room Isolation~~	~~Not Applicable~~

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. ~~Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low~~	
a. Safety Injection (ECCS)	$\leq (12.0)^{(5)}/(22.0)^{(4)}$
b. Reactor Trip (from SI)	$\leq (2.0)$
c. Feedwater Isolation	$\leq (7.0)^{(3)}$
d. Containment Isolation-Phase "A"	$\leq (17.0)^{(2)}/(27.0)^{(1)}$
e. Containment Vent and Purge Isolation	$\leq (25.0)^{(1)}/(10.0)^{(2)}$
f. Auxiliary Feedwater Pumps	$\leq (60.0)$
g. Essential Service Water System	$\leq (32.0)^{(2)}/(47.0)^{(1)}$
h. Steam Line Isolation	$\leq (9.0)^{(3)}$
i. Containment Cooling Fans	$\leq (55.0)^{(1)}/(40.0)^{(2)}$
j. Control Room Isolation	Not Applicable
7. Containment Pressure--High-~~High~~ 3	
a. Containment Spray	$\leq (45.0)^{(2)}/(57.0)^{(1)}$
b. Containment Isolation-Phase "B"	$\leq (65)^{(1)}/(75)^{(2)}$
7A Containment Pressure--High-2	
~~c.~~ a. Steam Line Isolation	$\leq$ ~~(9.0)~~$^{(3)}$ 7.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq (2.5)$
b. Feedwater Isolation	$\leq$ ~~(7.0)~~$^{(3)}$ 11.0
9. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq (60.0)$
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq (60.0)$
10. Containment Radioactivity - High	
a. Purge and Exhaust Isolation	$\leq (25.0)^{(1)}/(10.0)^{(2)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. RWST Level-Low -Low Coincident with ~~Containment Sump Level High and~~ Safety Injection	
a. Automatic Switchover to Containment Sump	≤ (250)(2)/(265)(1)
12. ~~Undervoltage RCP~~	
~~a. Turbine-driven Auxiliary Feedwater Pumps~~	~~≤ (60.0)~~
13. Station Blackout	
a. Auxiliary Feedwater Pumps	≤ (60.0)
14. Trip of Main Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
15 LOSS OF POWER (6.9KV Safeguards System Undervoltage)	
a. Preferred Offsite Source Undervoltage:	
Diesel Start	0.75
Source Bkr. Trip	0.5
b. Bus Undervoltage:	
1) Diesel Start	0.75
2) Initiation Solid State Safeguard System Sequencer	0.5

3-5

TABLE 3.~~35~~ (Continued)

TABLE NOTATION

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

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

(3) Air operated valves

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

(5) Diesel generator starting and sequence loading delays not included. RHR pumps not 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. Containment Pressure-High — 1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. ~~Differential Pressure Between Steam Lines--High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
f. ~~Steam Flow in Two Steam Lines--High Coincident With Either~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
~~1. T_{avg}--Low-Low, or~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
2. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. CONTAINMENT SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure--High ~~-High~~ 3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY 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) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure--High-~~High~~ 3	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-logical-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection	See 1 above for all 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.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure--High ~~High~~ 2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Flow in Two Steam Lines--High Coincident With Either~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
~~1. T_{avg}--Low-Low or~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
2. Steam Line Pressure--Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Negative Steam Pressure Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
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. AUXILIARY 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 - RCP~~	~~N.A.~~	~~R~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1~~
e. Safety Injection	See 1 above for all Safety Injection Surveillance Requirements							
f. Station Blackout	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2
AUTOMATIC Initiation of SWITCHOVER TO CONTAINMENT SUMP								
a. RSWT Level - Low-Low Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
~~Containment Sump Level High And~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
Safety Injection	See 1 above for all Safety Injection Surveillance Requirements							
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
8. LOSS OF POWER								
a. Preferred Offsite Source undervoltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Bus Undervoltage	N.A.	R	N.A.	R.	N.A.	N.A.	N.A.	1, 2, 3, 4

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.

INSTRUMENTATION

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. Fuel Storage Pool Area	1/Pool	**	≤ 15 mR/hr	$10^{-1} - 10^{4}$ mr/hr	25
~~i. Criticality Monitor~~	~~(1)~~	~~*~~	~~≤ 15 mR/hr~~	~~($10^{-1} - 10^{4}$) mR/hr~~	~~25~~
~~ii. Ventilation System Isolation~~	~~(1)~~	~~**~~	~~(≤ 2 x background)~~	~~($1 - 10^{5}$) cpm~~	~~27~~
~~b. Containment - Purge & Exhaust Isolation~~	~~(1)~~	~~6~~	~~(≤ 2 x background)~~	~~($1 - 10^{5}$) cpm~~	~~28~~
~~c. Control Room Isolation~~	~~(1)~~	~~All MODES~~	~~(≤ 2 x background)~~	~~($10^{-1} - 10^{4}$)mR/hr~~	~~29~~
~~d. Containment Area~~	~~2~~	~~1, 2, 3 & 4~~	~~( ) rad/hr~~	~~$1-10^{8}$ rad/hr~~	~~30~~
2. PROCESS MONITORS					
~~a. Fuel Storage Pool Area - Ventilation System Isolation~~					
~~i. Gaseous Activity~~	~~(1)~~	~~**~~	~~(≤ 2 x background)~~	~~($1 - 10^{5}$) cpm~~	~~27~~
~~ii. Particulate Activity~~	~~(1)~~	~~**~~	~~(≤ 2 x background)~~	~~($1 - 10^{5}$) cpm~~	~~27~~
b. Containment					
i. Gaseous Activity				$10^{-6} - 10^{-2}$ uci/cc	
a) Purge & Exhaust Isolation	~~(~~1~~)~~	6	*** (≤ ~~2 x background~~)	~~($1 - 10^{5}$)~~ cpm	28
b) RCS Leakage Detection	~~(~~1~~)~~	1, 2, 3 & 4	N/A	~~($1 - 10^{5}$)~~ cpm	26
ii. Particulate Activity				$5 \times 10^{-11} - 5 \times 10^{-7}$ uci/cc	
a) Purge & Exhaust Isolation	~~(~~1~~)~~	6	*** (≤ ~~2 x background~~)	~~($1 - 10^{5}$)~~ cpm	28
b) RCS Leakage Detection	~~(~~1~~)~~	1, 2, 3 & 4	N/A	~~($1 - 10^{5}$)~~ cpm	26

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
*** Alarm/trip setpoints will be per the OCDM when purge exhaust operations are in progress

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Radwaste Building Exhaust System	1	1, 2, 3 & 4	N.A.	$1\text{-}10^2$ uCi/cc	30
ii. Auxiliary Building Exhaust System	1	1, 2, 3 & 4	N.A.	$1\text{-}10^3$ uCi/cc	30
iii. Steam Safety Valve Discharge	1/valve	1, 2, 3 & 4	N.A.	$1\text{-}10^3$ uCi/cc	30
iv. Atmospheric Steam Dump Valve Discharge	1/valve	1, 2, 3 & 4	N.A.	$1\text{-}10^3$ uCi/cc	30
v. Shield Building Exhaust System	1	1, 2, 3 & 4	N.A.	$1\text{-}10^4$ uCi/cc	30
vi. Containment Purge & Exhaust System	1	1, 2, 3 & 4	N.A.	$1\text{-}10^5$ uCi/cc	30
vii. Condenser Exhaust System	1	1, 2, 3 & 4	N.A.	$1\text{-}10^5$ uCi/cc	30
i Plant Vent stacks	1	1, 2, 3, +4	N.A.	$10^{-6}\text{-}10^{5}$ µCi/cc	30
ii Main Steam Line	1/steam line	1, 2, 3, +4	N.A.	$1\text{-}10^3$ µCi/cc	30
j. Control Room Isolation	1	All MODES	≤ 2X background	——	29

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of OPERABLE channels less than 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 OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.4.6.1).

ACTION 27 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.12).

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.9).

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

ACTION 30 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel(s) to OPERABLE status within 7 days, or: ~~be in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours.~~

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 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. Fuel Storage Pool Area	S	R	M	**
~~i. Criticality Monitor~~	~~S~~	~~R~~	~~M~~	~~*~~
~~ii. Ventilation System Isolation~~	~~S~~	~~R~~	~~M~~	~~**~~
~~b. Containment - Purge & Exhaust Isolation~~	~~S~~	~~R~~	~~M~~	~~6~~
~~c. Control Room Isolation~~	~~S~~	~~R~~	~~M~~	~~All MODES~~
~~d. Containment Area~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3 & 4~~
2. PROCESS MONITORS				
~~a. Fuel Storage Pool Area - Ventilation System Isolation~~				
~~i. Gaseous Activity~~	~~S~~	~~R~~	~~M~~	~~**~~
~~ii. Particulate Activity~~	~~S~~	~~R~~	~~M~~	~~**~~
b. Containment				
i. Gaseous Activity				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4

~~*With fuel in the storage pool or building.~~
**With irradiated fuel in the storage pool.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
~~i. Radwaste Building~~ Exhaust System	S	R	M	1, 2, 3 & 4
ii. Auxiliary Building Exhaust System	S	R	M	1, 2, 3 & 4
iii. Steam Safety Valve Discharge	S	R	M	1, 2, 3 & 4
iv. Atmospheric Steam Dump Valve Discharge	S	R	M	1, 2, 3 & 4
v. Shield Building Exhaust System	S	R	M	1, 2, 3 & 4
vi. Containment Purge & Exhaust System	S	R	M	1, 2, 3 & 4
~~vii. Condenser Exhaust System~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3 & 4~~
i Plant Vent Stacks	S	R	M	1,2,3+4
ii Main Steam line	S	R	M	1,2,3+4
d Control Room Isolation	S	R	M	All Modes

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.

APPLICABILITY: 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, 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~~ determination of the detector Plateau Voltage 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}.

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With one or more 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.

* except instruments 2.a. and 2.b. which are inaccessible durning operation and will be calibrated during the next shutdown

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		1
a. Containment El. 814'-6"	± 1g	1
b. Containment EL. 1000-6"	± 1g	1
c. Electrical Manhole E1A2	± 1g	1
d.		1
2. Triaxial Peak Accelerographs		
a. Steam Generator	± 5g	1
b. Reactor Piping	± 5g	1
c. Component cooling water Heat Exchanger	± 5g	1
d.		1
e.		1
3. Triaxial Seismic Switches		
a. Containment El. 814'-6"	± 0.12g Vertical	1*
b.	± 0.10g Horizontal	1*
c.		1*
d.		1*
4. Triaxial Response-Spectrum Recorders		
a. Containment El. 814'-6"	±1g from 1.0 to 1.6 Hz	1*
b. Reactor Bldg Internal Str. El. 862'-0"	±2g from 2.0 to 32.0 Hz	1
c. Safeguards Bldg El. 773'-0"		1
d.		1
e.		1
f.		1

*With reactor control room indication

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			
a. Containment El. 814'-6"	M*	R	SA
b. Containment El. 1000'-6"	M*	R	SA
c. Electrical Manhole E 1A2	M*	R	SA
d.	M*	R	SA
2. Triaxial Peak Accelerographs			
a. Steam Generator	NA	R	NA
b. Reactor Piping	NA	R	NA
c. Component Cooling Water Heat Exchanger	NA	R	NA
d.	NA	R	NA
e.	NA	R	NA
3. Triaxial Seismic Switches			
a. Containment El. 814'-6"**	M	R	SA
b. **	M	R	SA
c. **	M	R	SA
d. **	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. Containment El. 814'-6"**	M	R	SA
b. Reactor Bldg Internal Str. El. 862'-0"	NA	R	SA
c. Safeguards Bldg El. 773'-0"	NA	R	SA
d.	NA	R	SA
e.	NA	R	SA
f.	NA	R	SA

*Except seismic trigger
**With reactor control room indications.

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

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. X-ST-4118,	Nominal Elev. 200 ft	*
b. X-ST-4117A,	Nominal Elev. 30 ft	* 1 of 3
c. X-ST-4117B, *	Nominal Elev. 30 ft	
2. WIND DIRECTION		
a. X-ZT-4116,	Nominal Elev. 200 ft	X
b. X-ZT-4115A,	Nominal Elev. 30 ft	* 1 of 3
c. X-ZT-4115B, *	Nominal Elev. 30 ft	
3. AIR TEMPERATURE - DELTA T		
a. X-TT-4126-1,	Nominal Elev. 200 ft	*
and ~~b.~~ X-TT-4125A-1	Nominal Elev. 30 ft	X
b X-TT-4126-2,	Nominal Elev. 200 ft	1 of 2
and X-TT-4125A-2,	Nominal Elev. 30 ft	

* mounted on backup tower

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 200 ft	D	SA
b. Nominal Elev. 30 ft	D	SA
c. Nominal Elev. 30 ft *	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 200 ft	D	SA
b. Nominal Elev. 30 ft	D	SA
c. Nominal Elev. 30 ft *	D	
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 200 ft	D	SA
and ~~b.~~ Nominal Elev. 30 ft	D	SA
b Nominal Elev. 200 ft	D	SA
and Nominal Elev. 30 ft	D	SA

* mounted on backup tower

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(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each 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 TURBINE-DRIVEN AFW PUMP-SPEED INDICATION	HSP	0-7000 RPM	1/PUMP
2 CHARGING PUMP TO CVCS CHARGING AND RCP SEALS - FLOW INDICATION	HSP	0-200 GPM	1
3 LETDOWN - FLOW INDICATION	HSP	0-200 GPM	1
4 WIDE RANGE REACTOR COOLANT TEMP. HOT LEG LOOP 1 & 2 (2 PEN REC.)	HSP	PEN 1, 0-700°F PEN 2, 0-700°F	1/LOOP (ALT. SHUTDOWN)
5 AFW TO STEAM GENERATOR - FLOW INDICATION	HSP	0-550 GPM	1/STEAM GENERATOR
6 TURBINE-DRIVEN AFW PUMP-SUCTION PRESSURE INDICATOR	HSP	25 PSIG	1/TURB. DRIVEN, AFW PUMP
7 TURBINE-DRIVEN AFW PUMP-DISCHARGE PRESSURE INDICATOR	HSP	0-200 PSIG X 10	1/TURB. DRIVEN, AFW PUMP.
8 MOTOR-DRIVEN AFW PUMP - SUCTION PRESSURE INDICATOR	HSP	25 PSIG	1
9 MOTOR-DRIVEN AFW PUMP - DISCHARGE PRESSURE INDICATOR	HSP	0-200 PSIG X 10	1
10 SSW - FLOW INDICATOR	HSP	0-20 GPM X 1000	1/TRAIN

TABLE 3.3-9
REMOTE SHUTDOWN MONITORING
INSTRUMENTATION - HSP

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
11. 6.9 KV BUS VOLTAGE - INDICATOR	HSP	0-9000V PT 7200 120V	1
12. 6.9 KV BUS FREQUENCY INDICATOR	HSP	55-60 65 Hz	1
13. 6.9 KV BUS PREFERRED OFFSITE SOURCE - AMPERES INDICATOR	HSP	0-1000 CT-1000 5A	1
14. 6.9 KV BUS ONSITE SOURCE - AMPERES INDICATOR	HSP	0-1000 CT-1000 5A	1
15. 6.9 KV BUS ALTERNATE OFFSITE SOURCE-AMPERES INDICATOR	HSP	0-1000 CT-1000 5A	1
16. STEAM GENERATOR - LEVEL INDICATOR	HSP	0-100 PCT	1/SG
17. STEAM GENERATOR - PRESSURE INDICATOR	HSP	0-1300 PSIG	1/SG
18. PRESSURIZER - PRESSURE INDICATOR	HSP	1700-2500 PSIG	1
19. PRESSURIZER - LEVEL INDICATOR	HSP	0-100 PCT	1
20. CONDENSATE STORAGE TANK - LEVEL	HSP	0-45 FT	1
21. WIDE RANGE RCS TEMP	HSP	0-700°F	1
22. SOURCE RANGE - NEUTRON PWR	HSP	$10^0 \sim 10^6$ COUNTS/SEC	1

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. TURBINE-DRIVEN AFW PUMP-SPEED INDICATION	M	R
2. CHARGING PUMP TO CVCS CHARGING AND RCP SEALS - FLOW INDICATION	M	R
3. LETDOWN - FLOW INDICATION	M	R
4. WIDE RANGE REACTOR COOLANT TEMP. HOT LEG LOOP 1 & 2 (2 PEN REC)	M	R
5. AFW TO STEAM GENERATOR -FLOW INDICATION	M	R
6. TURBINE-DRIVEN AFW PUMP-SUCTION PRESSURE INDICATOR	M	R
7. TURBINE-DRIVEN AFW PUMP-DISCHARGE PRESSURE INDICATOR	M	R
8. MOTOR-DRIVEN AFW PUMP-SUCTION PRESSURE INDICATOR	M	R
9. MOTOR-DRIVEN AFW PUMP-DISCHARGE PRESSURE INDICATOR	M	R
10. SSW - FLOW INDICATOR	M	R

TABLE 4.3-6
REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENT

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
11. 6.9 KV BUS VOLTAGE INDICATOR	M	R
12. 6.9 KV BUS FREQUENCY INDICATOR	M	R
13. 6.9 KV BUS PREFERRED OFFSITE SOURCE AMPERES INDICATOR	M	R
14. 6.9 KV BUS ONSITE SOURCE - AMPERES INDICATOR	M	R
15. 6.9 KV BUS ALTERNATE OFFSITE SOURCE-AMPERES INDICATOR	M	R
16. STEAM GENERATOR - LEVEL INDICATOR	M	R
17. STEAM GENERATOR-PRESSURE INDICATOR	M	R
18. PRESSURIZER - PRESSURE INDICATOR	M	R
19. PRESSURIZER - LEVEL INDICATOR.	M	R
20. CONDENSATE STORAGE TANK - LEVEL	M	R
21. WIDE RANGE RCS TEMP.	M	R
22. SOURCE RANGE - NEUTRON PWR	S(7)	R(6)

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels 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 instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT 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 - CATEGORY 1

INSTRUMENT ~~(Illustrational Only)~~	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure (Wide Range)	2	1
1a. Containment Pressure (Narrow Range)	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Condensate Storage Tank Level ~~Boric Acid Tank Solution Level~~	2	1
11. Auxiliary Feedater Flow Rate	2/steam generator	1/steam generator
12. Containment Radiation (High Range) ~~Reactor Coolant System Subcooling Margin Monitor~~	2	1
13. Neutron Flux (Source Range) ~~PORV Position Indicator~~	2	1
14. Neutron Flux (Intermediate Range) ~~PORV Block Valve Position Indicator~~	2	1
15. Containment Hydrogen Concentration ~~Safety Valve Position Indicator~~	2 (2 sensors per channel) ~~2/Valve~~	1 (1 sensor per channel) ~~1/Valve~~
~~16. Containment Water Level (Narrow Range)~~	~~2~~	~~1~~
17. Containment Water Level (Wide Range)	2	1
18. ~~In~~ Core ~~Thermocouples~~ Exit Temperature	~~4/core quadrant~~ 2 (4 thermocouples per core quadrant)	~~2/core quadrant~~ 1 (2 thermocouples per core quadrant)

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TABLE 4.3-7

(CATEGORY 1)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT ~~(Illustrational Only)~~	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Wide Range)	M	R
1a Containment Pressure (Narrow Range)	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Condensate Storage Tank Level ~~Boric Acid Tank Solution Level~~	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Containment Radiation (High Range) ~~Reactor Coolant System Subcooling Margin Monitor~~	M	R
13. Neutron Flux (Source Range) ~~PORV Position Indicator~~	M	R
14. Neutron Flux (Intermediate Range) ~~PORV Block Valve Position Indicator~~	M	R
15. Containment Hydrogen Concentration ~~Safety Valve Position Indicator~~	M	R
~~16. Containment Water Level (Narrow Range)~~	~~M~~	~~R~~
17. Containment Water Level (Wide Range)	M	R
18. ~~In~~ Core ~~Thermocouples~~ Exit Temperature	M	R

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CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

b. With both chlorine detection systems inoperable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVELLIANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 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 4.6.1.6).

~~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.~~

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

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 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.

4.3.3.8.2 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.

4.3.3.8.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION (Illustrative**)	MINIMUM INSTRUMENTS OPERABLE*		
	HEAT	FLAME	SMOKE
1. Containment			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
2. Control Room			
3. Cable Spreading			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
4. Computer Room	LATER		
5. Switchgear Room			
6. Remote Shutdown Panels			
7. Station Battery Rooms			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
8. Turbine			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
9. Diesel Generator			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
10. Diesel Fuel Storage			
11. Safety Related Pumps			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
12. Fuel Storage			
Zone 1 Elevation ____			
Zone 2 Elevation ____			

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

**List all detectors in areas required to insure the OPERABILITY of Safety related equipment and indicate instruments which automatically actuate fire suppression systems.

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACITON:

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

4.3.3.9 Each active channel of the loose-part detection systems shall be demonstrated OPERABLE by performance of:

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

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

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

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one stop valve or one control ~~governor~~ valve per high pressure turbine steam lead inoperable ~~and~~/or with one ~~reheat~~ stop valve or one control ~~reheat intercept~~ valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, ~~or close at least one valve in the affected steam lead(s)~~ or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

~~a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.~~

~~1. (Four) high pressure turbine stop valves.~~

~~2. (Four) high pressure turbine governor valves.~~

~~3. (Four) low pressure turbine reheat stop valves.~~

~~4. (Four) low pressure turbine reheat intercept valves.~~

see INSERT ON P 3/4 3-72

c ~~b~~. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

d ~~c~~. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

e ~~d~~. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

a. At least once per 14 days:

1) perform a test of the overspeed trip using the Automatic Turbine Tester (ATT)

2) cycle each of the valves in b.2) through at least one complete cycle from the running position using the ATT

b. or at least once per 7 days:

1) perform a manual test of the turbine overspeed trip

2) manually cycle each of the following valves through at least one complete cycle from the running position.

a) Four high pressure turbine stop valves

b) Four high pressure Turbine control valves

c) Four Low pressure turbine stop valves

d) Four Low pressure Turbine control valves.

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.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the Reactor Coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and Reactor Coolant pump,
2. Reactor Coolant Loop (B) and its associated steam generator and Reactor Coolant pump,
3. Reactor Coolant Loop (C) and its associated steam generator and Reactor Coolant pump,
4. Reactor Coolant Loop (D) and its associated steam generator and Reactor Coolant pump.

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

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the Reactor Coolant and/or residual heat removal (RHR) loops listed below shall be OPERABLE:

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

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

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

4. Reactor Coolant Loop (D) and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal Loop (A),

6. Residual Heat Removal Loop (B).

b. At least one of the above Reactor Coolant and/or RHR loops shall be in operation.**

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 (~~275~~) 295 °F unless ~~1) the pressurizer water volume is less than ____ cubic feet 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.

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.3.3 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

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

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

b. The secondary side water level of at least two steam generators shall be greater than (17)%.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##

ACTION:

a. With less than the above required loops OPERABLE 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 RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The required RHR loop shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

4.4.1.4.1.2 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.

4.4.1.4.1.3 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.

A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to ~~(275)~~ 295°F unless ~~1) the pressurizer water volume is less than ____ cubic feet 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 The required RHR loops shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

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

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.

ISOLATED LOOP (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.4.1.5 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops.

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

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's stop valves; either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F.

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated until:

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

b. The reactor is subcritical by at least 1 percent delta k/k.

APPLICABILITY: ALL MODES.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1 percent delta k/k within 30 minutes prior to opening the cold leg stop valve.

REACTOR COOLANT SYSTEM

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%~~.* +25, -50 PSIG

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.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ~~±1%~~.* +25, - 50 PSIG

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 1656 cubic feet, and at least two groups of pressurizer heaters each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With 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 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.

REACTOR COOLANT SYSTEM

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 inoperable 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. inoperable

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 power removed in order to meet the requirements of Specification 3.4.4.a.

4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

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 limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service, or repaired and is equal to (~~40~~)%* LATER 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.

9. Repair means the action of restoring a tube to useful service condition when shown acceptable, or plugging the tube

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

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~~ repair 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 montns 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

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

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 12 ~~3 N~~ % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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

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

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	repair ~~Plug~~ defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	~~Plug~~ repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	~~Plug~~ repair 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 de-fective tubes and inspect 2S tubes in each other S. G. repair 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 repair	N/A	N/A

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

$\frac{12}{n}\%$

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The containment atmosphere (gaseous or particulate) radioactivity monitoring system,

b. The containment pocket sump level and flow monitoring system, and

c. Either the (containment air cooler condensate flow rate) or a containment atmosphere (gaseous or particulate) 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. Containment atmosphere (gaseous and/or particulate) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment pocket sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,

c. ~~(Specify appropriate surveillance tests depending upon the type of leakage detection system selected.)~~
Containment air cooler condensate flowrate performance of CHANNEL FUNCTIONAL TEST 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. 40 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 containment atmosphere {gaseous or particulate} radioactivity monitor at least once per 12 hours.

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

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 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 pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. ~~At least once per 18 months.~~ Every refueling outage during startup

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

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

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

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

* In lieu of measuring leakrate, leak-tight integrity may be verified by absence of pressure build up in the test line down stream of the valve.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
8948 A, B, C, D	Accumulator Tank Discharge
8956 A, B, C, D	Accumulator Tank Discharge
8905 A, B, C, D	S.I. Hot Leg Injection
8949 A, B, C, D	S.I. Hot Leg Injection
8818 A, B, C, D	R.H.R. Cold Leg Injection
8819 A, B, C, D	S.I. Cold Leg Injection
8701 A, B	R.H.R. Suction Isolation
8702 A, B	R.H.R. Suction Isolation
RH-8705 A, B	R.H.R. Suction Isolation Relief

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

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	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} less than or equal to 250°F

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1.0 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 $100/\bar{E}$ microcuries 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 1-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.

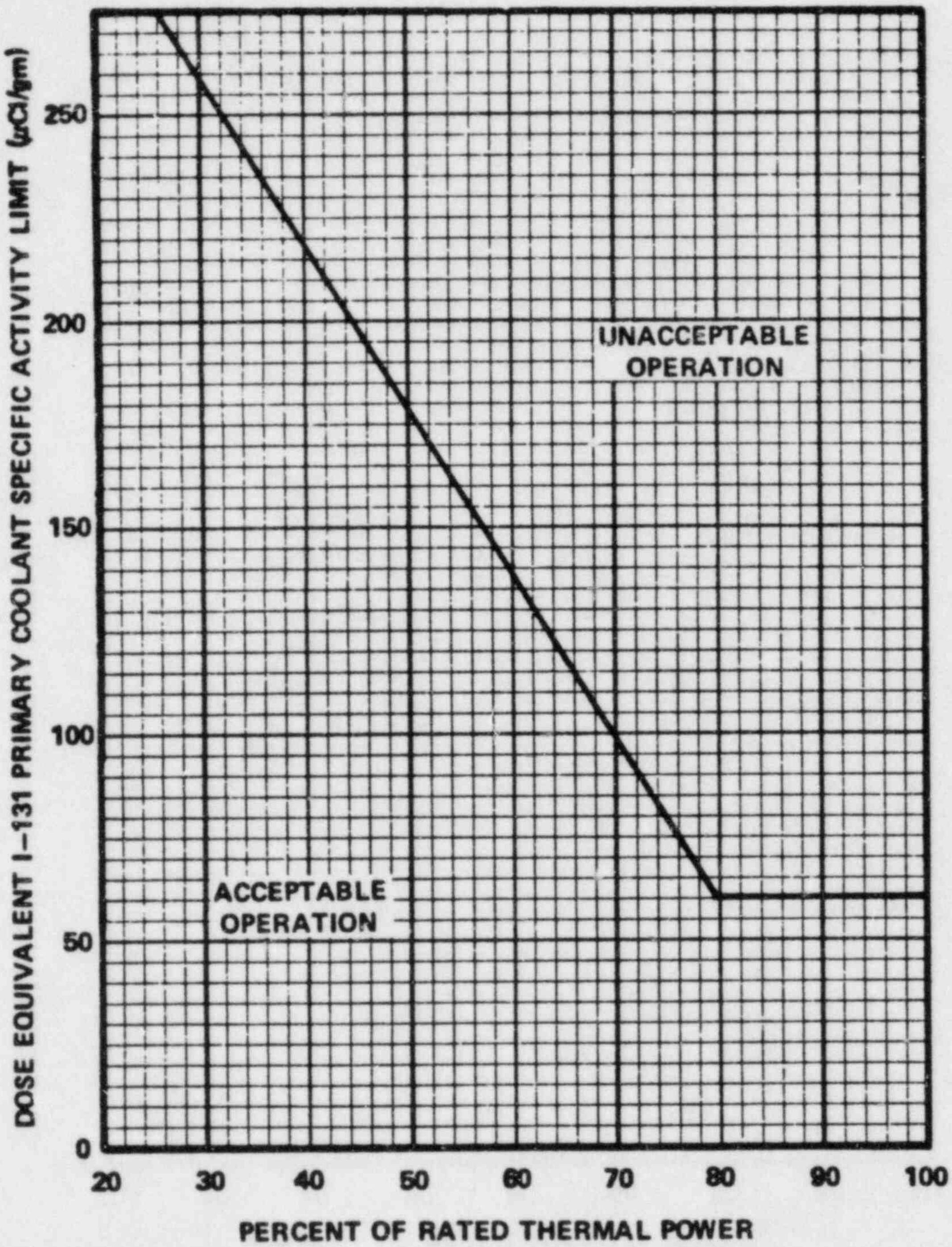


FIGURE 3.4-1

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 1-hour period.

b. A maximum cooldown of (100)°F in any 1-hour period.

c. A maximum temperature change of less than or equal to (10)°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)	ESTIMATED CAPSULE FLUENCE (n/cm^2)
U	58.5°	4.00	1st Refueling	3.6×10^{18}
Y	241°	3.69	5	1.33×10^{19} (a)
V	61°	3.69	9	2.38×10^{19} (b)
X	238.5°	4.00	15	4.31×10^{19}
W	121.5°	4.00	STANDBY	–
Z	301.5°	4.00	STANDBY	–

(a) Approximate fluence at $\frac{1}{4}$ wall thickness at EOL

(b) Approximate fluence at vessel inner wall at EOL

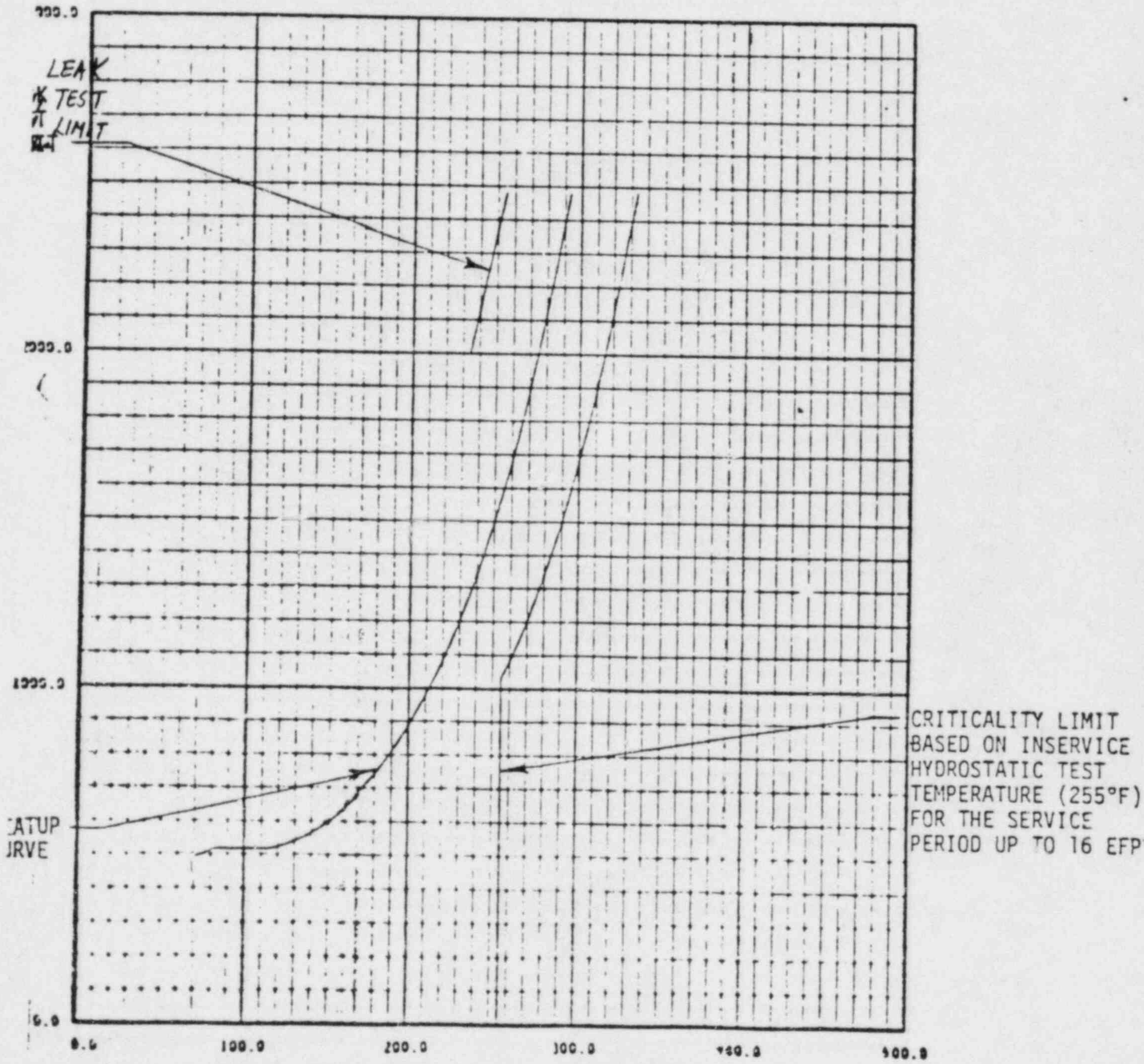
MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT% (ACTUAL CONTENT = 0.05 WT%)

RT_{NDT} INITIAL : CONSERVATIVELY ASSUMED TO BE 40°F (ACTUAL RT_{NDT} = 20°F)

RT_{NDT} AFTER 16 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



3.4-2

Figure 1 COMANCHE PEAK UNIT 1 Reactor Coolant System Heatup Limitations Applicable up to 16 EFPY

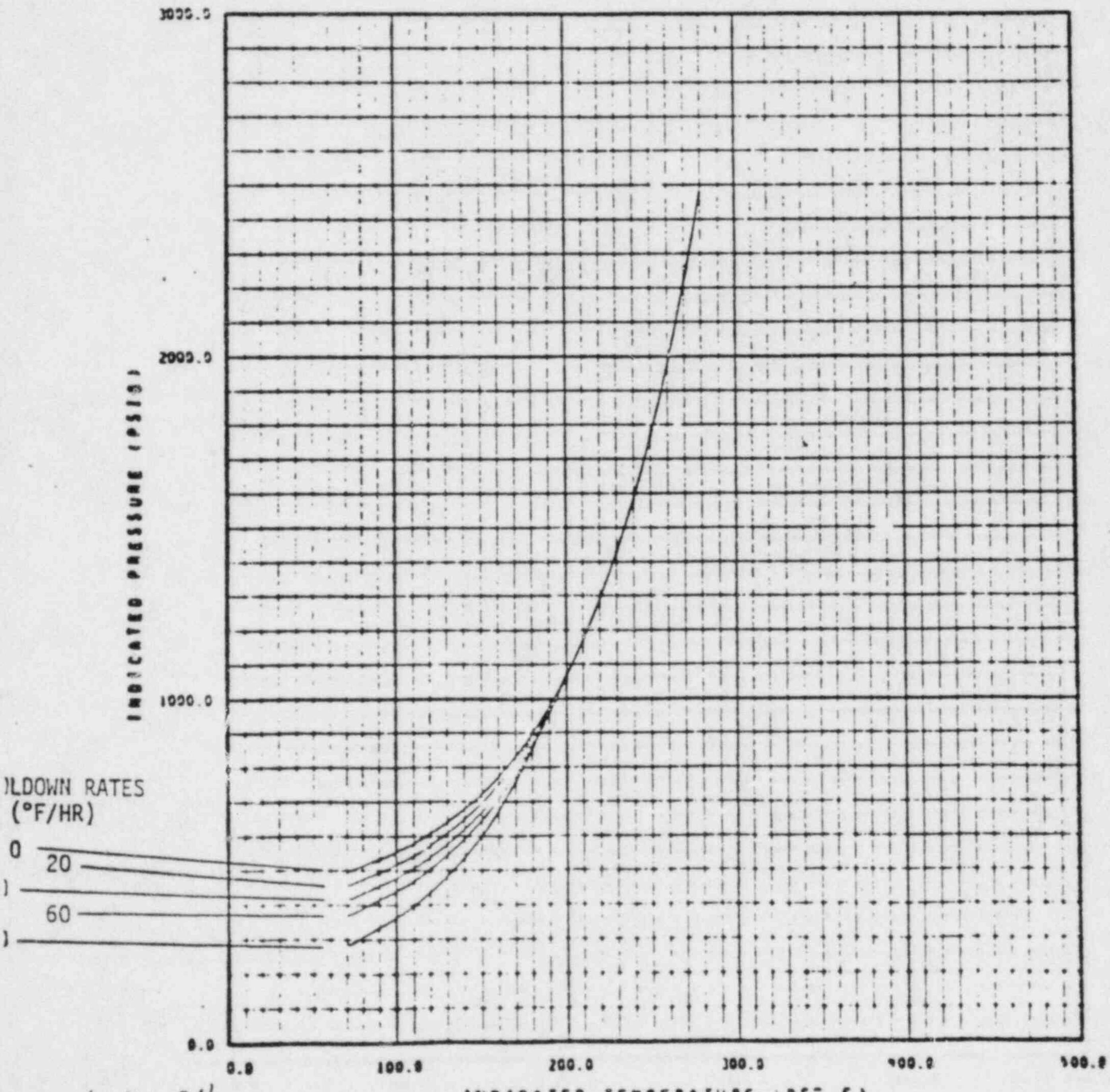
MATERIAL PROPERTY BASIS

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT% (ACTUAL CONTENT = 0.05 WT%)

RT_{NDT} INITIAL : CONSERVATIVELY ASSUMED TO BE 40°F (ACTUAL RT_{NDT} = 20°

RT_{NDT} AFTER 16 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS



3A 4-34 3.4-3

Figure A COMANCHE PEAK UNIT 1 Reactor Coolant System Cooldown Limitations Applicable up to 16 EFPY

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum spray water temperature differential of (~~320~~)°F.
625

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

OVERPRESSURE PROTECTION SYSTEMS

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 maximum setpoints ~~a lift setting of less than or equal to (450) psig, or~~ which vary with RCS temperature as shown on Figure 3.4-4 (selected setpoints are listed in Table 3.4-3).

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

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to ~~(275)~~ 295°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a LATER ( ) square inch vent(s) within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through a LATER ( ) square inch vent(s) within 8 hours.

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

d. The provisions of Specification 3.0.4 are not applicable.

Table ~~3.4.9.~~ 3.4-3

Power Operated Relief Valve Setpoints for Cold Overpressure Mitigation

Temperature (°F)	PORV Setpoint* (psig)
70	505
100	505
150	560
200	792
250	1335
295 ~~300~~	2485 ~~2500~~

* These setpoints are the maximum value versus temperature that will mitigate a ~~the~~ pressure transient ~~that for plant condition~~ to below the Appendix G limits for plant operating conditions as described in the Technical Specifications.

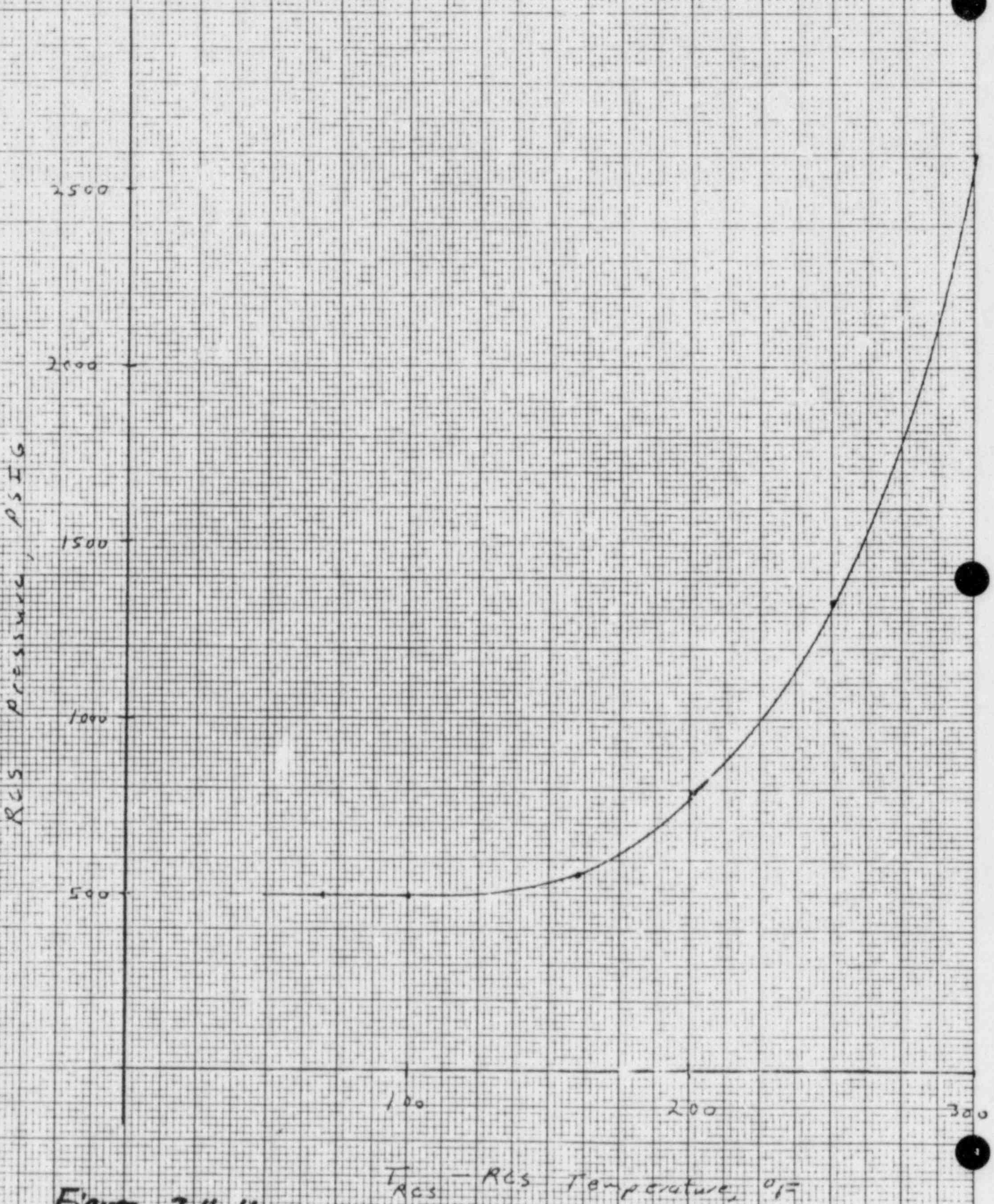


Figure 3.4-4

PORV Setpoints for Overpressure Mitigation

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a 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.

d. Testing pursuant to Specification 4.0.5.

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

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

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES.

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 6190 and 6560 gallons,

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

d. A nitrogen cover-pressure of between 603 and 686 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 1-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 1-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, by the absence of alarms, 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~~) 10 gal. 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 power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

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.

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

a. At least once per 31 days by the performance of a ANALOG CHANNEL OPERATIONAL TEST.

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

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 safety injection pump ~~(four loop plants only)~~,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. 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 containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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
a 8802 A+B	a SI pump to Hot Legs	a closed
b. 8808ABCD	b. Accum Discharge	b. OPEN
c. 8809 A+B	c. RHR TO Cold legs	c. OPEN
d. 8835	d. SI pump to cold Legs	d. OPEN
e 8840	e RHR TO Hot Legs	e closed

b. At least once per 31 days by:

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

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

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

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

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

d. At least once per 18 months by:

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

   a) with a simulated or actual Reactor Coolant System pressure signal greater than or equal to ~~425~~ 386 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 ~~600~~ 750 psig the interlocks will cause the valves to automatically close.

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on (safety injection actuation and automatic switchover to containment sump) test signals.

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

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

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

1. Centrifugal charging pump ≥ 2480 psig

2. Safety Injection pump ≥ 1490 psig

3. Residual heat removal pump ≥ 190 psig

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

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

2. At least once per 18 months.

CCP/SI ~~HPSI~~ System Valve Number	SI ~~LPSI~~ System Valve Number
a. 8810A	a. 8822 A
b. 8810B	b. 8822B
c. 8810C	c. 8822C
d. 8810D	d. 8822D

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 cold leg injection line flow rates, excluding the highest flow rate, is greater than or equal to 338 gpm, and

   b) The total pump flow rate is less than or equal to 550 gpm.

2. For safety injection pump lines, with a single pump running:

   a) The sum of the cold leg injection line flow rates, excluding the highest flow rate, is greater than or equal to 439 gpm, and

   b) The total pump flow rate is less than or equal to 650 gpm.

3. For residual heat removal pump lines, with a single pump running, the sum of the cold leg injection line flow rates is greater than or equal to 4844 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 upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

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

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or 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 ~~and one safety injection pump~~ shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~(275)~~°F.
295

SURVEILLANCE REQUIREMENTS

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

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to ~~(275)~~ 295°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of between ______ and ______ 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 1% 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 contained borated water volume of between 479,900 and 526,300 gallons,

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

c. A minimum water temperature of (~~35~~ 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 within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the (outside) air temperature is less than ~~35~~ 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 3.6.4.1, and the equipment hatch is closed and sealed.

b. By verifying that each containment air lock is OPERABLE per 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 (~~50~~ psig) 48.1 and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 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.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

   1. Less than or equal to L_a, (~~0.20~~ 0.10) percent by weight of the containment air per 24 hours at P_a, (~~50~~ 48.1 psig), or

   2. Less than or equal to L_t, (~~0.10~~ 0.05) percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (~~25~~ 24.05 psig).

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

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 or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (~~50~~ 48.1 psig) or at P_t (~~25~~ 24.05 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

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 (~~50~~ 48.1 psig) or P_t (~~25~~ 24.05 psig.)

d. Type B and C tests shall be conducted with gas at P_a (~~50~~ 48.1 psig) at intervals no greater than 24 months except for tests involving:

1. Air locks, and

2. Penetrations using continuous leakage monitoring systems~~, and~~

~~3. Valves pressurized with fluid from a seal system.~~

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

f. ~~Type B periodic tests are not required for penetrations continuously monitored by the Containment Isolation Valve and Channel Weld Pressurization Systems provided the systems are OPERABLE per Surveillance Requirement 4.6.1.4.~~

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SURVEILLANCE REQUIREMENTS (Continued)

g. ~~Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (55 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.~~

h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a (~~50~~ psig) at intervals no greater than once per 3 years.
48.1

i. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, ~~(50)~~ 48.1 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

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

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

3. Otherwise, be in at least HOT STANDBY within the next 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 containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to P_a (~~50~~ 48.1 psig) for at least 15 minutes.

b. By conducting overall air lock leakage tests at not less than P_a, (~~50~~ 48.1 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 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.
*Exemption to Appendix J of 10 CFR 50.

CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.1.4 The containment isolation valve and channel weld pressurization systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment isolation valve or channel weld pressurization system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The containment isolation valve pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a (55 psig) and has adequate capacity to maintain system pressure for at least 30 days.

4.6.1.4.2 The containment channel weld pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to P_a (50 psig) and has adequate capacity to maintain system pressure for at least 30 days.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment internal pressure shall be maintained between -1.5 and 3.0 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

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 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

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

Location

a. TE-5400, Dome area El. 1066'-0"
b. TE-5401, Containment Az. 270° EL. 911'-9"
c. TE-5402, Containment Az. 152° El. 866'-0"
d. TE-5403, Top of CRDM/RAMS EL. 865'-0"
e. TE-5404, Containment Az. 210° EL. 814'-0"

CONTAINMENT STRUCTURAL INTEGRITY (Prestressed concrete containment with ungrouted tendons and typical dome.)

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Tendons. The containment tendons' structural integrity 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 tendons' structural integrity shall be demonstrated by:

a. Determining that a representative sample* of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) each have a lift off force of between ___ (minimum) and ____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount: _____ log t and the minimum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount: ____ log t where t is the time interval in years from initial tensioning of the tendon to the current testing date. This test shall include an unloading cycle in which each of these tendons is detensioned to determined if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to their observed lift off

*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.

SURVEILLANCE REQUIREMENTS (Continued)

force, ±3%. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously. If the lift off force of any one tendon in the total sample population is out of the predicted bounds (less than minimum or greater than maximum), an adjacent tendon on each side of the defective tendon shall also be checked for lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. This single tendon shall be restored to the required level of integrity. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons checked for lift off force and change in elongation during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

b. Removing one wire or strand from each of a dome, vertical and hoop tendon checked for a lift off force and determining that over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. There are no changes in the presence or physical appearance of the sheathing filler grease.

3. A minimum tensile strength value of ___ psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.7.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

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

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

CONTAINMENT STRUCTURAL INTEGRITY (Prestressed concrete containment with ungrouted tendons and hemispherical dome.)

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Tendons. The containment tendons' structural integrity 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 tendons' structural integrity shall be demonstrated by:

a. Determining that a representative sample* of at least 4% but no less than 4, of the U tendons each have a lift off force of between _____ (minimum) and ______ (maximum) pounds at the first year inspection and that a representative sample* of at least 4%, but no less than 9, of the hoop tendons each have a lift off force of between _____ (minimum) and _____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift off forces shall be decreased from the value determined at the first year inspection by the amount: _____ log t and the minimum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount: ____ log t where t is the time interval in years from initial tensioning of the tendon to the current testing date. This test shall include an unloading cycle in which each of these tendons is detensioned to determined if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to their observed lift off

*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 (U and hoop) may be kept unchanged after the initial selection.

SURVEILLANCE REQUIREMENTS (Continued)

force, ±3%. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously. If the lift off force of any one tendon in the total sample population is out of the predicted bounds (less than minimum or greater than maximum), an adjacent tendon on each side of the defective tendon shall also be checked for lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. This single tendon shall be restored to the required level of integrity. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons checked for lift off force and change in elongation during subsequent tests may be reduced to a representative sample of at least 2%, but no less than 2, of the U tendons and a representative sample of at least 2%, but no less than 3, of the hoop tendons.

b. Removing one wire or strand from one U tendon and one hoop tendon checked for lift off force and determining that over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. There are not changes in the presence or physical appearance of the sheathing filler grease.

3. A minimum tensile strength value of ___ psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.7.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

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

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

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY (Reinforced concrete containment)

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The (~~42~~ 48 inch) containment purge supply and exhaust isolation valves shall be ~~sealed~~ locked closed. Operation with the (~~8~~ 18 inch) ~~purge supply and/or~~ pressure relief exhaust isolation valves open shall be limited to less than or equal to (~~90~~ 500) hours per 365 days.

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

ACTION:

With the (~~42~~ 48 inch) containment purge supply and/or exhaust isolation valve(s) open, or with the (~~8~~ 18 inch) ~~purge supply and/or exhaust~~ pressure relief isolation valve(s) open for more than (~~90~~ 500) hours per 365 days, close the open valve(s) 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

4.6.1.8.1 The (~~42~~ 48 inch) containment purge supply and exhaust isolation valves shall be verified to be:

a. Closed at least once per 24 hours.

b. ~~Sealed~~ Locked closed at least once per 31 days.

4.6.1.8.2 The cumulative time that the (~~8~~ 18 inch) ~~purge supply and exhaust~~ pressure relief isolation valves have been open during the past 365 days shall be determined at least once per 7 days.

4.6.1.8.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed (~~42~~ 48 inch) containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to ~~(~~0.05~~)~~ L_a.

4.6.1.8.4 At least once per 3 months each (~~8~~ 18 inch) containment ~~purge supply and exhaust~~ pressure relief isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to ~~(~~0.05~~)~~ L_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM (Credit taken for iodine removal)

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems (each having two pumps) shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; 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 containment spray system shall be demonstrated OPERABLE:

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

b. By verifying, Flow that on recirculation flow, each pump ~~develops~~ Provides a discharge ~~pressure~~ of greater than or equal to ~~?~~ 3600 ~~psig~~ gpm when tested pursuant to Specification 4.0.5.

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

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

2. Verifying that each spray pump starts automatically on a spray actuation or safety injection test signal.

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

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM (No credit taken for iodine removal)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment spray system inoperable and at least (four) containment cooling fans OPERABLE, restore the inoperable spray system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two containment spray systems inoperable and at least (four) containment cooling fans OPERABLE, restore at least one 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 both spray systems 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 containment spray system inoperable and one group of required containment cooling fans inoperable, restore either the inoperable spray system or the inoperable group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the folloiwng 30 hours. Restore both the inoperable spray system and the inoperable group of cooling fans 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

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

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

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

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

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

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

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

SPRAY ADDITIVE SYSTEM (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

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

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to ~~a~~ respective containment spray system pump flow.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

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

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

d. At least once per 5 years by verifying spray additive and RWST flow ~~each solution flow rate~~ (to be determined during pre-operational tests) ~~from the following drain~~ through the respective test loops ~~connections in~~ of the spray additive system:

~~1. (Drain line location) ____ ± ____ gpm~~

~~2. (Drain line location) ____ ± ____ gpm~~

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CONTAINMENT COOLING SYSTEM (OPTIONAL) (Credit taken for iodine removal by spray systems)

LIMITING CONDITIONS FOR OPERATION

3.6.2.3 (Two) independent groups of containment cooling fans shall be OPERABLE with (two) fan systems to each group. (Equivalent to 100% cooling capacity.)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling fans 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 containment cooling fans inoperable, and both containment spray systems OPERABLE, restore at least one group of cooling fans 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 fans 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 containment cooling fans inoperable and one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans 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 containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each fan group from the control room, and verifying that each fan group operates for at least 15 minutes.

2. Verifying a cooling water flow rate of greater than or equal to ___ gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a _____ test signal.

CONTAINMENT COOLING SYSTEM (OPTIONAL) (No credit taken for iodine removal by spray systems)

LIMITING CONDITION FOR OPERATION

3.6.2.3 (Two) independent groups of containment cooling fans shall be OPERABLE with (two) fan systems to each group. (Equivalent to 100% cooling capacity.)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling fans 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 containment cooling fans inoperable, and both containment spray systems OPERABLE, restore at least one group of cooling fans 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 fans 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 containment cooling fans inoperable and one containment spray system inoperable, restore either the inoperable group of containment cooling fans or 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 both the inoperable group of containment cooling fans and the inoperable spray system 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 containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each fan group from the control room and verifying that each fan group operates for at least 15 minutes.

   2. Verifying a cooling water flow rate of greater than or equal to ___ gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a ______ test signal.

3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent containment iodine cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.3 Each iodine cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal absorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ____ 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 _____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of 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 filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of _____ cfm ± 10%.

2. Verifying that the system starts on either a Safety Injection Test Signal or on a Containment Pressure -High Test Signal.

3. Verifying that the filter cooling bypass valves can be opened by operator action.

4. Verifying that the heaters dissipate _______ ± _______ kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the 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 _______ 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 _____ cfm ± 10%.

*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 Containment Ventilation ~~Purge and Exhaust~~ isolation test signal, each ~~Purge and Exhaust~~ ventilation valve actuates to its isolation position.

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.

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TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

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A. PHASE "A" ISOLATION VALVES

Valve No.	FSAR Table Reference No.(1)	Line or Service	Isolation Time (Seconds)	Note
HV-2333A	1	Main Steam From Steam Generator #1	5	#
HV-2333B	2	Main Steam From Steam Generator #1	5	#
HV-2409	3	Drain From Main Steam Line #1	5	#
HV-2334A	6	Main Steam From Steam Generator #2	5	#
HV-2334B	7	Main Steam From Steam Generator #2	5	#
HV-2410	8	Drain From Main Steam Line #2	5	#
HV-2335A	10	Main Steam From Steam Generator #3	5	#
HV-2335B	11	Main Steam From Steam Generator #3	5	#
HV-2411	12	Drain From Main Steam Line #3	5	#
HV-2336A	14	Main Steam From Steam Generator #4	5	#
HV-2336B	15	Main Steam From Steam Generator #4	5	#

Valve No.	FSAR Table Reference No.(1)	Line or Service	Isolation Time (Seconds)	Note
HV-2412	16	Drain From Main Steam Line #4	5	#
HV-2134	19	Feedwater to Steam Generator #1	5	#
1HV-2154	20	Feedwater Sample (FW to Stm Gen #1)	5	#
HV-2135	21	Feedwater to Steam Generator #2	5	#
1HV-2155	22	Secondary Sample (FW to Stm Gen #2)	5	#
HV-2136	23	Feedwater to Steam Generator #3	5	#
HV-2156	24	Feedwater Sample (FW to Stm Gen #3)	5	#
HV-2137	25	Feedwater to Steam Generator #4	5	#
HV-2157	26	Feedwater Sample (FW to Stm Gen #4)	5	#
HV-2399	27	Blowdown From Steam Generator #3	5	#
HV-2398	28	Blowdown From Steam Generator #2	5	#

Valve No.	FSAR Table Reference No. (1)	Line or Service	Isolation Time (Seconds)	Note
HV-2397	29	Blowdown From Steam Generator #1	5	#
HV-2400	30	Blowdown From Steam Generator #4	5	#
1-8152 1-8160	32 32	Letdown Line to Letdown Heat Exchanger	10	-
1-8809A	35	RHR to Cold Leg Loops #1 & #2	15	-
1-8890A	35	RHR to Cold Leg Loops #1 & #2	15	-
1-8809B	36	RHR to Cold Leg Loops #3 & #4	15	-
1-8890B	36	RHR to Cold Leg Loops #3 & #4	15	-
1-8047	41	Reactor Makeup Water to Pressure Relief Tank & RC Pump Stand Pipe	10	-
1-8802A	43	SI to RC System Hot Leg Loops #2 & #3	10	-
1-8881	43	SI to RC System Hot Leg Loops #2 & #3	10	-
1-8802B	44	SI to RC System Hot Leg Loops #1 & #4	10	-

Valve No.	FSAR Table Reference No.[1]	Line or Service	Isolation Time (Seconds)	Note
1-8824	44	SI to RC System Hot Leg Loops #1 & #4	10	-
1-8835	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	10	-
1-8823	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	10	-
1-8100	51	Seal Water Return and Excess Letdown	10	-
1-8112	51	Seal Water Return and Excess Letdown	10	-
1-7136	52	RCDT Heat Exchanger to Waste Hold Up Tank	10	-
LCV-1003 1-~~1003~~	52	RCDT Heat Exchanger to Waste Hold Up Tank	10	-
HV-5365	60	Demineralized Water Supply	5	-
HV-5366	60	Demineralized Water Supply	5	-
HV-5157	61	Containment Sump Pump Discharge	5	-
HV-5158	61	Containment Sump Pump Discharge	5	-

Valve No.	FSAR Table Reference No.(1)	Line or Service	Isolation Time (Seconds)	Note
HV-3487	62	Instrument Air to Containment	5	-
1-8840	63	RHR to Hot Leg Loops #2 & #3	15	-
1-8825	63	RHR to Hot Leg Loops #2 & #3	15	-
HV-2405	73	Sample From Steam Generator #1	5	-
HV-4170	74	RC Sample From Hot Legs	5	-
HV-4168	74	RC Sample From Hot Legs	5	-
HV-4169	74	RC Sample From Hot Legs	5	-
HV-2406	76	Sample From Steam Generator #2	5	-
HV-4167	77	Pressurizer Liquid Space Sample	5	-
HV-4166	77	Pressurizer Liquid Space Sample	5	-

Valve No.	FSAR Table Reference No.(1)	Line or Service	Isolation Time (Seconds)	Note
HV-4176	78	Pressurizer Steam Space Sample	5	-
HV-4165	78	Pressurizer Steam Space Sample	5	-
HV-2407	79	Sample From Steam Generator #3	5	-
HV-4175	80	Sample From Accumulators	5	-
HV-4171	80	Sample From Accumulators	5	-
HV-4172	80	Sample From Accumulators	5	-
HV-4173	80	Sample From Accumulators	5	-
HV-4174	80	Sample From Accumulators	5	-
HV-5544	94	Radiation Monitoring Sample	5	-
HV-5545	94	Radiation Monitoring Sample	5	-

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
HV-5546	102	Radiation Monitoring Sample Return	5	-
HV-5547	102	Radiation Monitoring Sample Return	5	-
1-8027	103	N2 Supply to Press. Relief Tank	10	-
1-8026	103	N2 Supply to Press. Relief Tank	10	-
1-8880	104	N2 Supply to Accumulators	10	-
1-7126	105	H2 Supply to RC Drain Tank	10	-
1-7150	105	H2 Supply to RC Drain Tank	10	-
HV-4710	111	CC Supply to Excess Letdown & RC Drain Tk Heat Exchanger	5	-
HV-4711	112	CC Return From Excess Letdown & RC Drain Tk Heat Exchanger	5	-
HV-3486	113	Service Air to Containment	5	-

Valve No.	FSAR Table Reference No. (1)	Line of Service	Isolation Time (Seconds)	Note
HV-4725	114	Containment CCW Drain Tank Pumps Discharge	5	-
HV-4726	114	Containment CCW Drain Tank Pumps Discharge	5	-
B. PHASE "B" ISOLATION VALVES				
HV=4777	54	Containment Spray to Spray Header (TR.B)	20	-
HV-4776	55	Containment Spray to Spray Header (TR.A)	20	-
HV-4708	117	CC Return From RCP's Motors	10	-
HV-4701	117	CC Return From RCP's Motors	10	-
HV-4700	118	CC Supply to RCP's Motors	10	-
HV-4709	119	CC Return From RCP's Thermal Barrier	10	-
HV-4696	119	CC Return From RCP's Thermal Barrier	10	-

C. CONTAINMENT VENTILATION ISOLATION VALVES

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
1-8871	83	Accumulator Test & Fill	10	-
1-8888	83	Accumulator Test & Fill	10	-
1-8964	83	Accumulator Test & Fill	10	-
HV-5556	84	Hydrogen Monitoring Sample	5	-
HV-5557	84	Hydrogen Monitoring Sample	5	-
HV-5550	85	Hydrogen Monitoring Sample	5	-
HV-5551	85	Hydrogen Monitoring Sample	5	-
HV-5552	88	Hydrogen Monitoring Sample	5	-
HV-5553	88	Hydrogen Monitoring Sample	5	-
HV-5554	91	Hydrogen Monitoring Sample	5	-
HV-5555	91	Hydrogen Monitoring Sample	5	-

Valve No.	FSAR Table Reference No.[1]	Line of Service	Isolation Time (Seconds)	Note
HV-5558	97	Hydrogen Monitoring Sample Return	5	-
HV-5559	97	Hydrogen Monitoring Sample Return	5	-
HV-5560	100	Hydrogen Monitoring Sample Return	5	-
HV-5561	100	Hydrogen Monitoring Sample Return	5	-
HV-5536	109	Containment Purge Air Supply	5	##
HV-5537	109	Containment Purge Air Supply	5	##
HV-5538	110	Containment Purge Air Exhaust	5	##
HV-5539	110	Containment Purge Air Exhaust	5	##
HV-6084	120	Chilled Water Supply to Containment Coolers	10	-
HV-6082	121	Chilled Water Return From Containment Coolers	10	-
HV-6083	121	Chilled Water Return From Containment Coolers	10	-

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
HV-5548	122	Containment Pressure Relief	5	-
HV-5549	122	Containment Pressure Relief	5	-
D. MANUAL VALVES				
1FW-158	20b	Chemical Feed to Steam Generator #1	N/A	#
1FW-106	20c	N2 Supply to Steam Generator #1	N/A	#
1FW-157	22b	Chemical Feed to Steam Generator #2	N/A	#
1FW-104	22c	N2 Supply to Steam Generator #2	N/A	#
1FW-156	24b	Chemical Feed to Steam Generator #3	N/A	#
1FW-102	24c	N2 Supply to Steam Generator #3	N/A	#
1FW-159	26b	Chemical Feed to Steam Generator #4	N/A	#
1FW-108	26c	N2 Supply to Steam Generator #4	N/A	-
1-8708B	33	RHR From Hot Leg Loop #4 (Relief)	N/A	

Valve No.	FSAR Table Reference No.[1]	Line of Service	Isolation Time (Seconds)	Note
1-8708A	34	RHR From Hot Leg Loop #1 (Relief)	N/A	-
1-7135	52	RCDT Heat Exchanger to Waste Holdup Tk	N/A	-
1SF-011	56	Refueling Water Purification to Refueling Cavity	N/A	-
1SF-012	56	Refueling Water Purification to Refueling Cavity	N/A	-
1SF-021	67	To Refueling Water Purification Pump	N/A	-
1SF-022	67	To Refueling Water Purification Pump	N/A	-
1SF-~~017~~ 053	71	Refueling Cavity Skimmer Pump	N/A	-
1SF-~~018~~ 054	71	Refueling Cavity Skimmper Pump	N/A	-
1CC-611	111	CC Supply to Excess Letdown & RC Drain Tk Heat Exchanger (Relief)	N/A	-
1CC-618	111	CC Supply to Excess Letdown & RC Drain Tk Heat Exchanger (Relief)	N/A	-

E. POWER OPERATED ~~RELIEF~~ Isolation VALVES

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
HV-2452-1	4	Main Steam to Aux. FPT From Steam Line #1	5	#
PV-2325	5	Main Steam From Steam Generator #1	N/A	#
PV-2326	9	Main Steam From Steam Generator #2	N/A	#
PV-2327	13	Main Steam From Steam Generator #3	N/A	#
HV-2452-2	17	Main Steam to Aux. FPT From Steam Line #4	5	#
PV-2328	18	Main Steam From Steam Generator #4	N/A	#
1HV-2491A	20a	Main & Auxiliary Feedwater to Steam Generator #1	10	#
1HV-2491B	20a	Main & Auxiliary Feedwater to Steam Generator #1	10	#

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
1HV-2492A	22a	Main & Auxiliary Feedwater to Steam Generator #2	10	#
1HV-2492B	22a	Main & Auxiliary Feedwater to Steam Generator #2	10	#
1HV-2493A	24a	Main & Auxiliary Feedwater to Steam Generator #3	10	#
1HV-2493B	24a	Main & Auxiliary Feedwater to Steam Generator #3	10	#
1HV-2494A	26a	Main & Auxiliary Feedwater to Steam Generator #4	10	#
1HV-2494B	26a	Main & Auxiliary Feedwater to Steam Generator #4	10	#
1-8701B	33	RHR From Hot Leg Loop #4	120	-
1-8701A	34	RHR From Hot Leg Loop #1	120	-

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
1-8801A	42	Boron Injection to Cold Leg Loops #1, #2, #3 and #4	10	-
1-8801B	42	Boron Injection to Cold Leg Loops #1, #2, #3 and #4	10	-
1-8105	46	Charging Line to Regenerative Heat Exchanger	10	-
1-8351A	47	Seal Injection to RC Pump (Loop #1)	10	-
1-8351B	48	Seal Injection to RC Pump (Loop #2)	10	-
1-8351C	49	Seal Injection to RC Pump (Loop #3)	10	-
1-8351D	50	Seal Injection to RC Pump (Loop #4)	10	-
HV-5542	58	Hydrogen Purge Supply	10	-
HV-5543	58	Hydrogen Purge Supply	10	-
HV-5563	58	Hydrogen Purge Supply	10	-

Valve No.	FSAR Table Reference No.[1]	Line of Service	Isolation Time (Seconds)	Note
HV-5540	59	Hydrogen Purge Exhaust	10	-
HV-5541	59	Hydrogen Purge Exhaust	10	-
HV-5562	59	Hydrogen Purge Exhaust	10	-
HV-2408	82	Sample From Steam Generator #4	5	-
1-8811A	125	Containment Recirc. Sump to RHR Pump (Train A)	20	-
1-8811B	126	Containment Recirc. Sump to RHR Pump (Train B)	20	-
HV-4782	127	Containment Recirc. Sump to Spray Pumps (Train A)	20	-
HV-4783	128	Containment Recirc. Sump to Spray Pumps (Train B)	20	-

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Insert to Section E of Table 3.6-1

Valve No.	FSAR Table Reference No. (1)	Line or Service	Isolation Time (Seconds)	Note
HV-2185	*	Feedwater Isolation Valve Bypass to Steam Generator #1	5	#
HV-2186	*	Feedwater Isolation Valve Bypass to Steam Generator #4		#
HV-2187	*	Feedwater Isolation Valve Bypass to Steam Generator #3		#
HV-2188	*	Feedwater Isolation Valve Bypass to Steam Generator #2		#
FV-2193	*	Feedwater Bypass to Steam Generator #1		#
FV-2194	*	" " " " " #2		#
FV-2195	*	" " " " " #3		#
FV-2196	*	" " " " " #4		#
1FP-591	*	FIRE PROTECTION	10	—
1FP-592	*	FIRE PROTECTION	10	—
1-8843	42	Boron Injection to cold leg Loops #1, #2, #3 and 4	by W	—
* WILL PROVIDE WHEN ADDED				
1-HV 7311	*	PASS	10	—
1-HV-7312	*	PASS	10	—

F. CHECK VALVES

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
1-8818A	35	RHR to Cold Leg Loops #1 and #2	N/A	-
1-8818B	35	RHR to Cold Leg Loops #1 and #2	N/A	-
1-8818C	36	RHR to Cold Leg Loops #3 and #4	N/A	-
1-8818D	36	RHR to Cold Leg Loops #3 and #4	N/A	-
1-8046	41	Reactor Makeup Water to Press. Relief Tank and RC Pump Stand Pipe	N/A	-
1-8815	42	Boron Injection to Cold Leg Loops #1, #2, #3 and #4	N/A	-
1-8905B	43	SI to RC System Hot Leg Loops #2 & #3	N/A	-
1-8905C	43	SI to RC System Hot Leg Loops #2 & #3	N/A	-

Valve No.	FSAR Table Reference No.[(1)]	Line of Service	Isolation Time (Seconds)	Note
1-~~8805A~~ 8905A	44	SI to RC System Hot Leg Loops #1 & #4	N/A	-
1-~~8805D~~ 8905D	44	SI to RC System Hot Leg Loops #1 & #4	N/A	-
1-8819A	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	N/A	-
1-8819B	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	N/A	-
1-8818C	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	N/A	-
1-8819D	45	SI to RC System Cold Leg Loops #1, #2, #3 & #4	N/A	-
1-8381	46	Charging Line to Regenerative Heat Exchanger	N/A	-
1CS-8368A	47	Seal Injection to RC Pump (Loop #1)	N/A	-
1CS-8368B	48	Seal Injection to RC Pump (Loop #2)	N/A	-

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
CS 1-8368C	49	Seal Injection to RC Pump (Loop #3)	N/A	-
CS 1-8368D	50	Seal Injection to RC Pump (Loop #4)	N/A	-
CS 1-8180	51	Seal Water Return and Excess Letdown	N/A	-
1CT-145	54	Containment Spray to Spray Header (TR.B)	N/A	-
1CT-142	55	Containment Spray to Spray Header (TR.A)	N/A	-
1CI-030	62	Instrument Air to Containment	N/A	-
1-8841A	63	RHR to Hot Leg Loops #2 & #3	N/A	-
1-8841B	63	RHR to Hot Leg Loops #2 & #3	N/A	-
1-8968	104	N2 Supply to Accumulators	N/A	-
1CA-016	113	Service Air to Containment	N/A	-

Valve No.	FSAR Table Reference No.(1)	Line of Service	Isolation Time (Seconds)	Note
1CC-629	117	CC Return From RCP's Motors	N/A	-
1CC-713	118	CC Supply to RCP's Motors	N/A	-
1CC-831	119	CC Return From RCP's Thermal Barrier	N/A	-
1CH-024	120	Chilled Water Supply to Containment Coolers	N/A	-

Notes

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

* May be opened on an intermittent basis under administrative control (normally closed manual valves only)

\# Not subject to Type C leakage tests

\#\# The provisions of Specification 3.0.4 are not applicable.

CONTAINMENT SYSTEMS

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

channels (with at least one monitor per channel)

3.6.5.1 Two independent containment hydrogen ~~monitors~~ shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

channel / channel

With one hydrogen monitor channel inoperable, restore the inoperable channel ~~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 channel shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. Two ~~One~~ volume percent hydrogen, balance nitrogen.

b. Six ~~Four~~ volume percent hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.5.2 Two independent containment 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.

CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM (If less than 2 hydrogen recombiners available)

LIMITING CONDITION FOR OPERATION

3.6.5.3 A containment hydrogen purge cleanup system shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the containment hydrogen purge cleanup system inoperable, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is _______ 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 ______ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of 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 filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of ___ cfm ± 10%.

2. Verifying that the filter cooling bypass valves can be manually opened.

3. Verifying that the heaters dissipate _____ ± _____ kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the 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 _____ 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 _______ cfm ± 10%.

*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.

CONTAINMENT SYSTEMS

HYDROGEN MIXING SYSTEM (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen mixing 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

4.6.5.4 Each hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by starting each system from the control room and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least ________ cfm.

CONTAINMENT SYSTEMS

3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.6 Two independent containment penetration room exhaust air cleanup systems shall be OPERABLE.

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

ACTION:

With one containment penetration room exhaust air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 Each containment penetration room exhaust air cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

  1. Verifying that with the system operating at a flow rate of ____ cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)

  2. 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 ____ cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. 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.

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

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

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of _____ cfm ± 10%.

2. Verifying that the system starts on a Safety Injection Test Signal.

3. Verifying that the filter cooling bypass valves can be manually opened.

4. Verifying that the heaters dissipate _____ ± ______ kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the 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 _____ 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 ________ cfm ± 10%.

* 99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.

3/4 6.7 VACUUM RELIEF VALVES (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.7 The primary containment to atmosphere vacuum relief valves shall be OPERABLE with an actuation set point of less than or equal to ___ psid.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one primary containment to atmosphere vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator of a~~n~~ ~~unisolated~~ reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. ~~With (n) 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 (n-1) 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.

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* In MODE 3 a maximum of 19 safety valves may be made inoperable to permit insitu testing of the operable safety valve as required by Specification 4.7.1.1.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING N 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)~~ 87
2	~~(64)~~ 64
3	~~(42)~~ 43

~~TABLE 3.7-2~~

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING N-1 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~~	~~(52)~~
~~2~~	~~(38)~~
~~3~~	~~(25)~~

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

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE
a. 1MS-021, 058, 093, 129	1185 psig	16 in^2
b. 1MS-022, 059, 094, 130	1195 psig	16 in^2
c. 1MS-023, 060, 095, 131	1205 psig	16 in^2
d. 1MS-024, 061, 096, 132	1215 psig	16 in^2
e. 1MS-025, 062, 097, 133	1235 psig	16 in^2

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

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a total dynamic head ~~discharge pressure~~ of greater than or equal to ___ ~~psig~~ 3274 feet at a flow of greater than or equal to 470 gpm.

2. Verifying that the steam turbine driven pump develops a total dynamic head ~~discharge pressure~~ of greater than or equal to ___ 3438 feet psig at a flow of greater than or equal to 985 gpm when the secondary steam supply pressure is greater than 1185 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

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.

4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

b. ` least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts* as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

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*When the secondary steam supply pressure is greater than 110 psig for the steam turbine-driven pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least 282,540 gallons of 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 (alternate water source) as a backup supply to the auxiliary 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 auxiliary feedwater pumps.

~~4.7.1.3.2 The (alternate water source) shall be demonstrated OPERABLE at least once per 12 hours by (method dependent upon alternate source) whenever the (alternate water source) is the supply source for the auxiliary feedwater pumps.~~

PLANT SYSTEMS

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 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 (after receipt of signal to close) 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 both the primary and secondary coolants in the steam generators shall be greater than (70)°F when the pressure of either coolant in the steam generator is greater than (200) psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to (200) psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F.

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.

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 ________ test signal, as appropriate.

safety injection test signal or Containment isolation phase B

3/4.7.4 STATION SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent station service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one station 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 station 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 ______ test signal.

safety injection

3/4.7.5 ULTIMATE HEAT SINK (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation (770 ft) Mean Sea Level, USGS datum, and

b. An service ~~average~~ water intake temperature of less than or equal to (115)°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIRMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

3/4.7.6 FLOOD PROTECTION (OPTIONAL*)

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Squaw Creek Reservoir (SCR) ~~(usually the ultimate heat sink)~~ exceeds 778 ft Mean Sea Level USGS datum. ~~at~~ ______

APPLICABILITY: At all times.

ACTION:

With the water level ~~at~~ ______ above elevation 778 ft Mean Sea Level USGS datum:

~~a. (Be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours) and~~

~~b.~~ Initiate and complete within 2 hours, the following flood protection measures:
Verify that any equipment which is to be opened for maintenance is isolated from SCR by (1) elevation above 790 ft, (2) Isolation valves or (3) Stop-gates

SURVEILLANCE REQUIREMENTS

4.7.6 The water level ~~at~~ ______ shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is ~~below~~ above elevation 775 ft, Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 777 ft, Mean Sea Level USGS datum.

~~*This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.~~

PLANT SYSTEMS

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room emergency air cleanup systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one control room emergency air cleanup 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 emergency air cleanup 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

4.7.7 Each control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to ~~(120)~~ 80°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 10 hours with the heaters on.

for the recirculation system and 800 cfm ± 10% for the emergency pressurization system.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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:

   ~~1. Verifying that with the system operating at a flow rate of ___ cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)~~

   2. 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 8,000 cfm ± 10%

   3. 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.

   4. Verifying a system flow rate of 8000 cfm ± 10% during recirculation system operation and 800 cfm ± 10% during emergency pressurization 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 filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of 8,000 cfm ± 10% and 800 cfm ± 10%, respectively

   2. Verifying that on a containment phase A isolation 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 the system maintains ~~the~~ control room at a positive pressure of greater than or equal to 1/8 ~~(1/4)~~ inch W.G. relative to the outside atmosphere during system operation.

   4. Verifying that the heaters in the emergency pressurization system dissipate 20 ± 1 kw when tested in accordance with ANSI N510-1975.

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f. After each complete or partial replacement of a recirculation 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 8,000 cfm ± 10%.

g. After each complete ~~or partial~~ replacement of ~~a~~ the recirculation 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 8000 cfm ± 10%.

h. After each complete or partial replacement of an emergency pressurization 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 800 cfm ± 10%.

i. After each complete replacement of the emergency pressurization charcoal adsorber banks 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 800 cfm + 10%.

* ~~99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.~~

PLANT SYSTEMS

3/4.7.8 ~~ECCS PUMP ROOM~~ ESF EXHAUST AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ~~ECCS pump room~~ ESF exhaust air cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ~~ECCS pump room~~ ESF exhaust air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

SURVEILLANCE REQUIREMENTS

4.7.8 Each ECCS pump room exhaust air cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

~~1. Verifying that with the system operating at a flow rate of ____ cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)~~

~~2.~~ 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 15,000 cfm ± 10%. per cleanup train (30,000 cfm ± 10% total)

SURVEILLANCE REQUIREMENTS (Continued)

3. 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.

4. Verifying a system flow rate of 15,000 cfm ± 10% per cleanup train (30,000 cfm ± 10% total) 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 filters and charcoal adsorber banks of less than ~~(6)~~ inches Water Gauge while operating the system at a flow rate of 15,000 cfm ± 10%. per cleanup train.

2. Verifying that the system starts on a Safety Injection Test Signal.

3. Verifying that the filter cooling bypass valves can be manually opened.

4. Verifying that the heaters dissipate 100 ± 5 kw per cleanup train when tested in accordance with ANSI N510-1975.

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 one cleanup train ~~the system~~ at a flow rate of 15,000 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 one cleanup train ~~the system~~ at a flow rate of 15,000 cfm ± 10%.

* ~~99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.~~

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 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:

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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 4.7.9.~~c~~ on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program 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 [redacted] 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.

c. Refueling Outage Inspections

at least once per 18 months ~~During each refueling outage~~ 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 Specification ~~s 4.7.7.d or 4.7.7.e, as applicable.~~ 4.7.9.5. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and ~~cannot~~ 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 ~~4.7.9.d or 4.7.9.e~~ 4.7.9.5, 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~~ 4.7-1, "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification ~~4.7.9.d or 4.7.9.e~~ 4.7.9.5. 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~~ 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type ~~design~~ shall be functionally tested. If at any time the point plotted falls in

e. Functional Tests (Continued)

may

the "Accept" region, testing of that type of snubber ~~shall~~ 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.

shall

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 ~~should~~ 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.

attached to

g. Functional Test Failure Analysis (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components ~~which are supported by the inoperable snubbers.~~ The purpose of this engineering evaluation shall be to determine if the components ~~supported by the inoperable snubbers~~ were adversely affected by the inoperability of the snubbers in order to ensure that the ~~supported~~ 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 ~~design~~ 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. 4.7.9.e

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.

service life is not exceeded

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the ~~seals do not fail~~ between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be ~~estimated~~ based on engineering information and the seals shall be replaced so that the maximum ~~expected~~ service life ~~does not expire~~ 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.

determined and established

will not be exceeded

j. Exemption From Visual Inspection or Functional Tests

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.

move to B3/4 7-5

FIGURE 4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST all new

Table 3.7-4a

Safety-Related Hydraulic Snubbers*

(Manufacturer)

System	Size (Kips) Small		Medium		Large	
	( )	( )	( )	( )	( )	( )
Subtotal-1						
Subtotal-2						
TOTAL						

---

*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.

all new

Table 3.7-4b

Safety-Related Mechanical Snubbers*

(Manufacturer)

System	Small		Size (Kips) Medium		Large	
	( )	( )	( )	( )	( )	( )
Subtotal-1						
Subtotal-2						
TOTAL						

---

*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.

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or ~~5~~ 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1. With a half-life greater than 30 days (excluding Hydrogen 3), and

   2. In any form other than gas.

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.

4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

d. Devices containing only tritium and devices held in storage in the original container prior to initial installation do not require testing.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. ~~[~~two~~]~~ fire suppression pumps, each with a capacity of ~~[~~2500~~]~~ gpm, with their discharge aligned to the fire suppression header,

~~b. Separate water supplies, each with a minimum contained volume of ____ gallons, and~~

b. ~~c.~~ An OPERABLE flow path capable of taking suction from the Safe Shutdown Impoundment (SSI) ~~tank and the ____ tank~~ and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.5 and 3.7.11.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump ~~and/or one water supply~~ inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, ~~in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to 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: ← site

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~~

~~ACTION: (Continued)~~

~~c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. ~~At least once per 7 days by verifying the contained water supply volume.~~

b. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by starting ~~each~~ the electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

d. (At least once per 6 months by performance of a system flush.)

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each pump develops at least ~~(2500)~~ gpm at a system head of ~~(250)~~ 320 feet,

   3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 80 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

Except the containment standpipe system, which will undergo an air flow test to insure the flow path to each hose station is unobstructed during each cold shutdown exceeding 24 hrs. unless performed in the previous six months

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.2 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 400 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-~~65~~, 1975 is within the acceptable limits specified in Table 1 of ASTM D975-~~74~~, 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.

4.7.11.1.3 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

3.7.11.2 The ~~following~~ spray and/or sprinkler systems listed in table 3.7-2 shall be OPERABLE:

~~a. (Plant dependent - to be listed by name and location.)~~

~~b.~~

~~c.~~

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 a hourly fire watch patrol. ~~Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.~~

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 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.

TABLE 3.7-2

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
FUEL	RAILROAD LOADING & UNLOADING AREA EL. 810'-0"	99 a.
ELECTRICAL CONTROL	CABLE SPREADING RM. NORTH HALF EL. 807'-0"	64
	CABLE SPREADING RM. SOUTH HALF EL. 807'-0"	64
	CABLE SPREADING RM. SOUTH HALF EL. 807'-0"	63
	CABLE SPREADING RM. NORTH HALF EL. 807'-0"	63
	DELUGE SYSTEM FOR EMERGENCY FILTRATION UNIT-CPX-VAFUPK-23 EL. 854'-4"	73
	DELUGE SYSTEM FOR EMERGENCY PRESSURIZATION UNIT-CPX-VAFUPK-21 EL. 854'-4"	73

TABLE 3.7-2 (CONTINUED)

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
ELECTRICAL CONTROL	DELUGE SYSTEM FOR EMERGENCY PRESSURIZATION UNIT-CPX-VAFUPK-22 EL. 854'-4"	74
↓	DELUGE SYSTEM FOR EMERGENCY FILTRATION UNIT-CPX-VAFUPK-24 EL. 854'-4"	74
DIESEL GENERATOR	DIESEL DAY TANK ENCLOSURE-TRAIN B EL. 844'-0"	13
	DIESEL GENERATOR RM.-TRAIN B EL. 810'-6"	12
	DIESEL DAY TANK ENCLOSURE-TRAIN A EL. 844'-0"	11
↓	DIESEL GENERATOR RM.-TRAIN A EL. 810'-6"	10

TABLE 3.7-2 (CONTINUED)

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
SAFEGUARD'S	CORRIDOR & PIPE PENETRATION AREA EL. 831'-6"	15 & 144
	ELECTRICAL EQUIP. RM. EL. 831'-6"	16
	ELECTRICAL EQUIP. RM. EL. 852'-6"	18
	CORRIDOR & CHEM. TANK RM. EL. 790'-6"	10, 4, 142, 5, 6
	CORRIDOR EL. 810'-6"	8
	ELECTRICAL EQUIPMENT RM. EL. 810'-6"	9

TABLE 3.7-2 (CONTINUED)

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
AUXILIARY	DELUGE SYSTEM FOR CPX-VAFUPK - 02, -04, -06, -08, -10, -12, -14, -16, -20	39 (EL. 873'-6
	DELUGE SYSTEM FOR CPX-VAFUPK - 01, 03, 05, 07, 09, 11, 13, 15, 19	40 (EL. 886'-6'
SAFEGUARDS	PIPE PENETRATION AREA & ELEC. EQUIP. RM. EL. 852'-6", 880'-6"	17a, 17c

TABLE 3.7-2 (CONTINUED)

** FIRE AREA IN E+C BUILDING

* FIRE AREA IN SAFEGUARDS BUILDING

(5)

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
AUXILIARY	STAIR AREA A-11 FILTER STORAGE EL. 831'-6", EL. 873'-6" & EL. 852'-6"	21d, 19*, 21f
	South Half Corr. EL. 852'-6"	21f
	NORTH HALF CORR. EL. 852'-6"	21f, 73**
	MECH. EQUIP. AREA 873'-6" STAIR AREA A-10 EL. 831'-6", 842'-0"	38, 21f, 21d
	SOUTH HALF CORR. EL. 831'-6"	21d
	NORTH HALF CORR. EL. 831'-6"	21d
	NORTH HALF CORR. EL. 810'-6"	21b
	SOUTH HALF CORR. EL. 810'-6"	21b

TABLE 3.7-2 (CONTINUED)

** FIRE AREA IN E+C BUILDING

BUILDING	AREA DESCRIPTION	FIRE AREAS AFFECTED
AUXILIARY	HEAT EXCHANGE & TUBE REMOVAL AREA EL. 790'-6"	21g
	NORTH AUX. CORR. EL. 790'-6" CHILLER EQ. AREA (UNIT 2) EL. 778'-0". BATT. RM. CORR. EL. 792'-0" & MECH. EQ. AREA EL. 778'-0"	43**, 44**, 21a, 54**, 154**
	SOUTH AUX. CORR. EL. 790'-6" CHILLER EQ. AREA (UNIT 1) EL. 778'-0" BATT. RM. CORR. EL-792'-0" & MECH. EQ. AREA EL. 778'-0".	43**, 47**, 21a, 57**, 153**
SERVICE WATER INTAKE STRUCTURE	DIESEL FIRE PUMP AREA EL. 796'-0"	103
	SWIS (GENERAL AREA) EL. 796'-0", EL. 810'-6"	104b, 104c
CONTAINMENT	CPX-VAFUPK-17	101e
	CPX-VAFUPK-18	101e

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a ________ test signal, and
   manually initiated

   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 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following high pressure and low pressure CO_2 systems shall be OPERABLE.

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 systems 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.

4.7.11.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than ____ and pressure to be greater than ____ 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."

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.3.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the CO_2 storage tank weight to be at least 90% of full charge weight.

b. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The following Halon systems shall be OPERABLE.

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight (or level) and pressure to be at least 90% of full charge pressure.

c. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.5 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, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit 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

4.7.11.5 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 operations to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   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 #

ATTACHED

*List all Fire Hose Stations required to ensure the OPERABILITY of safety-related equipment.

TABLE 3.7-5 (CONTINUED)

BUILDING	HOSE STATION TAG No.	ELEVATION	AREAS AFFECTED
3. DIESEL GEN.	CPI-FPFESH-16	810'-6"	10a, 10b, 11
	CPI-FPFESH-17	844'-0"	12a, 12b, 13
4. SAFEGUARDS	CPI-FPFESH-12	773'-6"	1a, 3
	CPI-FPFESH-11	773'-6"	2a, 4
	CPI-FPFESH-02	790'-6"	1c, 1d, 3, 4
	CPI-FPFESH-13	790'-6"	1c, 4
	CPI-FPFESH-01	790'-6"	2c, 2d, 4
	CPI-FPFESH-14	790'-6"	2c, 4, 5, 6, 7
	CPI-FPFESH-05	810'-6"	3, 8
	CPI-FPFESH-15	810'-6"	2e, 2f, 8, 142, 143
	~~CPI-FPFESH-03~~	~~810'-6"~~	~~9~~
	CPI-FPFESH-04	810'-6"	9, 157
	CPI-FPFESH-07	831'-6"	3, 15, 16
	CPI-FPFESH-08	831'-6"	2g, 3, 14, 15, 16, 144, 15
	CPI-FPFESH-06	831'-6"	16, 157
	CPI-FPFESH-18	852'-6"	17a, 17c
	CPI-FPFESH-09	852'-6"	17a, 18, 158
	CPI-FPFESH-10	852'-6"	18, 157
	CPI-FPFESH-19	873'-6"	3, 19, 20
5. SERVICE WATER INTAKE STRUC.	CPX-FPFEIH-01	796'-0"	104C, 104B, 103

TABLE 3.7-5

BUILDING	HOSE STATION TAG NO	ELEVATION	AREAS AFFECTED
1. FUEL	CPX-FPFEFH-03	810'-6"	96, 97, 99a, 99b
	CPX-FPFEFH-02	810'-6"	99a, 99c
	~~CPX-FPFEFH-0~~	~~810'-6"~~	~~98, 99c, 99b~~
	~~CPX-FPFEFH-0~~	~~841'-0"~~	~~98, 99a, 100~~
	~~CPX-FPFEFH-05~~	~~841'-0"~~	~~98, 99a~~
	~~CPX-FPFEFH-07~~	~~860'-0"~~	~~99c~~
	~~CPX-FPFEFH-0~~	~~860'-0"~~	~~99c~~
2. ELECTRIC CONTROL	~~CP1-FPFEEH-0~~	~~778'-0"~~	~~43, 47, 150~~
	~~CP1-FPFEEH-0~~	~~778'-0"~~	~~43, 44, 45, 149~~
	CP2-FPFEEH-0	792'-0"	45, 48, 50, 52, 54, 55 58, 59
	CP1-FPFEEH-02	792'-0"	49, 51, 53, 56, 57, 60, 61, 62
	CP1-FPFEEH-03	807'-0"	62, 64, 75
	~~CP2-FPFEEH-03~~	~~807'-0"~~	~~63~~
	~~CP1-FPFEEH-04~~	~~807'-0"~~	~~64~~
	CP2-FPFEEH-02	807'-0"	63, 45, 161
	CPX-FPFEEH-03	831'-6"	45, 62, 65, 66, 67, 68, 69, 162, 163
	CPX-FPFEEH-01	831'-6"	65
	CPX-FPFEEH-02	831'-6"	65
	~~CPX-FPFEEH-0~~	~~840'-6"~~	~~140, 141, 65, 70, 71, 72, 45~~
	CP2-FPFEEH-04	854'-4"	45, 73, 75, 161
	CP1-FPFEEH-07	854'-4"	45, ~~74~~, 75

TABLE 3.7-5 (CONTINUED)

* THESE FIRE AREAS ARE LOCATED IN E+C BUILDING

BUILDING	HOSE STATION TAG No.	ELEVATION	AREAS AFFECTED
6. AUXILIARY	CPX-FPFEXH-01	790'-6"	21a, 154*
	~~CPX-FPFEXH-03~~	~~790'-6"~~	~~21a~~
	~~CPX-FPFEXH-18~~	~~790'-6"~~	~~21a~~
	CPX-FPFEXH-02	790'-6"	21a, 153*
	~~CPX-FPFEXH-19~~	~~790'-6"~~	~~21a~~
	CPX-FPFEXH-05	810'-6"	21b, 21c, 27, 28, 29, 30, 33
	~~CPX-FPFEXH-04~~	~~810'-6"~~	~~21b, 27, 31a, 33~~
	CPX-FPFEXH-06	810'-6"	21b, 21c, 23, 24, 25, 26, 32
	~~CPX-FPFEXH-07~~	~~810'-6"~~	~~21b, 22~~
	CPX-FPFEXH-21	831'-6"	21d
	CPX-FPFEXH-09	831'-6"	21d, 34, 36
	CPX-FPFEXH-08	831'-6"	21b, 21h, 31b, 35, 37
	CPX-FPFEXH-20	831'-6"	21d
	CPX-FPFEXH-22	842'-0"	21d, 21e
	CPX-FPFEXH-23	842'-0"	21d, 21e
	CPX-FPFEXH-11	852'-6"	21F
	CPX-FPFEXH-10	852'-6"	21F
	~~CPX-FPFEXH-24~~	~~852'-6"~~	~~21F~~
	~~CPX-FPFEXH-15~~	~~873'-6"~~	~~38~~
	~~CPX-FPFEXH-12~~	~~873'-6"~~	~~39~~
	~~CPX-FPFEXH-14~~	~~873'-6"~~	~~38~~
	CPX-FPFEXH-13	873'-6"	39
	CPX-FPFEXH-17	886'-6"	40
	~~CPX-FPFEXH-16~~	~~886'-6"~~	~~40~~

TABLE 3.7-5 (CONTINUED)

BUILDING	HOSE STATION TAG No.	ELEVATION	AREAS AFFECTED
7. CONTAINMENT - UNIT 1	CP1-FPFECH-01	808'-0"	101 b
	CP1-FPFECH-02	808'-0"	101 b
	CP1-FPFECH-03	808'-0"	101 b
	CP1-FPFECH-04	832'-6"	101 d
	CP1-FPFECH-05	832'-6"	101 d
	CP1-FPFECH-06	832'-6"	101 d
	CP1-FPFECH-07	852'-6"	101 f
	CP1-FPFECH-08	852'-6"	101 f
	CP1-FPFECH-09	852'-6"	101 f
	CP1-FPFECH-10	905'-0"	101 h
	CP1-FPFECH-11	905'-0"	101 h

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.6 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.6 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.

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TABLE 3.7-6

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT TAG NUMBER
adjecent to Service water intake structure	CPX-FPFEHY-01

*List all Yard Fire Hydrants and Hydrant Hose Houses required to ensure the OPERABILITY of safety-related equipment.

PLANT SYSTEMS

3/4.7.12 FIRE RATED ASSEMBLES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire rated 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 and ventilation duct 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 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

4.7.12.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by: Performing a visual inspection of

a. Performing a visual inspection of the exposed surfaces of each fire rated assemblies.

b. ~~Performing a visual inspection of~~ each fire window/fire damper/ and associated hardware.

c. ~~Performng 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. Samples shall be selected such that each penetration seal will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 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:

d ~~a~~. That ~~The position of~~ each unlocked ~~closed~~ fire door without electrical supervision is closed at least once per 24 hours.

c ~~b~~. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test at least once per 18 months.

b ~~c~~. That ~~The position of~~ each locked closed fire door is closed at least once per 7 days.

a ~~d~~. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days.

3/4.7.13 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.13 The temperature of each area shown in Table 3.7-7 shall be maintained within 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 within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.13 The temperature in each of the areas shown in Table 3.7-7 shall be determined to be within its limit at least once per 24 hours.

TABLE 3.7-7

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. Control Room	80
2. Fuel Handling Building (Normal access areas)	104
3. Safeguards Building (Normal access areas)	104
4. auxilary Building (normal access areas)	104
5. Electrical + Control Building (Normal access areas)	104
6. Diesel Generator Building	122
7. Service water Intake Structure	122
8. Turbine Building - Switchgrear area	115
9. Containment (outside Missile Shield)	120
CRDM Shroud	163
Detector Well	135
Inside Missile Shield	140
10. RHR Pump Rooms	122
11. SIS Pump Rooms	122
12. CCWS Pump Rooms	122
13. Centrifugal Charging Pump Rooms	122
14. UPS / Battery Room Areas	104
15. Spent Feul Pool cooling + cleanup Pump and Heat Exchanger Rooms	122
16. AFW Pump Rooms	122
17. CSS Pump Rooms	122

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. Separate day and engine-mounted fuel tanks containing a minimum volume of 1440 gallons of fuel,

2. A separate fuel storage system containing a minimum volume of 88175 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 1 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 1 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.

c. With one diesel generator inoperable in addition to 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

ACTION: (Continued)

(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 1 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 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) ~~unit~~ the 6.9KV safeguards bus power supply from the Preferred offsite ~~normal~~ circuit to the alternate offsite 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 and engine-mounted fuel tank,

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day and engine-mounted tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least (~~900~~) 450 rpm in less than or equal to (10) seconds. The generator voltage and frequency shall be (~~4160~~) 6900 + (~~420~~) 690 volts and (60) + (1.2) Hz within (10) seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.

   b) Simulated loss of preferred offsite power by itself. Start-up transformer secondary winding undervoltage.

   c) Simulated ~~loss of offsite power in conjunction with an ESF actuation test signal.~~ Safeguards bus undervoltage.

   d) ~~An ESF~~ Safety Injection actuation test signal by itself.

   e) Safety Injection actuation test signal in conjunction with loss of preferred offsite power

5. Verifying the generator is synchronized, loaded to greater than or equal to (~~continuous rating~~) 7000 KW in less than or equal to (60) seconds, and operates with a load greater than or equal to (continuous rating) for at least 60 minutes,

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

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day ~~and engine-mounted fuel~~ tank~~s~~.

c. At least once per 92 days and from new fuel oil prior to addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @ 40°C of greater than or equal to 1.9 but less than or equal to 4.1 when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

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 791 load of greater than or equal to (~~largest single emergency load~~) kw while maintaining voltage at 6900 ± 690, volts and frequency at (60) ± (1.2) Hz (less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal whichever is less).

3. Verifying the generator capability to reject a load of (7000 ~~continuous rating~~) kw without tripping. The generator voltage shall not exceed (~~4784~~ 7590) 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 (~~4160~~ 6900) ± (~~420~~ 690) volts and (60) ± (1.2) Hz during this test.

5. Verifying that on an ~~ESF~~ SI 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 (~~4160~~ 6900) ± (~~420~~ 690) volts and (60) ± (1.2) Hz within (10) seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of the preferred and alternate offsite power sources in conjunction with an ~~ESF~~ SI actuation test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within (10) seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at ~~(4160)~~ 6900 $\pm$ 690 ~~(420)~~ volts and (60) $\pm$ (1.2) Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon ~~loss of voltage on the emergency bus concurrent with~~ a safety injection actuation signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 7700 ~~(2 hour rating)~~ kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 7000 ~~(continuous rating)~~ kw. The generator voltage and frequency shall be 6900 ~~(4160)~~ $\pm$ 690 ~~(420)~~ volts and (60) $\pm$ (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.7.b.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the ~~2000-hour~~ continuous rating of 7000 kw.

10. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

12. Verifying that ~~the~~ each fuel transfer pump transfers fuel from ~~each~~ the fuel storage tank to the day ~~and engine-mounted~~ tank of ~~each~~ its associated diesel via the installed ~~cross-connection~~ lines.

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

14. Verifying that only the following diesel generator lockout features prevent diesel generator starting ~~only when required:~~ by an S.I signal

~~a)~~ (~~turning gear engaged~~) a) Barring device engaged (PS-13B closed)

~~b)~~ ~~(emergency stop)~~ b) Maintenance Lockout made

c) Local/Remote Emergency stop

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least (~~900~~) 450 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 hypoclorite solution, 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
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 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.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day ~~and engine-mounted fuel~~ tank~~s~~ containing a minimum volume of 1440 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 88175 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to (~~1~~) LATER square inch vent. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the 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.

ELECTRICAL POWER SYSTEMS

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. Train A ~~(250/~~125)-volt Battery banks BT1ED1 and BT1ED3 and at least one ~~No. 1, and its associated~~ full capacity chargery associated with each battery.

b. Train B ~~(250/~~125)-volt Battery banks BT1ED2 and BT1ED4 and at least one ~~No. 2, and its associated~~ full capacity chargery associated with each battery.

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 both ~~one~~ of the ~~required~~ full capacity chargers of one battery bank inoperable, demonstrate the OPERABILITY of the ~~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.

~~This specification is intended for use on plants with two divisions of D.C. power only. Modifications may be necessary, on a plant-unique basis, to accommodate different designs.~~

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each ~~(250/~~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 ~~(258/129)~~ 128-volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below (~~220~~/110)-volts, or battery overcharge with battery terminal voltage above (~~300~~/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
LATER (~~150 x 10^-6~~) ohms, and

3. The average electrolyte temperature of (~~a representative number~~ 12) of connected cells is above (~~60~~ 70 F).

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

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

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to (~~150 x 10^-6~~) ohms, and
LATER

4. The battery charger will supply at least (~~400~~ 300) amperes at (125~~/250~~)-volts for at least (~~8~~ 12) hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

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 $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	$\geq$ 1.200[b]	$\geq$ 1.195 Average of all connected cells > 1.205	Not more than .020 below the average of all connected cells Average of all connected cells $\geq$ 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.

~~Numbers in paretheses assume a manufacturer's recommended full charge specific gravity of 1.215.~~

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

of one train

3.8.2.2 As a minimum, ~~one~~ (~~250/~~125)-volt battery banks and ~~its~~ one associated full capacity chargers shall be OPERABLE.

for each battery bank

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; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a ( ) square inch vent. LATER

b. With either 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 associated batterys inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required (~~250/~~125)-volt battery bank and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

ELECTRICAL POWER SYSTEMS

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 no tie breakers ~~open (both)~~ between redundant busses within the unit (and between units at the same station):

a. ~~Division #1~~ Train A A.C. Emergency Busses consisting of:
   1. ~~(4160)~~ 6900 volt Emergency Bus # 1EA1
   2. ~~(480)~~ volt Emergency Bus # 1EB1 from transformer T1EB1
   3. 480 volt Emergency Bus # 1EB3 from transformer T1EB3

b. ~~Division #2~~ Train B A.C. Emergency Busses consisting of:
   1. ~~(4160)~~ 6900 volt Emergency Bus # 1EA2
   2. ~~(480)~~ volt Emergency Bus # 1EB2 from transformer T1EB2
   3. 480 volt Emergency Bus # 1EB4 from transformer T1EB4

c. ~~(120)~~ 118 volt A.C. ~~Vital~~ Instrument Busses # 1PC1, 1PC3 and 1EC1 energized from ~~its~~ their associated inverters connected to D.C. Bus # 1ED1*.

d. ~~(120)~~ 118 volt A.C. ~~Vital~~ Instrument Busses # 1PC2, 1PC4 and 1EC2 energized from ~~its~~ their associated inverters connected to D.C. Bus # 1ED2*.

e. ~~(120)~~ 118 volt A.C. ~~Vital~~ Instrument Bus # 1EC5 energized from ~~its~~ associated inverter connected to D.C. Bus # 1ED3.

f. ~~(120)~~ 118 volt A.C. ~~Vital~~ Instrument Bus # 1EC6 energized from ~~its~~ associated inverter connected to D.C. Bus # 1ED4.

g. Train A 125 volt D.C. Busses 1ED1 and 1ED3 energized from Battery Banks BT1ED1 and BT1ED3 respectively.

h. Train B 125 volt D.C. Busses 1ED2 and 1ED4 energized from Battery Banks BT1ED2 and BT1ED4 respectively.

*Two inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

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 and inverters.
ac switchgrear and dc switchboard

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

ACTION:

a. With one of the required ~~divisions~~ Trains of A.C. Emergency busses not fully energized, re-energize the ~~division~~ Train 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~~ Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. ~~Vital~~ Instrument Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) re-energize the A.C. ~~Vital~~ Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank, 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.

ONSITE POWER DISTRIBUTION

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 ~~division~~ Train of A.C. Emergency Busses consisting of one ~~(4160)~~ 6900 volt and ~~one~~ (two) {480} volt A.C. Emergency Busses.

b. Two ~~{120}~~ 118 volt A.C. ~~Vital~~ Instrument Busses (channel-oriented), ~~energized from their~~ associated inverters connected to their respective D.C. Busses.

d. One Train of dc busses consisting of ~~One~~ two ~~{250/125}~~ volt D.C. Busses energized from ~~its~~ their associated battery banks. Busses shall be of the same train as items a. and c.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours depressurize and vent the RCS through a ( ) square inch vent.

LATER

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 ac switchgear and dc switchboard busses and inverters.

c. One Train of A.C. Instrument Busses consisting of two 118 volt A.C. Instrument Busses energized from their associated inverters connected to their respective D.C. Busses. Busses shall be of the same train as items a. and d.

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 containment penetration conductor overcurrent protective device(s) shown in Table 3.8-1 inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker within 72 hours, declare the affected System or component inoperable, and verify the backup circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 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 (~~4-15~~ 6.9 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least ~~10% of the~~ one circuit breakers ~~of each voltage level~~ of each Current rating, 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.

(c) For each circuit one breaker found inoperable during these functional tests, ~~an~~ additional ~~representative sample of at least 10% of all~~ the circuit breaker~~s~~ 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. For the lower voltage circuit breakers the nominal trip setpoint and short circuit response times are listed in Table 3.8-1. 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 SIZE ~~type~~ of fuse rated more than 8 amps on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that SIZE ~~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~~ SIZE shall be functionally tested until no more failures are found or all fuses of that ~~type~~ SIZE have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Device Number and Location	Trip Setpoint (Amperes)	Response Time (sec/cycles)	System Powered
1. 6900 VAC (Primary breaker) (Back-up breaker)			Reactor Coolant pump 1 2 3 4
2. 480 VAC from MOAD Centers List all; primary breakers Back-up breakers " "			
3. 480 VAC from MCC List all; primary breakers Back-up breakers " "			
4. 125V DC Lighting List all; primary breakers Back-up breakers " "			
5. 440 VAC CRDM Power Primary breakers Back-up breakers " "			

See following PAGES

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number and Location	Setpoint	Test Point (Amperes)	Response Time (Seconds)	System Powered
1. 6.9 KVAC from Switchgears				
A. Switchgear Bus 1A1				RCP #11
1. Primary Breaker 1PCPX1				
a. Relay 50 M1-51	Current - 720A Time Dial - #6 Instantaneous - 4960A	Long Time - 3600A Instantaneous - 5456A	<25 sec. Instantaneous	
b. Relay 26	Current - 1280A Time Dial - #10	Long Time - 6400A	<2.5 sec.	
c. Relay 86M	N/A	N/A	Instantaneous	
2. Backup Breakers 1A1-1 or 1A1-2				
a. Relay 51M2	Current - 800A Time Dial - #7	Long Time - 4000A	<30 sec.	
b. Relay 51 for 1A1-1	Current - 3200A Time Dial - #10	Long Time - 16000A	<2.5 sec.	
c. Relay 51 for 1A1-2	Current - 3200A Time Dial - #10	Long Time - 16000A	<2.5 sec.	
d. Relay 86/1A1	N/A	N/A	Instantaneous	
B. Switchgear Bus 1A2				RCP #12
1. Primary Breaker 1PCPX2				
a. Relay 50M1-51	Current - 720A Time Dial - #6 Instantaneous - 4960A	Long Time - 3600A Instantaneous - 5456A	<25 sec Instantaneous	

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

Device Number and Location	Setpoint	Test Point (Amperes)	Response Time (Seconds)	System Powered
b. Relay 26	Current - 1280A Time Dial - #10	Long Time - 6400A	<2.5 sec.	
c. Relay 86M	N/A	N/A	Instantaneous	
2. Backup Breakers 1A2-1 or 1A2-2				
a. Relay 51M2	Current - 800A Time Dial - #7	Long Time - 4000A	<30 sec.	
b. Relay 51 for 1A2-1	Current - 3200A Time Dial - #10	Long Time - 16000	<2.5 sec.	
c. Relay 51 for 1A2-2	Current - 5200 Time Dial - #10	Long Time - 16000	<2.5 sec.	
d. Relay 86/1A2	N/A	N/A	Instantaneous	
C Switchgear Bus 1A3				
1. Primary Breaker 1PCPX3				RCP #13
a. Relay 50M1-51	Current - 720A Time Dial - #6 Instantaneous - 4960A	Long Time - 3600A Instantaneous - 5456A	<~~23~~ 25 sec. Instantaneous	
b. Relay 26	Current - 1280A Time Dial - #10	Long Time - 6400A	<2.5 sec.	
c. Relay 86M	N/A	N/A	Instantaneous	
2. Backup Breakers 1A3-1 or 1A3-2				
a. Relay 51M2	Current - 800A Time Dial - #7	Long Time - 4000A	<30 sec.	

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Device Number and Location	Setpoint	Test Point (Amperes)	Response Time (Seconds)	System Powered
b. Relay 51 for 1A3-1	Current - 3000A Time Dial - #10	Long Time - 15000A	<2.5 sec.	
c. Relay 51 for 1A3-2	Current - 3000A Time Dial - #10	Long Time - 15000A	<2.5 sec.	
d. Relay 86/1A3	N/A	N/A	Instantaneous	
D. Switchgear Bus 1A4				
1. Primary Breaker 1PCPX4				RCP #14
a. Relay 50M1-51	Current - 720A Time Dial - #6 Instantaneous - 4960A	Long Time - 3600A Instantaneous - 5456A	<25 sec. Instantaneous	
b. Relay 26	Current - 1280A Time Dial - #10	Long Time - 6400	<2.5 sec.	
c. Relay 86M	N/A	N/A	Instantaneous	
2. Backup Breakers 1A4-1 or 1A4-2				
a. Relay 51M2	Current - 800A Time Dial #7	Long Time - 4000A	<30 sec.	
b. Relay 51 for 1A4-1	Current - 3000A Time Dial - #10	Long Time - ~~1500~~ 15000A	<2.5 sec.	
c. Relay 51 for 1A4-2	Current - 3000A Time Dial - #10	Long Time - ~~1500~~ 15000A	<2.5 sec.	
d. Relay 86/1A4	N/A	N/A	Instantaneous	

TABLE 3.8-1

CONTAINMENT PENTRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number
and Location

2. 480 VAC from Switchgears

2.1 System Powered - Containment Recirc. Fans and CRDM Vent Fans

Device Location - 480V Switchgears 1EB1, 1EB2, 1EB3 and 1EB4

A. Primary Breakers - 1FNAV1, 1FNAV2, 1FNAV3, 1FNAV4, 1FNCB1 and 1FNCB3

Device Number		Setpoints	Test Point (Amperes)	Response Time (Seconds)
Westinghouse Amptector II (for each primary breaker)	Long time pick-up	180A (0.9x)	Long time -1000A	< 20
	Long time delay	15 ~~22~~ secs	Instant. - 3600A	<.06
	Inst. pick-up	1800A (9x)		

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

B. Backup Breakers - 1EB1-1, 1EB2-1, 1EB3-1 and 1EB4-1

Device Number*	Setpoint	Test Point (Amperes)	Response Time (Seconds)
1. Long time & Instantaneous Relays			
$\frac{50/51}{1FNAV1}$ (1EB1-1) $\frac{50/51}{1FNAV2}$ (1EB2-1)			
$\frac{50/51}{1FNAV3}$ (1EB3-1) $\frac{50/51}{1FNAV4}$ (1EB4-1)	Current - 300A Time Dial #5 Instant. 3000A	Long Time 1500A Instant. 3300A	<23 sec. Instantaneous
$\frac{50/51}{1FNCB1}$ (1EB3-1) $\frac{50/51}{1FNCB2}$ (1EB4-1)			

*Associated circuit breaker shown in parentheses; e.g. 1EB3-1 is back-up to 1FNAV3 and 1FNCB1

2. Time Delay Relays	Setpoint	Test Point (Amperes)	Response Time (Seconds)
$\frac{62}{1FNAV1}$ (1EB1-1) $\frac{62}{1FNAV2}$ (1EB2-1)			
$\frac{62}{1FNAV3}$ (1EB3-1) $\frac{62}{1FNAV4}$ (1EB4-1)	Time Delay - 0.3 sec.	N/A	<0.33 sec.
$\frac{62}{1FNCB1}$ (1EB3-1) $\frac{62}{1FNCB2}$ (1EB4-1)			

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2.2 System Powered - Containment Polar Crane
Device Location - 480V Switchgear 1EB4

A. Primary Breaker - 1SCCP1

Device Number		Setpoint	Test Point (Amperes)	Response Time (Seconds)
Westinghouse Amptector II	Long Time Pick-up	225A (0.75x)	Long Time - 900A	< 18 sec.
	Long Time Delay	8 ~~[illegible]~~ sec.	Instantaneous - 2400A	<.06 sec.
	Inst. Pick-up	1200A (4x)		

B. Backup Breaker 1EB4-1

Device Number		Setpoint	Test Point (Amperes)	Response Time (Seconds)
1. 51 / 1SCCP1	Long Time Pick-up	480A,	2400A	<1.6
	Time Dial	#1		
2. ~~[illegible]~~ 62 / 1SCCP1	Time Delay Contact	.2 sec.	N/A	<0.22 sec.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

3. 480VAC from Motor Control Centers

3.1 Device Location - MCC 1EB1-2, Compt. Numbers listed below.

Primary and Backup Breakers - Both primary and backup breakers have identical trip ratings and are located in the same MCC Compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements. Short circuit response time of THED and THFK breakers shall be less than 0.021 and 0.025 ~~0.027~~ seconds respectively. The breaker type and overload response time are listed below.

MCC 1EB1-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
4G	THED	15AT	40.7	45	Motor Operated Valve 1-TV-4691
4M	THED	15AT	40.7	45	Motor Operated Valve 1-TV-4693
3F	THED	15AT	40.7	45	Containment Drain Tank Pump-03
9H	THED	15AT	40.7	45	Reactor Cavity Sump Pump-01
9M	THED	15AT	40.7	45	Reactor Cavity Sump Pump-02
7H	THED	15AT	40.7	45	Containment Sump #1 Pump-01
7M	THED	15AT	40.7	45	Containment Sump #1 Pump-02
6H	THED	15AT	40.7	45	RCP #11 Motor Space Heater-01
6M	THED	15AT	40.7	45	RCP #13 Motor Space Heater-03
8B	THED	15AT	40.7	45	Incore Detector Drive "A"
8D	THED	15AT	40.7	45	Incore Detector Drive "B"

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

MCC 1EB1-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
7B	THED	15AT	40.7	45	Incore Detector Drive "F"
5D	THED	15AT	40.7	45	Fuel Transfer System Reactor Side Cont. Pnl FDR-01
3B	THED	15AT	40.7	45	Stud Tensioner Hoist Outlet-01
7D	THED	15AT	40.7	45	Hydraulic Deck Lift-01
4B	THED	15AT	40.7	45	Reactor Coolant Pump Motor Hoist Receptacles-42
8H	THED	40AT	40.7	120	RC Pipe Penetration Cooling Unit-01
8M	THED	40AT	40.7	120	RC Pipe Penetration Cooling Unit-02
5H	THED	40AT	40.7	120	RCP #11 Oil Lift Pump-01
5M	THED	40AT	40.7	120	RCP #13 Oil Lift Pump-03
10B	THED	40AT	40.7	120	Preaccess Filter Train Package Receptacles-17
5B	THED	45AT	40.7	135	Containment Ltg XFMR-14 (PNL-C3)
6D	THED	50AT	154	150	Refueling Machine (Manipilator Crane-01)
2M	THED	70AT	154	210	RC Drain Tank Pump No. 1
2F	THED	70AT	154	210	Containment Ltg XFMR-16 (PNL C7, C9)
1M	THED	70AT	154	210	Containment Ltg XFMR-12 (PNL C1 & C5)
3M	THED	100AT	132	300	Preaccess Fan No. 11

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

3. 480VAC from Motor Control Centers

3.2 Device Location - MCC 1EB2-2 Compt. Numbers listed below.

Primary and Backup Breakers - Both primary and backup breaker have identical trip ratings and located in the same MCC compt. These breaker are General Electric type THED or THFK with thermal-magnetic trip elements. Short circuit response time of THED and THEK breakers shall be less than 0.021 and ~~0.027~~ 0.025 seconds respectively. The breaker type and overload response time are listed below:

MCC-1EB2-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than Sec. at	Amp.	System Powered
4G	THED	15AT	40.7	45	Motor Operated Valve 1-TV-4692
4M	THED	15AT	40.7	45	Motor Operated Valve 1-TV-4694
3F	THED	15AT	40.7	45	Containment Drain Tank Pump-04
7H	THED	15AT	40.7	45	Containment Sump No. 2 Pump-03
7M	THED	15AT	40.7	45	Containment Sump No. 2 Pump-04
6H	THED	15AT	40.7	45	RCP No. 12 Motor Space Heater-02
6M	THED	15AT	40.7	45	RCP No. 14 Motor Space Heater-04
5B	THED	15AT	40.7	45	Incore Detector Drive "C"
2B	THED	15AT	40.7	45	Incore Detector Drive "D"
7B	THED	15AT	40.7	45	Incore Detector Drive "E"
5D	THED	15AT	40.7	45	Containment Fuel Storage Crane-01
3B	THED	15AT	40.7	45	Stud Tensioner Hoist Outlet-02
4B	THED	15AT	40.7	45	Containment Solid Rad Waste Compactor-01

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

MCC-1EB2-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ____Sec. at	____Amp.	System Powered
10B	THED	15AT	40.7	45	RCC Change Fixture Hoist Drive-01
10F	THED	30AT	40.7	90	Refueling Cavity Skimmer Pump-01
8H	THED	40AT	40.7	120	RC Pipe Penetration Fan-03
8M	THED	40AT	40.7	120	RC Pipe Penetration Fan-04
5H	THED	40AT	40.7	120	RCP No. 12 Oil Lift Pump-02
5M	THED	40AT	40.7	120	RCP No. 14 Oil Lift Pump-04
12H	THED	40AT	40.7	120	Preaccess Filter Train Package Receptacles - 18
6D	THED	40AT	40.7	120	Containment Auxiliary Upper Crane-01
2F	THED	45AT	40.7	135	Containment Ltg. XFMR-13
7D	THED	50AT	154	150	Containment Elevator-01
2D	THED	60AT	154	180	Containment Access Rotating Platform-01
2M	THED	70AT	154	210	Reactor Coolant Drain Tank Pump-02
9F	THED	70AT	154	210	Containment Ltg. XFMER-17 (PNL C8 & C10)
9M	THED	70AT	154	210	Containment Ltg. XFMR-15 (PNL C4 & C6)
3M	THED	100AT	132	300	Preaccess Fan-12
1G	THFK	150AT (Inst. setting ~~1250~~A) 1500	78 ~~.007~~ .025	450 ~~1375~~ 2000	Containment Welding Machine Power Supply Unit

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

3. 480VAC from Motor Control Centers

3.3 Device Location - MCC 1EB3-2, Compt. numbers listed below

Primary and Backup Breakers - Unless noted otherwise, both primary and backup breakers have identical trip ratings and are located in the same MCC compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements short circuit response time of THED and THFK breakers shall be less that 0.021 and ~~0.027~~ 0.025 seconds respectively. The breaker type and overload response time are listed below:

MCC-1EB3-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
8RF	THED	15AT	40.7	45	JB-1S-1005Ø for Altern. Feed to Motor Operated Valve 1-8702A
1G	THED	15AT	40.7	45	Motor Operated Valve 1-8112
9G	THED	15AT	40.7	45	Motor Operated Valve 1-8701A
9M	THED	15AT	40.7	45	Motor Operated Valve 1-8701B
5M	THED	15AT	40.7	45	Motor Operated Valve 1-8000A
5G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-6074
4G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-6076
4M	THED*	15AT*	40.7	45	Motor Operated Valve 1-HV-6078

*Primary protection is provided by GE type QMR fusible switch with 3.2A fuse.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

MCC-1EB3-2 Compt. No.	G.E Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
2G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-4696
2M	THED	15AT	40.7	45	Motor Operated Valve 1-HV-4701
3G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-5541
3M	THED	15AT	40.7	45	Motor Operated Valve 1-HV-5543
1M	THED	15AT	40.7	45	Motor Operated Valve 1-HV-6083
9RF	THED	30AT	40.7	~~45~~ 90	Motor Operated Valve 1-HV-4782
9RM	THED	40AT	40.7	~~45~~ 120	Motor Operated Valve 1-HV-8811A
6F	THED	50AT	154	150	Motor Operated Valve 1-HV-8808A
6M	THED	50AT	154	150	Motor Operated Valve 1-HV-8808C
7M	THED	70AT	154	210	Containment Ltg. XFMR-18
8M	THED	100AT	132	300	Neutron Detector Well Fan-09
7F	THFK	125AT	78	375	Electric H2 Recombiner Power Supply PNL-01
		(Inst. Setting 1250A)	~~.027~~ .025	~~1375~~ 1500	

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

3. 480VAC from Motor Control Centers

3.4 Device Location - MCC 1EB4-2, Compt. numbers listed below

Primary and Backup Breakers - Unless noted otherwise, both primary and backup breakers have identical trip ratings and are located in the same MCC compt. These breakers are General Electric type THED or THFK with thermal-magnetic trip elements. Short circuit response time of THED and THFK breakers shall be less than 0.021 and ~~0.027~~ 0,025 seconds respectively. The breakers type and overload response time are listed below.

MCC 1EB4-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
1M	THED	15AT	40.7	45	JB-1S-1230G, Altern. Power Supply Feed to Mov 1-8701B
8G	THED	15AT	40.7	45	Motor Operated Valve 1-8702A
8M	THED	15AT	40.7	45	Motor Operated Valve 1-8702B
4M	THED	15AT	40.7	45	Motor Operated Valve 1-8000B
4G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-6075
3G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-6077
3M	THED*	15AT*	40.7	45	Motor Operated Valve 1-HV-6079
2G	THED	15AT	40.7	45	Motor Operated Valve 1-HV-5562
2M	THED	15AT	40.7	45	Motor Operated Valve 1-HV-5563
8RF	THED	30AT	40.7	90	Motor Operated Valve 1-HV-4873
8RM	THED	40AT	40.7	120	Sump to #2 RHR Pump MOV 1-8811B

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

MCC 1EB4-2 Compt. No.	G.E. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	System Powered
5F	THED	50AT	154	150	Accumulator Iso. VLV, Mov-1-8808B
5M	THED	50AT	154	150	Accumulator Iso. VLV, Mov-1-8808D
6M	THED	70AT	154	~~150~~ 210	Containment Ltg. XFMR-19 (PNL SC2 & SC4)
7M	THED	100AT	132	~~200~~ 300	Neutron Detector Well Fan-10
6F	THFK	125AT	78	375	Elect. H_2 Recombiner Power Supply PNL-02
		(Inst. Setting - 1250A)	~~.027~~ .025	~~1375~~ 1500	

*Primary protection is provided by G.E. type QMR fusible switch with 3.2A fuse.

4. 480VAC from Panelboards for Pressurizer Heaters

System Powered - Pressurizer Heaters

A. Primary Breakers - General Electric Type TJJ Thermal Magnetic breaker

Breaker No. & Location - Ckt. Nos. 2 thru 4 of Panel boards 1EB1-1, 1EB1-2, 1EB2-2, 1EB3-2, 1EB4-1, 1EB4-2 and Ckt. Nos. 2 thru 5 of Panelboards 1EB2-1 and 1EB3-1

Set Points	Test Points (Amperes)	Response Time (Seconds)
Nominal Trip Setpoint and Rating - 125A	Long Time 250A	< 340 seconds
Instantaneous Pick-up - 375 Amp. (LO Magnetic Setting)	Instantaneous 1375	<.027 seconds

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

B. Backup Breakers - General Electric Type THJS with longtime and inst. Solid State trip device with 400 Amp. Sensor

Breaker No. & Location - Ckt. No. 1 of Panelboards 1EB1-1, 1EB1-2, 1EB2-1, 1EB2-2, 1EB3-1, 1EB3-2, 1EB4-1 and 1EB4-2.

Breaker Settings - For Ckt. No. 1 of Panelboards 1EB1-1, 1EB1-2, 1EB2-2, 1EB3-2, 1EB4-1 and 1EB4-2:

Setpoint		Test Point (Amperes)	Response Time (Seconds)
Long Time Pick-up	280A (0.7X)		
Long Time Trip Band	- Maximum	Long Time 840	<110
Inst. Trip Point	2800A (7X)	Instantaneous 3080A	<.28

For Ckt. No. 1 of Panelboards 1EB2-1 and 1EB3-1:

Setpoint		Test Point (Amperes)	Response Time (Seconds)
Long Time Pick-up	- 360A (0.9X)	Long Time 1080	< 55
Long Time Trip Band	- Intermediate	Instantaneous 3960	<.28
Inst. Trip Point	- 3600A (9X)		

5. DC Power from Rod Control Power Cabinets

System Powered - Rod Control
Fuse Location - Rod Control power Cabinets 1AC, 1BD, 2AC, 2BD and SCDE

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

A. Primary Fuses:

Fuse Location & Number	Fuse Rating Amp.	Fuse Resistance Greater Than ____ OHM	System Powered
FU13 to FU20	10	~~.006~~ .008	Stationary Gripper Coils
FU21 to FU24	10	~~.006~~ .008	Moving Gripper Coils
FU25 to FU32	10	~~.006~~ .008	Stationary Gripper Coils
FU33 to FU36	10	~~.006~~ .008	Moving Gripper Coils
FU37 to FU44	10	~~.006~~ .008	Stationary Gripper Coils
FU45 to FU52	10	~~.006~~ .008	Moving Gripper Coils
A51/FU1 & FU2 to A58/FU1 & FU2	50	~~.004~~ .0012	Lift Coils

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

B. Backup Fuses:

Fuse Location & Number	Fuse Rating Amp.	Fuse Resistance Greater Than ___ OHM	System Powered
FU1 to FU9	30	.004	Stationary Gripper Coils
Movable Bus. Duct Plug-in Unit A102-FU1 to FU3	30	.004	Moving Gripper Coils
Lift Bus-Duct Plug-in Unit A101-FU1 to FU3	150	~~.0002~~ .0003	Lift Coils

6. <u>120V Space Heater Circuits from 480V Switchgears:</u>

System Powered - Containment Recirc. Fan and CRDM Vent. Fan Motor Space Heaters.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Primary Breakers:

Bkr. Location & Number	Westinghouse Bkr Type	Nominal Trip Setpoint & Rating Amp	Overload Response Time Less Than Sec. at	Amp.	Inst. Response Time Less Than Sec. at	Amp.
Swgr. 1EB1, Cubicle 3A CP1-VAFNAV-01 Space Heater Bkr.	EB1010	10	28	30	.02	1000
Swgr. 1EB2, Cubicle 3A CP1-VAFNAV-02 Space Heater Bkr.	EB1010	10	28	30	.02	1000
Swgr. 1EB3, Cubicle 9A CP1-VAFNAV-03 Space Heater Bkr.	EB1010	10	28	30	.02	1000
Swgr. 1EB4, Cubicle 9A, CP1-VAFNAV-04 Space Heater Bkr.	EB1010	10	28	30	.02	1000
Swgr. 1EB3, Cubicle 8A CP1-VAFNCB-01 Space Heater Bkr.	EB1010	10	28	30	.02	1000
Swgr. 1EB4, Cubicle 8A CP1-VAFNAV-02 Space Heater Bkr.	EB1010	10	28	30	.02	1000

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

Backup Breakers:

Bkr. Location & Number	G.E. Bkr. Bkr. Type	Nominal Trip Setpoint & Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	Instant. Response Time Less Than ___Sec. at	___Amp.
Panel 1EC3-2 Ckt. No. 3	TED	~~45~~ 25	40.7	~~135~~ 75	.02	~~4500~~ 2500
Panel 1EC3-2 Ckt. No. 4	TED	~~25~~ 25	40.7	~~135~~ 75	.02	~~4500~~ 2500
Panel 1EC4-2 Ckt. No. 3	TED	~~45~~ 25	40.7	~~135~~ 75	.02	~~4500~~ 2500
Panel 1EC4-2 Ckt. No. 4	TED	~~45~~ 25	40.7	~~135~~ 75	.02	~~4500~~ 2500

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

7. 120V Space Heater Circuits from 480V MCC's:

Primary Fuses

Location - Each MCC Starter Compartment MCC's 1EB1-2, 1EB2-2, 1EB3-2 and 1EB4-2.

Backup Fuses -

Fuse Location and Number	Fuse Rating Amp.	Fuse Resistance Greater Than ___ Ohms	System Powered
MCC 1EB1-2 Compt. 12E, 1FU	20	.006 ~~.0094~~ ~~[illegible]~~	Space Heater Circuits from MCC 1EB1-2
MCC 1EB2-2 Compt. 12F, 1FU	20	.006 ~~.0094~~ ~~[illegible]~~	Space Heater Circuits from MCC 1EB2-2
MCC 1EB3-2 Compt. 7C, 1FU	20	.006 ~~.0127~~ ~~[illegible]~~	Space Heater Circuits from MCC 1EB3-2
MCC 1EB4-2 Compt. 6C, 1FU	20	.006 ~~.0127~~	Space Heater Circuits from MCC 1EB4-2

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

8. 125V DC Lighting:

System Powered - Emergency DC Lighting

Primary Breaker -

Breaker Location and Number	GE Bkr. Type	Nominal Trip Setpoint and Rating Amp.	Overload Response Time Less Than ___Sec. at	___Amp.	Inst. Response Time Less Than ___Sec. at	___Amp.
DC Panelboard 1D2-1, Ckt #6	TFJ	80 Amp (Inst. Set at LO-680A)	86	240	0.024	1050

Backup Fuse -

Fuse Location and Number	Fuse Rating ~~Rating~~ Amp	Fuse Resistance Greater Than ___Ohm
DC Switchboard 1D2, Ckt. #1-2	200A	.0002

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

9. 125V DC Control Power:

System Powered - Various

Primary Devices - 3 Amp fuses in termination cabinets listed below with backup devices

Backup Breakers:

Cab. No.	Panelboard No.	Ckt. No.	General Electric Breaker Type	Nominal Trip Rating (amp)	Overload Response Time Less Than ___ Sec. at	___ Amp.
01	XED1-1	1	TED	30	40.7	90
02	XED2-1	1	TED	30	40.7	90
03	XD2-3	8	TED	20	40.7	60
04	XED1-1	1	TED	30	40.7	90
05	1ED2-1	17	TED	20	40.7	60
06	XD2-3	8	TED	20	40.7	60
07	1ED1-1	14	TED	20	40.7	60
08	XED2-1	3	TED	30	40.7	90
09	1D2-3	10	TED	20	40.7	60
10	1ED1-1	14	TED	20	40.7	60
11	1ED2-1	17	TED	20	40.7	60
13	1ED1-1	17	TED	20	40.7	60
39A	XD2-1	11	TED	20	40.7	60

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 value 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)
1-HV-2480	Motor Driven AFW Pump A Service Water Supply Valve	YES *
1-HV-2481 ✓	Motor-Driven AFW Pump B Service Water Supply Valve	YES *
1-HV-2482	Turbine-Driven AFW Pump B Service Water Supply Valve	YES *
1-HV-2484	Condensate Storage Tank Isolation Valve	YES *
1-HV-2485	Condensate Storage Tank Isolation Valve	YES *
1-HV-2491A	Aux Feedwater Isolation Valve-Stm Gen No.1 (turbine pump)	YES *
1-HV-2491B	Aux Feedwater Isolation Valve -Stm Gen. No.1 (Motor Pump)	YES *
1-HV-2492A	Aux Feedwater Isolation Valve - Stm. Gen. No.2 (Turbine Pump)	YES *
1-HV-2492B	Aux Feedwater Isolation Valve - Stm. Gen. No.2 (Motor Pump)	YES *
1-HV-2493A	Aux Feedwater Isolation Valve - Stm. Gen. No.3 (Motor Pump)	YES *
1-HV-2493B	Aux Feedwater Isolation Valve - Stm. Gen. No.3 (Turbine Pump)	YES *
1-HV-2494A	Aux Feedwater Isolation Valve-Stm. Gen. No.4 (Motor Pump)	YES *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-HV-2494B	Aux Feedwater Isolation Valve - Stm. Gen. No. 4 (Turbine Pump)	YES *
1-HV-4512	Seal Steam Header Drain Valve	YES *
1-HV-4513		YES *
1-HV-4514		YES *
1-HV-4515		YES *
1-HV-4524		YES *
1-HV-4525		YES *
1-HV-4526		YES *
1-HV-4527		YES *
1-HV-4572		YES *
1-HV-4573		YES *
1-HV-4574		YES *
1-HV-4575		YES *
1-HV-4696		YES *
1-HV-4699	CCW Supply to Recirc. Pumps Coolers Control Valve	YES *
1-HV-4700		YES *
1-HV-4701		YES *
1-HV-4708		YES *
1-HV-4709		YES *
1-FV-4772-1		YES *
1-FV-4772-2		YES *
1-FV-4773-1		YES *
1-FV-4773-2		YES *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-HV-4758	Refueling Water to Containment Spray Pump 01 and 03 Control Valve	YES *
1-HV-4759		YES *
1-HV-4776	Containment Spray HX-01 Outlet Control Valve	YES *
1-HV-4777		YES *
1-HV-4782	Isolation Tank Control Valve	YES *
1-HV-4783		YES *
1-LV-4754	Chemical Additive Tank Discharge Isolation Valve	YES *
1-LV-4755		YES *
1-HV-4286	SSW Auto Strainer Inlet Valve	YES *
1-HV-4287	SSW Auto Strainer Inlet Valve	YES *
1-HV-4393	Diesel Generator Pack A to Service Water Control Vlv.	YES *
1-HV-4394	Diesel Generator Pack B to Service Water Control Vlv.	YES *
1-HV-4395	SSW Train A to Aux. FW Pumps Control Vlv.	YES *
1-HV-4396	SSW Train B to Aux. FW Pumps Control Vlv.	YES *
1-8000A	Pressurizer Relief Isolation Valve	YES *
1-8000B	Pressurizer Relief Isolation Valve	YES *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-8100	Reactor Coolant Pump Seal Water Isolation	Yes *
1-8112	RCP Seal Water Isolation Valve	Yes *
1-8351A	CVCS Seal Water Injection Isolation Valve	Yes *
1-8351B	CVCS Seal Water Injection Isolation Valve	Yes *
1-8351C	CVCS Seal Water Injection Isolation Valve	Yes *
1-8351D	CVCS Seal Water Injection Isolation Valve	Yes *
1-LCV-112B	Volume Control Tank Outlet Isolation Valve	Yes *
1-LCV-112C	Volume Control Tank Outlet Isolation Valve	Yes *
1-LCV-112D	Refueling Water Storage Tank To Charging Pump Vlv.	Yes *
1-LCV-112E	Refueling Water Storage Tank To Charging Pump Vlv.	Yes *
1-8104	Boric Acid Tank Filter To Charging Pump	Yes *
1-8105	Charging Pump To Reactor Coolant System Isolation	Yes *
1-8106	Charging Pump To Reactor Coolant System Isolation	Yes *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-8110	Centrifugal Charging Pump Mini-flow Isolation Vlv.	YES *
1-8111	Centrifugal Charging Pump Mini-flow Isolation Vlv.	YES *
1-FCV-610	RHR Pump 1 Miniflow Valve	YES *
1-FCV-611	RHR Pump 2 Miniflow Valve	YES *
1-8701A	RHR Loop 1 Inlet Isolation Valve	YES *
1-8701B	RHR Loop 2 Inlet Isolation Valve	YES *
1-8702A	RHR Loop 1 Inlet Isolation Valve	YES *
1-8702B	RHR Loop 2 Inlet Isolation Valve	YES *
1-8716A	RHR Cross Connect Valve	YES *
1-8716B	RHR Cross Connect Valve	YES *
1-8801A	Safety Injection Isolation Valve	YES *
1-8801B	Safety Injection Isolation Valve	YES *
1-8804A	RHR Pump 1 To SI Pump	YES *
1-8804B	RHR Pump 2 To SI Pump	YES *
1-8807A	Suction Header Cross Connect	YES *
1-8807B	Suction Header Cross Connect	YES *
1-8924	Safety Injection Charging Suction Cross Connection	YES *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-8808A	Accumulator Isolation Valve	YES *
1-8808B	Accumulator Isolation Valve	YES *
1-8808C	Accumulator Isolation Valve	YES *
1-8808D	Accumulator Isolation Valve	YES *
1-8802A	Safety Injection Pump To Hot Legs Isolation Valve	YES *
1-8802B	Safety Injection Pump To Hot Legs Isolation Valve	YES *
1-8806	Refueling Water Storage Tank To Safety Injection Pumps	YES *
1-8809A	RHR System To Cold Leg Isolation Valve	YES *
1-8809B	RHR System To Cold Leg Isolation Valve	YES *
1-8811A	Containment Sump To RHR Pump 1	YES *
1-8811B	Containment Sump To RHR Pump 2	YES *
1-8812A	Refueling Water Storage Tank to RHR Pump 1 Isolation Valve	YES *
1-8812B	Refueling Water Storage Tank to RHR Pump 2 Isolation Valve	YES *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-8813	Safety Injection Pump Miniflow Isolation Valve	Yes *
1-8814A	Safety Injection Pump Miniflow Isolation Valve	Yes *
1-8814B	Safety Injection Pump Miniflow Isolation Valve	Yes *
1-8821A	Safety Injection Pump Cross Connection Valve	Yes *
1-8821B	Safety Injection Pump Cross Connection Valve	Yes *
1-8835	Safety Injection Pump to Cold Legs Isolation Valve	Yes *
1-8840	RHR To Hot Leg Isolation Valve	Yes *
1-8923A	No. 1 Safety Injection Pump Soction Valve	Yes *
1-8923B	No. 2 Safety Injection Pump Suction Valve	Yes *
1-HV-5540	H2 Controlled Purge Exhaust	Yes *
1-HV-5541	H2 Controlled Purge Exhaust	Yes *
1-HV-5542	H2 Controlled Purge Supply	Yes *
1-HV-5543	H2 Controlled Purge Supply	Yes *
1-HV-5562	H2 Controlled Purge Exhaust	Yes *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
1-HV-5563	H2 Controlled Purge Supply	Yes *
X-HV-5825A	Motor Operated Damper	Yes *
X-HV-5826	Motor Operated Damper	Yes *
X-HV-5828A	Motor Operated Damper	Yes *
X-HV-5829	Motor Operated Damper	Yes *
X-HV-5831	Motor Operated Valve	Yes *
X-HV-5834	Motor Operated Damper	Yes *
X-PV-5855	Motor Operated Damper	Yes *
X-PV-5856	Motor Operated Damper	Yes *
X-HV-5857	Motor Operated Damper	Yes *
X-HV-5858	Motor Operated Damper	Yes *
1-HV-6082	Chilled Wtr Return Header/Chilled Wtr	Yes *
1-HV-6083	Header/Chilled Wtr Chilled Wtr Return	Yes *
1-HV-6084	Header/Chilled Wtr	Yes *

TABLE 3.8-2 (continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
X-HV-5579	H2 Purge Exh Bypass Vlv.	YES *
X-HV-5526	Cont't H2 Purge Air Exh to Fan 01 Damper	YES *
X-HV-5529	Cont't H2 Purge Air Exh to Fan 02 Damper	YES *
X-HV-5580	H2 Purge Exhaust Bypass Valve	YES *
X-HV-5825E	Control Bldg. Control Room Make-Up Air Supply Fan 37 Exhaust Damper	YES *
X-HV-5828E	Control Bldg. Control Room Make-Up Air Supply Fan 38 Exhaust Damper	YES *

* Permanently bypassed - provides alarm only

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less

b. A boron concentration of greater than or equal to (2000) ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to ( ) gpm of a solution containing greater than or equal to ( ) 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.

30

7000

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.

*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.

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. A ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A 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 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

    1. Closed by an isolation valve, blind flange, or manual valve, or

    2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the Containment 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~~ REFUELING MACHINE

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 (2750) pounds, and

2. An overload cutoff 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 (610) pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of (600) 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 (2750) pounds and demonstrating an automatic load cutoff 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 (610) pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE ~~POOL BUILDING~~ AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2100 pounds shall be prohibited from travel over fuel assemblies in the storage ~~pool~~ areas.

APPLICABILITY: With fuel assemblies in the storage ~~pool~~ Areas.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2100 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor 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 [illegible] containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 The required RHR loop shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

4.9.8.1.2 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.

3800

*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

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet, with irradiated fuel in the reactor pressure vessel.

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 vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

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3/4.9.9 CONTAINMENT VENTILATION ~~PURGE AND EXHAUST~~ ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation ~~Purge and Exhaust~~ Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Containment Ventilation ~~Purge and Exhaust~~ Isolation System inoperable, close each of the ~~Purge and Exhaust~~ Ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation ~~Purge and Exhaust~~ Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ~~Purge and Exhaust~~ Ventilation isolation occurs on manual initiation and on a high radiation test signal from each of the containment air radiation monitor~~ing~~ ~~instrumentation channels~~.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel 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

4.9.10 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.

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

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

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 STORAGE POOL AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel storage pool air cleanup systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel storage pool air cleanup system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel storage pool air cleanup system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no fuel storage pool air cleanup system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one spent fuel storage pool air cleanup system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel storage pool air cleanup systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of ____ cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)

2. 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 ____ cfm ± 10%.

3. 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.

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

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

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of ____ cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to (1/4) inches Water Gauge relative to the outside atmosphere during system operation.

4. Verifying that the filter cooling bypass valves can be manually opened.

5. Verifying that the heaters dissipate _____ ± _____ kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the 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 ______ 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 _____ cfm ± 10%.

*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99% when a filter efficiency of 90% is assumed.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided:

~~a.~~ ~~R~~reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

~~b. All part-length rods are withdrawn to at least the 180-step position and OPERABLE.~~

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 ~~or the part-length rods not within their withdrawal limits~~, 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 ~~and part-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.

~~4.10.1.3 The part-length rods shall be demonstrated OPERABLE by moving each part length rod greater than or equal to 10 steps within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.~~

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.1.3.7,~~ 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, ~~3.1.3.7~~, 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 Specfications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specification 4.2.2.2 and 4.2.2.3

b. Specification 4.2.3.2

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, 3.1.3.6, and ~~3.1.3.7~~ may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to (~~531~~)°F.

APPLICABILITY: MODE 2. 541

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than (531)°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a 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 (~~531~~)°F at least once per 30 minutes during PHYSICS TESTS.

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 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 a 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.

*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.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION Statements and whose occurrence would violate the intent of a 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 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 Containment 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 the required Containment Spray Systems are inoperable, within one hour measure 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 within 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.

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3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of (1.6%) delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value (-~~3.9~~ 4.0) x 10^{-4} delta k/k/°F. The MTC value of (-~~3.0~~ 3.1) x 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 (-~~3.9~~ 4.0) x 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 (~~541~~ 551)°F. This limitation is required to ensure 1) the moderator temperature coefficient is within it analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) ~~associated heat tracing systems, and~~ 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires ~~(5106)~~ 17,000 gallons of 7000 ~~20,000~~-ppm borated water from the boric acid storage tanks or 132,000 ~~(52,622)~~ 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~~ 295)°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either (920) gallons of 7000 ~~20,000~~-ppm borated water from the boric acid storage tanks or (7120) 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 and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod

55 drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to (~~541~~)°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.

~~(ALTERNATE if required by DNB considerations)~~

The restriction prohibiting part length rod insertion ensures that adverse power shapes and rapid local power changes which may affect DNB considerations do not occur as a result of part-length rod insertion during operation.

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 as near to full power as practicable ~~with the part-length control rods withdrawn from the core.~~ The full-length rods may be positioned within the core in accordance with their respective insertion limits, however they ~~and~~ should be inserted near their normal position for steady-state operation at high power levels (less than 215 steps withdrawn). 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 $\pm$(5)% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure (3.2-1) while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for 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 $\pm$ 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.

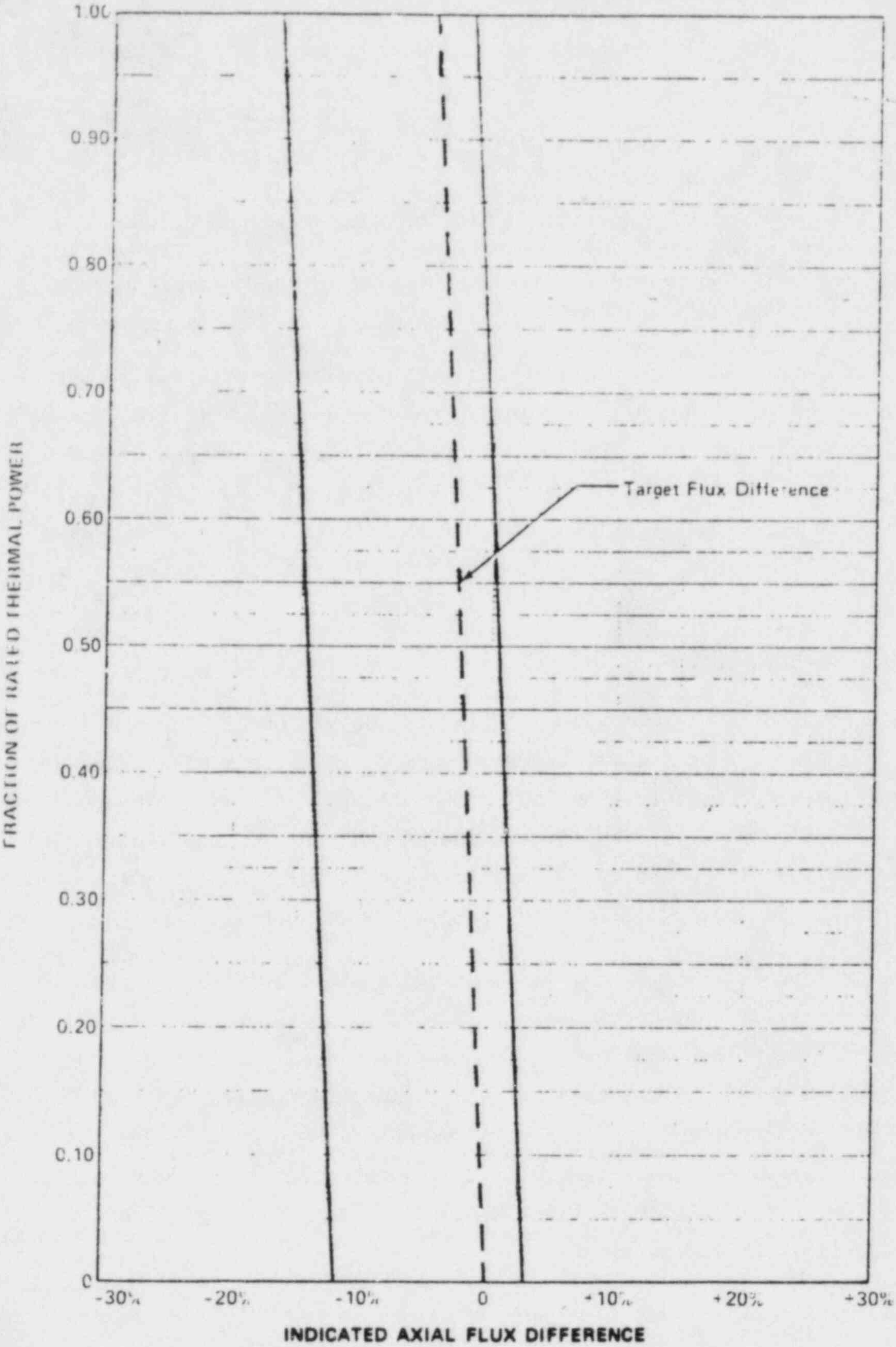

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figure~~s~~ 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 the "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.

~~Fuel rod bowing reduces the value of DNB ratio. Sufficient credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% and 3% margin in the difference between the 1.3 DNBR safety limit and the minimum DNBR calculated for the Complete Loss of Flow event.~~ 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).

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.

BASES

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 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.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. ~~A limiting tilt of (1.025) can be tolerated before the margin for uncertainty in F_Q is depleted~~. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 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.

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.

3/4.3.1 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 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 Feature 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)

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 on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from 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 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.8 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.

3/4.3.3.9 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.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. ~~Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment, or structures.~~ Although the orientation of the turbine is such that potentially damaging missles which could impact and damage safety related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.

3/4.4 REACTOR COOLANT SYSTEM

BASES

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 (~~275~~ 295)°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 (   )°F above each of the RCS cold leg temperatures.

~~(OPTIONAL)~~

~~The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.~~

~~REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)~~

~~(OPTIONAL)~~

~~Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water-induced reactivity transient.~~

3/4.4.2 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 20,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. 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 setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 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.

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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 {CPSES} 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°hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the ~~NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975.~~" [INSERT A]

- [INSERT B] -

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 (~~12~~) effective full power years of service life. The (~~12~~ 16) EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

REACTOR COOLANT SYSTEM

BASES

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, ~~and~~ copper content and phosphorus content of the material in question, can be predicted using Figure 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.~~" [INSERT C] 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 (~~12~~ 16) EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 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 ~~WCAP-7924-A.~~ the followin 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 Section III as the reference flaw, amply

INSERTS for Page B 3/4 4-7

[INSERT A]

1972 Winter Addenda to section III of the ASME Boiler and Pressure Vessel Code.

[INSERT B]

The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 are applicable to Comanche Peak Unit 1 for up to 16 EFPY. The most limiting material in Comanche Peak Unit 1 has an initial RT_{NDT} of 20°F and a copper content of 0.05 WT%. Whereas, the heatup and cooldown curves in Figures 3.4-2 and 3.4-3 are based on an initial RT_{NDT} of 40°F and a copper content of 0.10 WT%. As a result, Figures 3.4-2 and 3.4-3 are conservatively applicable to Comanche Peak Unit 1.

INSERTS For P B 3/4 4-8

[INSERT C]

the largest value of ΔRT_{NDT} computed by either the Regulatory Guide 1.99 Trend Curves from the 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.

TABLE B 3/4.4-1

COMANCHE PEAK UNIT #1 REACTOR VESSEL FRACTURE TOUGHNESS PROPERTIES

COMPONENT	GRADE	CODE NO	Cu %	P %	NDT °F	35 MIL TEMP. °F	RT_{NDT} °F	MWD SHELF ENERGY FT-LB	NMWD SHELF ENERGY FT-LB
CLOSURE HD. DOME	A533B, CL 1	R1110-1	—	—	10	100	40	—	126.0
" " TORUS	"	R1111-1	—	—	-50	30	-30	—	116.5
" " FLANGE	A508 CL.2	R1102-1	—	—	40	100	40	—	119.0
VESSEL FLANGE	"	R1101-1	—	—	10	70	10	—	97.0
INLET NOZZLE	"	R1105-1	—	—	-10	50	-10	—	147.0
" "	"	R1105-2	—	—	-20	40	-20	—	136.5
" "	"	R1105-3	—	—	-10	50	-10	—	134.0
" "	"	R1105-4	—	—	-10	50	-10	—	156.5
OUTLET NOZZLE	"	R1106-1	—	—	-20	40	-20	—	135.0
" "	"	R1106-2	—	—	-10	50	-10	—	111.0
" "	"	R1106-3	—	—	-20	50	-10	—	135.5
" "	"	R1106-4	—	—	-20	40	-20	—	117.5
UPPER SHELL	A533B, CL 1	R1104-1	.07	.012	-30	100	40	—	83.0
" "	"	R1104-2	.08	.011	-50	100	40	—	75.0
" "	"	R1104-3	.05	.010	-20	70	10	—	107.5
INTER SHELL	"	R1107-1	.06	.010	-20	70	10	111.5	93.5
" "	"	R1107-2	.06	.010	-20	50	-10	123.5	103.0
" "	"	R1107-3	.05	.007	-20	70	10	131.0	88.0
LOWER SHELL	"	R1108-1	.08	.008	-20	60	0	119.0	85.0
" "	"	R1108-2	.05	.006	-30	80	20	124.5	78.0
" "	"	R1108-3	.07	.008	-30	60	0	122.0	98.0
BOTTOM HD. TORUS	"	R1112-1	—	—	-50	50	-10	—	112.0
" " DOME	"	R1113-1	—	—	-50	70	10	—	90.0
[illegible]		G1.67	.04	.008	-70	-10	-70	—	150.0

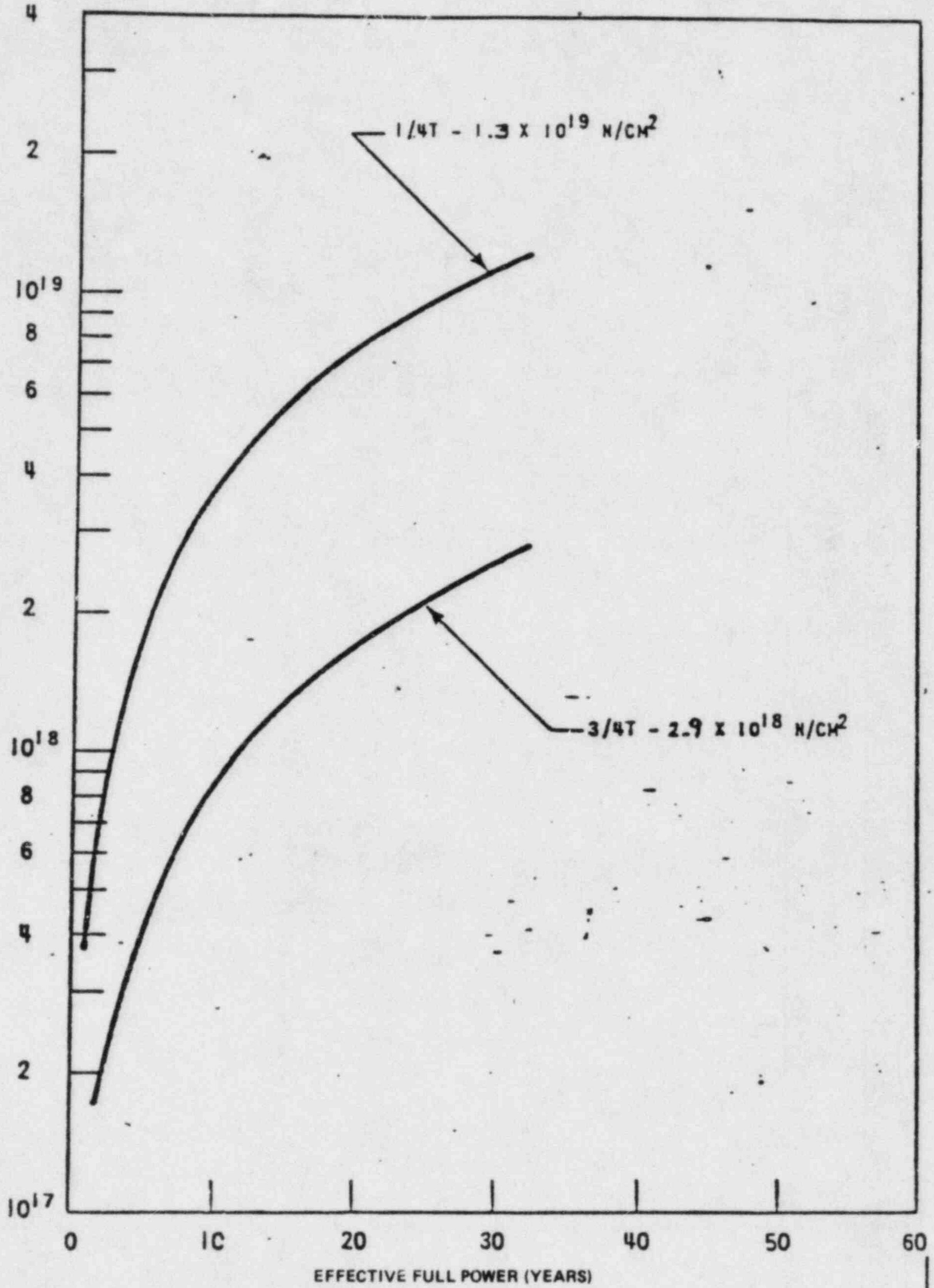


Figure B 3/4 4.1 Fast Neutron Fluence (E> 1mev) as a Function of Full Power Service Life

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

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.

STEAM GENERATORS (Continued)

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 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 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

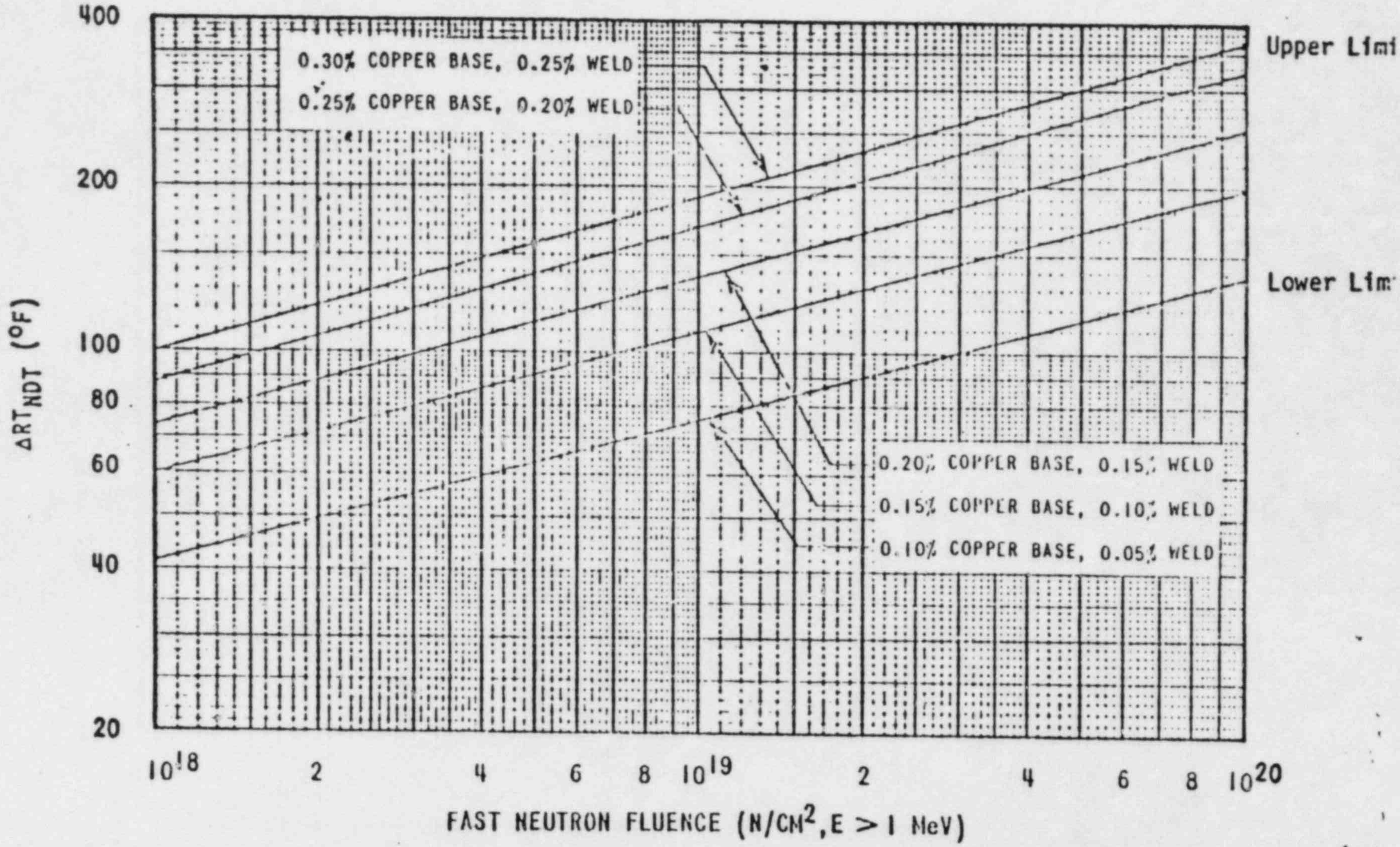
The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds (40) GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6



B 3/4.4-2

FIGURE A Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550°F

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 heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

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.

LATER

The OPERABILITY of two PORVs or an RCS vent opening of greater than ( ) 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 ~~(275)~~°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 ( )°F above the RCS cold leg temperatures or (2) the start of a ~~HPSI~~ pump and its injection into a water solid RCS.

295

50

Charging

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 and Addenda through Summer 1975

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 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 and one safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable below ~~(275)~~ 295°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 RCS water

REFUELING WATER STORAGE TANK (Continued)

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.

BASES

FOR

SECTION 3/4.6A

CONTAINMENT SYSTEMS SPECIFICATIONS

FOR

WESTINGHOUSE

ATMOSPHERIC TYPE CONTAINMENT

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 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals 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 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS~~ (OPTIONAL)

The OPERABILITY of the ~~isolation valve and~~ containment channel weld pressurization systems is ~~required~~ to ~~meet the~~ restrictions on overall containment leak ~~rate~~ assumed in the accident analyses. ~~The surveillance Requirements~~ for determing OPERABILITY are consistent with Appendix ~~"J" of~~ 10 ~~CFR 50.~~

3/4.6.1.5 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of (~~3.0~~) 5.0 psig and 2) the containment peak pressure does not exceed the design pressure of (~~54~~) 50 psig during (LOCA or steam line break conditions).

The maximum peak pressure expected to be obtained from a (LOCA or steam line break) event is (~~48~~) 44.2 psig. The limit of (3) psig for initial positive containment pressure will limit the total pressure to 48.1 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.6 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a (LOCA or steam line break accident).

3/4.6.1.7 CONTAINMENT STRUCTURAL INTEGRITY

~~(Prestressed concrete containment with ungrouted tendons.)~~

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of (48) psig in the event of a (LOCA or steam line break accident). The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, 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 wire or strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the ~~tendon's operating conditions and environment.)~~

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", January 1976.

(Reinforced concrete containment.)

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

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

will withstand the maximum pressure of 48.1 psig in the event of a (LOCA or steam line break accident). A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.8 CONTAINMENT VENTILATION SYSTEM

The (~~42~~ 48-inch) containment ~~purge supply and exhaust~~ ventilation 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.

The use of the containment purge lines is restricted to the (~~8-inch~~ 18-inch) Pressure relief ~~purge supply and exhaust~~ isolation valves to ensure that the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of a loss-of-coolant accident durng purging operations.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a (LOCA or steam line break). The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

~~(Credit taken for iodine removal)~~

The containment spray system which is composed of two redundant trains ~~and the containment cooling system are redundant to each other in~~ provides post accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

~~(No credit taken for iodine removal)~~

~~The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the containment spray system, the allowable out-of-service time requirements for the containment spray system and containment cooling system have been interrelated and adjusted to reflect this additional redundancy in cooling capability.~~

3/4.6.2.2 SPRAY ADDITIVE SYSTEM ~~(OPTIONAL)~~

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between (8.5) and (11.0) for the

SPRAY ADDITIVE SYSTEM (Continued)

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 water volume limit includes an allowance for water 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 CONTAINMENT COOLING SYSTEM (OPTIONAL)

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

(Credit taken for iodine removal by spray systems)

The containment cooling system and the containment spray system are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out-of-service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

(No credit taken for iodine removal by spray systems)

The containment cooling system and the containment spray system are redundant to each other in providing post accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the containment spray system, the allowable out-of-service time requirements for the containment cooling system and containment spray system have been interrelated and adjusted to reflect this additional redundancy in cooling capacity.

3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)

The OPERABILITY of the containment iodine 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. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and 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 containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.5 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 containment 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.** ~~(Cumulative operation of the purge system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters)~~. 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.~~

~~3/4.6.6 PENETRATION ROOM EXHAUST AIR FILTRATION SYSTEM~~ (OPTIONAL)

The OPERABILITY of the penetration room exhaust system ensures that radioactive materials leaking from the containment atmosphere through containment penetrations following a LOCA are filtered and adsorbed prior to reaching the environment. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the LOCA analyses.

3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than ___ psig. This condition is necessary to prevent exceeding the ~~containment design limit for internal vacuum of ___ psig.~~

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% (~~1100~~ 1303.5 psig) of its design pressure of (~~1000~~ 1185) psig during the most severe anticipated system operational transient. The ~~maximum~~ rated 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~~ 1974 Edition. The total rated relieving capacity for all valves on all of the steam lines is (18,190,884) lbs/hr which is (120) percent of the total secondary steam flow of (15,140,016) 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 ~~N~~ 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

~~For N-1 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (76)}$$

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 (~~N~~ 4) loop operation

~~(76) = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for (N-1) 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 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than (350)°F from normal operating conditions in the event of a total loss of offsite power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (~~350~~ 470) gpm at a pressure of (~~1133~~ 1221) psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of (~~700~~ 940) gpm at a pressure of (~~1133~~ 1221) psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than (350)°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for (18.22) 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 containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the 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 a steam generator RT_{NDT} of (60)°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the 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 ~~(OPTIONAL)~~

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 limitation~~s~~ on minimum water level ~~and maximum temperature are~~ is 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. The limitation on maximum temperature is based on the recommended cooling for the diesel generators.

3/4.7.6 FLOOD PROTECTION ~~(OPTIONAL)~~

The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation (778 Mean Sea Level is based on the maximum elevation at which facility flood control measures may need to be augmented to provide protection to safety-related equipment in the Electrical and Control building.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP 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. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The 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.

3/4.7.8 ESF ~~ECCS PUMP ROOM~~ EXHAUST AIR FILTRATION SYSTEM

The OPERABILITY of the ESF ~~ECCS pump room~~ exhaust air filtration system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspectior or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. Selection of a representative sample according to the expression $35 (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly 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 not intended to affect plant operation.

R.G. 8.21 and 10CFR 35.14
(b) 5 and (e) 1 and 10CFR 31.5
(c) 2

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on ~~10 CFR 70.39(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.

3/4.7.11 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~~, ~~Halon~~, 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.

FIRE SUPPRESSION SYSTEMS (Continued)

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.

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.

3/4.7.12 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.

3/4.7.13 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 ( )°F.~~

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.

rated over 8Amps

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 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.

Small fuses rated less than 8 amps used in control circuits, are sized well below the ampacity of minimum size control penetration conductors (12 AWG) they protect,

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.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 ~~MANIPULATOR CRANE~~ REFUELING MACHINE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of drive rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE ~~BUILDING~~ AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage rocks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT ~~PURGE AND EXHAUST~~ VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 STORAGE POOL VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN 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 occuring 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.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure (5.1-1).

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure (5.1-2).

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 135 feet.

b. Nominal inside height = 195 feet. (DOME 67.5 feet - Total = 262.5 feet)

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 12 feet.

f. Nominal thickness of steel liner, wall = 3/8 inches.

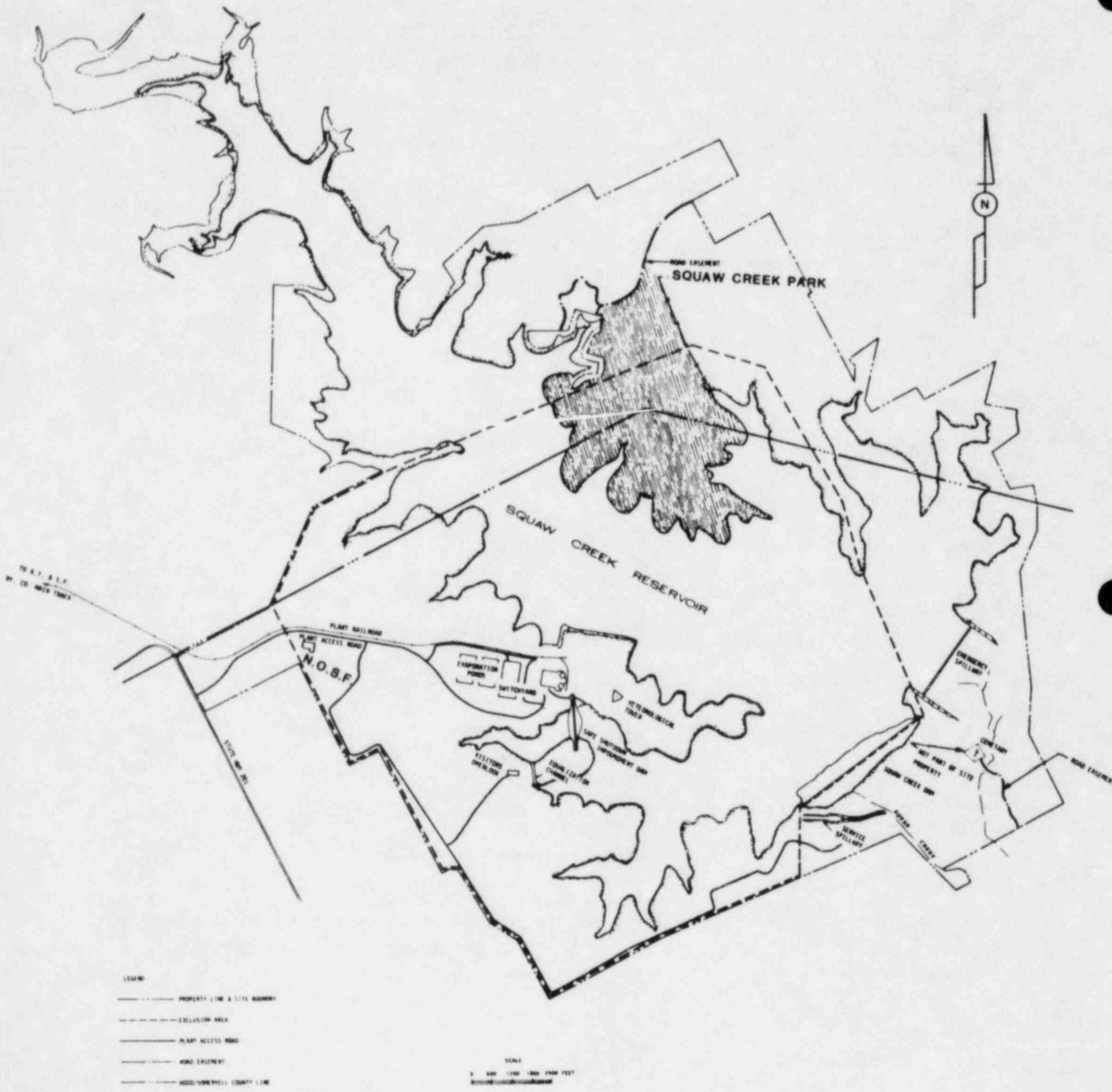
g. Net free volume = 2.985 X10⁶ cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 280°F.

Dome = 1/2 inches

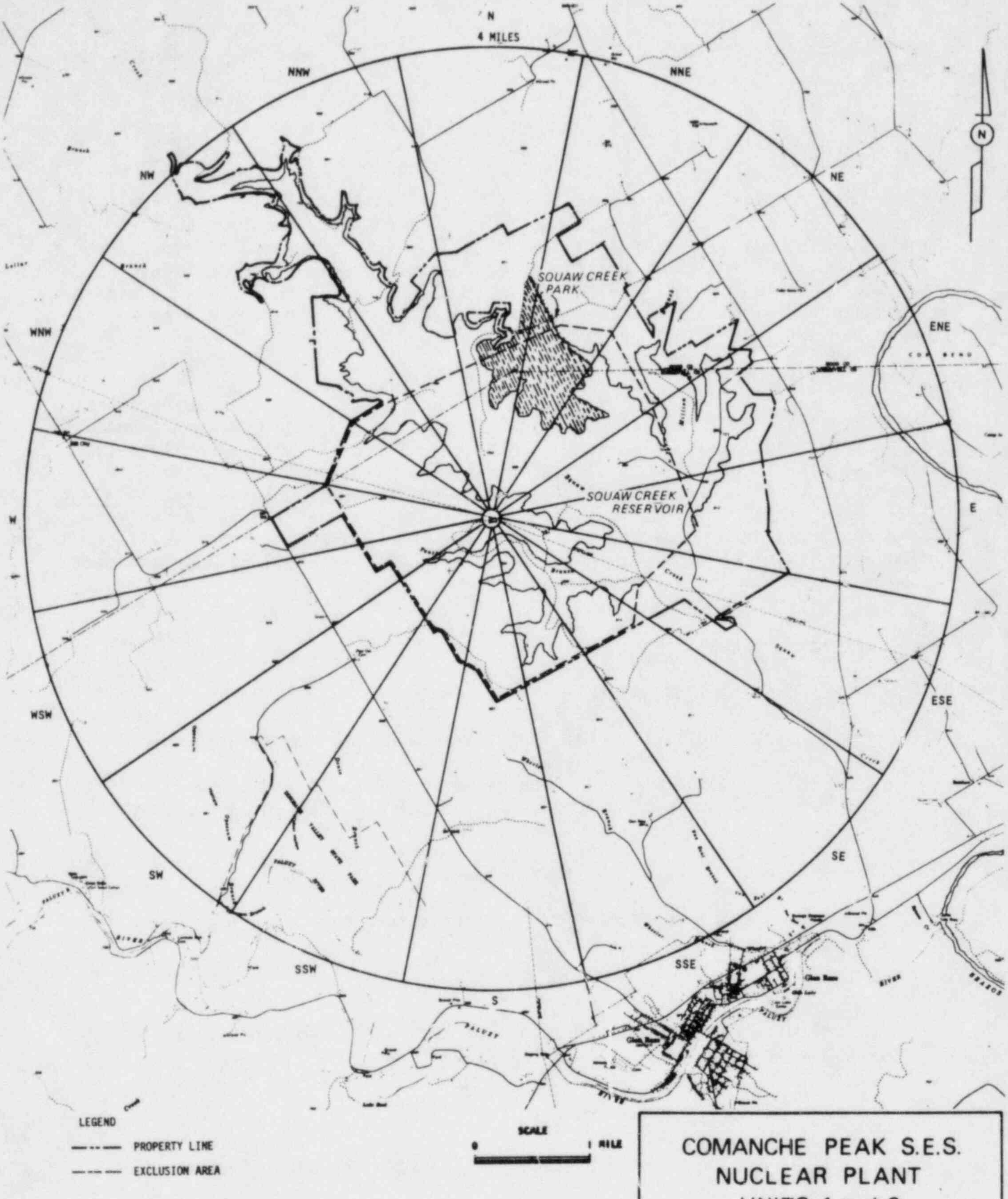
BaseMat = 1/4 inches



COMANCHE PEAK S.E.S.
NUCLEAR PLANT
UNITS 1 and 2

EXCLUSION AREA

FIGURE **5.1-1**



COMANCHE PEAK S.E.S.
NUCLEAR PLANT
UNITS 1 and 2

LOW POPULATION ZONE

FIGURE **5.1-2**

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 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 3.15 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 53 full length ~~and ____ part length~~ control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. ~~The part length control rod assemblies shall contain a nominal 36 inches of absorber material at their lower ends~~. The nominal values of absorber material shall be ~~80~~ 95.5 percent natural hafnium ~~silver, 15 percent indium and 5 percent cadmium.~~ All control rods shall be clad with stainless steel tubing. ~~The balance of the void length in the part length rods shall contain aluminum oxide.~~

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 12,500 ± 100 cubic feet at a nominal T_{avg} of (~~525~~)°F.
588.5

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The Primary meteorological tower shall be located as shown on Figure (5.1-1).

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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, with fuel assemblies of the highest anticipated enrichment in place ~~which includes a conservative allowance of (2.6%) delta k/k for uncertainties as described in Section (4.3) of the FSAR.~~

b. A nominal (~~21~~ 16) inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed (0.98) when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 854ft

CAPACITY

~~5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ___ 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.

5.6.3 Spent fuel storage pool No. 1 is designed and shall be maintained with a storage capacity limited to no more than 556 fuel assemblies, based on 16 inch center-to-center rack design. Spent fuel storage pool No. 2 shall remain without racks and a storage capacity of 0 fuel assemblies until additional capacity is required, at which time an appropriate design will be selected.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 (~~250~~) heatup cycles at $\leq$ 100°F/hr and 200 (~~250~~) cooldown cycles at < 100°F/hr.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 (~~250~~) pressurizer cooldown cycles at $\leq$ 200°F/hr.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 (~~100~~) loss of load cycles, without immediate turbine or reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 (~~50~~) cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical ESF Electrical System.
	80 (~~100~~) cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 (~~500~~) reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	~~(~~10~~)~~ inadvertent auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 ~~(50)~~ leak tests.	Pressurized to $\geq$ ~~(~~2485~~)~~ psig.
	10 ~~(5)~~ hydrostatic pressure tests.	Pressurized to $\geq$ (~~3100~~) 3106 psig.
Secondary System	~~(~~1~~)~~ steam line break.	Break in a > 6 inch steam line.
	10 ~~(5)~~ hydrostatic pressure tests.	Pressurized to $\geq$ (~~1350~~) 1481 psig.

STANDARD

TECHNICAL SPECIFICATIONS

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The MANAGER, PLANT OPERATIONS ~~(Plant Superintendent)~~ 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 (or during his absence from the Control Room, AN ASSISTANT SHIFT SUPERVISOR OR REACTOR OPERATOR ~~a designated individual)~~ shall be responsible for the Control Room command function. A management directive to this effect, signed by the VICE PRESIDENT, NUCLEAR ~~(highest level of corporate management)~~ 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 in Figure 6.2-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed 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. A Person qualified in radiation protection procedures

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 (2) other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

#The RADIATION PROTECTION ~~health physics~~ technicians 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.

This figure shall show the organizational structure and lines of responsibility for the offsite groups that provide technical and management support for the unit. The organizational arrangement for performance and monitoring Quality Assurance activities should also be indicated.

LATER

Figure 6.2-1

OFFSITE ORGANIZATION

This figure shall show the organizational structure and lines of responsibility for the unit staff. Positions to be staffed by licensed personnel should be indicated.

LATER

Figure 6.2-2

UNIT ORGANIZATION

INSERT ATTACHED TABLE 6.2-1

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION
SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1	1
~~SRO~~ Ass'T S.S. (SRO)	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
SRO - Individual 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 Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

Table 6.2-1a

MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH A COMMON CONTROL ROOM

WITH UNIT 2 IN MODE 5 OR 6 OR DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
~~SRO~~ Ass't SS (SRO)	1	None
RO	2	1
AO	2	2[b]
STA	1	None

WITH UNIT 2 IN MODES 1, 2, 3 OR 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 or 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1[a]	None
RO	2[b]	1
AO	2[b]	1
STA	1[a]	None

a/ Individual may fill the same position on Unit 2

b/ One of the two required individuals may fill the same position on Unit 2.

INSERT

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH A COMMON CONTROL ROOM

MODE	UNIT LICENSED TO OPERATE	
	UNIT 1	UNIT 1 AND 2
ONE OR BOTH UNITS IN MODE 1, 2, 3, or 4	1 S. S. 1 Ass't. S. S. 2 R. O. 2 A. O. 1 STA	1 S. S. 1 Ass't S. S. 3 R. O. 3 A. O. 1 STA
TOTAL	~~6~~ 7	~~8~~ 9
BOTH UNITS IN MODE 5 or 6	1 S. S. 1 R. O. 1 A. O.	1 S. S. 2 R. O. 3 A. O.
TOTAL	3	6

POSITION (1)		USNRC LICENSE
SHIFT SUPERVISOR	- S. S.	SRO
ASSISTANT SHIFT SUPERVISOR	- Ass't S. S.	SRO
REACTOR OPERATOR	- R. O.	RO
AUXILIARY OPERATOR	- A. O.	NONE
SHIFT TECHNICAL ADVISOR	- S.T.A.	NONE

(1) Any qualified and USNRC Senior Licensed member of management may be used to satisfy the minimum Shift Supervisor or Assistant Shift Supervisor requirement. Any qualified and USNRC Licensed individual may be used to satisfy the Reactor Operator requirement.

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 at least five, dedicated, full-time engineers located ~~on site.~~ IN THE NUCLEAR OPERATIONS SUPPORT FACILITY

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for ~~maintaining surveillance~~ REVIEWS 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 revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to ~~(a high level corporate official in a technically oriented position who is not in the management chain for power production).~~ THE OPERATIONS SUPPORT SUPERINTENDENT.

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).

Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing (an ANSI Standard agreed to by the NRC staff) or alternately by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of a unique organizational structure.

*Not responsible for sign-off function.

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6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the ~~(position title)~~ DIRECTOR, NUCLEAR TRAINING and shall meet or exceed the requirements and recommendations of Section ( ) 5.5.1 AND 5.5.2, RESPECTIVE of ~~(an~~ ANSI N-18.1-1971 ~~Standard agreed to by the NRC staff)~~ and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG.

6.5 REVIEW AND AUDIT

The method by which independent review and audit of facility operations is accomplished may take one of several forms. The licensee may either assign this function to an organizational unit separate and independent from the group having responsibility for unit operation or may utilize a standing committee composed of individuals from within and outside the licensee's organization.

Irrespective of the method used, the licensee shall specify the details of each functional element provided for the independent review and audit process as illustrated in the following example specifications.

6.5.1 ~~UNIT REVIEW GROUP (URG)~~ STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The ~~(Unit Review Group)~~ STATION OPERATIONS REVIEW COMMITTEE shall function to advise the ~~(Plant Superintendent)~~ MANAGER, PLANT OPERATIONS on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The ~~(Unit Review Group)~~ STATION OPERATIONS REVIEW COMMITTEE shall be composed of the:

Chairman:	~~(Plant Superintendent)~~ MANAGER, PLANT OPERATIONS
Member:	~~(Operations Supervisor)~~ OPERATIONS SUPERINTENDENT
Member:	~~(Technical Supervisor)~~ ENGINEERING SUPERINTENDENT
Member:	~~(Maintenance Supervisor)~~ MAINTENANCE SUPERINTENDENT
Member:	~~(Plant Instrument and Control Engineer)~~ QUALITY ASSURANCE SUPERVISOR
Member:	~~(Plant Nuclear Engineer)~~ RADIATION PROTECTION ENGINEER
Member:	~~(Health Physicist)~~ Administrative Superintendent

ALTERNATES

6.5.1.3 ~~All alternate members shall be appointed in writing by the (URG) Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in (URG) activities at any one time.~~
Each SORC member shall appoint an alternate member. Each alternate member shall be approved by the SORC Chairman

MEETING FREQUENCY

6.5.1.4 The ~~(URG)~~ SORC shall meet at least once per calendar month and as convened by the ~~(URG)~~ SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the ~~(URG)~~ SORC necessary for the performance of the RC ~~(URG)~~ responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and ~~four~~ three members ~~including~~ or alternates.

RESPONSIBILITIES

6.5.1.6 The ~~(Unit Review Group)~~ STATION OPERATIONS REVIEW COMMITTEE shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) all programs required by Specification 6.8 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the ~~(Plant Superintendent)~~ MANAGER, ~~NU~~ PLANT OPERATIONS to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety and are not described in the Final Safety Analysis Report.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence ~~to the (Superintendent of Power Plants) and~~ to the ~~(Company Nuclear Review and Audit Group).~~ OPERATIONS REVIEW COMMITTEE

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the ~~(Plant Superintendent) or the (Company Nuclear Review and Audit Group).~~ OPERATIONS REVIEW COMMITTEE

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the ~~(Company Nuclear Review and Audit Group)~~. OPERATIONS Review Committee

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the ~~(Company Nuclear Review and Audit Group).~~ OPERATIONS REVIEW COMMITTEE.

AUTHORITY

Station Operations Review Committee

6.5.1.7 The (~~Unit Review Group~~) shall:

Manager, Plant Operations

a. Recommend in writing to the (~~Plant Superintendent~~) approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

~~c. Provide written notification within 24 hours to the (Superintendent of Power Plants) and the (Company Nuclear Review and Audit Group) of disagreement between the (URG) and the (Plant Superintendent); however, the (Plant Superintendent) shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.~~

RECORDS

Station Operations Review Committee

6.5.1.8 The (~~Unit Review Group~~) shall maintain written minutes of each (~~URG~~ SORC) meeting that, at a minimum, document the results of all (~~URG~~ SORC) activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the (~~Superintendent of Power Plants~~ Manager, Nuclear Operations) and the (~~Company Nuclear Review and Audit Group~~ Operations Review Committee).

~~6.5.2 COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNRAG)~~

~~FUNCTION~~

~~6.5.2.1 The (Company Nuclear Review and Audit Group) 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~~

ADD INSERT

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h. quality assurance practices

i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

COMPOSITION

6.5.2.2 The (CNRAG) shall be composed of the:

Director:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the (CNRAG) Director to serve on a temporary basis; however, no more than two alternates shall participate as voting members in (CNRAG) activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the (CNRAG) Director to provide expert advice to the (CNRAG).

MEETING FREQUENCY

6.5.2.5 The (CNRAG) 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 The minimum quorum of the (CNRAG) necessary for the performance of the (CNRAG) review and audit functions of these Technical Specifications shall consist of the Director or his designated alternate and (at least 4 CNRAG) members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

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REVIEW

6.5.2.7 The (CNRAG) 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 of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. 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 (Unit Review Group).

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the (CNRAG). These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

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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 (CNRAG) or the (Vice President 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 an outside fire protection 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.

AUTHORITY

6.5.2.9 The (CNRAG) shall report to and advise the (Vice President 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 (CNRAG) activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each (CNRAG) meeting shall be prepared, approved and forwarded to the (Vice President-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 (Vice President-Operations) within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the (Vice President-Operations) and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

 (INSERT)

6.5.2 OPERATIONS REVIEW COMMITTEE (ORC)

FUNCTION

6.5.2.1 The Operations Review Committee shall function to provide independent review of designated activities in the following areas:

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. administrative controls and quality assurance practices
i. emergency preparedness
j. other appropriate fields associated with the unique characteristics of CPSES

COMPOSITION

6.5.2.2 The Operations Review Committee shall be composed of five or more members of whom no more than a minority are members having line responsibility for operations at CPSES. Members will be appointed by the TUGCO Vice President, Nuclear.

ALTERNATES

6.5.2.3 Alternate members may be appointed by the Vice President, Nuclear or by the regular members subject to the approval of the Vice President, Nuclear. In the regular members' absence, alternate members may act with the full authority of regular members. Alternate members must be designated in advance and their participation shall be restricted to legitimate absences of regular members. The Secretary, ORC, will maintain a list of current members and their alternates.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the ORC Chairman to provide expert advice to the ORC.

MEETING FREQUENCY

6.5.2.5 The ORC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading at at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum for a formal or for an unscheduled meeting of the ORC shall consist of not less than a majority of the appointed members (or their alternates) including the chairman or his designated alternate. No more than a minority of the quorum shall have line responsibility for operation of CPSES.

REVIEW

6.5.2.7 The Operations Review Committee shall be responsible for the independent review of the following:

a. Written evaluations of modifications to CPSES facilities which involve changes to safety-related systems, changes to procedures which have been identified by CPSES as safety-related, and those tests or experiments not described in the CPSES FSAR and which are completed without prior NRC approval under provisions of 10CFR50.59(a)(1). This review is to verify that such changes, tests, or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59(a)(2).
b. Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an unreviewed safety question as defined in 10CFR50.59(c).
c. Proposed changes to the Technical Specifications and license amendments will be reviewed prior to submittal to the NRC to verify the advisability of such changes.
d. Violations, deviations, and reportable events, which require reporting to the NRC in writing such as the following:

    1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
    2. Significant operating abnormalities or deviations from normal or expected performance of plant nuclear safety-related structures, systems, or components.
    3. Reportable events which require reporting to the NRC in writing within 24 hours, as defined in the plant Technical Specifications.

REVIEW (continued)

The purpose for the reviews of the above events is to verify the adequacy of any investigations of such events and the corrective actions taken to prevent or reduce the probability of recurrence of such events.

e. All reports of audits conducted by the NRC and selected reports of audits conducted by TUGCO Quality Assurance Division, and the CPSES Quality Assurance Station personnel to verify compliance with the CPSES Operations Administrative Control and Quality Assurance Plan and CPSES license requirements, and to detect trends that may be detrimental to safe station operation.

f. Reports/meeting minutes of the SORC and selected reports and documentation, as specified by the ORC, which pertain to reviews of safety-related activities conducted by the CPSES Operating Staff. The purpose is to verify the adequacy of the on-site review process and to detect trends that may be detrimental to safe operations.

g. Other safety-related matters deemed appropriate by ORC members or referred to the ORC by the Station Operations Review Committee or by other functional organization units within CPSES, TUGCO, or TUSI.

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.

AUDITS

6.5.2.8 Audits of designated CPSES activities shall be performed under the cognizance of the ORC. These audits shall encompass areas such as the following:

a. The conformance of CPSES 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 station staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in CPSES 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", 10CFR50, at least once per 24 months.

AUDITS (continued)

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 CPSES operation considered appropriate by the ORC or the Vice President, Nuclear.
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 an outside fire protection firm.
j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside fire consultant at intervals no greater than 3 years.

AUTHORITY

6.5.2.9 The Operations Review Committee shall report to and advise the Executive Vice President and General Manager, TUGCO on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of ORC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each ORC meeting shall be prepared, approved and forwarded to the Executive Vice President and General Manager, TUGCO 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 Executive Vice President and General Manager, TUGCO within 14 days following completion of the review.
c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Executive Vice President and General Manager, TUGCO and to the management positions responsible for the areas audited within 14 days after completion of the audit report.

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 SORC ~~(URG)~~ and submitted to the ~~(CNRAG)~~ ORC and the ~~(Superintendent of Power Plants)~~ 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 ~~(Superintendent of Power Plants)~~ Manager, Nuclear Operations and the ~~(CNRAG)~~ ORC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC ~~(URG)~~. 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 ~~(CNRAG)~~ ORC and the ~~(Superintendent of Power Plants)~~ 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 implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the ~~(URG)~~ SORC and approved by the ~~(Plant Superintendent)~~ Manager, Plant Operations prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the ~~(URG)~~ SORC and approved by the ~~(Plant Superintendent)~~ Manager, Plant Operations within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include (the ~~recirculation~~ containment spray, safety injection, chemical and volume control, ~~gas stripper~~ residual heat removal, and ~~hydrogen recombiners~~ primary sampling). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) ~~Integrated leak test requirements~~ Operating Pressure leak test inspections 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 ~~(PWRs only)~~

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, which shall include 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, and

(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. Backup Method for Determining Subcooling Margin
(PWRs with a single channel of monitoring instrumentation)

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. 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 of personnel,

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.5

Written procedures shall be established, implemented and maintained to limit the working hours of plant staff who perform safety-related functions (e.g., licensed senior operators, licensed operators, radiation protection personnel, auxiliary operators and key maintenance personnel). Adequate shift coverage shall be maintained without routine heavy use of overtime. Any deviations from these written procedures shall be authorized by the Manager, Plant Operations and the basis documented for granting the deviation. These procedures shall be consistent with NRC guidelines in NUREG 0737 on "Shift Manning".

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP 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 FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details 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 REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance); waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

~~b. The results of the core barrel movement monitoring activities performed during the report period. (CE units only).~~

~~c. (Any other unit unique reports required on an annual basis.)~~

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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 or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty 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 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

RADIAL PEAKING FACTOR LIMIT REPORT ~~(W only)~~

6.9.1.10 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulations, 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 will 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

Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of 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 required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the ~~(URG)~~ SORC and the ~~(GNRAG).~~ ORC

l. Records of the service lives of all snubbers listed in 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.

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~~ REQUIREMENTS OF 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 RWP.

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 SUPERVISOR ~~Shift Foreman~~ on duty and/or RADIATION PROTECTION ~~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 and the maximum allowable stay time for individuals in that area. 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 may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

*RADIATION PROTECTION ~~Health Physics~~ personnel or personnel escorted by RADIATION PROTECTION ~~Health Physics~~ personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

**Measurement made at 18" from source of radioactivity.

NUREG-0472
REVISION 2

RADIOLOGICAL EFFLUENT TECHNICAL

SPECIFICATIONS FOR PWR'S

JULY 1979

JAN 1 1980

FEB 1 1980

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

NOTE: Add 3/4.3.3.9 and 3/4.3.3.10 with appropriate page numbers to Index section for Monitoring Instrumentation and its Bases; also add 5.1.3 and 5.1.4 to Section 5.0 Index.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

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BASES

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CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

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

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

DOSE EQUIVALENT I-131

1.19 The 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."

SOURCE CHECK

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

PROCESS CONTROL PROGRAM (PCP)

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

SOLIDIFICATION

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.32 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 of gaseous and liquid effluent monitoring alarm/trip setpoints.

WASTE PROCESSING
GASEOUS ~~RADWASTE TREATMENT~~ SYSTEM

WASTE PROCESSING
1.33 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.

VENTILATION EXHAUST TREATMENT SYSTEM

1.34 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.

PURGE - PURGING

1.35 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.

VENTING

1.36 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.

MEMBER(S) OF THE PUBLIC

1.37 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their normal job function, occassionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

SITE BOUNDARY

1.38 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

UNRESTRICTED AREA

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

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.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

10

3.3.3.~~9~~ 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, without delay ~~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. If the inoperable instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluents Release Report why the inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	(1)	28
~~b. Steam Generator Blowdown Effluent Line~~	~~(1)~~	~~29~~
c. Turbine Building (Floor Drains) Sumps Effluent Line	(1)	30
2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	(1)	30
~~b. Component Cooling Water System Effluent Line~~	~~(1)~~	~~30~~
~~3.~~ CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
~~a. Steam Generator Blowdown Effluent Line~~	~~(1)~~	~~29~~
~~b. Turbine Building Sumps Effluent Line~~	~~(1)~~	~~30~~
4. FLOW RATE MEASUREMENT DEVICES *		
a. Liquid Radwaste Effluent Line	(1)	31
~~b. Circulating Water Discharge Canal~~	~~(1)~~	~~31~~
~~c. Steam Generator Blowdown Effluent Lines~~	~~(1)~~	~~31~~

* Pump curves may be used to estimate flow; in such cases ACTION 31 is not required.

TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
~~5.~~ RADIOACTIVITY RECORDERS (*)		
a. Liquid Radwaste Effluent Line	(1)	33
b. Steam Generator Blowdown Effluent Line	(1)	34
~~6.~~ TANK LEVEL INDICATING DEVICES (for tanks outside plant buildings)		
~~a. ________~~	~~(1)~~	~~32~~
~~b. ________~~	~~(1)~~	~~32~~
~~c. ________~~	~~(1)~~	~~32~~
~~d. ________~~	~~(1)~~	~~32~~

~~(*Required only if alarm/trip set point is based on recorder-controller)~~

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - 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 individuals ~~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 29 - 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 or 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 30 - 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 24 ~~8~~ hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml, Cs-137.

ACTION 31 - 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.

~~ACTION 32 - 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.~~

TABLE 3.3-12 (Continued)

TABLE NOTATION

~~ACTION 33~~ - ~~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 14 days provided the gross radioactivity level is determined at least once per 4 hours during actual releases.~~

~~ACTION 34~~ - ~~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 the gross radioactivity level is determined at least once per 4 hours during actual release.~~

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line	D (5)	P	R(3)	Q(1)
~~b. Steam Generator Blowdown Effluent Line~~	~~D~~	~~M~~	~~R(3)~~	~~Q(1)~~
c. Turbine Building (Floor Drains) Sumps Effluent Line	D (5)	M	R(3)	Q(1)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D (5)	M	R(3)	Q(2)
~~b. Component Cooling Water System Effluent Line~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~
~~3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR~~				
~~a. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Turbine Building Sumps Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~

TABLE 4.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
~~b. Steam Generator Blowdown Effluent Line~~	~~D(4)~~	~~N.A.~~	~~R~~	~~Q~~
~~c. Circulating Water Discharge Canal~~	~~D(4)~~	~~N.A.~~	~~R~~	~~Q~~
~~5.~~ RADIOACTIVITY RECORDERS				
~~a. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Liquid Radwaste Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~6. TANK LEVEL INDICATING DEVICES (for tanks outside the building)~~				
~~a. ______~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~b. ______~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~c. ______~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~d. ______~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~

TABLE 4.3-12 (Continued)

TABLE NOTATION

* During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL 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. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL 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. Circuit failure.
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 intial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)

(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.

(5) CHANNEL CHECK shall consist of verifying indication of count rate and does not require flow through the monitor.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.~~10~~ 11 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, without delay ~~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. If the inoperable instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluents Release Report why the inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. PLANT VENT STACK ~~WASTE GAS HOLDUP SYSTEM~~			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	(1)	*	~~35~~ 37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Effluent System Flow Rate Measuring Device	(1)	*	36
e. Sampler Flow Rate Measuring Device	(1)	*	36
~~2A. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)~~			
~~a. Hydrogen Monitor~~	~~(1)~~	~~**~~	~~39~~
~~b. Hydrogen or Oxygen Monitor~~	~~(1)~~	~~**~~	~~39~~
2B. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems not designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor	~~(2)~~ 1	**	40
b. ~~Hydrogen or~~ Oxygen Monitor	~~(2)~~ 1	**	40

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~3.~~	~~CONDENSER EVACUATION SYSTEM~~			
~~a.~~	~~Noble Gas Activity Monitor~~	~~(1)~~	~~*~~	~~37~~
~~b.~~	~~Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c.~~	~~Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d.~~	~~Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~e.~~	~~Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
4.	VENT ~~HEADER~~ DUCT SYSTEM			
a.	Noble Gas Activity Monitor	(1)	*	37
~~b.~~	~~Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c.~~	~~Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d.~~	~~Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~e.~~	~~Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~5.~~	~~CONTAINMENT PURGE SYSTEM~~			
~~a.~~	~~Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release~~	~~(1)~~	~~*~~	~~38~~
~~b.~~	~~Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c.~~	~~Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~5. CONTAINMENT PURGE SYSTEM (Continued)~~			
~~e.~~ Flow Rate Monitor	(1)	*	36
~~f. Sampler Flow Rate Monitor~~	(1)	*	36
~~6. AUXILIARY BUILDING VENTILATION SYSTEM~~			
~~a.~~ Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
~~e.~~ Sampler Flow Rate Monitor	(1)	*	36
~~7. FUEL STORAGE AREA VENTILATION SYSTEM~~			
~~a.~~ Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
~~e.~~ Sampler Flow Rate Monitor	(1)	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~8.~~ RADWASTE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36
~~9.~~ STEAM GENERATOR BLOWDOWN VENT SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

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

~~ACTION 25 - 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 36 - 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~~ 24 hours.

ACTION 37 - 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 38 - 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 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this waste gas holdup system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.~~

ACTION 40 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue ~~for up to 14 days. With (two) channels inoperable, be in at least HOT STANDBY within 6 hours.~~ provided another method of ascertaining oxygen or hydrogen concentrations such as grab sample analyses, is implemented to provide measurements at least once per 4 hours.

ACTION 41 - 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.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. ~~WASTE GAS HOLDUP SYSTEM~~ PLANT VENT STACK					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	~~P~~ 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 Monitor	~~P~~ D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	~~P~~ W	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(4)	~~M~~ N/A	**
~~b. Hydrogen Monitor (alternate)~~	~~D~~	~~N.A.~~	~~Q(4)~~	~~M~~	~~**~~
c. Oxygen Monitor	D	N.A.	Q(5)	~~M~~ N/A	**
~~d. Oxygen Monitor (alternate)~~	~~D~~	~~N.A.~~	~~Q(5)~~	~~M~~	~~**~~

* Also prior to any release from waste gas holdup system or containment purge

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
~~3. CONDENSER EVACUATION SYSTEM~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
4. VENT ~~HEADER~~ DUCT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~d. Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
~~CONTAINMENT PURGE SYSTEM~~					
~~a.~~ Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	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 Monitor	D	N.A.	R	Q	*
~~e.~~ Sampler Flow Rate Monitor	D	N.A.	R	Q	*
~~AUXILIARY BUILDING VENTILATION SYSTEM~~					
~~a.~~ Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
~~e.~~ Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
~~FUEL STORAGE AREA VENTILATION SYSTEM~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
~~RADWASTE AREA VENTILATION SYSTEM~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
9. ~~STEAM GENERATOR BLOWDOWN VENT~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-13 (Continued)

TABLE NOTATION

* At all times.

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

(1) The CHANNEL FUNCTIONAL 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. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL 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. Circuit failure.
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. (Operating plants may substitute previously established calibration procedures for this requirement.)

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

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

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

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

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 to unrestricted areas (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 to unrestricted areas exceeding the above limits, ~~immediately~~ restore the concentration to within the above limits. without delay

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.~~

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of radioactivity analysis shall be used in accordance with the 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.

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] Liquid Waste Processing System	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			~~P-32~~	~~1×10^{-6}~~
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e] Turbine Building Floor Drains	D ~~Continuous[e]~~ Grab sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	D ~~Continuous[e]~~ Grab sample	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			~~P-32~~	~~1×10^{-6}~~
	D ~~Continuous[e]~~ Grab sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample 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):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

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),

2.22×10^6 is the number of transformations per minute per microcurie,

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. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 4.11-1 (Continued)

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 collected continuously in proportion to the rate of flow of the effluent stream.~~ Prior to analyses, all samples taken for the composite shall be throughly 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. Nuclides which are below the LLD for the analyses shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

g. Analyses for Fe-55 will be performed for a period of two years. These analyses will then be discontinued unless this radionuclide is shown to potentially contribute as much as 10% of the total dose due to effluents released from the site to unrestricted areas.

RADIOACTIVE EFFLUENTS

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 reduce the releases ~~of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrem to the total body and 10 mrem to any organ.~~ and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits. (This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.*)

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

*Applicable only if drinking water supply is taken from the receiving water body.

RADIOACTIVE EFFLUENTS

LIQUID WASTE ~~TREATMENT~~ PROCESSING

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid ~~radwaste treatment~~ waste processing system components as specified in the ODCM 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 each reactor at the site to unrestricted areas (see Figure 5.1-4) when ~~averaged~~ accumulated 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, during periods in which discharge of liquid effluents containing radioactive materials to unrestricted areas occurs or is expected.

~~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 ___ 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.~~

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary ~~of the following~~ tank~~s~~ shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

~~a.~~ ________
~~b.~~ ________
~~c.~~ ________
~~d. Outside temporary tank~~

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive without delay material in any of the above listed tanks exceeding the above limit, ~~immediately~~ suspend all additions of radioactive material to the tank and within 48 hours either reduce the tank contents to within the limit, or provide prompt notification to the

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

Commission pursuant to Specification 6.9.1.12. The written followup shall include a schedule and description of activities planned and/or taken to reduce the tank contents to within the above limits.

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.

*Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (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~~ I-131, tritium, and for all radioactive materials in particulate form ~~and radionuclides (other than noble gases)~~ 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~~ without delay 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 radioactive materials, other than noble gases, 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 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 (Gaseous)	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge	P Each Purge[b] Grab Sample (Gaseous)	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. ~~(List other release points where gaseous effluents are discharged from the facility)~~ Plant Vent	M[b,c,e] Grab Sample (Gaseous)	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
~~D. All Release Types as listed in A, B, C above.~~	Continuous[f]	W[d] Charcoal Sample	I-131	1×10^{-12}
			~~I-133~~	~~1×10^{-10}~~
	Continuous[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	~~Continuous[f]~~	~~Noble Gas Monitor~~	~~Noble Gases Gross Beta & Gamma~~	~~1×10^{-6}~~

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample 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):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

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),

2.22×10^6 is the number of transformations per minute per microcurie,

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. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed 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. Nuclides which are below the LLD for the analyses shall be reported as "less than" the nuclide's LLD and shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

to areas at and beyond the SITE BOUNDARY

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.

~~(The dose design objectives shall be reduced based on predicted noble gas releases from the turbine building if effluent sampling is not provided. The dose design objectives shall also be reduced based on expected public occupancy of areas, e.g., beaches and visitor centers within the site boundary.)~~

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 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 reduce the releases ~~of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose is within (10) mrad for gamma radiation and (20) mrad for beta radiation.~~

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.

and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with LCO 3.11.2.2.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131 ~~RADIOIODINES~~, TRITIUM, AND RADIOACTIVE MATERIALS IN PARTICULATE FORM, ~~AND RADIONUCLIDES OTHER THAN NOBLE GASES~~

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from I-131 ~~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, from the site (see Figure 5.1-3) to areas at and beyond the SITE BOUNDARY 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 I-131 ~~radioiodines~~, tritium or radioactive materials in particulate form, ~~or radionuclides (other than noble gases)~~ 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 reduce the releases ~~of radioiodines and radioactive materials in particulate form, and radionuclides (other than nobles gases) with half-lives greater than 8 days in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters~~, ~~so that the cumulative dose or dose commitment to an individual from these releases is within (15) mrem to any organ.~~ and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with LCO 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.

as specified in the ODCM

RADIOACTIVE EFFLUENTS

GASEOUS ~~RAD~~WASTE ~~TREATMENT~~ PROCESSING

to areas at and beyond the SITE BOUNDARY

LIMITING CONDITION FOR OPERATION

PROCESSING

3.11.2.4 ~~The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE.~~ The appropriate portions of the GASEOUS ~~RAD~~WASTE ~~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 each reactor at the site (see Figure 5.1-3), when accumulated ~~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 each reactor at the site (see Figure 5.1-3) when ~~averaged~~ accumulated 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 site shall be projected at least once per 31 days, in accordance with the ODCM, during periods in which discharge of gaseous effluents containing radioactive materials to areas at and beyond the SITE BOUNDARY occurs or is expected.

~~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 ___ minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.~~

EXPLOSIVE GAS MIXTURE (Systems designed to withstand a hydrogen explosion)

LIMITING CONDITION FOR OPERATION

~~3.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be limited to less than or equal to 4% by volume.~~

~~APPLICABILITY: At all times.~~

~~ACTION:~~

~~a. With the concentration of hydrogen or oxygen in the waste gas holdup system exceeding the limit, restore the concentration to within the limit within 48 hours.~~

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

SURVEILLANCE REQUIREMENTS

~~4.11.2.5 The concentration of hydrogen or 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 or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.~~

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE (Systems not designed to withstand a hydrogen explosion)

LIMITING CONDITION FOR OPERATION

3.11.2.5A The concentration of hydrogen and/or oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen and/or 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 hydrogen and/or oxygen to within the limit within 48 hours.

b. With the concentration of hydrogen and/or oxygen in the waste gas holdup system greater than 4% by volume, ~~immediately~~ without delay suspend all additions of waste gases to the system and reduce the concentration of hydrogen and/or oxygen to less than or equal to 2% within ~~one~~ 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5A The concentrations of hydrogen and/or 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/or oxygen monitors required OPERABLE or alternative provisions specified by Table 3.3-13 of Specification 3.3.3.10.

EXPLOSIVE GAS MIXTURE (Hydrogen rich systems not designed to withstand a hydrogen explosion)

LIMITING CONDITION FOR OPERATION

3.11.2.5B 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 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume within one hour.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5B The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 200,000 curies 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, without delay ~~immediately~~ suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, or provide prompt notification to the Commission, in lieu of any other report, pursuant to Specification 6.9.1.8. The written report shall include a description of activities planned and/or taken to reduce the tank contents to within the above 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~~ 31 ~~hours~~ days 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~~ 92 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.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

limitations of 40 CFR 190 shall be achieved by complying with the limits of Technical Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3

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 a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, 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 other sections of this technical specification.

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.

pursuant to Specification 6.9.2 and limit subsequent releases such that the dose or dose commitment to any real individual is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over the calendar year. This Special Report shall include an analysis which demonstrates that radiation exposures to any real individual from all effluent pathways and direct radiation are less than the 40 CFR 190 Standard. Otherwise obtain a variance from the Commission to permit releases which exceed the 40 CFR 190 Standard

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:

or after confirmation, whichever is later,

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.

confirmed*, measured

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

* A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis, but in any case within 60 days.

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.

Footnotes are added following table

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [a]

Exposure Pathway and/or Sample	Number of Samples and Sample Locations~~**~~ (b)	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
Radioiodine and Particulates	(Locations 1-~~8~~ 9)	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.	Radioiodine canister. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION	(Locations ~~8-45~~ 10-52) $\geq$ 2 dosimeters or $\geq$ 1 instrument for continuously measuring and recording dose rate at each location.	At least once per 31 days. or At least once per 92 days. (Read-out frequencies are determined by type of dosimeters selected.)	Gamma dose. At least once per 31 days. or Gamma dose. At least once per 92 days.

~~** Sample locations are given on the figure and table in the ODCM.~~

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM a

Exposure Pathway and/or Sample	Number of Samples and Sample Locations~~**~~ (b)	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	(Locations 53 - 55 ~~A6 and A7~~)	Monthly grab ~~Composite*~~ sample ~~collected over a period of ≤ 31 days.~~	Gamma isotopic analysis monthly of each ~~composite~~ sample. Tritium analysis of composite sample at least once per 92 days.
b. Ground	(Locations 56 ~~A8~~ and 57 ~~A9~~)	At least once per 92 days.	Gamma isotopic and tritium analyses of each sample.
c. Drinking	(Locations 57 ~~50-52~~)	Monthly grab ~~Composite*~~ sample ~~collected over a period of ≤ 14 days, if I-131 analysis is performed; or~~	I-131 analysis of each ~~composite~~ sample; and
		~~Composite* sample collected over a period of ≤ 31 days.~~	Gross beta and gamma isotopic analysis of each ~~composite~~ sample. Tritium analysis of composite sample at least once per 92 days.
d. Sediment from Shoreline	(Locations 58 and 59 ~~53~~)	At least once per 184 days.	Gamma isotopic analysis of each sample.

~~*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.~~
~~**Sample locations are shown on the figure in the ODCM.~~

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [a]

Exposure Pathway and/or Sample	Number of Samples and Sample Locations ~~**~~ (b)	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	(Locations ~~54-57~~ 60-63)	At least once per 15 days when animals are on pasture; at least once per 31 days at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish ~~and Invertebrates~~	(Locations ~~58~~ 64 and ~~59~~ 65)	~~One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species:~~ Semi-annual (184 days) of recreationally important species that are collected during each sample time ~~1.~~ ____ ~~2.~~ ____	Gamma isotopic analysis on edible portions.
c. Food Products or Broadleaf Vegetation	(Locations ~~60-62~~ 66-71)	At time of harvest or Monthly. One sample of ~~each of the following classes of food products~~: Vegetables grown in site vicinity. ~~1.~~ ____ ~~2.~~ ____ ~~3.~~ ____	Gamma isotopic analysis on edible portion.
	~~(Location 63)~~	~~At time of harvest. One sample of broad leaf vegetation.~~ Indicator: Broad leaf vegetation grown offsite near unrestricted area boundary in highest calculated annual average ground level D/Q sector. Monthly or during harvest.[d,e] Control: Similar broad leaf vegetation grown 10 to 20 miles distant in least prevalent wind direction. Monthly or during harvest.[d,e]	I-131 analysis.

~~**Sample locations are shown on the figure in the ODCM.~~

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATION

a. Deviations are permitted from the specified sampling schedule if specimen is unobtainable due to hazardous conditions, inclement weather, seasonable unavailability, malfunctions of equipment or other legitimate reason.

b. Sample locations are shown in the ODCM.

c. For the purpose of this table, a thermoluminescent dosimeter (TLD) is considered to be one chip, and two or more chips in a packet or a multiple area TLD are considered as two or more dosimeters.

d. Where access to edible green leafy vegetables is not possible, non-edible plants with similar leaf characteristics from the same sector may be substituted.

e. When broad leaf vegetation grown offsite near unrestricted area boundary is not available, a garden census will be conducted.

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)	Broadleaf Vegetation ~~Food Products~~ (pCi/Kg, wet)
H-3	2×10^4 (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2	0.9		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			3×10^2	

(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}				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1[b]	7×10^{-2}		1	60	
Cs-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{-2}	150	18	80	180
Ba-140	60			60		
La-140	15			15		

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample 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 picocurie per unit mass or volume),

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 transformation 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 (for environmental samples, not plant effluent 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 shall 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 an a posteriori (after-the-fact) limit for a particular measurement.

TABLE 4.12-1 (Continued)

TABLE NOTATION

b. LLD for drinking water.

c. Other peaks 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 permanent 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. ~~(For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three 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).~~

a revised figure in the ODCM reflecting the new locations shall be submitted to the Commission as an inclusion in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.12.

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~~ 60 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~~ calendar year between the dates of ~~(June 1 and October 1)~~ April 1 and August 1 using ~~that information which will provide the best results, such as by a~~ any of the following methods: door-to-door survey, aerial survey, visual 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.

RADIOLOGICAL ENVIRONMENTAL MONITORING

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, or a program selected by the licensee in the absence of an approved program.

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.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT 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.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT 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.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents 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 ~~rad~~waste ~~treatment~~ processing systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE PROCESSING ~~TREATMENT~~

The OPERABILITY of the liquid ~~rad~~waste processing ~~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 ~~rad~~waste processing ~~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.

3/4.11.2 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 an infant via the cow-milk-infant pathway to less than or equal to 1500 mrem/year for the nearest cow to the plant.

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.

IODINE-131 TRITIUM AND

3/4.11.2.3 DOSE - ~~RADIOIODINES~~, RADIOACTIVE MATERIALS IN PARTICULATE FORM ~~AND RADIONUCLIDES OTHER THAN NOBLE GASES~~

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 radionuclides other than noble gases 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 ~~RAD~~WASTE ~~TREATMENT~~ PROCESSING

The OPERABILITY of the GASEOUS ~~RAD~~WASTE ~~TREATMENT~~ Processing 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, automatic diversion to recombiners, or injection of dilutants 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 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

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, visual 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.

BASES

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.

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~~ unrestricted area boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 135 feet.

b. Nominal inside height = 195 feet.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 12 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 2,985,000 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 50 psig and a temperature of 280°F.

This figure shall consist of a map of the site area and provide, at a minimum, the information described in Section (2.1.2) of the FSAR and meteorological tower location.

EXCLUSION AREA

FIGURE 5.1-1

This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

LOW POPULATION ZONE

FIGURE 5.1-2

This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where gaseous effluents are released. If on-site land areas subject to radioactive materials in gaseous waste are utilized by the public for recreational or other purposes, then these areas shall be identified by occupancy factors and the licensee's method of occupancy control. The figure shall be sufficiently detailed to allow identification of structures and release point elevations, and areas within the site boundary that are accessible by members of the general public. See NUREG-0133 for additional guidance.

SITE BOUDARY FOR GASEOUS EFFLUENTS

FIGURE 5.1-3

This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where liquid effluent leaves the site. If on-site water areas containing radioactive wastes are utilized by the public for recreational or other purposes, the points of release to these water areas shall be identified. The figure shall be sufficiently detailed to allow identification of structures near the release points and areas within the site boundary where ground and surface water is accessible by members of the general public. See NUREG-0133 for additional guidance.

SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

ALL STS-I

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.5.1 ~~UNIT REVIEW GROUP (URG)~~ STATION OPERATIONS REVIEW COMMITTEE (SORC)

RESPONSIBILITIES

6.5.1.6 The ~~URG~~ Station Operations Review Committee (SORC) shall be responsible for:

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the ~~(Superintendent of Power Plants)~~ Vice President-Nuclear and to the ~~(Company Nuclear Review and Audit Group).~~ Operations Review Committee

l. Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and major changes to the ~~rad~~waste ~~treatment~~ processing systems.

6.5.2 ~~COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNRAG)~~ OPERATIONS REVIEW COMMITTEE

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the ~~(CNRAG).~~ ORC These audits shall encompass:

l. The radiological environmental monitoring program and the results thereof at least once per 12 months.

m. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

n. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

~~o. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, December 1977 at least once per 12 months.~~

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. ~~Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, December 1977.~~ Conduct of a program of Interlaboratory Comparison of radiological analyses.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3/]

6.9.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

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[3/]

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.

3/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

Calculating the dose to the likely most exposed member of the public from liquid and gaseous effluents.

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 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 period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports.~~ ~~The~~ meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 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 Operatin. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

calendar year

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 (specifiy whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

in accordance with Specification 3.11.4

Historical annual-average meteorology or

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 the 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 bases of PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience 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 by the (Unit Review Group).

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12

~~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

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. Cause(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.

g. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specifications 3.11.1.1 or 3.11.2.1.

6.10 RECORD RETENTION

6.10.2

1. Records of analyses required by the radiological environmental monitoring program.

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 ~~(URG)~~. SORC.

2. Shall become effective upon review and acceptance by the ~~(URG)~~. SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluents Release ~~(Monthly Operating~~ Report within ~~90 days~~ 6 months 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 ~~(URG)~~. SORC

2. Shall become effective upon review and acceptance by the ~~(URG)~~. SORC

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE ~~TREATMENT~~ PROCESSING SYSTEMS (Liquid, Gaseous and solid)

6.15.1 Licensee initiated major changes to the radioactive waste processing systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the ~~Monthly Operating~~ Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~(Unit Review Group)~~ SORC. The discussion of each change shall contain:

   a. A summary of the evaluation 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 differ 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 ~~(URG)~~ SORC.

2. Shall become effective upon review and acceptance by the ~~(URG)~~ SORC.